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FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix A
Environmental Justice Analysis

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Appendix A
Environmental Justice Analysis

A.1 Introduction

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low Income Populations,* directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations. Executive Order 12898 also directs the Administrator of the Environmental Protection Agency to convene an interagency Federal Working Group on Environmental Justice. The Working Group is directed to provide guidance to Federal agencies on criteria for identifying disproportionately high and adverse human health or environmental effects on minority and low-income populations. The Working Group has not yet issued the guidance directed by Executive Order 12898, although it has developed working draft definitions. The definitions used in this analysis are based on the draft working definitions. Further, in coordination with the Working Group, DOE is in the process of developing internal guidance on implementing the Executive Order. Because both the Working Group and DOE are still in the process of developing guidance, the approach taken in this analysis may depart somewhat from whatever guidance is eventually issued.

This appendix addresses environmental justice for the acceptance of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. Analyses of environmental justice concerns are provided in three areas: (1) potential ports of entry, (2) potential transportation routes from candidate ports of entry to interim management sites, and (3) areas surrounding potential interim management sites. These analyses lead to the conclusion that the alternatives analyzed in this Environmental Impact Statement (EIS) would result in no disproportionately high and adverse effects on minority populations or low-income communities surrounding the candidate ports, transport routes, or interim management sites.

A.2 Concerns and Definitions

Public comments show a widespread concern for public health and safety because spent nuclear fuel is radioactive. Two related environmental documents (DOE, 1995 and DOE, 1994) have been published recently which address the safety and potential health issues due to transportation and storage of spent nuclear fuel. Analyses of radiological health effects in those documents as well as this EIS (see Chapter 4) demonstrate that the expected health effects are small. In the case of spent nuclear fuel from foreign research reactors, no fatalities are expected due to radiological exposure or traffic accidents. No significant health effects are expected for the general population. Consequently, there would be no disproportionately high or adverse human health effects imposed on any population segment. In the sections below, minority and low-income populations are identified in the areas near potential candidate ports of entry, potential interim management sites, and potential transportation routes. The 1990 census data were used in this appendix as the basis of the analysis (DOC, 1992). This allows equal comparison of data between ports, sites, and routes in different states.

The analysis uses the following draft definitions:

- *Minority* — Individuals classified by the U.S. Bureau of the Census as Negro/Black/African American, Hispanic, Asian and Pacific Islander, American Indian, Eskimo, Aleut, and other non-White persons. The minority population in an affected area is the number of individuals residing in the area who are members of a minority group.

- *Low-Income Community* — An area for which the median household income is 80 percent or below the median household income for the metropolitan statistical area (urban) or county (rural). While "80 percent" is used in this analysis based on definitions used by the U.S. Department of Housing and Urban Development, this percentage may change in the final guidelines under preparation by the Working Group and the Department of Energy.

- *Disproportionately High and Adverse Human Health Effects* — Any human health effects, including cumulative or synergistic effects, on minority or low-income populations which substantially exceed generally accepted levels of risk. This is a draft definition prepared by the Working Group which might change during preparation of the final guidelines.

- *Substantially Affect Human Health* — To impact human health such that there is a measurable incidence of any specific illness, disease, or disorder significantly higher than the national average. This is also a draft definition developed by the Working Group which might change during preparation of final guidelines.

A.3 Environmental Justice in Areas Near the Candidate Ports of Entry

Under normal port activities associated with receipt of the spent nuclear fuel shipments — including harbor activities, unloading the ship, transfer of the spent nuclear fuel casks to truck or train, and movement out of the port city — the dominant radiological impacts were shown in Section 4.2.2 to be the exposures received by the workers in the immediate vicinity of the shipping cask. These individuals include inspectors, shipping cask handlers, and truck drivers. Since the intensity of the radiation from the cask falls off with distance, the doses that might be received by other workers and members of the general population can theoretically be calculated, but would not generally be measurable or distinguishable from natural background radiation.

Potential radiological impacts to people residing near the port are associated with low probability (less than one in a million) accidents that are so severe that the spent nuclear fuel casks rupture and a fire would burn long enough around the cask that some of the radioactive material would be released. In this case, some of the radioactive spent nuclear fuel might be vaporized and lifted by the heat of the fire and carried downwind of the accident location. Where and how far this radioactive material would go before being deposited on the ground would depend on how high the heat from the fire lofts it and the particular weather conditions at the time. Most of this vaporized spent nuclear fuel would be expected to be deposited in the first few miles downwind of the fire but small amounts could be carried out for several tens of miles.

Because the particular details of both the accident conditions (such as the severity of the fire) and the weather conditions at the time of an accident could vary widely, a range of accident conditions and wind directions, wind speeds, and other weather conditions were examined during the evaluation of accident effects (see Section 4.2.2.3). Population impact evaluations were performed for distances out to 80 km (50 mi). Risks of latent cancer deaths were found to range from about 0.003 to 0.000003 latent cancer fatalities (LCF). No latent cancer fatalities would be expected due to accidents at ports.

Containerized spent nuclear fuel casks shipped under the proposed policy would be transferred from the ship at commercial or military ports by personnel experienced in handling containerized cargo, and shipped by truck or rail to one of the five candidate interim management sites. Candidate ports may handle thousands of standard containers each month, unloaded from vessels which can carry up to several thousand casks. The number of casks to be handled would be small in comparison to routine cargo handling, thus having a negligible impact on normal port activities.

As part of the environmental justice analysis, distributions of minority populations and low-income households surrounding candidate ports of entry were estimated from 1990 census data. Although radiological health effects resulting from an accident are calculated at distances up to 80 km (50 mi), the largest radiological effects would usually be expected to occur within roughly a 16-km (10-mi) radius of the accident site. Thus, the distribution of minority and low-income populations is described for circular areas defined by a 16-km (10-mi) radius, centered at each candidate port of entry.

A.3.1 Distribution of Minority Populations Near the Candidate Ports

The minority population characteristics within 16 km (10 mi) of candidate ports of entry for foreign research reactor spent nuclear fuel are presented in Table A-1. For comparison, this table lists minority population features for regions surrounding the ports and for counties which lie partially within the 16-km (10-mi) radius centered at the port. Population characteristics shown in the table were extracted from 1990 census data available from the U.S. Bureau of the Census. The data resolves population characteristics at the "block group level," which generally consists of between 250 and 550 housing units.

With the exception of the Port of Wilmington and 2 military ports, MOTSU (Military Ocean Terminal, SUnny Point) and NWS (Naval Weapons Station) Concord, the percentage of minority populations residing within 16 km (10 mi) of candidate ports exceeds the percentage of minority populations residing within the state. Similarly, the percentage of minority populations residing near the candidate ports exceeds the percentage of minorities residing in counties surrounding the candidate ports. Ports at MOTSU, NWS Concord, Portsmouth, and Newport News are exceptions with larger percentages of minority populations in the surrounding counties.

The racial and ethnic composition of minority populations residing near the candidate ports is shown in Table A-2. In the case of candidate ports located on the east coast, African Americans compose the largest portion of the minority population. Minority populations residing near the candidate ports on the west coast are comprised of a more uniform mixture of African Americans, Asians, Hispanics, and Native Americans. The minority population residing near the Port of Galveston on the Gulf of Mexico is predominately African American and Hispanic.

The spatial distribution of minority populations residing within 16 km (10 mi) of each of the candidate ports is shown in the maps of those ports as presented in Figures A-1 to A-11. The circle shown in each figure has a 16-km (10-mi) radius, centered on the port. As indicated in the legend of each figure, geographical areas are shaded according to the percentage of minority population within the area. Resolution in the figures is at the census block group level. Due to variations in the populations of block groups, the geographical size of any particular block group area is not necessarily proportional to the numerical population. As an example, for ease of enumeration, the U.S. Bureau of the Census may define block group boundaries which actually extend into oceans, bays, or lakes. This allows inclusion in the census data of individuals who reside on boats or offshore houses, a situation particularly predominant in locations such as Galveston (see Figure A-3).

Table A-1 Minority Populations Residing Near the Candidate Ports

Candidate Port	Total Population Residing within 16 km of Port	Minority Population Residing within 16 km of Port	% Minority Population Residing within 16 km of Port	Total Population Residing in Surrounding Counties	Minority Population Residing in Surrounding Counties	% Minority Population Residing in Surrounding Counties	Population Residing in Surrounding State(s)	Minority Population Residing in Surrounding State(s)	% Minority Population Residing in Surrounding State(s)
Charleston, SC:									
Wando Terminal	233,424	82,271	35.2	423,815	145,534	34.3	3,486,703	1,094,792	31.4
NWS Terminal	209,188	73,437	35.1	423,815	145,534	34.3	3,486,703	1,094,792	31.4
Galveston, TX	73,322	36,375	49.6	217,445	72,133	33.2	19,986,510	6,665,631	33.4
Hampton Roads, VA:									
Newport News	430,757	161,317	37.4	1,010,296	400,061	39.6	6,187,358	1,484,501	24.0
Norfolk	681,864	300,179	44.0	1,010,296	400,061	39.6	6,187,358	1,484,501	24.0
Portsmouth	665,700	248,099	37.3	1,010,296	400,061	39.6	6,187,358	1,484,501	24.0
Jacksonville, FL	334,212	123,336	36.9	758,647	203,833	26.9	12,937,926	3,449,230	26.7
MOTSU, NC	7,995	1,496	18.7	50,985	9,835	19.3	6,628,637	1,651,356	24.9
NWS Concord, CA	381,070	110,969	29.1	1,145,248	375,442	32.8	29,760,021	12,666,060	42.6
Portland, OR	356,064	54,704	15.4	1,395,233	138,500	9.9	2,842,321	261,730	9.2
Savannah, GA	155,166	80,361	51.8	344,677	128,206	37.2	9,964,919	3,023,249	30.3
Tacoma, WA	511,575	85,341	16.7	2,123,421	347,788	16.4	4,866,692	637,561	13.1
Wilmington, NC	115,057	27,301	23.7	200,124	44,757	22.4	6,628,637	1,651,356	24.9

Table A-2 Racial and Ethnic Composition for Populations Residing Within 16 km of the Candidate Ports

Candidate Port	Total Pop.	Total Minority Pop.	% Minority Pop.	Amer. Indian, Eskimo or Aleut Pop.	% Amer. Indian, Eskimo or Aleut	Asian or Pacific Islander Pop.	% Asian or Pacific Islander	African Amer. Pop.	% African Amer.	Hispanic Origin Pop.	% Hispanic Origin	Other Race	% Other Race	White	% White
Charleston, SC:															
Wando Terminal	233,424	82,271	35.2	531	0.2	1,804	0.8	76,783	32.9	3,042	1.3	109	0.05	151,143	64.8
NWS Terminal	209,188	73,437	35.1	767	0.4	3,496	1.7	64,961	31.1	4,099	2.0	115	0.05	135,751	64.9
Galveston, TX	73,322	36,375	49.6	262	0.4	1,271	1.7	19,737	26.9	15,012	20.5	90	0.12	36,946	50.4
Hampton Roads, VA:															
Newport News	430,757	161,317	37.4	1,932	0.4	7,872	1.8	138,920	32.3	12,300	2.9	292	0.07	269,441	62.6
Norfolk	681,864	300,179	44.0	2,971	0.4	10,697	1.6	270,729	39.7	15,308	2.2	471	0.07	346,410	50.8
Portsmouth	665,700	248,099	37.3	2,763	0.4	9,612	1.4	221,200	33.2	14,069	2.1	453	0.07	322,815	48.5
Jacksonville, FL	334,212	123,336	36.9	960	0.3	5,456	1.6	108,641	32.5	8,149	2.4	128	0.04	210,815	63.1
MOTSU, NC	7,995	1,496	18.7	32	0.4	7	0.1	1,359	17.0	90	1.1	6	0.08	6,498	81.3
NWS Concord, CA	381,070	110,969	29.1	2,769	0.7	42,788	11.2	26,452	6.9	38,498	10.1	460	0.12	270,102	70.9
Portland, OR	356,064	54,704	15.4	4,086	1.1	12,617	3.5	27,012	7.6	10,632	3.0	355	0.10	301,359	84.6
Savannah GA	155,166	80,361	51.8	370	0.2	1,578	1.0	76,583	49.4	1,734	1.1	94	0.06	74,805	48.2
Tacoma, WA	511,575	85,341	16.7	7,095	1.4	28,321	5.5	32,687	6.4	16,779	3.3	457	0.09	426,231	83.3
Wilmington, NC	115,057	27,301	23.7	534	0.5	587	0.5	25,360	22.0	796	0.7	22	0.02	87,755	76.3

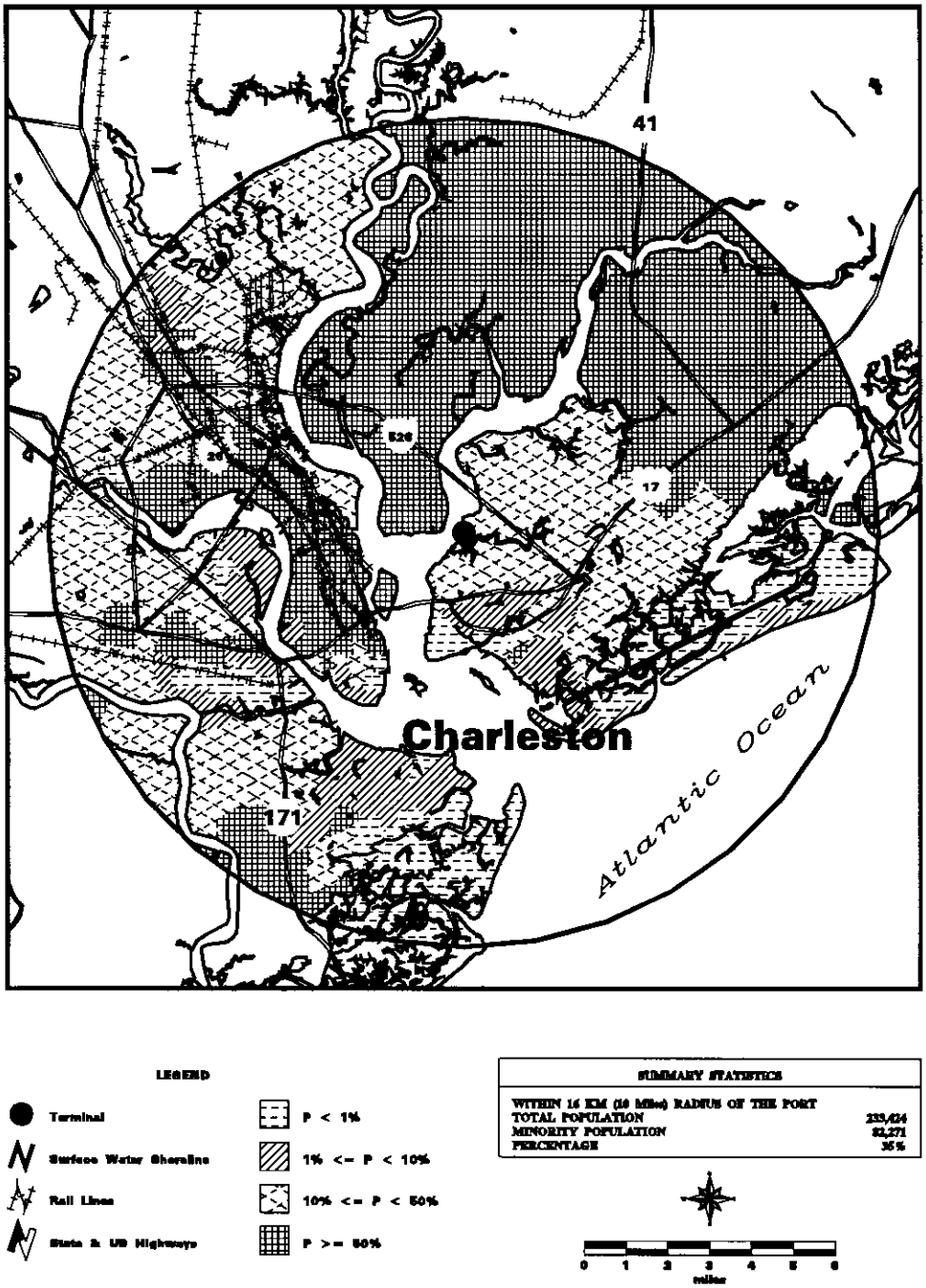


Figure A-1 Distribution of the Minority Population Residing within 16 km of the Wando Terminal, Port of Charleston, South Carolina

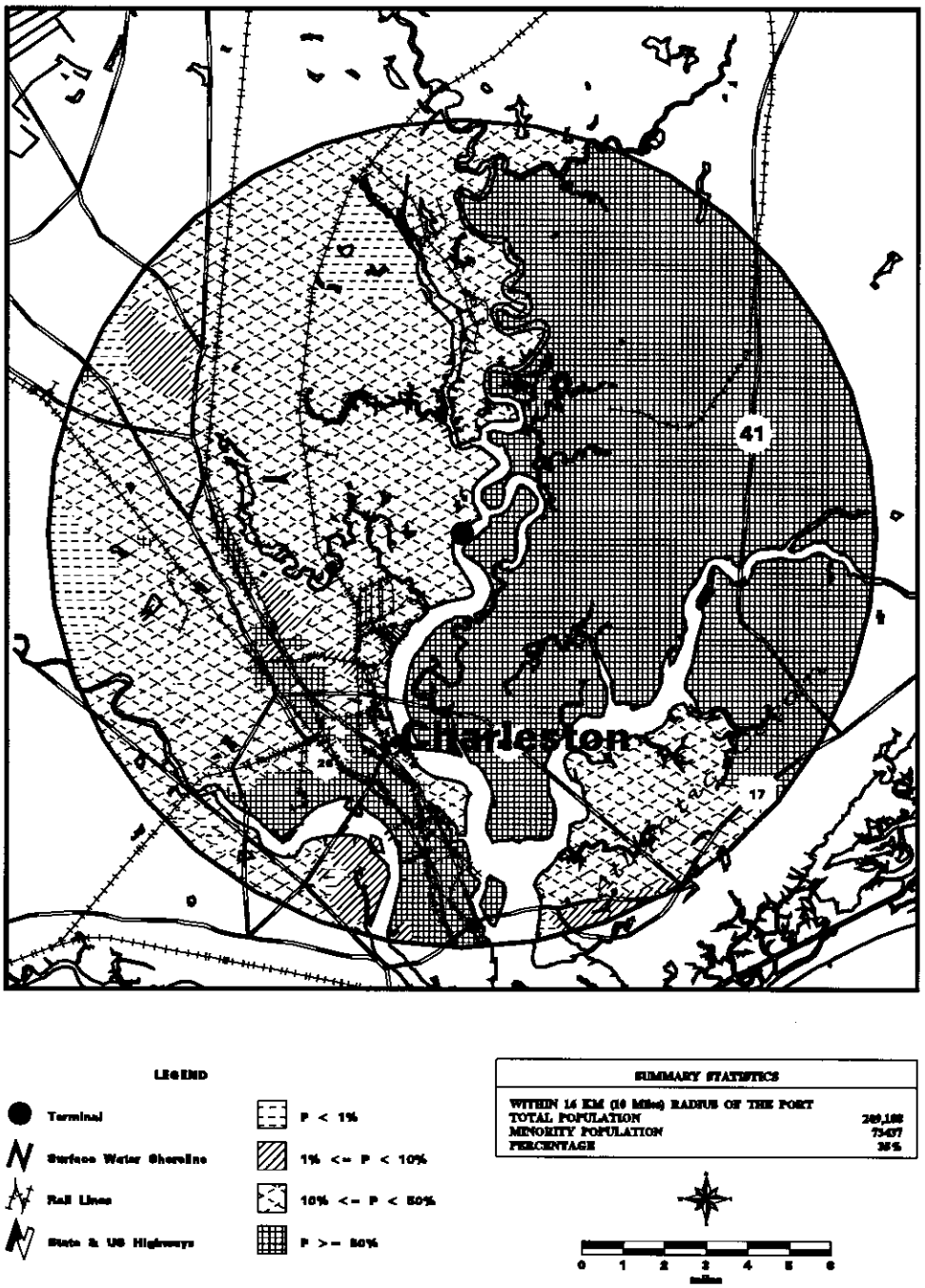


Figure A-2 Distribution of the Minority Population Residing within 16 km of the NWS Charleston, South Carolina

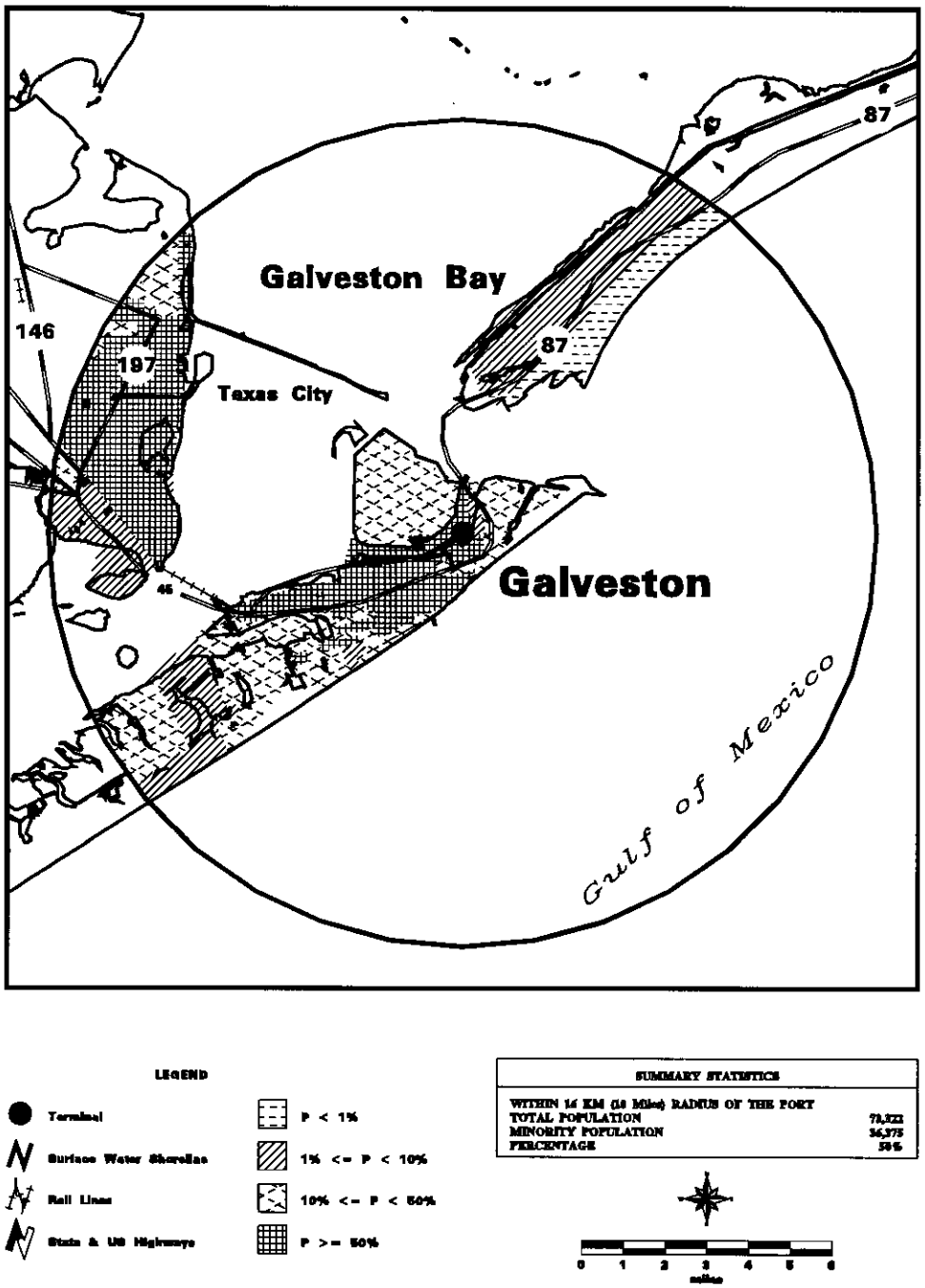


Figure A-3 Distribution of the Minority Population Residing within 16 km of the Port of Galveston, Texas

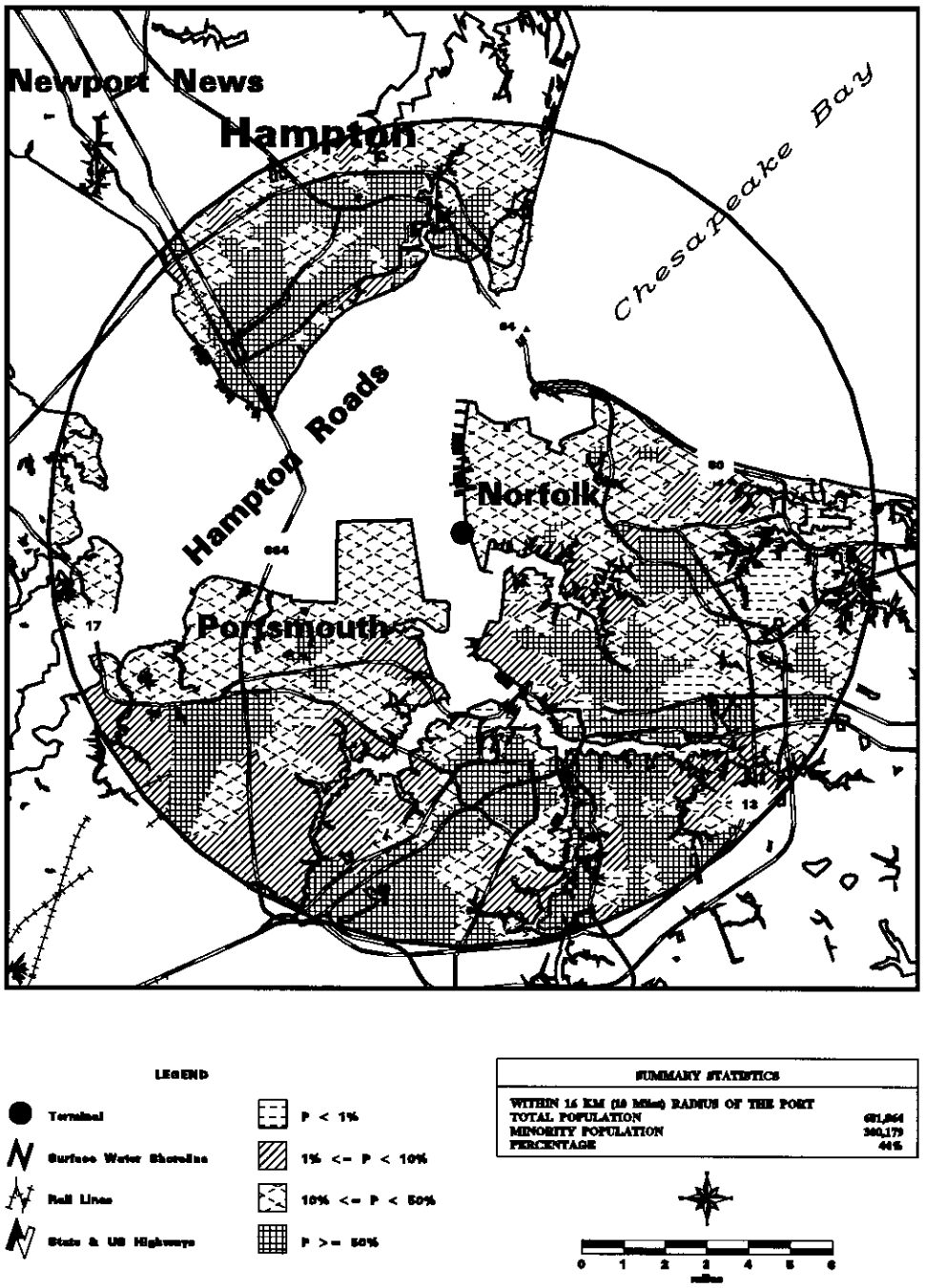


Figure A-4 Distribution of the Minority Population Residing within 16 km of the Port of Hampton Roads: Newport News, Norfolk, and Portsmouth, Virginia Terminals

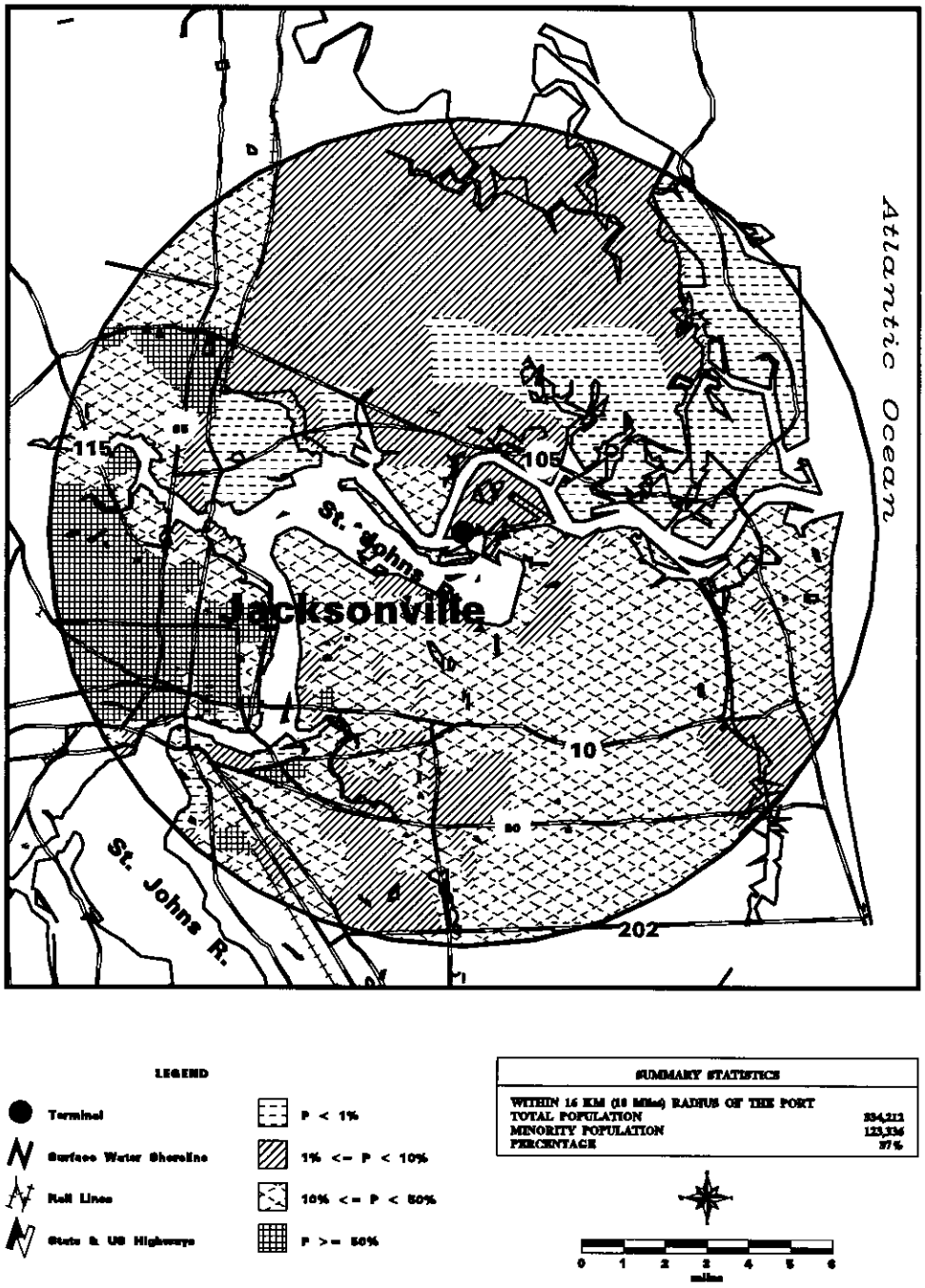
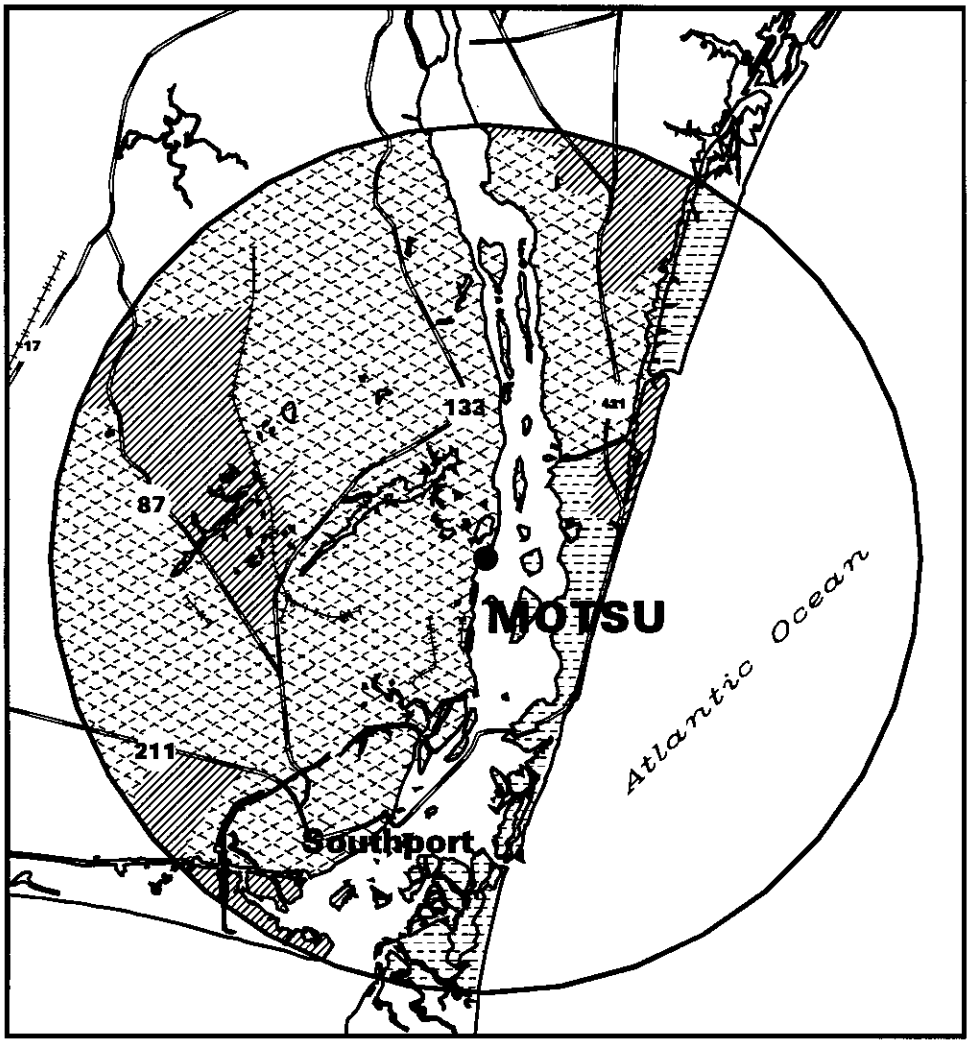


Figure A-5 Distribution of the Minority Population Residing within 16 km of the Port of Jacksonville, Florida



LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 10%
- 10% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 16 KM (10 Miles) RADIUS OF THE PORT	
TOTAL POPULATION	7,995
MINORITY POPULATION	1,456
PERCENTAGE	19%

0 1 2 3 4 5 6
miles

Figure A-6 Distribution of the Minority Population Residing within 16 km of the Port of MOTSU, North Carolina

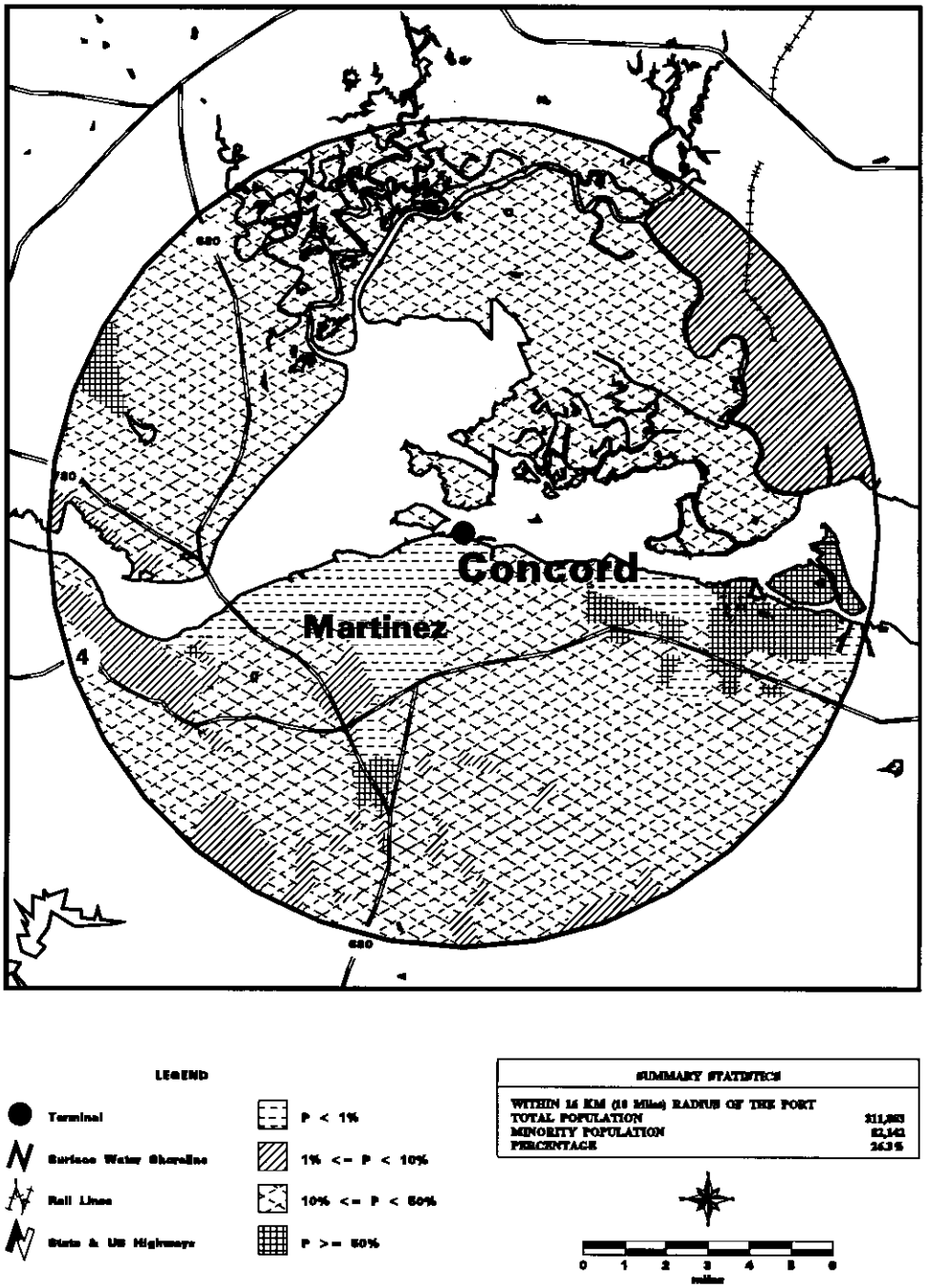


Figure A-7 Distribution of the Minority Population Residing within 16 km of the Port of NWS Concord, California

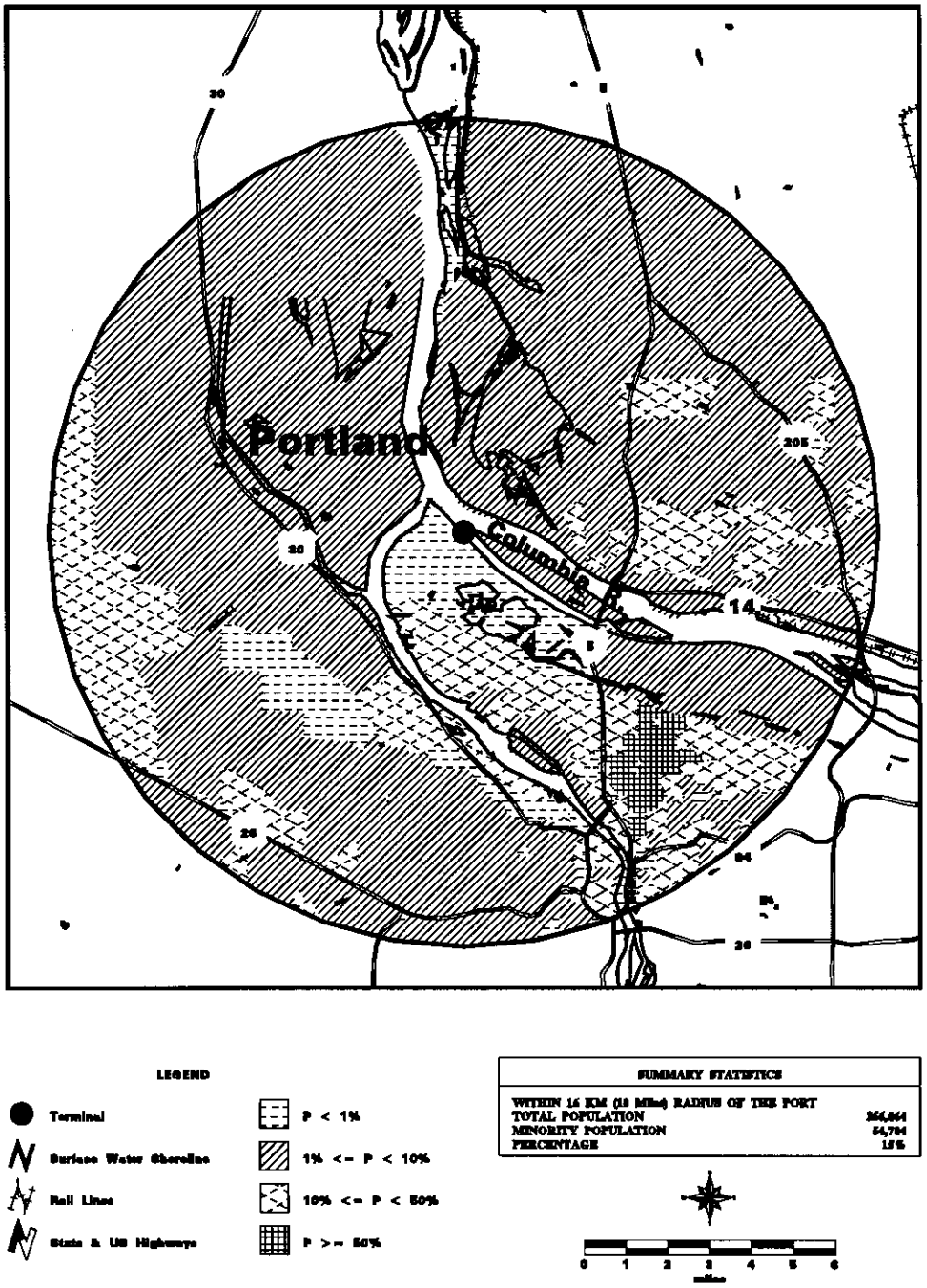


Figure A-8 Distribution of the Minority Population Residing within 16 km of the Port of Portland, Oregon

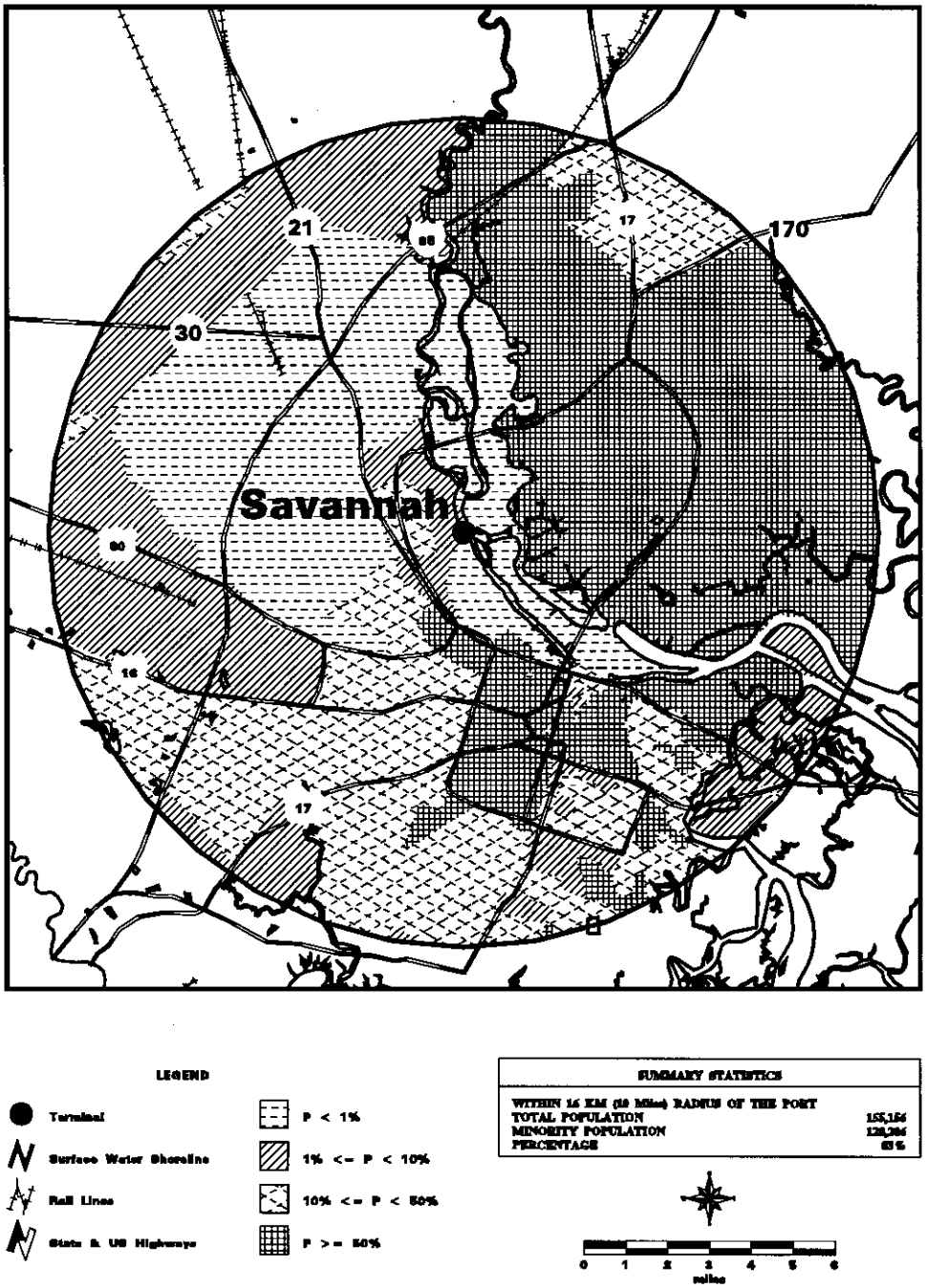


Figure A-9 Distribution of the Minority Population Residing within 16 km of the Port of Savannah, Georgia

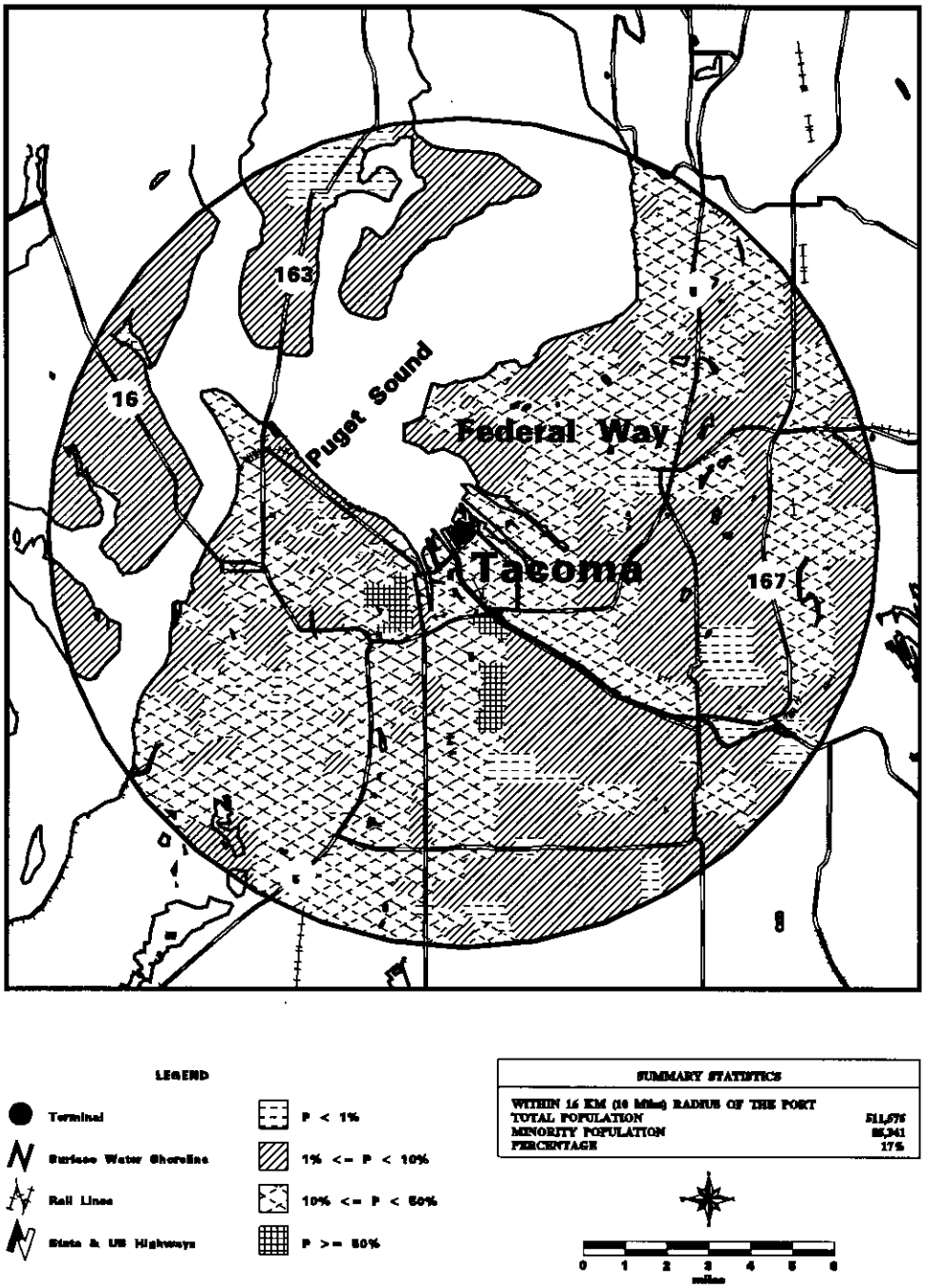


Figure A-10 Distribution of the Minority Population Residing within 16 km of the Port of Tacoma, Washington

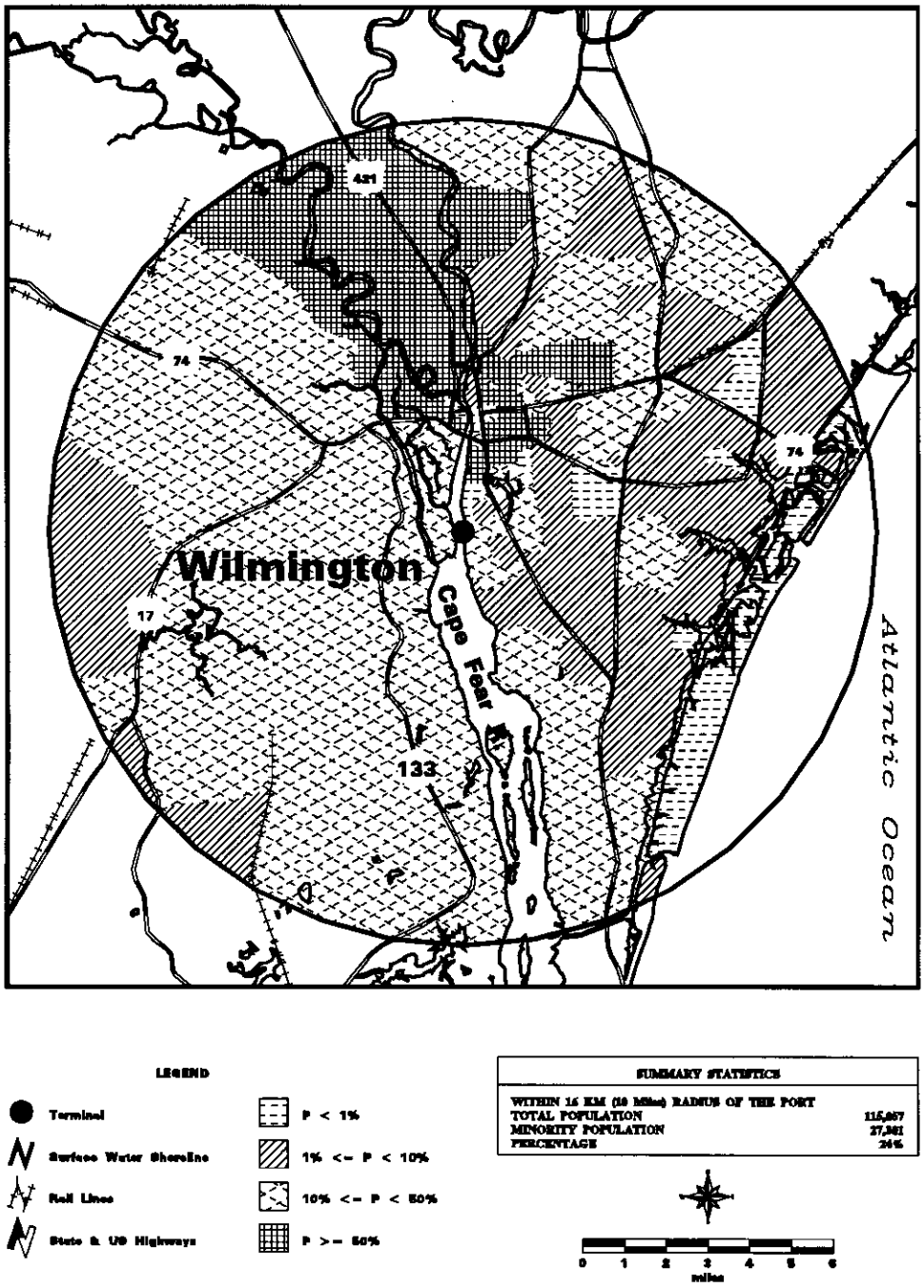


Figure A-11 Distribution of the Minority Population Residing within 16 km of the Port of Wilmington, North Carolina

A.3.2 Distribution of Low-Income Households Near the Candidate Ports

The number of low-income households near the candidate ports is shown in Table A-3. Except for the ports of MOTSU and Hampton Roads, the percentage of low-income households immediately surrounding the port is larger than the percentage of low-income households in the surrounding counties. Similarly, for most of the candidate ports, the percentage of low-income households near the port exceeds the percentage of low-income households in the surrounding state, although the ports of Charleston, MOTSU, Newport News, and NWS Concord are exceptions.

Distributions of low-income households near the candidate ports are shown in the maps of the ports presented in Figures A-12 through A-22. In these figures, geographical areas defined by census block group boundaries are shaded according to the percentage of low-income households within the block group. Since the number of households within a block group varies, the size of a shaded area is not necessarily proportional to the population within that area.

A.4 Environmental Justice Along Transportation Routes

The dominant radiological impacts associated with the normal or incident-free (accident-free) transportation activities would be the exposures received by the workers in the immediate vicinity of the cask, principally the truck drivers or train personnel. These individuals would be the only people receiving a measurable exposure during a routine spent nuclear fuel shipment.

The dose received by an individual near a spent nuclear fuel cask during shipment would be proportional to both the distance from the cask and the time of exposure. As discussed in Chapter 4 and Appendix E, the radiation dose rate from a cask containing spent nuclear fuel decreases with distance from the cask. Individuals living along the transportation routes would therefore be expected to receive low exposures because of both their distance from the cask and their short time of their exposure. While it is possible to make estimates of the collective dose of the population along a route, as in Chapter 4 and Appendix E, these minuscule doses would only be meaningful in the collective sense.

Ground and barge transportation accidents would be expected to result in no additional radiological impacts to the population in the vicinity of the accident. Potential radiological impacts from low probability accidents, which vary considerably, would be dependent on the accident conditions (such as the severity of an associated fire) and the weather conditions at the time of an accident. Since shipping accidents could occur at any location along the routes, it is not possible to identify the racial and economic composition of the populations that might be impacted. In general, however, the principal radiological impacts would be limited to the area within a few miles of the accident location and could be expected to impact a broad mixture of the population in the area.

Tables A-4 and A-5 show minority populations and low-income households, respectively, residing in 800-m (0.5-mi) wide corridors on each side of the road, rail, or barge routes from each of the candidate ports of entry to the Idaho National Engineering Laboratory and the Savannah River Site, both of which could receive spent nuclear fuel in the near term. In these tables, a county is called a "surrounding" county if its boundaries lie at least partially within the 800-m (0.5-mi) corridor. Routes used for this analysis are described in Appendix E.

As a general observation, percentages of minority populations residing along ground transportation routes (Column 7 of Table A-4) from candidate ports on the west coast to the Idaho National Engineering Laboratory are noticeably less than those for transportation from candidate east coast ports to the Savannah River Site. In addition, a higher percentage of minority individuals were found to reside along rail transportation routes than along truck transportation routes. The percentages varied from a minimum

Table A-3 Low-Income Households Residing Near the Candidate Ports

Candidate Port	*Total Households Residing within 16 km of Port*	*Low-Income Households Residing within 16 km of Port*	*% Low-Income Households Residing within 16 km of Port*	*Households Residing in Surrounding Counties*	*Low-Income Households Residing in Surrounding Counties*	*% Low-Income Households Residing in Surrounding Counties*	*Households Residing in Surrounding State(s)*	*Low-Income Households Residing in Surrounding State(s)*	*% Low-Income Households Residing in Surrounding State(s)*
Charleston, SC:									
Wando Terminal	85,851	36,904	43.0	149,358	62,552	41.9	1,258,783	545,937	43.4
NWS Terminal	72,765	32,020	44.0	149,358	62,552	41.9	1,258,783	545,937	43.4
Galveston, TX	29,360	16,607	56.6	81,417	34,984	43.0	6,079,341	2,815,886	46.3
Hampton Roads, VA:									
Newport News	126,789	51,055	40.3	336,638	137,129	40.7	2,294,722	937,123	40.8
Norfolk	206,464	90,723	43.9	336,688	137,129	40.7	2,294,722	937,123	40.8
Portsmouth	175,994	75,147	42.7	336,688	137,129	40.7	2,294,722	937,123	40.8
Jacksonville, FL	125,930	61,052	48.5	290,999	125,610	43.2	5,138,360	2,087,579	40.6
MOTSU, NC	3,071	1,166	38.0	20,094	8,455	42.1	2,517,098	1,067,345	42.4
NWS Concord, CA	143,676	58,344	40.6	415,223	167,426	40.3	10,399,700	4,307,948	41.4
Portland, OR	146,047	66,186	45.3	542,696	222,075	40.9	1,105,362	453,038	41.0
Savannah, GA	57,266	28,960	50.6	125,693	52,772	42.0	3,625,358	1,566,725	43.2
Tacoma, WA	198,458	83,101	41.9	843,736	338,779	40.2	1,875,508	736,285	39.3
Wilmington, NC	45,537	19,491	42.8	79,175	33,226	42.0	2,517,098	1,067,345	42.4

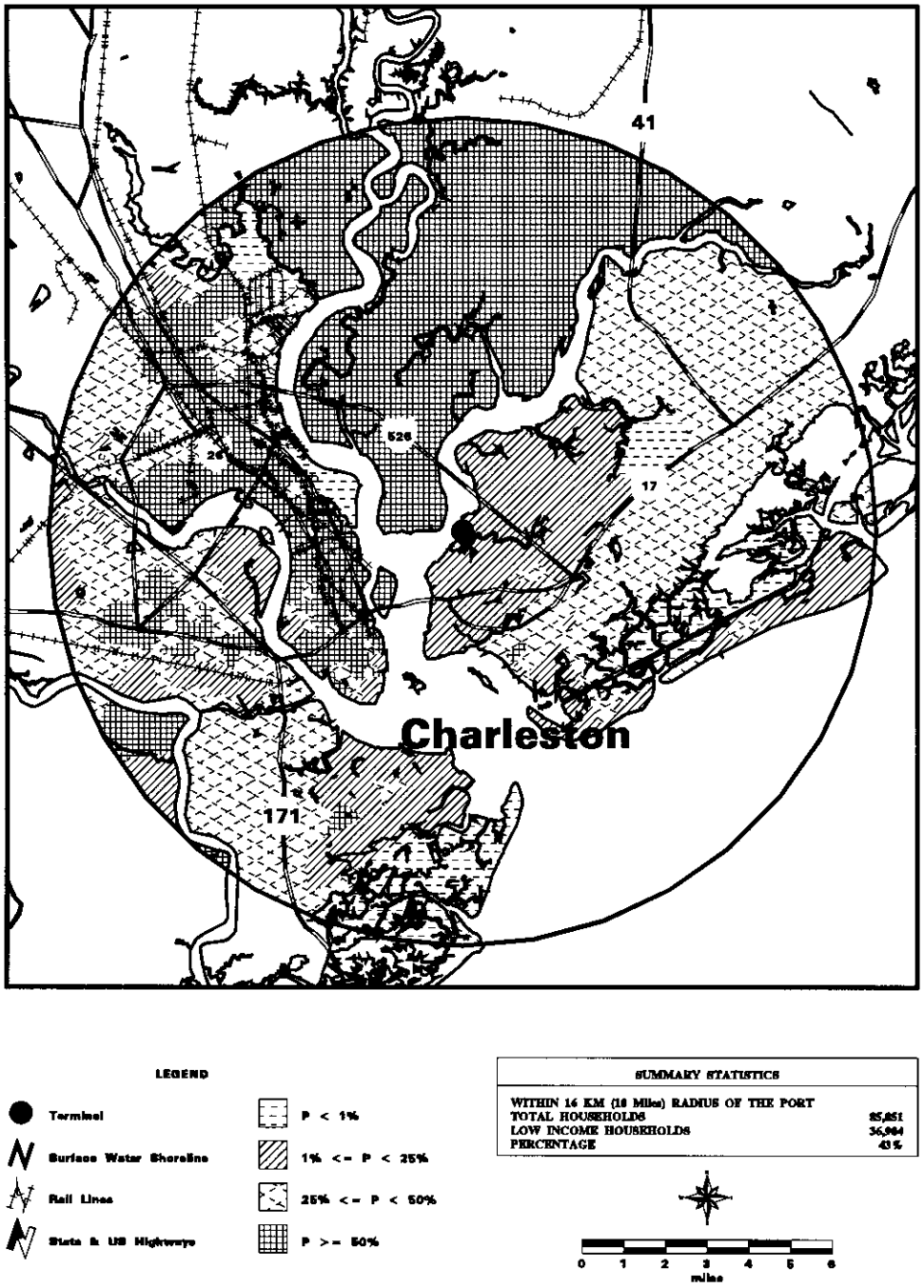
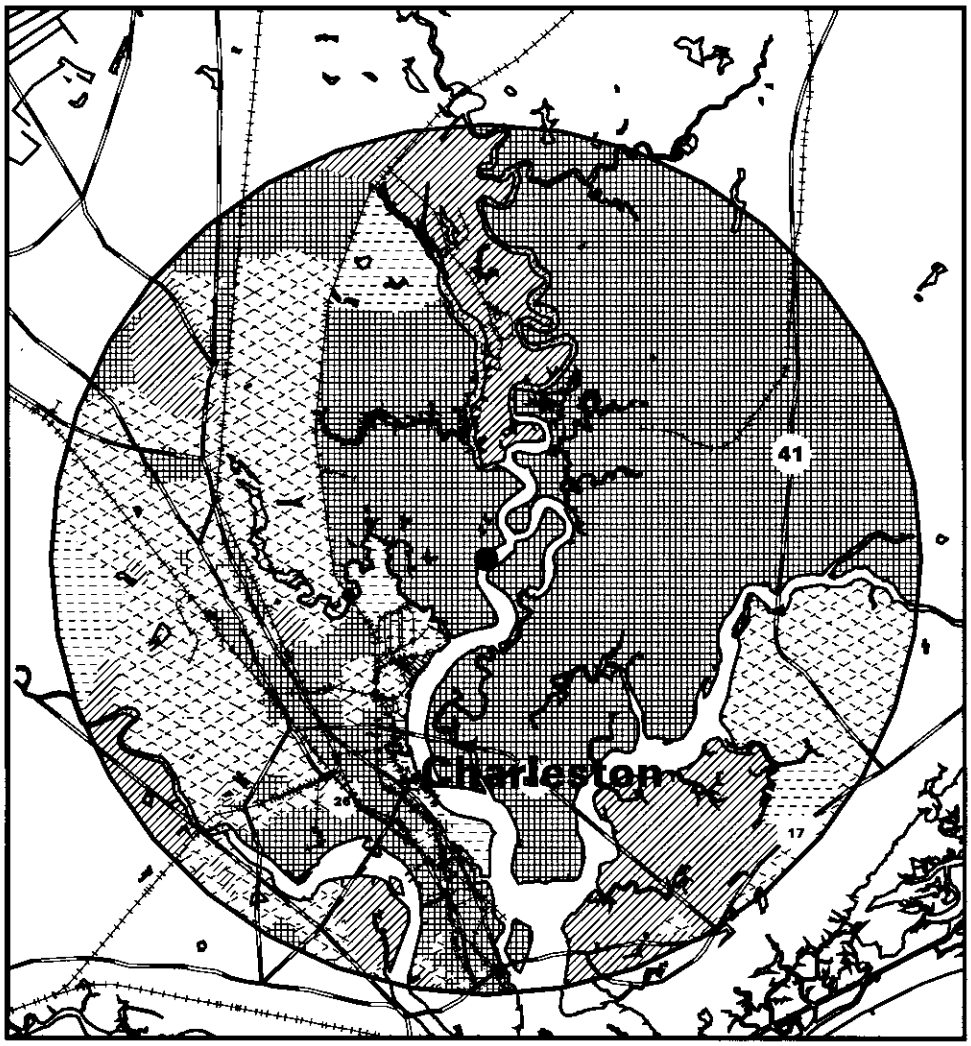


Figure A-12 Distribution of Low-Income Households Residing within 16 km of the Wando Terminal, Port of Charleston, South Carolina



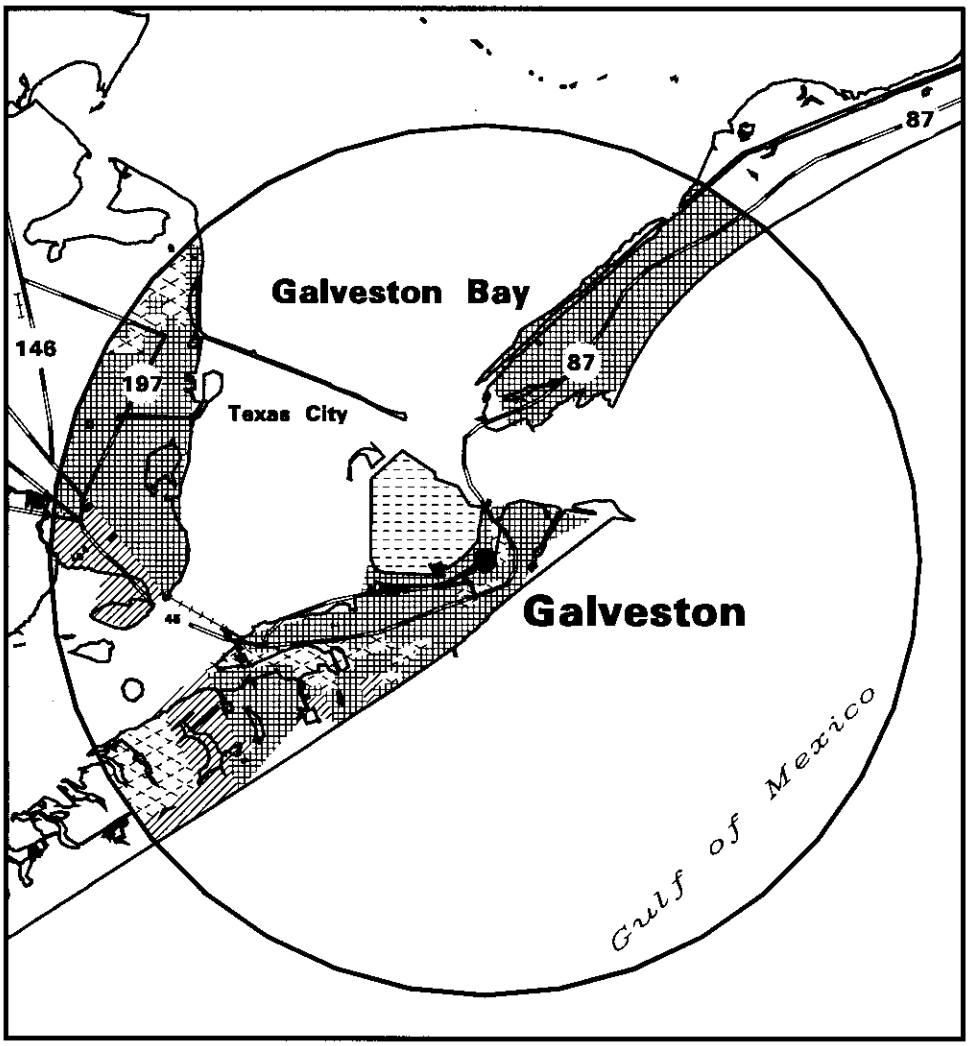
LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 25%
- 25% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 16 KM (10 Miles) RADIUS OF THE PORT	
TOTAL HOUSEHOLDS	72,765
LOW INCOME HOUSEHOLDS	32,028
PERCENTAGE	44%

0 1 2 3 4 5 6
miles

Figure A-13 Distribution of Low-Income Households Residing within 16 km of the NWS Charleston, South Carolina



LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 25%
- 25% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 16 KM (10 Miles) RADIUS OF THE PORT	
TOTAL HOUSEHOLDS	29,360
LOW INCOME HOUSEHOLDS	16,687
PERCENTAGE	57%

0 1 2 3 4 5 6
miles

Figure A-14 Distribution of Low-Income Households Residing within 16 km of the Port of Galveston, Texas

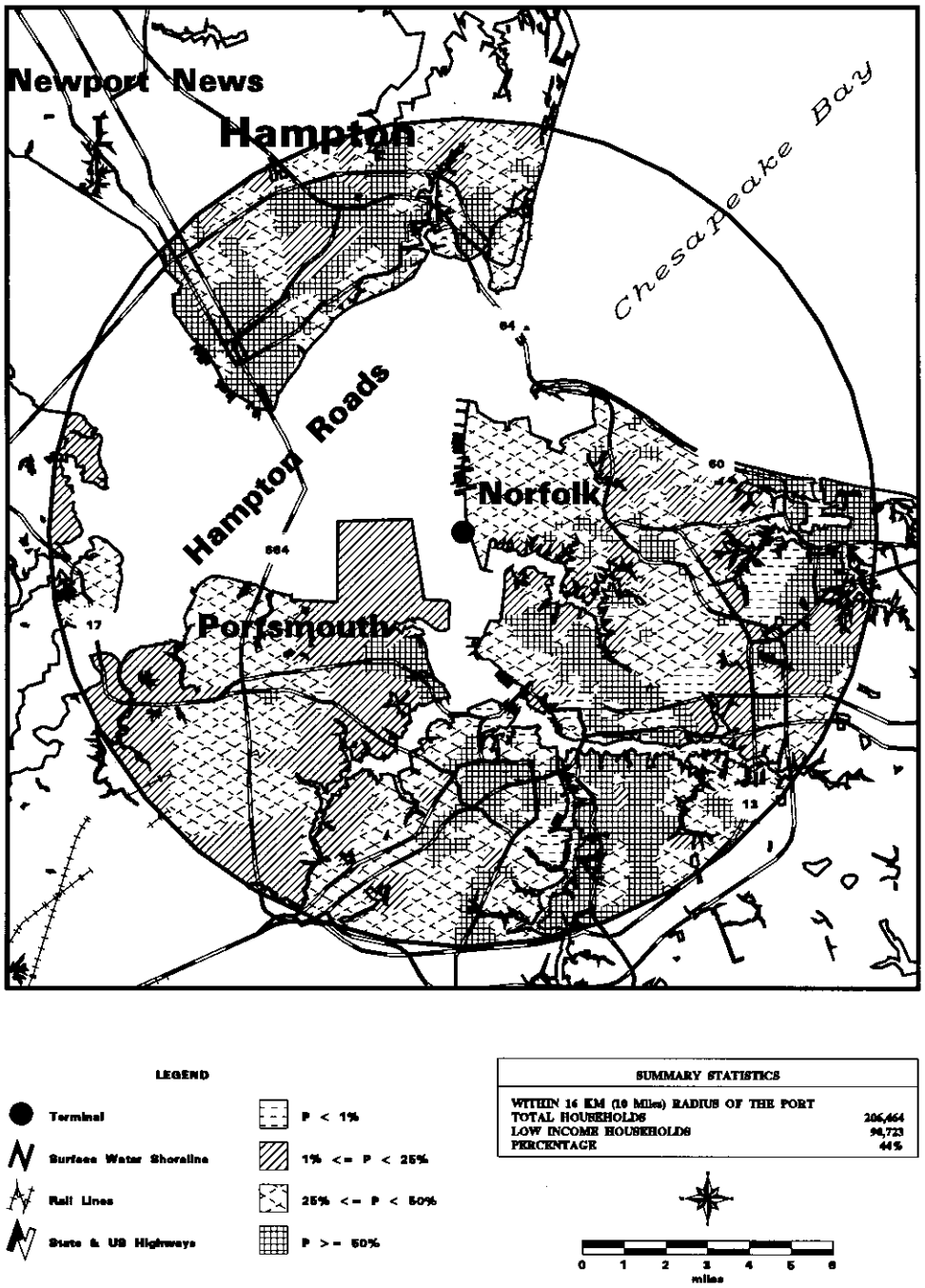
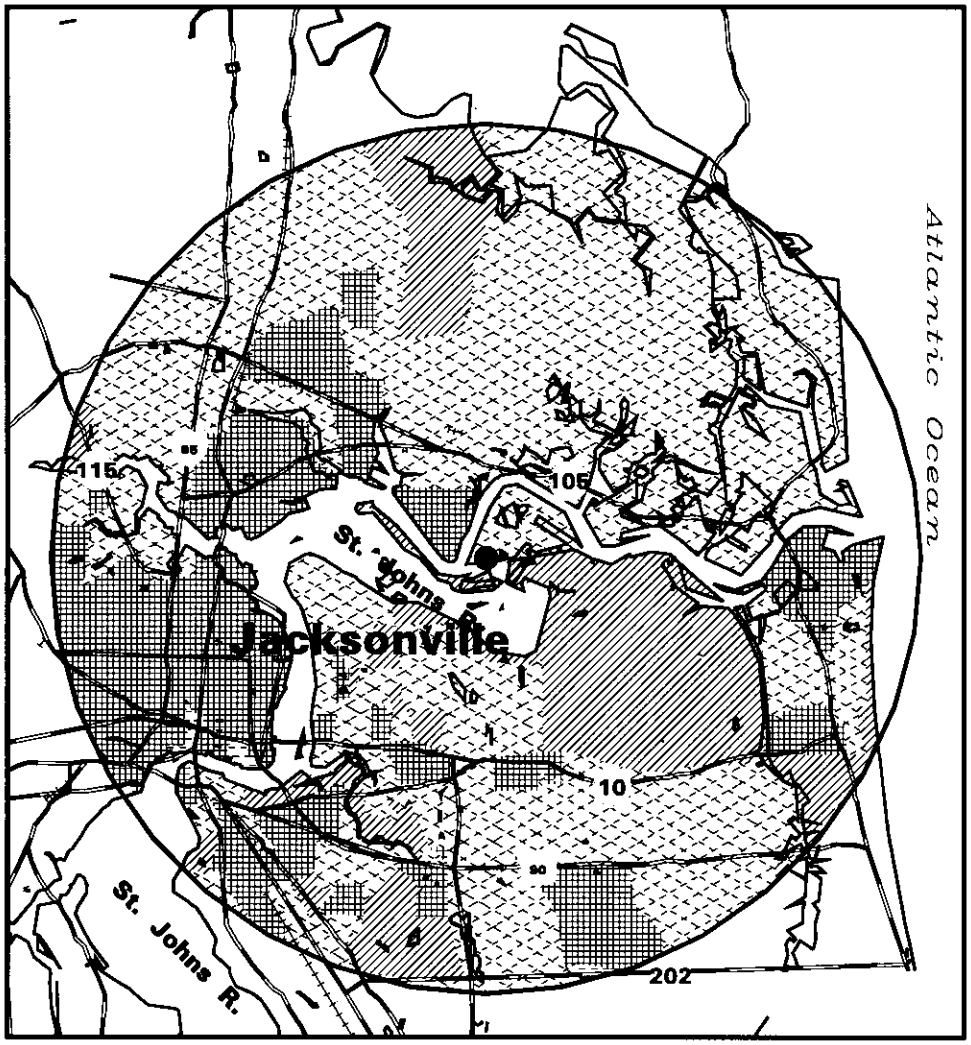


Figure A-15 Distribution of Low-Income Households Residing within 16 km of the Port of Hampton Roads: Newport News, Norfolk, and Portsmouth, Virginia Terminals



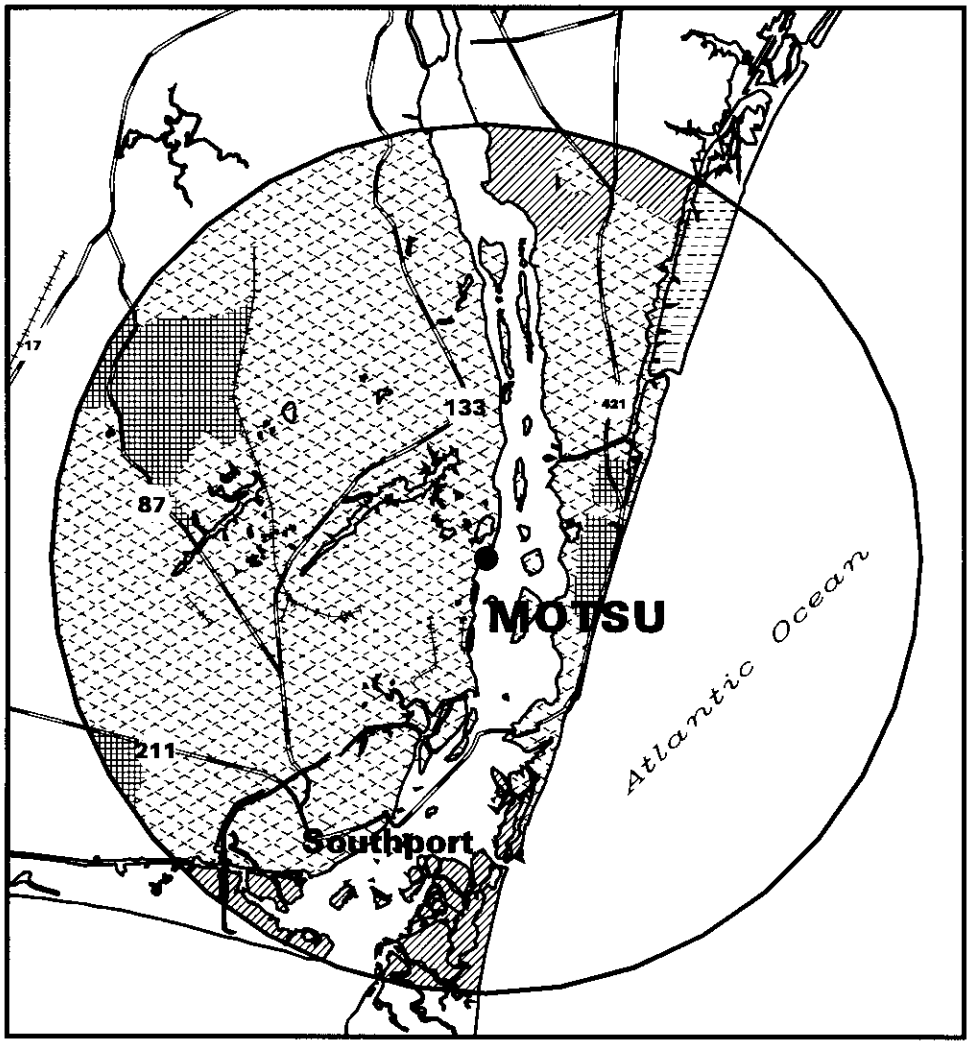
LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 25%
- 25% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 16 KM (10 Miles) RADIUS OF THE PORT	
TOTAL HOUSEHOLDS	135,930
LOW INCOME HOUSEHOLDS	61,852
PERCENTAGE	46%

0 1 2 3 4 5 6
miles

Figure A-16 Distribution of Low-Income Households Residing within 16 km of the Port of Jacksonville, Florida



LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 25%
- 25% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 16 KM (10 Miles) RADIUS OF THE PORT	
TOTAL HOUSEHOLDS	3,071
LOW INCOME HOUSEHOLDS	1,168
PERCENTAGE	38%

0 1 2 3 4 5 6
miles

Figure A-17 Distribution of Low-Income Households Residing within 16 km of the Port of MOTSU, North Carolina

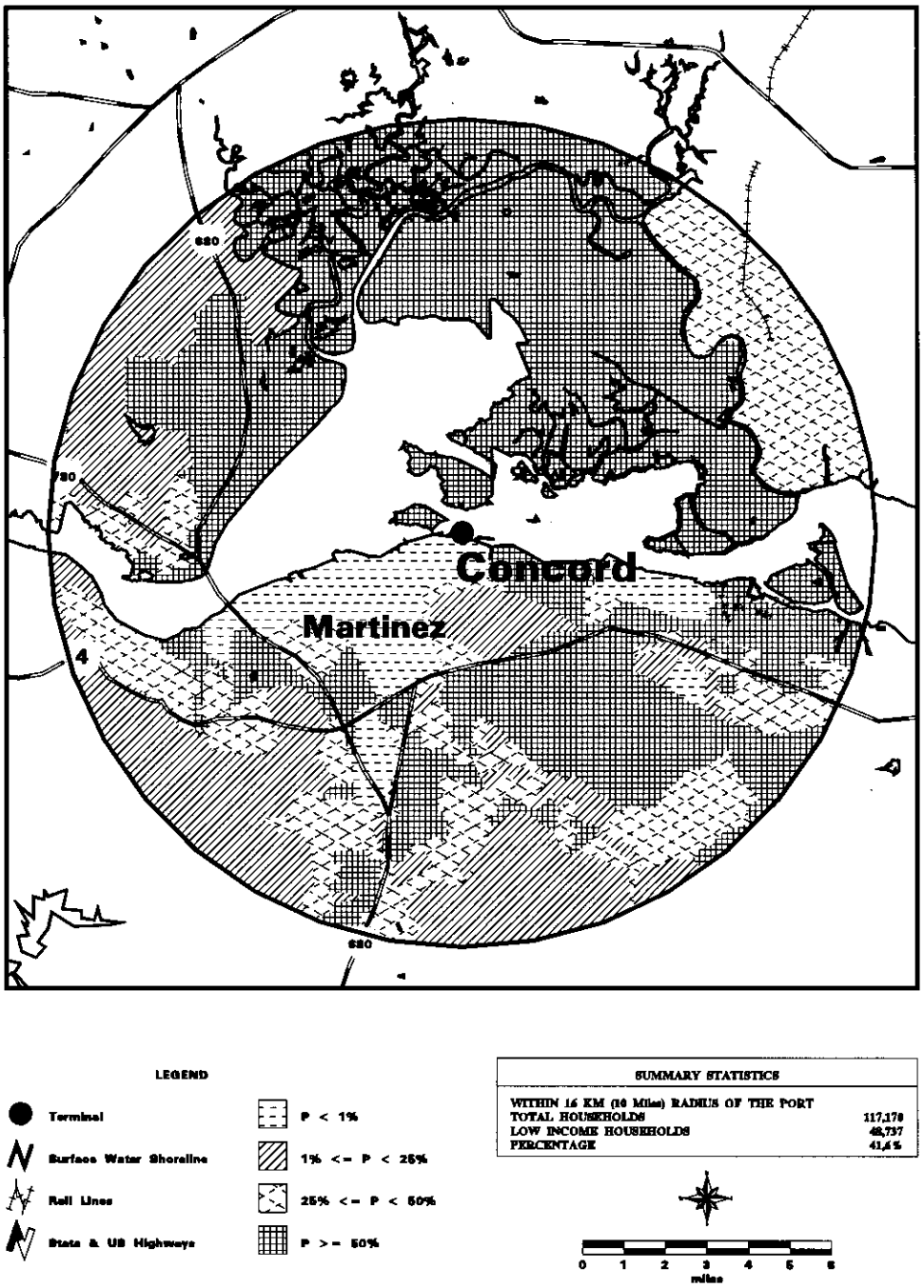


Figure A-18 Distribution of Low-Income Households Residing within 16 km of the Port of NWS Concord, California

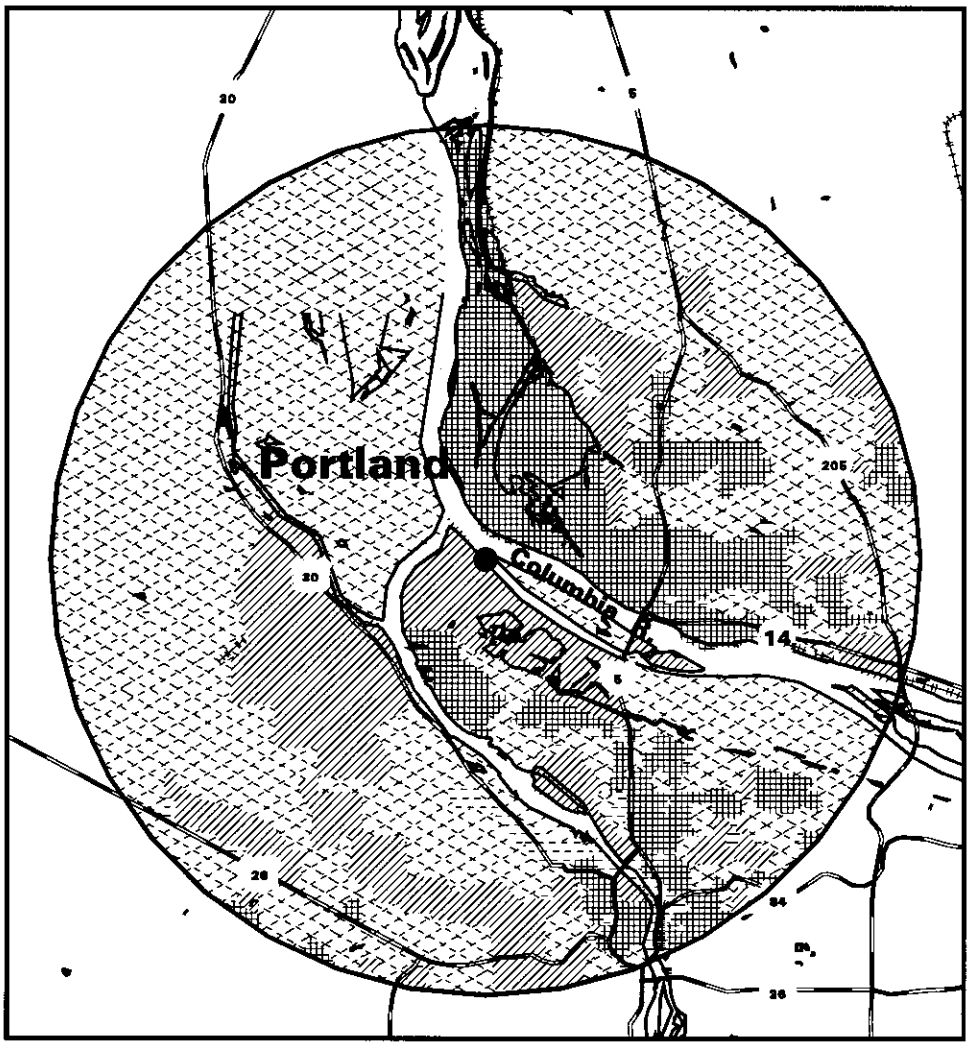
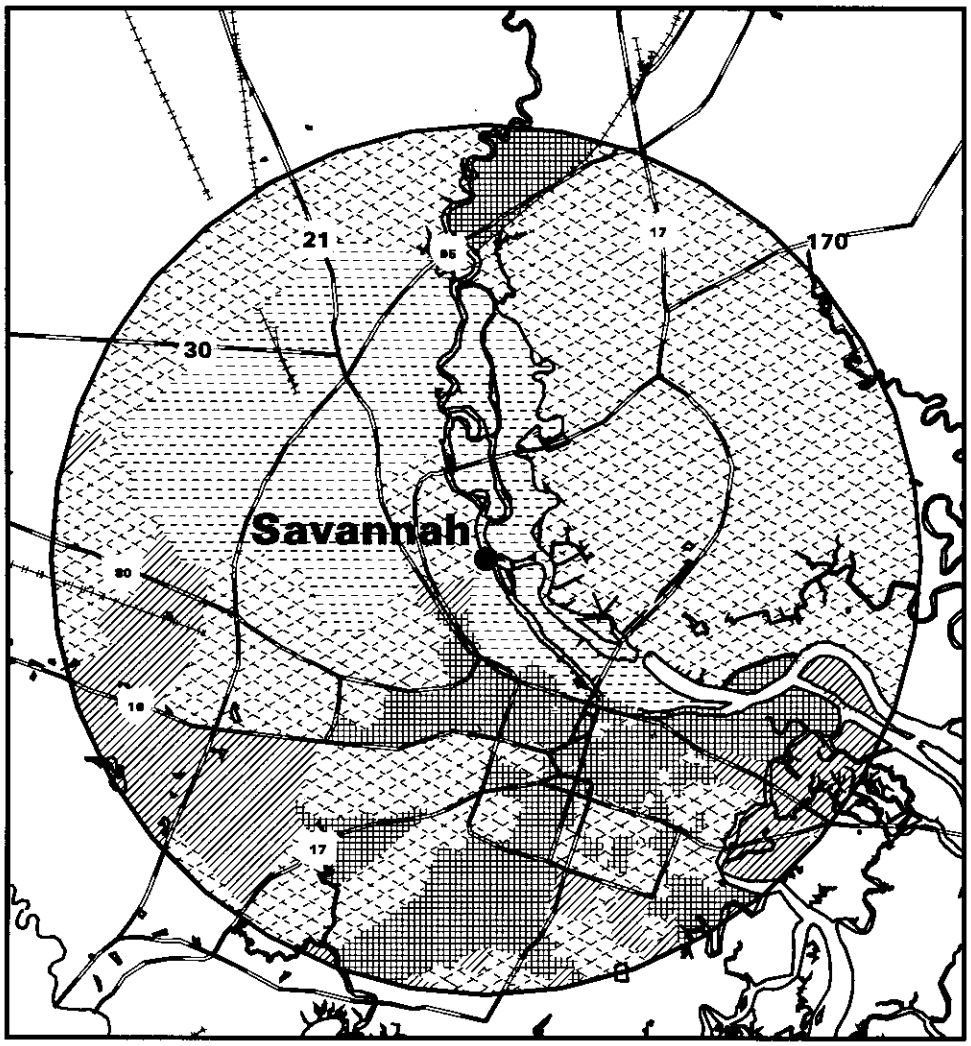


Figure A-19 Distribution of Low-Income Households Residing within 16 km of the Port of Portland, Oregon



LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 25%
- 25% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 16 KM (10 Miles) RADIUS OF THE PORT	
TOTAL HOUSEHOLDS	57,264
LOW INCOME HOUSEHOLDS	28,968
PERCENTAGE	51%

0 1 2 3 4 5 6
miles

Figure A-20 Distribution of Low-Income Households Residing within 16 km of the Port of Savannah, Georgia

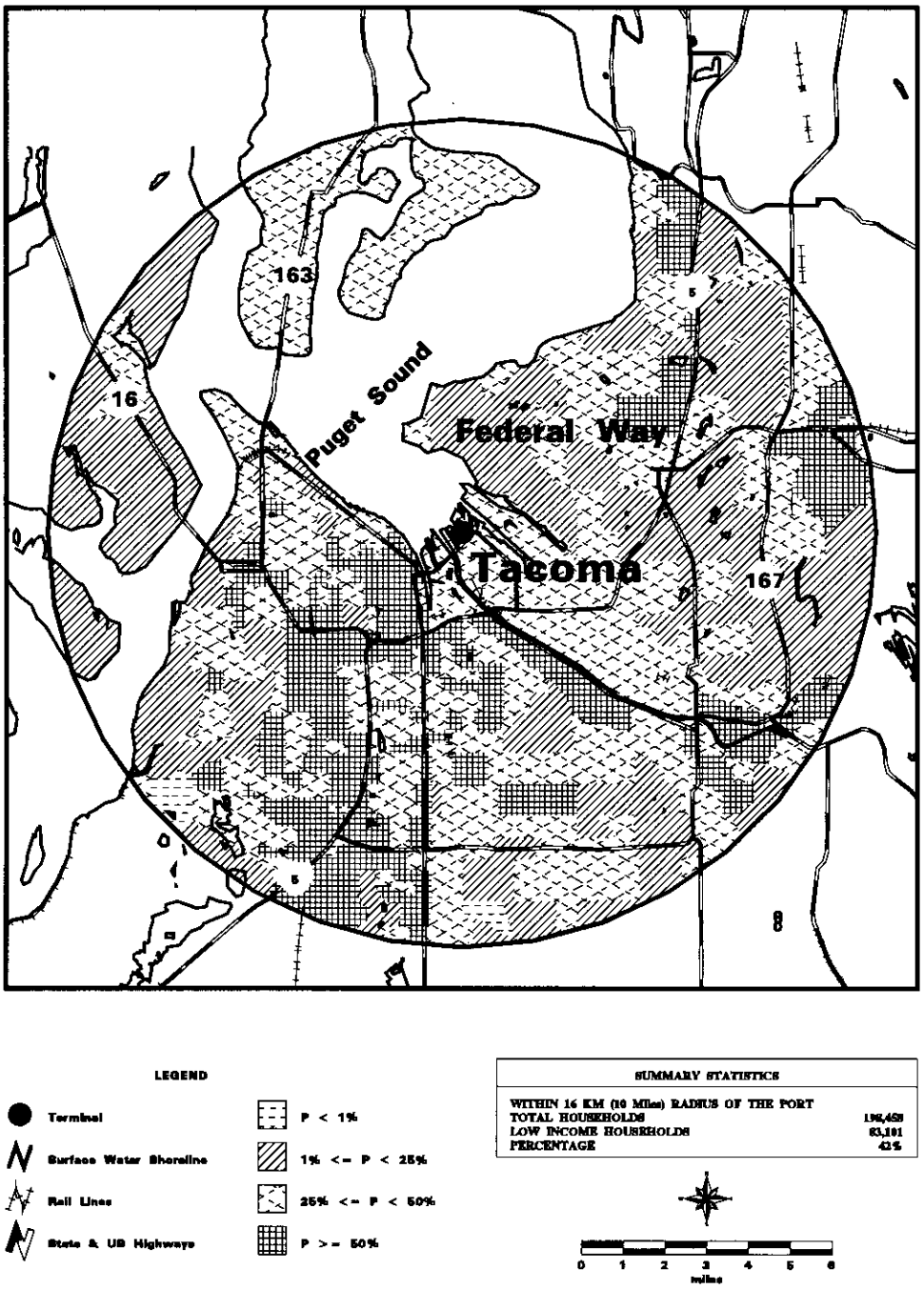
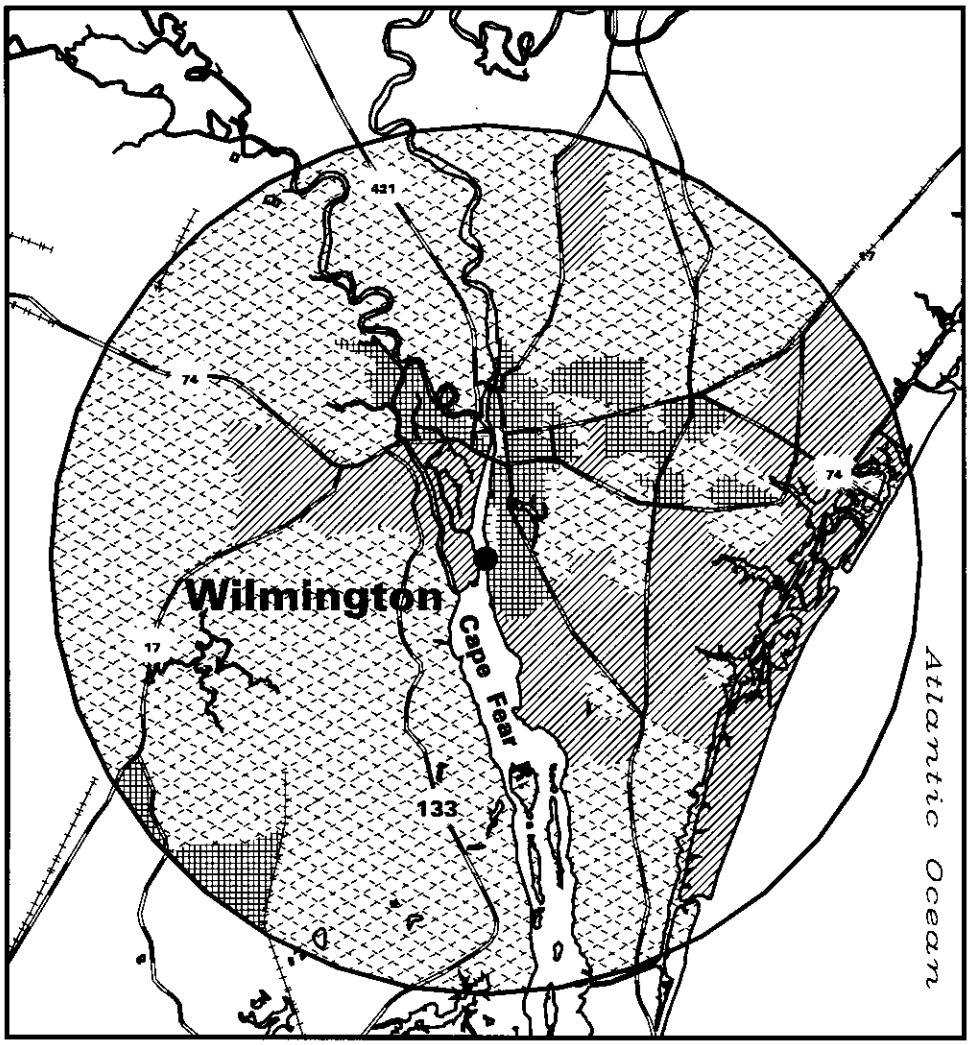


Figure A-21 Distribution of Low-Income Households Residing within 16 km of the Port of Tacoma, Washington



LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 25%
- 25% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 16 KM (10 Miles) RADIUS OF THE PORT	
TOTAL HOUSEHOLDS	45,537
LOW INCOME HOUSEHOLDS	19,491
PERCENTAGE	43%

0 1 2 3 4 5 6
miles

Figure A-22 Distribution of Low-Income Households Residing within 16 km of the Port of Wilmington, North Carolina

Table A-4 Minority Populations Residing Near Ground and Barge Transportation Routes

Route No.	Port	Destination	Mode	Total Pop. Along Route	Minority Pop. Along Route	% Minority Pop. Along Route	Total Pop. in Surrounding Counties	Minority Pop. in Surrounding Counties	% Minority Pop. in Surrounding Counties
	Charleston, SC:								
1	Wando Terminal	INEL	Rail	709,863	177,890	25.1	8,562,589	2,027,753	23.7
2	Wando Terminal	INEL	Truck	514,213	90,978	17.7	8,205,925	1,508,320	18.4
3	Wando Terminal	SRS	Rail	19,633	8,783	44.7	693,370	246,297	35.5
4	Wando Terminal	SRS	Truck	84,729	36,613	43.2	1,180,381	403,989	34.2
5	NWS Terminal	INEL	Rail	709,863	177,890	25.1	8,562,589	2,027,753	23.7
6	NWS Terminal	INEL	Truck	503,081	79,646	15.8	8,205,925	1,508,320	18.4
7	NWS Terminal	SRS	Rail	19,633	8,783	44.7	693,370	246,297	35.5
8	NWS Terminal	SRS	Truck	75,476	31,538	41.8	1,180,381	403,989	34.2
9	Galveston, TX	INEL	Rail	390,876	124,553	31.9	3,794,113	1,006,498	26.5
10	Galveston, TX	INEL	Truck	600,239	205,700	34.3	9,292,668	2,858,758	30.8
11	Galveston, TX	SRS	Rail	528,014	286,872	54.3	8,577,378	3,424,688	39.9
12	Galveston, TX	SRS	Truck	429,057	189,407	44.1	7,965,572	3,038,319	38.1
13	Hampton Roads, VA	INEL	Rail	1,005,972	354,386	35.2	11,758,163	2,982,158	25.4
14	Hampton Roads, VA	INEL	Truck	603,551	115,107	19.1	10,248,206	1,954,088	19.1
15	Hampton Roads, VA	SRS	Rail	221,375	107,613	48.6	2,655,287	984,457	37.1
16	Hampton Roads, VA	SRS	Truck	212,286	98,584	46.4	2,266,251	880,358	38.8
17	Jacksonville, FL	INEL	Rail	697,964	200,596	28.7	8,106,565	1,742,778	21.5
18	Jacksonville, FL	INEL	Truck	622,326	167,802	27.0	10,644,968	2,317,151	21.8
19	Jacksonville, FL	SRS	Rail	52,145	36,707	70.4	1,217,454	388,456	31.9
20	Jacksonville, FL	SRS	Truck	72,821	30,887	42.4	2,154,525	678,172	31.5
21	MOTSU, NC	INEL	Rail	753,535	184,459	24.5	9,084,840	2,087,107	23.0
22	MOTSU, NC	INEL	Truck	515,468	91,325	17.7	9,446,043	1,856,873	19.7
23	MOTSU, NC	SRS	Rail	75,932	33,173	43.7	1,302,260	499,281	38.3
24	MOTSU, NC	SRS	Truck	93,987	40,073	42.6	1,518,891	517,912	34.1
25	NWS Concord, CA	INEL	Rail	344,524	72,137	20.9	4,655,756	973,603	20.9
26	NWS Concord, CA	INEL	Truck	267,109	57,926	21.7	4,655,756	973,603	20.9
27	NWS Concord, CA	SRS	Rail	1,443,296	796,105	55.2	30,242,508	13,433,482	44.4
28	NWS Concord, CA	SRS	Truck	1,240,640	536,731	43.3	28,254,357	11,932,215	42.2

Table A-4 Minority Populations Residing Near Ground and Barge Transportation Routes (Continued)

Route No.	Port	Destination	Mode	Total Pop. Along Route	Minority Pop. Along Route	% Minority Pop. Along Route	Total Pop. in Surrounding Counties	Minority Pop. in Surrounding Counties	% Minority Pop. in Surrounding Counties
29	Portland, OR	HS	Barge	28,430	3,599	12.7	1,099,340	140,872	12.8
30	Portland, OR	INEL	Rail	162,678	26,252	16.1	1,300,552	154,630	11.9
31	Portland, OR	INEL	Truck	124,067	15,463	12.5	1,563,392	172,864	11.1
32	Portland, OR	SRS	Rail	950,116	193,322	20.3	15,663,396	3,695,119	23.6
33	Portland, OR	SRS	Truck	671,113	130,098	19.4	11,149,104	2,395,779	21.5
34	Savannah, GA	INEL	Rail	680,075	178,683	26.3	7,696,157	1,661,208	21.6
35	Savannah, GA	INEL	Truck	574,641	140,293	24.4	9,931,415	2,135,851	21.5
36	Savannah, GA	SRS	Barge	1,715	520	30.3	124,099	56,936	45.9
37	Savannah, GA	SRS	Rail	13,835	11,330	81.9	369,053	151,266	41.0
38	Savannah, GA	SRS	Truck	51,065	18,916	37.0	1,135,620	402,488	35.4
39	Tacoma, WA	INEL	Rail	255,650	45,100	17.6	2,427,856	290,185	12.0
40	Tacoma, WA	INEL	Truck	178,532	29,787	16.7	3,141,728	488,589	15.6
41	Tacoma, WA	SRS	Rail	753,535	184,459	24.5	17,228,536	3,946,748	22.9
42	Tacoma, WA	SRS	Truck	514,955	91,051	17.7	8,850,682	1,470,687	16.6
43	Wilmington, NC	INEL	Rail	753,535	184,459	24.5	9,084,840	2,087,107	23.0
44	Wilmington, NC	INEL	Truck	514,955	91,051	17.7	9,446,043	1,856,873	19.7
45	Wilmington, NC	SRS	Rail	75,932	33,173	43.7	1,302,260	499,281	38.3
46	Wilmington, NC	SRS	Truck	102,951	37,735	36.7	1,518,891	517,912	34.1

INEL = Idaho National Engineering Laboratory, SRS = Savannah River Site, HS = Hanford Site

Table A-5 Low-Income Households Residing Near Ground and Barge Transportation Routes

Route No.	Port	Destination	Mode	Total Households Along Route	Low-Income Households Along Route	% Low-Income Households Along Route	Total Households in Surrounding Counties	Low-Income Households in Surrounding Counties	% Low-Income Households in Surrounding Counties
	Charleston, SC:								
1	Wando Terminal	INEL	Rail	279,468	141,864	50.8	3,256,143	1,359,530	41.8
2	Wando Terminal	INEL	Truck	199,269	81,749	41.0	3,082,221	1,266,095	41.1
3	Wando Terminal	SRS	Rail	7,305	3,358	46.0	242,968	101,094	41.6
4	Wando Terminal	SRS	Truck	31,040	15,664	50.5	419,616	175,993	41.9
5	NWS Terminal	INEL	Rail	279,468	141,864	50.8	3,256,143	1,359,530	41.8
6	NWS Terminal	INEL	Truck	193,945	79,186	40.8	3,082,221	1,266,095	41.1
7	NWS Terminal	SRS	Rail	7,305	3,358	46.0	242,968	101,094	41.6
8	NWS Terminal	SRS	Truck	27,339	13,019	47.6	419,616	175,993	41.9
9	Galveston, TX	INEL	Rail	157,276	83,829	53.3	1,410,581	568,315	40.3
10	Galveston, TX	INEL	Truck	230,042	110,020	47.8	3,471,001	1,440,001	41.5
11	Galveston, TX	SRS	Rail	189,537	103,279	54.5	3,135,368	1,315,512	42.0
12	Galveston, TX	SRS	Truck	157,216	74,900	47.6	2,912,738	1,217,498	41.8
13	Hampton Roads, VA	INEL	Rail	372,127	184,477	49.6	4,350,161	1,774,235	40.8
14	Hampton Roads, VA	INEL	Truck	234,717	98,570	42.0	3,895,065	1,611,870	41.4
15	Hampton Roads, VA	SRS	Rail	83,505	42,094	50.4	963,965	403,155	41.8
16	Hampton Roads, VA	SRS	Truck	80,245	37,427	46.6	782,526	332,806	42.5
17	Jacksonville, FL	INEL	Rail	271,994	140,535	51.7	3,097,768	1,304,381	42.1
18	Jacksonville, FL	INEL	Truck	240,520	106,683	44.4	4,036,596	1,686,985	41.8
19	Jacksonville, FL	SRS	Rail	18,576	10,158	54.7	449,914	193,017	42.9
20	Jacksonville, FL	SRS	Truck	27,436	14,176	51.7	784,169	332,060	42.3
21	MOTSU, NC	INEL	Rail	294,953	148,549	50.4	3,458,175	1,443,995	41.8
22	MOTSU, NC	INEL	Truck	199,145	82,005	41.2	3,591,483	1,507,703	42.0
23	MOTSU, NC	SRS	Rail	28,089	13,791	49.1	465,649	197,699	42.5
24	MOTSU, NC	SRS	Truck	35,670	16,610	46.6	547,984	235,188	42.9
25	NWS Concord, CA	INEL	Rail	128,843	66,118	51.3	1,643,159	675,769	41.1
26	NWS Concord, CA	INEL	Truck	97,241	44,810	46.1	1,643,159	675,769	41.1
27	NWS Concord, CA	SRS	Rail	488,799	257,198	52.6	10,625,645	4,386,618	41.3
28	NWS Concord, CA	SRS	Truck	436,446	184,959	42.4	10,022,988	4,153,610	41.4

Table A-5 Low-Income Households Residing Near Ground and Barge Transportation Routes (Continued)

Route No.	Port	Destination	Mode	Total Households Along Route	Low-Income Households Along Route	% Low-Income Households Along Route	Total Households in Surrounding Counties	Low-Income Households in Surrounding Counties	% Low-Income Households in Surrounding Counties
29	Portland, OR	HS	Barge	11,458	4,582	40.0	428,477	178,142	41.6
30	Portland, OR	INEL	Rail	63,053	29,047	46.1	497,875	206,447	41.5
31	Portland, OR	INEL	Truck	49,120	20,461	41.7	595,780	247,985	41.6
32	Portland, OR	SRS	Rail	368,054	183,842	49.9	5,842,902	2,398,677	41.1
33	Portland, OR	SRS	Truck	261,130	106,936	41.0	4,245,248	1,761,997	41.5
34	Savannah, GA	INEL	Rail	265,122	133,766	50.5	2,932,786	1,231,245	42.0
35	Savannah, GA	INEL	Truck	224,123	97,864	43.7	3,767,606	1,570,456	41.7
36	Savannah, GA	SRS	Barge	595	248	41.7	42,983	18,577	43.2
37	Savannah, GA	SRS	Rail	4,868	2,669	54.8	134,291	56,688	42.2
38	Savannah, GA	SRS	Truck	19,340	9,235	47.8	408,100	171,672	42.1
39	Tacoma, WA	INEL	Rail	98,487	48,094	48.8	917,614	377,168	41.1
40	Tacoma, WA	INEL	Truck	68,303	29,265	42.8	1,212,119	493,235	40.7
41	Tacoma, WA	SRS	Rail	448,663	225,672	50.3	6,455,529	2,633,674	40.8
42	Tacoma, WA	SRS	Truck	223,282	99,101	44.4	3,387,084	1,375,781	40.6
43	Wilmington, NC	INEL	Rail	294,953	148,549	50.4	3,458,175	1,443,995	41.8
44	Wilmington, NC	INEL	Truck	198,884	81,839	41.1	3,591,483	1,507,703	42.0
45	Wilmington, NC	SRS	Rail	28,089	13,791	49.1	465,649	197,699	42.5
46	Wilmington, NC	SRS	Truck	39,759	18,538	46.6	547,984	235,188	42.9

INEL = Idaho National Engineering Laboratory, SRS = Savannah River Site, HS = Hanford Site

of 12.5 percent for transportation by truck from Portland, Oregon to Idaho National Engineering Laboratory to a maximum of 81.9 percent for rail transportation from Savannah, Georgia to the Savannah River Site.

As shown in Column 7 of Table A-5, similar observations are true for percentages of low-income households residing along ground transportation routes. In the case of low-income households, percentages varied from a minimum of 41.0 percent for truck transportation from Portland, Oregon and Charleston, South Carolina to the Savannah River Site and Idaho National Engineering Laboratory, respectively, to a maximum of 54.8 percent for rail transportation from Savannah, Georgia to the Savannah River Site.

Populations residing within 1.6 km of barge routes are numerically very small in comparison with those residing near ground transportation routes. Percentages of minority populations and low-income households residing near barge routes are similar to the percentages for ground transportation modes.

A.5 Environmental Justice in Areas Near the Candidate Management Sites

Under normal management site activities associated with receipt and storage of the spent nuclear fuel, the dominant radiological impacts have been shown to be the exposures received by the site workers in the immediate vicinity of the spent nuclear fuel cask. These individuals would be principally those working within the spent nuclear fuel storage facility. The racial and economic composition of these individuals at each management site that would receive the majority of the dose could vary considerably. Health effects due to normal operations and accidents at the five candidate management sites are presented in Section 4.2.4. No latent cancer fatalities or other fatalities would be expected to result from the handling and storage of spent nuclear fuel from foreign research reactors at the sites. At none of the sites would the radiological impacts of either normal releases or low probability accidental releases of spent nuclear fuel be expected to significantly affect the general population outside the management site boundary, including minority and low-income populations. Consequently, there are no adverse impacts of the proposed action on these groups.

A.5.1 Distribution of Minority Populations Near the Candidate Management Sites

The distribution of minority populations residing in various areas surrounding the candidate interim management sites is presented in Table A-6. This table shows minority populations within an 80-km (50-mi) radius centered at the interim management site. For comparison, minority populations are also shown for the counties surrounding each site. A county was included in the analysis if its boundaries lie at least partially within this circle. As shown in the table, minority populations surrounding the Nevada Test Site and the Idaho National Engineering Laboratory are numerically small in comparison with those surrounding the Hanford Site and the Savannah River Site. The minority population surrounding the Nevada Test Site is relatively large because the boundary of the county containing Las Vegas, NV is within 80 km (50 mi) of the site. The Savannah River Site has the largest percentage of minorities in the surrounding area and surrounding counties.

The racial and ethnic composition of minorities surrounding the candidate interim management sites is illustrated in Table A-7. Hispanics composed nearly 81 percent of the minority population surrounding the Hanford Site at the time of the 1990 census. The Hanford Site is also surrounded by a relatively large percentage (about 8 percent) of Native Americans due to the presence of the Yakama Indian Reservation and tribal headquarters in the State of Washington. The area surrounding the Idaho National Engineering Laboratory has the second smallest percentage of minorities of all the sites. The surrounding minority composition is primarily Hispanic, Native American, and Asian. The Fort Hall Indian Reservation lies

Table A-6 Minority Populations Residing Near the Candidate Interim Management Sites

Candidate Management Site	*Population within 80 km of Site*	*Minority Population within 80 km of Site*	*% Minority Population within 80 km of Site*	*Population in Counties Surrounding Site*	*Minority Population in Counties Surrounding Site*	*% Minority Population in Counties Surrounding Site*
Savannah River Site	566,823	214,016	37.8	944,982	330,078	34.9
Idaho National Engineering Laboratory	176,311	15,449	8.8	265,823	21,828	8.2
Hanford Site	383,934	95,042	24.8	565,871	116,610	20.6
Oak Ridge Reservation	863,758	53,185	6.2	1,220,355	65,346	5.4
Nevada Test Site	12,421	2,005	16.1	777,797	186,714	24.0

largely within 80 km (50 mi) of the candidate management site at the Idaho National Engineering Laboratory. Hispanics and African Americans compose nearly 85 percent of the minority population surrounding the Nevada Test Site. The total and minority populations residing within 80 km (50 mi) of the Nevada Test Site are ten times smaller than those of each of the other sites. The Oak Ridge Reservation is surrounded by the smallest percentage of minorities among the five candidate management sites. Minorities residing within 80 km (50 mi) of the site comprise approximately 6 percent of the total population, and African Americans make up nearly 75 percent of this minority population. The Savannah River Site has the largest surrounding minority population of the five candidate interim management sites: African Americans compose approximately 94 percent of the minority population residing within 80 km (50 mi) of this site.

Figures A-23 to A-27 show the distribution of minorities residing within 80 km (50 mi) of each of the candidate management sites. These illustrations were obtained from an analysis of 1990 census data using a geographical information system. The data were obtained from U.S. Bureau of the Census Tiger Line files which contain political boundaries and geographical features, and Summary Tape Files which contain demographic information. Data were resolved to the block group level, usually 250 to 550 household units. In the legend of each figure, "P" denotes the percentage of the total population within block groups comprised of minority members. The most heavily shaded areas shown in these figures indicate block groups for which the minority population exceeds 50 percent.

The minority population residing near the Hanford Site is spread throughout the area with concentrations in directions northeast, southeast, and southwest of the site. By contrast, the minority population surrounding the Idaho National Engineering Laboratory resides in quadrants northeast and southeast of the site. None of the block groups located within 80 km (50 mi) of the Nevada Test Site contained 50 percent of minority residents during the 1990 census. Due to the sparse population surrounding the site, block groups would be relatively large in geographical area. Minorities within 80 km (50 mi) of the Savannah River Site reside throughout the area with concentrations south of the site. As discussed above, no significant radiological health effects are expected for workers or the general population surrounding the five candidate interim management sites, including minority or low income workers.

A.5.2 Distribution of Low-Income Households Near the Candidate Management Sites

Table A-8 demonstrates the number of low-income households in areas surrounding the candidate interim management sites. Except for the Nevada Test Site, the number of low-income households immediately surrounding the sites is typical of the corresponding number for surrounding counties. In the case of the Nevada Test Site, the percentage of low-income households in the area surrounding the site is noticeably larger than that for the relatively affluent nearby counties.

Table A-7 Racial and Ethnic Composition of Minority Populations Residing Within 80 km of Candidate Management Sites

Candidate Management Sites	*Total Pop.*	*Total Minority Pop.*	*% Minority Pop.*	*Amer. Indian, Eskimo, or Aleut Pop.*	*% Amer. Indian, Eskimo, or Aleut*	*Asian or Pacific Islander Pop.*	*% Asian or Pacific Islander*	*African Amer. Pop.*	*% African Amer.*	*Hispanic Origin Pop.*	*% Hispanic Origin*	*Other Race*	*% Other Race*	*White*	*% White*
SRS	566,823	214,016	37.8	1,136	0.2	5,557	1.0	201,302	35.5	5,874	1.0	145	0.0	352,807	62.2
INEL	176,311	15,449	8.8	3,977	2.3	1,753	1.0	505	0.3	9,075	5.1	138	0.1	160,862	91.2
HS	383,934	95,042	24.8	7,913	2.1	5,296	1.4	4,331	1.1	76,933	20.0	568	0.1	288,891	75.2
ORR	863,758	53,185	6.2	2,985	0.3	4,695	0.5	40,695	4.7	4,518	0.5	290	0.0	810,573	93.8
NTS	12,421	2,005	16.1	273	2.2	49	0.4	634	5.1	1,048	8.4	0	0.0	10,415	83.8

SRS = Savannah River Site, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, ORR = Oak Ridge Reservation, NTS = Nevada Test Site

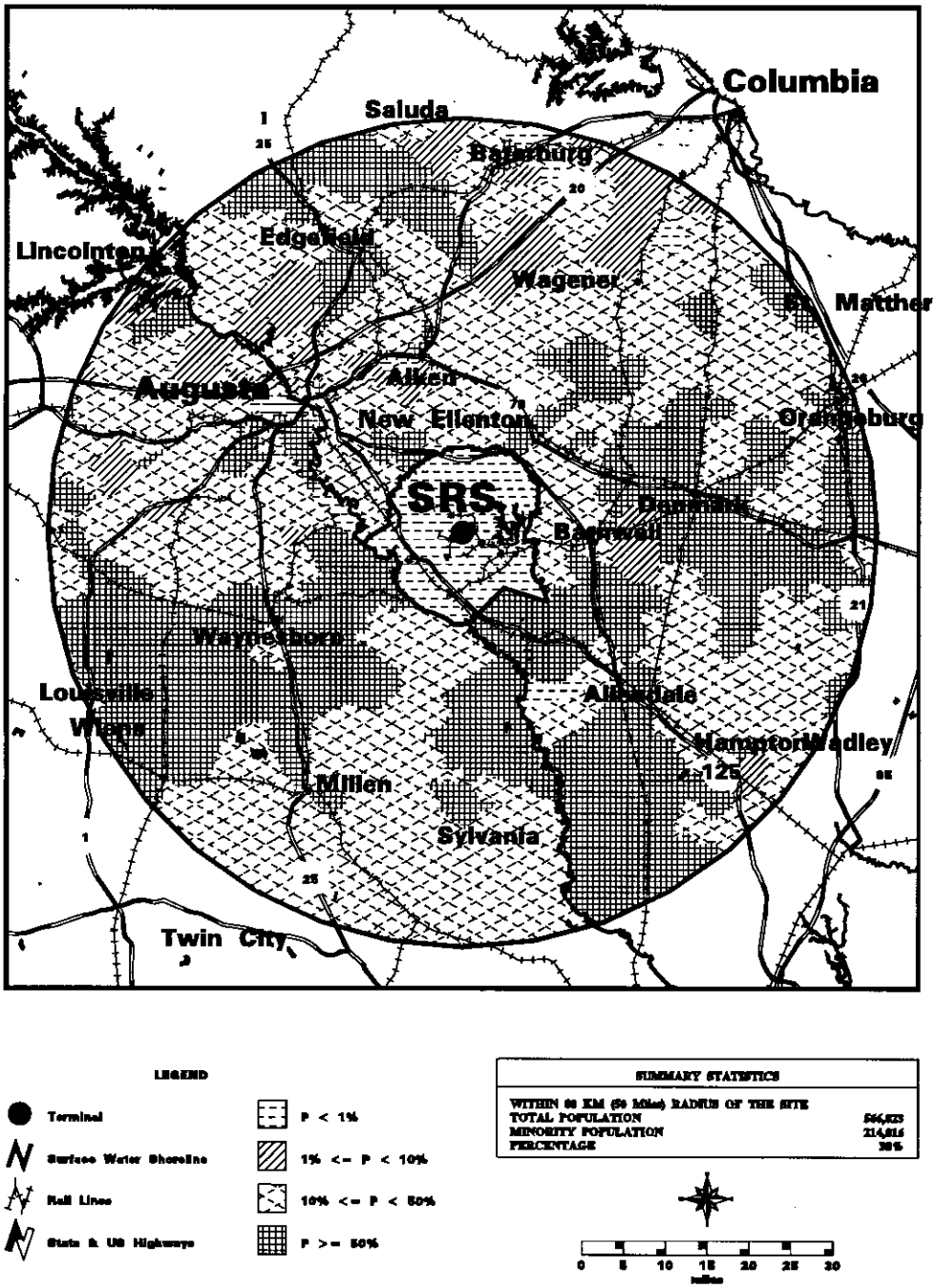
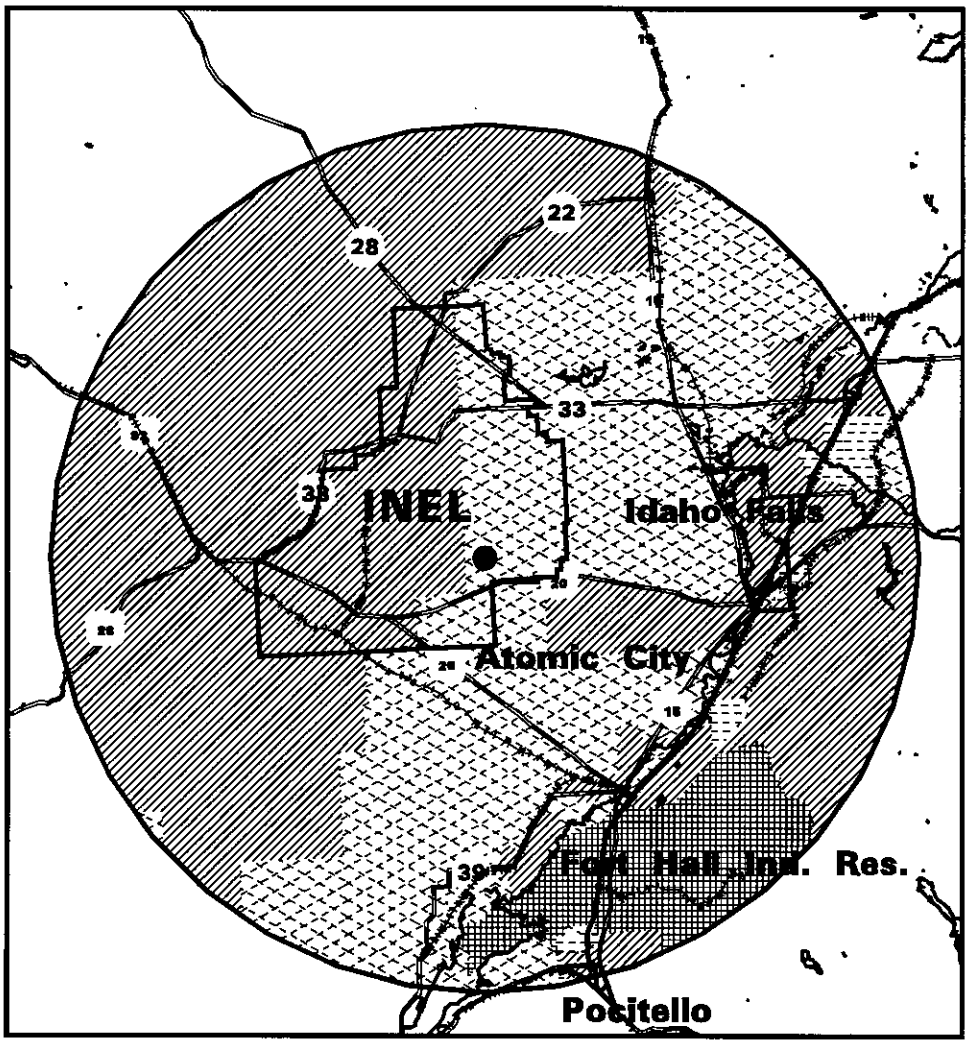


Figure A-23 Distribution of the Minority Population Residing within 80 km of the Savannah River Site



LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 10%
- 10% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 80 KM (50 Miles) RADIUS OF THE SITE	
TOTAL POPULATION	176,311
MINORITY POPULATION	15,449
PERCENTAGE	9%

0 5 10 15 20 25 30 miles

Figure A-24 Distribution of the Minority Population Residing within 80 km of the Idaho National Engineering Laboratory

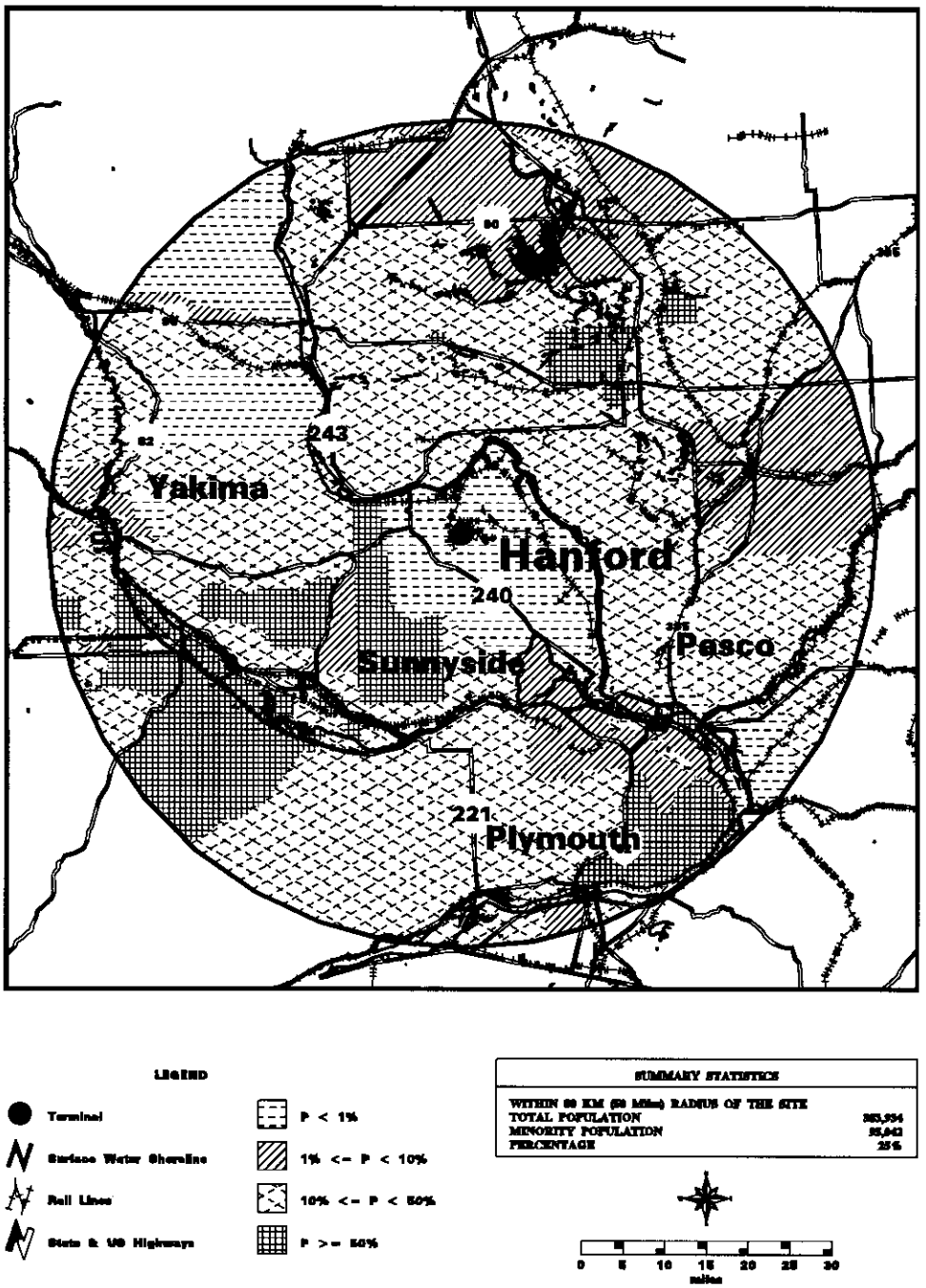


Figure A-25 Distribution of the Minority Population Residing within 80 km of the Hanford Site

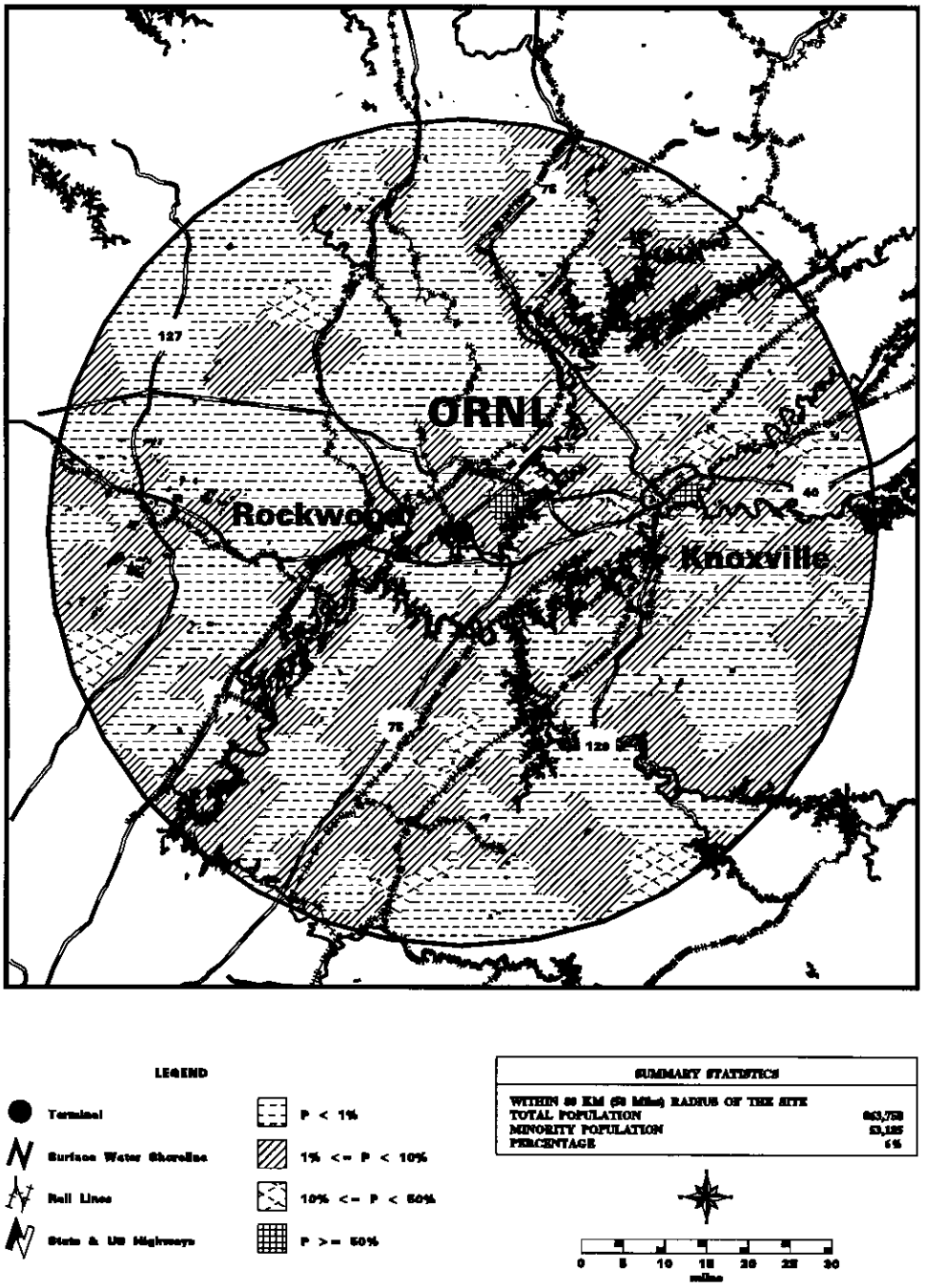


Figure A-26 Distribution of the Minority Population Residing within 80 km of the Oak Ridge Reservation

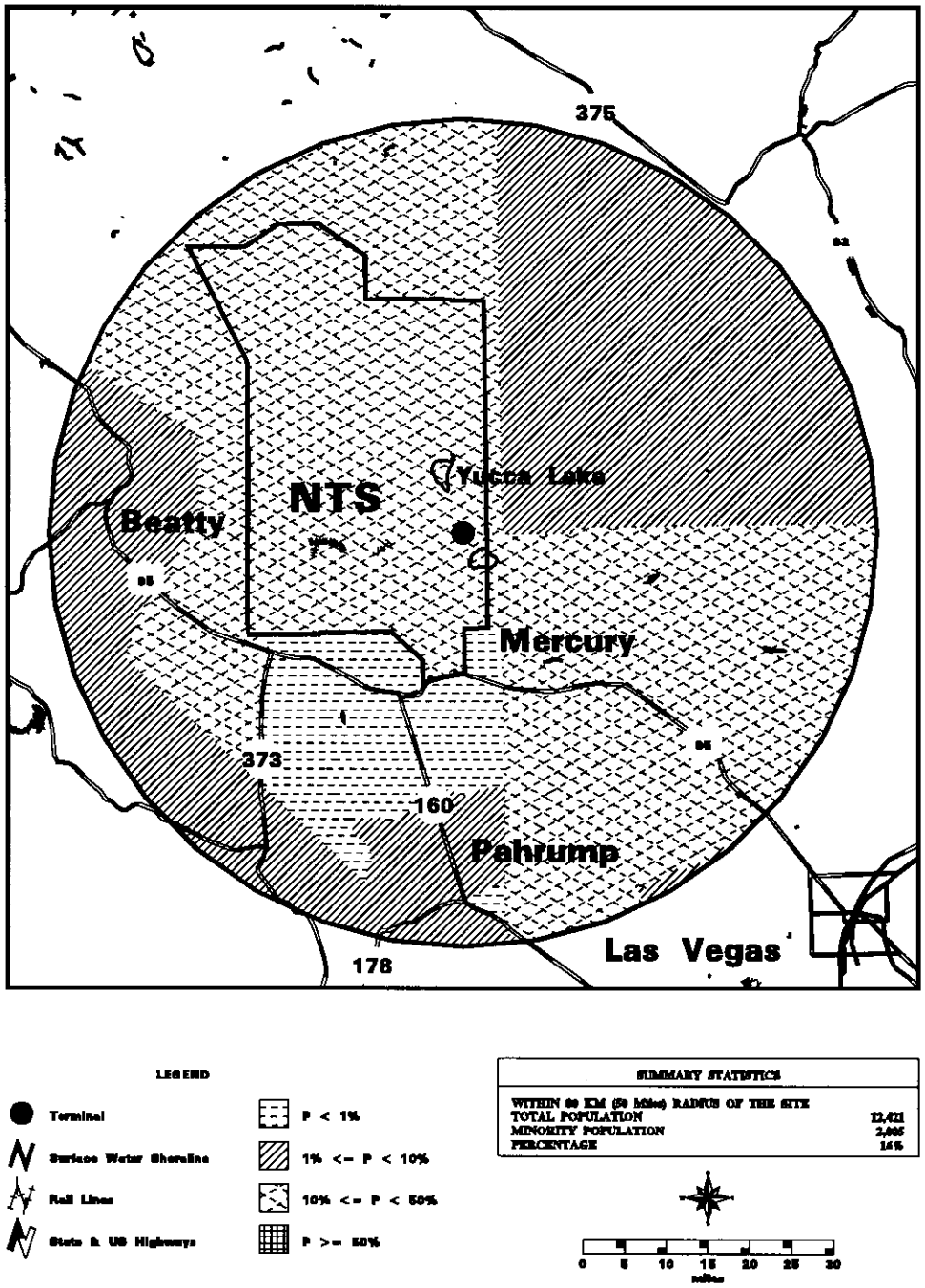


Figure A-27 Distribution of the Minority Population Residing within 80 km of the Nevada Test Site

Table A-8 Low-Income Households Near the Candidate Interim Management Sites

Candidate Management Site	*Households within 80 km of Site*	*Low-Income Households within 80 km of Site*	*% Low-Income Households within 80 km of Site*	*Households in Counties Surrounding Site*	*Low-Income Households in Counties Surrounding Site*	*% Low-Income Households in Counties Surrounding Site*
Savannah River Site	197,937	82,930	41.9	332,193	137,883	41.5
Idaho National Engineering Laboratory	55,109	22,452	40.7	87,723	36,821	42.0
Hanford Site	136,496	57,667	42.2	204,501	86,693	42.4
Oak Ridge Reservation	335,589	147,537	44.0	468,276	206,898	44.2
Nevada Test Site	4,194	2,024	48.3	301,810	119,625	39.6

Figures A-28 through A-32 show the distribution of low-income households within 80 km (50 mi) of each of the candidate interim management sites. The symbol "P" in each legend represents the percentage of low-income households. The heaviest shading indicates where these households total 50 percent or more.

For the Hanford Site, the Idaho National Engineering Laboratory, and the Nevada Test Site, block groups containing 50 percent or more low-income households lie largely south of the site. Low-income households reside throughout the 80-km (50-mi) radius, centered at the Savannah River Site. For the proposed action, no disproportionately high adverse effects are projected for low-income households in the vicinity of the interim management sites.

Characterization of minority and low-income populations residing within a geographical area is sensitive to the basic definitions and assumptions used in conducting the analysis to identify them. Both the Interagency Working Group and DOE are in the process of preparing final guidelines for use in the evaluation of environmental justice. In the absence of final guidance, the definitions and approaches being used by and within Federal agencies could vary. For example, this Final EIS and the Programmatic SNF&INEL EIS present demographic characterizations obtained from the same Census Bureau data base, but use different definitions and assumptions.

The differences in the definitions and assumptions between the Programmatic SNF&INEL EIS and the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Final EIS are as follows:

1. Although both of these EISs use the same 1990 U.S. Census Bureau data base, the Programmatic SNF&INEL EIS uses data aggregated at the census tract level (2,500 to 8,000 persons) while this Final EIS uses data aggregated at the block group level (250 to 550 housing units).

2. In some cases, census blocks or tracts lie partly within the area being analyzed (i.e., within the 80-km (50-mi) radius circle around a potential spent nuclear fuel management site). Since the exact distribution of the populations within such blocks or tracts is not available, the data is insufficient to allow a precise count. To address this situation, the Programmatic SNF&INEL EIS includes a low-income or minority population in its analyses if 50 percent or more of the tract falls within an 80-km (50-mi) radius around the site being considered. In similar situations, this Final EIS assumes that the general population and the minority population are distributed uniformly throughout a block group, and includes the fraction of the low-income or minority population that corresponds to the fraction of the census block group area that falls within the 80-km (50-mi) radius circle.

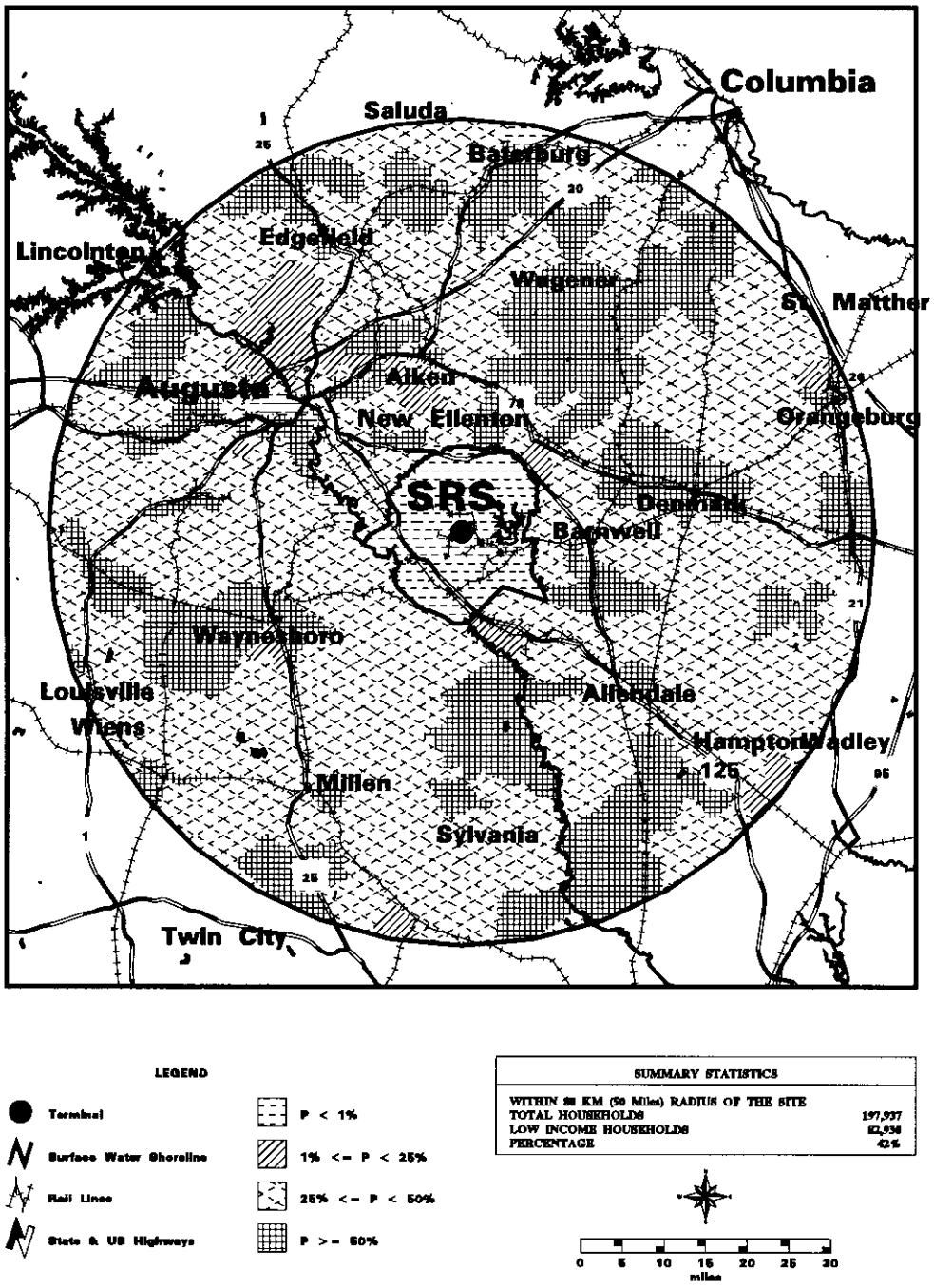


Figure A-28 Distribution of Low-Income Households Residing within 80 km of the Savannah River Site

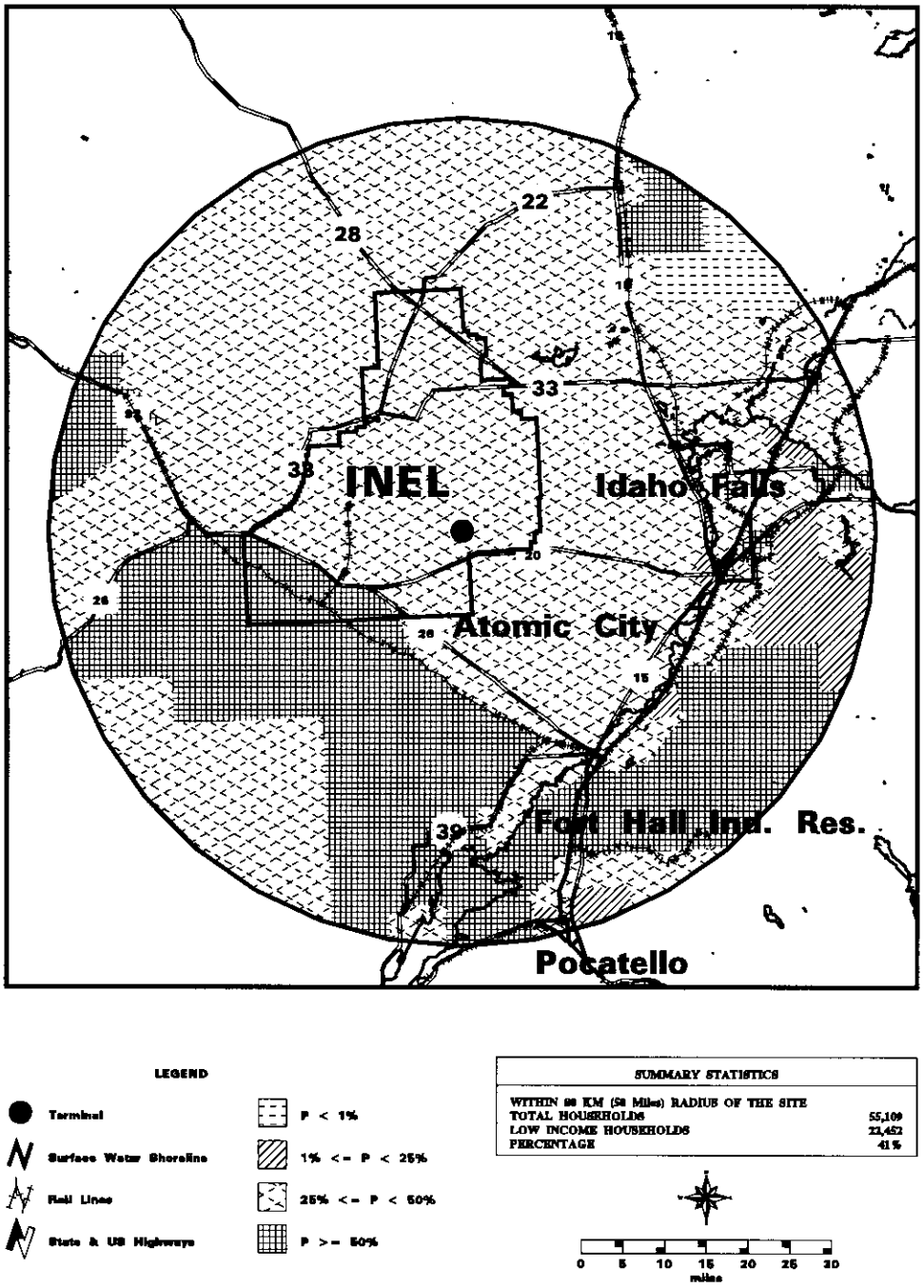
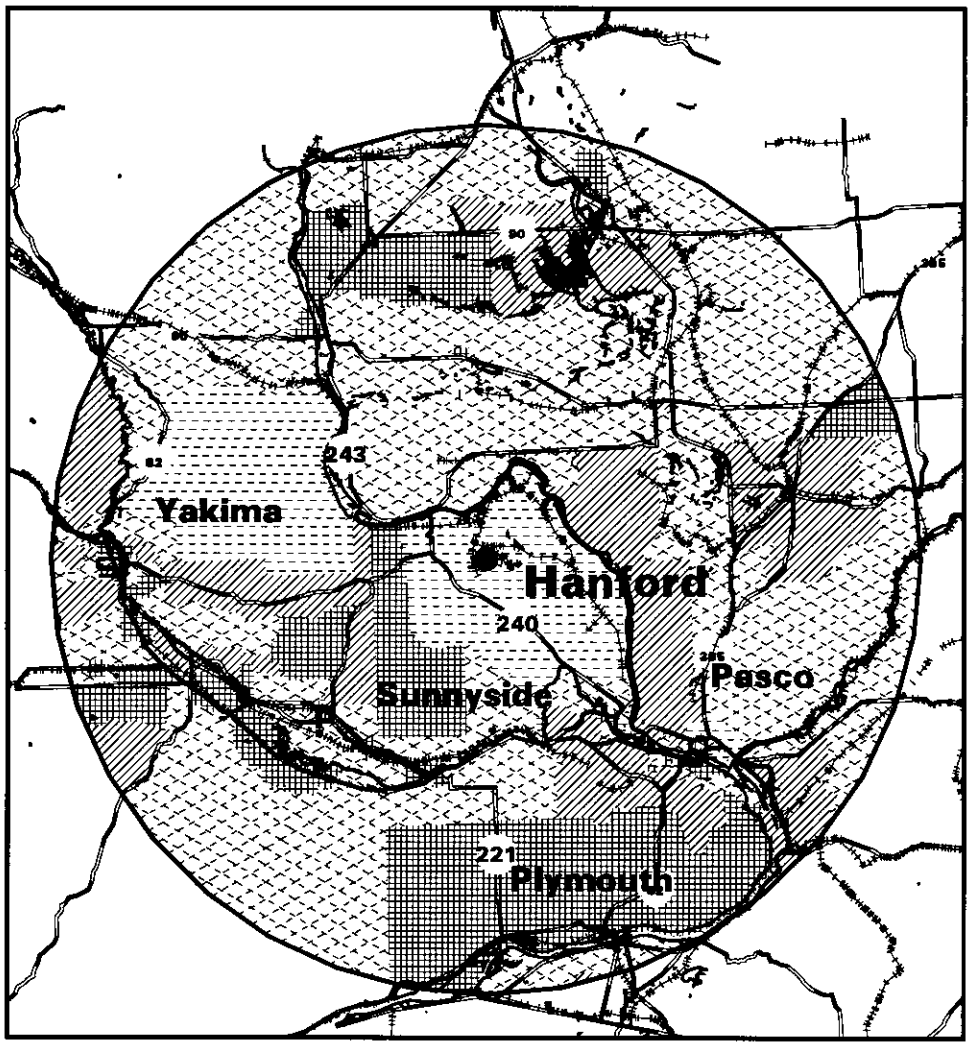


Figure A-29 Distribution of Low-Income Households Residing within 80 km of the Idaho National Engineering Laboratory



LEGEND

- Terminal
- Surface Water Shoreline
- Rail Lines
- State & US Highways
- P < 1%
- 1% <= P < 25%
- 25% <= P < 50%
- P >= 50%

SUMMARY STATISTICS	
WITHIN 80 KM (50 Miles) RADIUS OF THE SITE	
TOTAL HOUSEHOLDS	136,496
LOW INCOME HOUSEHOLDS	57,667
PERCENTAGE	42%

0 5 10 15 20 25 30
miles

Figure A-30 Distribution of Low-Income Households Residing within 80 km of the Hanford Site

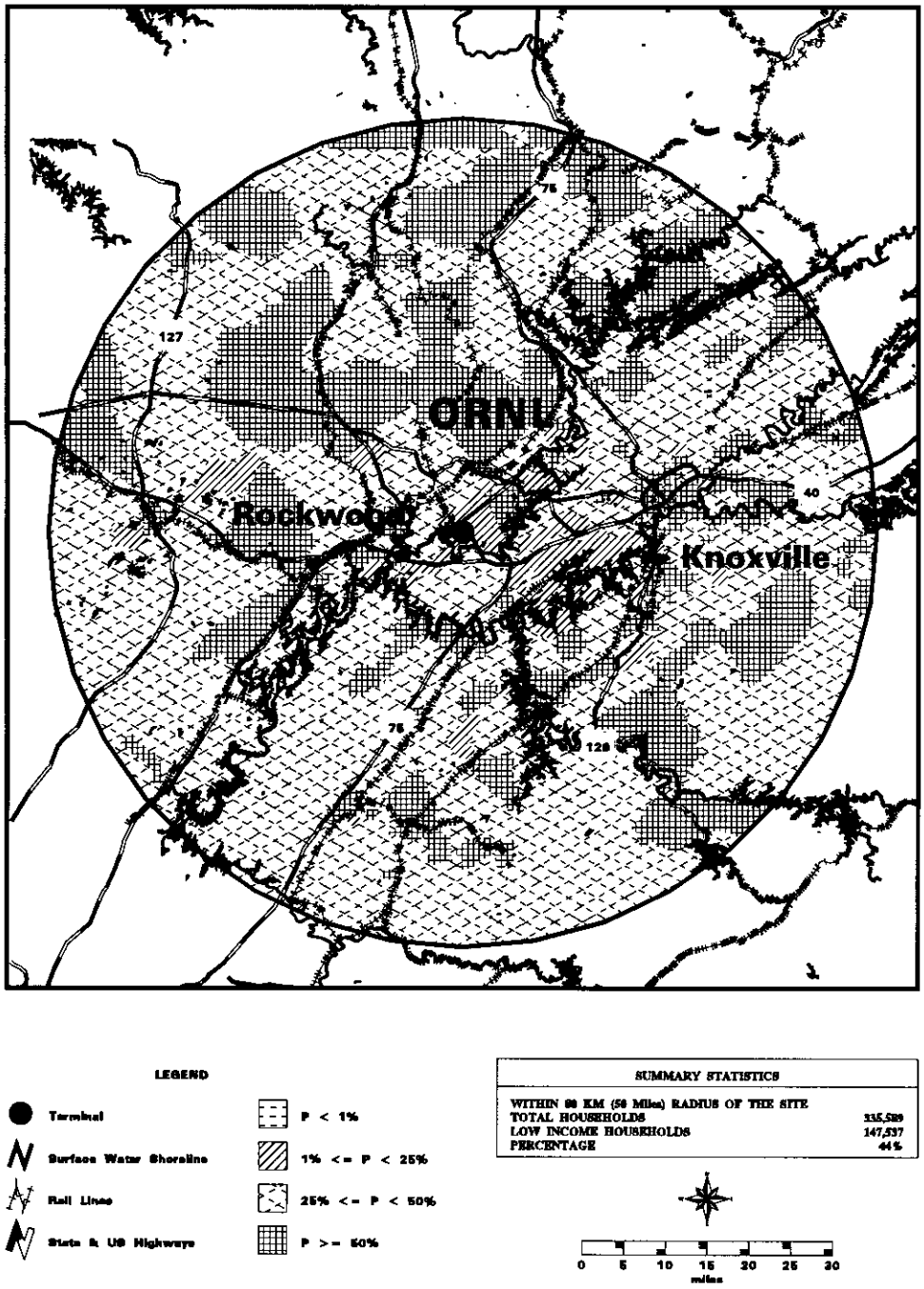


Figure A-31 Distribution of Low-Income Households Residing within 80 km of the Oak Ridge Reservation

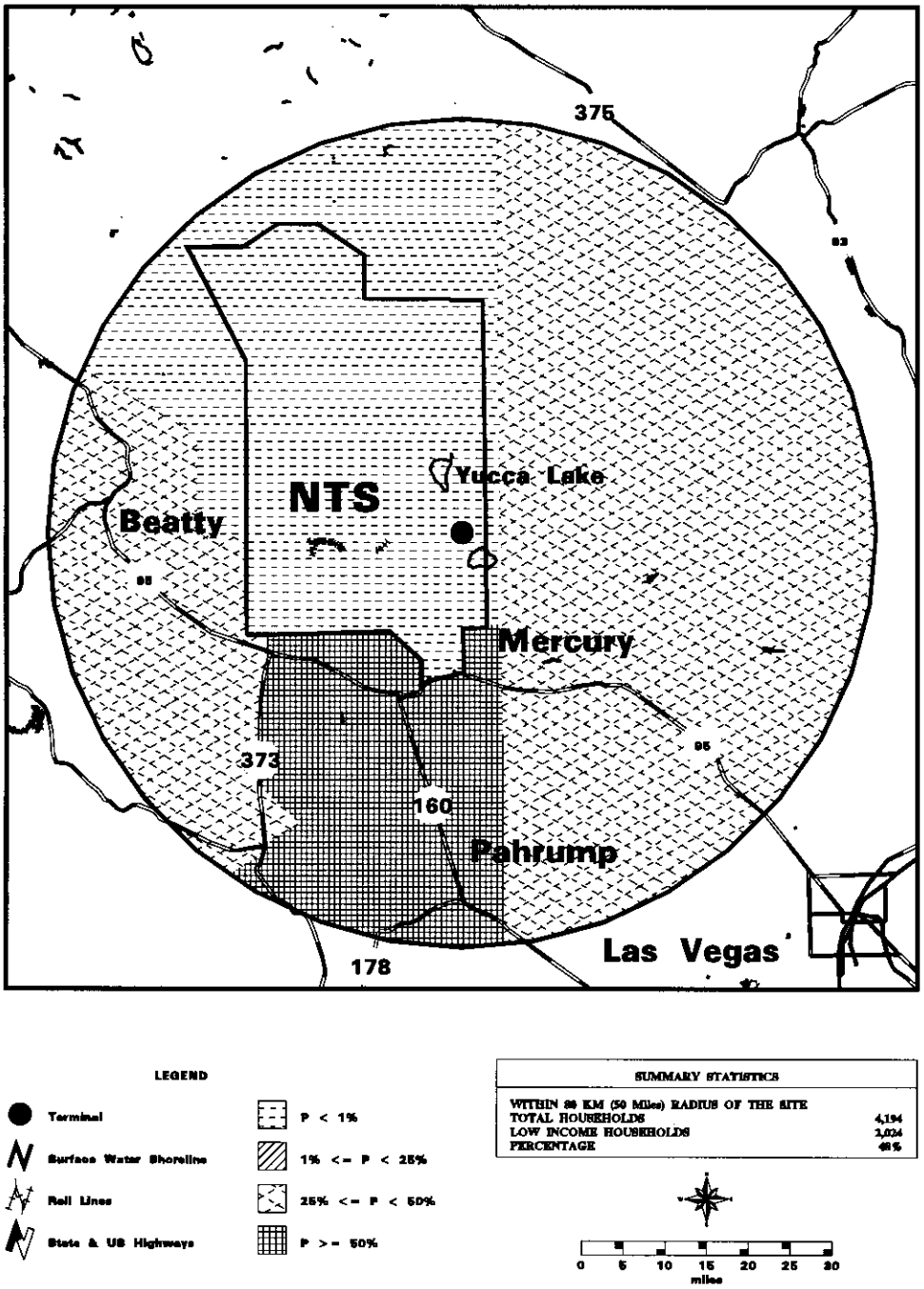


Figure A-32 Distribution of Low-Income Households Residing within 80 km of the Nevada Test Site

3. The Programmatic SNF&INEL EIS defines low-income populations as those in a poverty status as determined annually by the U.S. Census Bureau, based on the Consumer Price Index, and aggregated by the thresholds set forth by the Census Bureau (i.e., a group of people and/or a community experiencing common conditions of exposure or impact, in which 25 percent or more of the population is characterized as living in poverty), a method used by the U.S. Environmental Protection Agency. This Final EIS uses the definition of low-income community established by the U.S. Department of Housing and Urban Development (given in Section A.2 above). Both definitions are permitted under the draft guidance developed by the Interagency Working Group.

These different definitions and assumptions have resulted in differences in the characterization of low-income and minority populations. The two sets of data are summarized in Tables A-9 and A-10 and the most significant differences are discussed below.

Table A-9 Comparison of the Programmatic SNF&INEL EIS's and the FRR SNF Final EIS's Minority Characterization Results

Candidate Management Site	*Total Individuals Residing within 80 km (50 mi)*		*Minority Individuals Residing within 80 km (50 mi)*		*% of Minority Individuals Residing within 80 km (50 mi)*	
	Programmatic SNF&INEL EIS	*FRR SNF Final EIS*	*Programmatic SNF&INEL EIS*	*FRR SNF Final EIS*	*Programmatic SNF&INEL EIS*	*FRR SNF Final EIS*
Savannah River Site	619,959	566,823	233,955	214,016	37.7	37.8
Idaho National Engineering Laboratory	172,366	176,311	11,722	15,449	6.8	8.8
Hanford Site	370,807	383,934	75,381	95,042	20.3	24.8
Oak Ridge Reservation	867,231	863,758	49,742	53,185	5.7	6.2
Nevada Tests Site	11,918	12,421	759	2,005	6.4	16.1

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

Table A-10 Comparison of the Programmatic SNF&INEL EIS's and the FRR SNF Final EIS's Low-Income Characterization Results

Candidate Management Site	*Total Population Residing within 80 km (50 mi)*		*Low-Income Group Residing within 80 km (50 mi)*		*% of Low-Income Group Residing within 80 km (50 mi)*	
	Programmatic SNF&INEL EIS (Individuals)	*FRR SNF Final EIS (Households)*	*Programmatic SNF&INEL EIS (Individuals)*	*FRR SNF Final EIS (Households)*	*Programmatic SNF&INEL EIS*	*FRR SNF Final EIS*
Savannah River Site	619,959	197,937	107,764	82,930	17.4	41.9
Idaho National Engineering Laboratory	172,366	55,109	23,416	22,452	13.6	40.7
Hanford Site	370,807	136,496	65,584	57,667	17.7	42.2
Oak Ridge Reservation	867,231	335,589	134,661	147,537	15.5	44.0
Nevada Tests Site	11,918	4,194	1,474	2,024	12.4	48.3

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

The minority populations identified are reasonably consistent between the Programmatic SNF&INEL EIS and the FRR SNF Final EIS, except for results obtained at the Nevada Test Site (the largest proportional difference) and the Hanford Site (the largest difference in numbers of individuals), as shown in Table A-9. The range in results for both locations is due to the different aggregations of the demographic data used (census tracts vs. blocks), and the differences in the methods used to account for the population of tracts or groups lying only partly within the area being analyzed, as discussed above. For example, both sites are

located in rural or sparsely populated regions so that census tracts surrounding the sites are relatively large in geographical area. In addition, the outskirts of Las Vegas, Nevada begin approximately 80 km (50 mi) from the Nevada Test Site, making the analysis particularly sensitive to differences in treatment of census tracts or block groups that lie partly within a circle of an 80-km (50-mi) radius centered at that site. Most areas within the zone of impact of the Nevada Test Site are restricted access and unpopulated land.

As a result of the different definitions used for identification of low-income populations, the results of these analyses are markedly different, as shown in Table A-10. Both sets of data are correct. They simply reflect the fact that different definitions and assumptions can result in different characterizations of low-income populations.

The approach to evaluating environmental justice used in this document may change as a result of future guidance issued by the Interagency Working Group or DOE. Nevertheless, as demonstrated by the different approaches discussed above, the conclusions are not expected to change because the impacts resulting from the proposed action under all alternatives present no significant risk to the potentially affected populations. As a result, no disproportionately high and adverse effects would be expected for any particular segment of the population, including minority and low-income populations.

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DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix B
Foreign Research Reactor Spent Nuclear Fuel Characteristics and Transportation Casks

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

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Appendix B
Foreign Research Reactor Spent Nuclear Fuel Characteristics and Transportation Casks

B.1 Spent Nuclear Fuel Characteristics

This section presents relevant characterization and other information on foreign research reactor spent nuclear fuel that could be managed under the proposed action. The information includes:

- Estimated amounts of spent nuclear fuel;
- A list of research reactors and foreign countries from which the spent nuclear fuel would originate;
- A description of fuel type design along with important characteristics regarding fuel design, geometry, and burnup;
- A description of the radionuclide inventories for the bounding spent nuclear fuel type; and
- An estimation of the number of foreign research reactor spent nuclear fuel shipments.

B.1.1 Estimated Amount of Spent Nuclear Fuel

The proposed action is for the U.S. Department of Energy (DOE) and Department of State to adopt a policy to manage foreign research reactor spent nuclear fuel which contains uranium enriched in the United States in a manner consistent with the goals of the U.S. nuclear weapons nonproliferation policy (see Chapter 2). The amount of spent nuclear fuel from foreign research reactors that would be managed during the policy period (1995-2005) is approximately 19.2 metric tons of heavy metal (MTHM) with a volume of approximately 110 m^3 (4,100 ft^3) representing approximately 22,700 elements[1] (see Tables B-1 and B-2). Tables B-1 and B-2 provide an estimate of the total amount of spent nuclear fuel that is currently stored or could be generated in each country by late 2005 (Matos, 1994). These tables also provide the estimated number of shipments expected from each country. The number of shipments is a key parameter in evaluating the potential risks associated with the handling and transportation of foreign research reactor spent nuclear fuel (see Section B.1.6). It should be noted that the number of spent nuclear fuel elements and the number of shipments presented for each country in this appendix are estimates based on projections of the numbers of elements to be discharged from foreign research reactors in each country listed over a 10-year period into the future. These estimates are intended to conservatively bound the total number of foreign research reactor spent nuclear fuel elements and shipments associated with the proposed policy. However, the actual distribution of elements and shipments among the listed countries might change, within the limits of the total numbers of elements and shipments listed, based on actual experience gained during the lifetime of any policy that may be established.

---

1 *Various fuel forms and geometries are used in the foreign research reactors (see Section B.1.3). In order to reduce confusion, each individual spent nuclear fuel is called a spent nuclear fuel "element." An element could be an assembly, a rod, a pin, or a cluster of rods or pins.*

Table B-1 Estimated Number of Aluminum-Based Spent Nuclear Fuel Elements of U.S.-Origin Generated by Foreign Research Reactor Operators by January 2006

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium (kg)*[b]	*Estimated Number of Shipments*
Argentina[a]	283	71	9
Australia	975	427	9
Austria	157	191	5
Belgium	1,766	730	59
Brazil[a]	155	99	5
Canada	2,831	4,478	116
Chile[a]	58	12	2
Colombia[a]	16	2	1
Denmark	660	529	22
France	1,962	3,442	149
Germany	1,504	909	49
Greece[a]	239	113	8
Indonesia[a]	198	236	6
Iran[a]	29	6	1
Israel	192	111	6
Italy	150	43	5
Jamaica[a]	2	1	1
Japan	2,981	3,134	99
Korea (South)[a]	168	321	7
Netherlands	1,488	1,404	49
Pakistan[a]	82	16	3
Peru[a]	29	39	1
Philippines[a]	50	24	2
Portugal[a]	88	54	3
South Africa[a]	50	10	2
Spain (from Scotland)[c]	40	16	1
Sweden	1,113	1,374	37
Switzerland	159	128	5
Taiwan	127	66	4
Thailand[a]	31	5	1
Turkey[a]	69	89	2
United Kingdom	12	4	1
Uruguay[a]	19	18	1
Venezuela[a]	120	82	4
Total	**17,803**	**18,184**	**675**

[a] *Countries other than high-income economies (World Bank, 1994). These are considered to be "developing" countries.*

[b] *To derive uranium mass in pounds, multiply the amount by 2.2.*

[c] *40 spent nuclear fuel elements of Spain's JEN-1 Reactor core are stored in Dounreay, Scotland.*

In addition, in this Environmental Impact Statement (EIS), DOE is considering potential management of highly-enriched uranium (HEU) and low enriched uranium (LEU) target materials from three countries: Canada, Belgium, and Indonesia. These countries have used, and will be using, target fuels which contain U.S.-origin enriched uranium to produce molybdenum-99 (^{99}Mo), which decays to technetium-99 (^{99}Tc), a medical isotope. The amount of target materials that is expected to be brought back to the United States would contain about 556 kg of uranium in 56 to 140 shipments (see Section B.1.5 for detail).

Table B-2 Estimated Number of TRIGA Reactor Spent Nuclear Fuel Elements of U.S.-Origin Generated by Foreign Research Reactor Operators by January 2006

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium (kg)*[b]	*Estimated Number of Shipments*
Austria	106	20	3
Bangladesh[a]	100	49	3
Brazil[a]	75	14	3
Finland	171	33	6
Germany	358	68	12
Indonesia[a]	245	47	8
Italy	386	72	13
Japan	326	62	11
Korea (South)[a]	336	64	11
Malaysia[a]	94	47	3
Mexico[a]	186	35	6
Philippines[a]	128	79	4
Romania[a]	1,451	189	48
Slovenia[a]	393	75	13
Taiwan	144	86	5
Thailand[a]	136	35	4
Turkey[a]	79	15	2
United Kingdom	90	17	3
Zaire[a]	136	26	4
Total	**4,940**	**1,033**	**162**

[a] *Countries other than high-income economies (World Bank, 1994). These are identified as "developing" countries.*

[b] *To derive uranium mass in pounds, multiply the amount by 2.2.*

The information provided in Tables B-1 and B-2, with regards to the number of spent nuclear fuel elements and the amount of initial mass of uranium, is based on the following assumptions and considerations as compiled by J. Matos of Argonne National Laboratory (Matos, 1994).

B.1.1.1 Fuel Type

Under the "Offsite Fuels Policy" that was in effect during 1988, DOE accepted aluminum-based and Training, Research, Isotope, General Atomic (TRIGA) research reactor fuels[2] for disposition (DOE 1986, and 1987). The "Offsite Fuels Policy" and the current proposed policy pertain to irradiated fuels from foreign nuclear research reactors other than those involved in the conduct of research and development activities leading to demonstration of the practical value of such reactors for industrial or commercial purposes. Specifically, the "Offsite Fuels Policy" and the proposed policy apply solely to the following types of reactor fuels:

2 *Aluminum-based fuel is aluminum-clad and has an active fuel region that consists of an alloy of uranium and aluminum or a dispersion of uranium-bearing compound (e.g., $UA1_x$, U_3O_8, U_3Si_2, U_3Si) in aluminum. TRIGA fuel consists of an alloy of uranium and zirconium and is clad in either aluminum, incoloy, or stainless steel.*

1. Aluminum-clad reactor fuels where the uranium-235 (^{235}U) content is equal to or greater than 20 percent, by weight, of the total uranium content (i.e., HEU fuel). The active fuel region of these fuels may be configured as uranium-aluminum alloy, uranium-oxide[3] or uranium-aluminide. Spent nuclear fuels containing significant quantities of uranium-233 (^{233}U) are excluded from receipt.

2. Aluminum-clad reactor fuels where the ^{235}U content is less than 20 percent by weight of the total uranium content (i.e., LEU fuel). The active fuel regions of these fuels may be configured as uranium-silicide, uranium-aluminide or uranium-oxide. Fuels containing significant quantities of ^{233}U are excluded from receipt.

3. Aluminum-, incoloy-, or stainless steel-clad, uranium-zirconium hydride (other than ^{233}U) TRIGA fuel types.

In addition to the aluminum-based and TRIGA fuel types discussed above, U.S.-origin enriched uranium is also used in the fuel elements of several fast reactors and other special purpose reactors, in the UO_2 rodded fuel assemblies of several thermal research reactors, and in thermal homogeneous liquid and solid fueled reactors. The enrichment of the uranium ranges from 2 percent to 93 percent. These fuels do not qualify for management under the proposed policy because they were not included in the fuel types that were eligible for return to the United States under the "Offsite Fuels Policy" that was in effect in 1988.

B.1.1.2 Data Sources and Assumptions

Information on current spent nuclear fuel inventories containing U.S.-origin enriched uranium at foreign research reactors and temporary storage facilities was obtained from several sources: (1) questionnaires sent out by DOE and returned by foreign research reactor organizations in 1993 and 1994, (2) data summarized from irradiated fuel questionnaires sent out by and returned to the International Atomic Energy Agency in 1993 and 1994, and (3) Reduced Enrichment for Research and Test Reactors (RERTR) Program information on foreign research reactor fuel inventories, operation, and fuel cycles. Additional information on reactor fuel characteristics and reactor operation was obtained from directories of nuclear research reactors published by the International Atomic Energy Agency (IAEA, 1989).

Beginning with irradiated fuel inventory data, several assumptions were made, first to normalize the data to a common starting date of January 1996, and then to estimate the number of irradiated fuel elements in reactor cores and the number of spent nuclear fuel elements that could be generated during the 10-year policy period (1995-2005). These assumptions are:

1. Most foreign research reactors will continue operation during the 10-year policy period. If a permanent shutdown date has been specified by the research reactor operator, irradiated fuel was accumulated to that date only.

2. The number of irradiated fuel elements in each reactor core was determined from available reports and publications, or estimated. The estimated number of spent nuclear fuel elements covered under the proposed policy includes the inventory within the core of each research reactor at the end of the policy period. This would account for fuel elements in the reactor core of research reactors that shut down during, or at the end of, the policy period.

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3 *This uranium-oxide composition refers to aluminum-clad fuel plates or tubes containing dispersions of U_3O_8 in aluminum. It does not include fuels containing UO_2 pellets clad in aluminum, zircaloy, stainless steel, or other materials.*

3. Known current and planned shutdowns for prolonged periods of maintenance and refurbishment have been incorporated into the estimates.

4. Dates for conversion from HEU to LEU fuel have been estimated, and the enrichment change was incorporated into the inventory data.

5. Estimated irradiated fuel inventories have been included for reactors that are under construction and plan to begin operation before the Record of Decision date (assumed here to be December 31, 1995) of the proposed policy using U.S.-origin enriched uranium.

6. Spent nuclear fuel from previously shutdown reactors with fuel in temporary storage has been included.

B.1.1.3 Foreign Research Reactors Eligible for Inclusion in this EIS

There are 104 research and test reactors located in 41 foreign countries that possess aluminum-based and TRIGA fuels containing U.S.-origin enriched uranium. These foreign research reactors are listed in Tables B-3 through B-5. Table B-3 lists 76 reactors that possess aluminum-based fuel only. These foreign research reactors are arranged in a number of categories that depend on each reactor's LEU conversion status. Table B-4 lists 25 foreign research reactors that possess TRIGA fuel only. Table B-5 lists three foreign research reactors that were converted from HEU aluminum-based fuel to LEU TRIGA fuel and thus possess both aluminum-based and TRIGA spent nuclear fuels.

B.1.1.4 Developing Countries

For purposes of this EIS, developing countries are defined as countries having other than high-income economies, on the basis of per capita Gross Domestic Product, by the World Bank (World Bank, 1994). Two countries, Zaire and Taiwan, were not listed in the World Bank report. Zaire is considered here to have a low-income economy; and Taiwan, with an estimated per capita Gross Domestic Product of $10,900 (1994), is considered to have a high-income economy. The countries shown below qualify as developing countries according to this criterion:

List of Developing Countries

Low Income Economies	*Lower Middle Income Economies*		*Upper Middle Income Economies*	
Bangladesh	Chile	Romania	Argentina	Slovenia
Indonesia	Colombia	Thailand	Brazil	South Africa
Pakistan	Iran	Turkey	Greece	South Korea
Zaire	Jamaica		Malaysia	Uruguay
	Peru		Mexico	Venezuela
	Philippines		Portugal	

B.1.2 General Characteristics of Nuclear Fuels and Spent Nuclear Fuel

Nuclear fuels consist of fissile materials that produce a net increase in neutrons when they absorb neutrons, and fertile materials that produce fissile material when they absorb neutrons. The principal fissile materials are ^{235}U, Plutonium-239 (^{239}Pu), and ^{233}U (Plutonium-241 or ^{241}Pu is also of some importance). The principal fertile materials are uranium-238 (^{238}U) and Thorium-232 (^{232}Th) (Plutonium-240 or ^{240}Pu and uranium-234 or ^{234}U also play roles as fertile materials). The only fissile

Table B-3 Foreign Research and Test Reactors that Possess Only Aluminum-Based Fuel Containing HEU and LEU of U.S.-Origin

	Reactor	Country	Power, MW	Fuel Geometry	Initial Enrichments[a] Wt-% ^{235}U, U.S. Origin			Comment (see Note)
					Enr.1	Enr.2	Enr.3	
HEU Reactors Fully- or Partially-Converted to LEU Fuel								
1	RA-3	Argentina	3	Plates	90	-	-	(1)
2	ASTRA	Austria	10	Plates	93	45	20	
3	IEA-R1	Brazil	2	Plates	93	-	20	
4	NRU	Canada	125	Pin Cluster	93	-	20	
5	DR-3	Denmark	10	Tubes	93	85	20	
6	OSIRIS	France	70	Plates	-	-	20	
7	FRG-1	Germany	5	Plates	93	-	20	
8	NRCRR	Iran	5	Plates	93	-	-	(2)
9	JMTR	Japan	50	Plates	93	45	20	
10	PARR	Pakistan	5	Plates	92	-	-	(2)
11	R2	Sweden	50	Plates	93	-	20	
HEU Reactors that Have Ordered LEU Fuel Elements for Conversion								
12	GRR-1	Greece	5	Plates	93	-	20	(3)
13	HOR	Netherlands	2	Plates	93	-	20	(3)
14	TR-2	Turkey	5	Plates	93	-	20	(3)
HEU Reactors that Can Be Converted to LEU Fuel								
15	RA-6	Argentina	0.5	Plates	90	-	-	
16	HIFAR	Australia	10	Tubes	80	60	20	(3)
17	SAR-GRAZ	Austria	0.01	Plates	90	-	20	
18	MNR	Canada	2	Plates	93	-	20	
19	Slowpoke - Alberta	Canada	0.02	Pin Bundle	93	-	-	
20	Slowpoke - Halifax	Canada	0.02	Pin Bundle	93	-	-	
21	Slowpoke - Montreal	Canada	0.02	Pin Bundle	93	-	-	
22	Slowpoke - Saskatchewan	Canada	0.02	Pin Bundle	93	-	-	
23	Slowpoke - Toronto	Canada	0.02	Pin Bundle	93	-	-	
24	LA REINA	Chile	5	Plates	80	-	-	
25	IAN-R1	Colombia	0.03	Plates	90	-	-	
26	EOLE	France	0.01	Plates	93	-	-	
27	MINERVE	France	0.003	Plates	93	-	-	
28	SCARABEE	France	20	Plates	93	-	-	
29	Strasbourg - Cronenbourg	France	0.1	Plates	90	-	-	
30	Ulyssee - Saclay	France	0.1	Plates	90	-	-	
31	BER-II	Germany	10	Plates	93	-	20	(3)
32	FRJ-2	Germany	23	Tubes	80	-	20	(3)
33	FRM	Germany	4	Plates	93	45	-	
34	IRR-1	Israel	5	Plates	93	-	20	(3)
35	Slowpoke	Jamaica	0.02	Pin Bundle	93	-	-	
36	JMTRC	Japan	0	Plates	93	45	-	
37	JRR-4	Japan	3.5	Plates	93	-	20	(3)
38	KUCA	Japan	0	Plates	93	45	-	
39	KUR	Japan	5	Plates	93	-	20	(3)
40	UTR Kinki	Japan	0	Plates	90	-	-	
41	HFR Petten	Netherlands	45	Plates	93	-	20	(3)
42	LFR	Netherlands	0.03	Plates	93	-	-	
43	RPI	Portugal	1	Plates	93	-	20	
44	SAFARI	S. Africa	20	Plates	93	-	-	(4)

	Reactor	Country	Power, MW	Fuel Geometry	Initial Enrichments[a] Wt-% ^{235}U, U.S. Origin			Comment (see Note)
					Enr.1	Enr.2	Enr.3	
45	R2-0	Sweden	1	Plates	90	-	-	
46	ZPRL	Taiwan	0.01	Plates	93	-	20	
HEU Operating Reactors that Cannot be Converted with Current Technology								
47	BR-2	Belgium	60	Tubes	90-93	-	-	
48	ORPHEE	France	14	Plates	93	-	-	
49	RHF	France	57	Involute Plates	93	-	-	
HEU Operating Reactors Announced to be Shutdown								
50	SILOE	France	35	Plates	93	45	20	
51	SILOETTE	France	0.1	Plates	93	-	-	
52	FMRB	Germany	1	Plates	93	-	-	
53	FRG-2	Germany	15	Plates	90 - 93	-	20	
54	JRR-2	Japan	10	Plates	93	45	-	
55	UTR 300	U. K.	0.3	Plates	90	-	-	
Shutdown Reactors Possessing HEU Fuel								
56	MOATA	Australia	-	Plates	90	-	-	
57	BR-02	Belgium	-	Tubes	90	-	-	
58	NRX	Canada	-	Pin Cluster	93	-	-	
59	PTR	Canada	-	Plates	93	-	-	
60	Slowpoke - Kanata	Canada	-	Pin Bundle	93	-	-	
61	MELUSINE	France	-	Plates	93	-	-	
62	GALILEO	Italy	-	Plates	89	-	-	
63	ISPRA-1	Italy	-	Plates	90	-	-	
64	RANA	Italy	-	Plates	90	-	20	
65	JEN-1	Spain	-	Plates	79	-	20	(5)
66	SAPHIR	Switzerland	-	Plates	93	45	20	
LEU Operating Reactors Possessing Only LEU Fuel								
67	RA-0	Argentina	0.01	Plates	-	-	20	
68	Argonauta	Brazil	0.2	Plates	-	-	20	
69	RSG-GAS30	Indonesia	30	Plates	-	-	20	
70	JRR-3M	Japan	20	Plates	-	-	20	
71	TTR-1	Japan	0.1	Plates	-	-	20	
72	RP-10	Peru	10	Plates	-	-	20	
73	KMRR	S. Korea	30	Pin Cluster	-	-	20	(6)
LEU Shutdown Reactors Possessing Only LEU Fuel								
74	THAR	Taiwan	-	Plates	-	-	20	
75	RU-1	Uruguay	-	Plates	-	-	20	
76	RV-1	Venezuela	-	Plates	-	-	20	

[a] *Initial enrichments, in weight-% ^{235}U, of the fuels possessed or anticipated to be possessed by each reactor. Only fuels containing uranium of U.S.-origin are included.*

Note:

(1) *Converted to LEU fuel of Soviet origin.*

(2) *Converted to LEU fuel of Chinese origin.*

(3) *Use of fuel containing LEU of U.S.-origin is anticipated to begin before 2001.*

(4) *Currently uses HEU of South African origin.*

(5) *JEN-1 fuel is currently being stored in Dounreay, Scotland.*

(6) *The KMRR reactor in South Korea began operation using LEU aluminum-based fuel in January 1995.*

Table B-4 Foreign Research and Test Reactors that Possess Only TRIGA Fuel Containing HEU and LEU of U.S.-Origin

	Reactor	Country	Power, MW	Fuel Geometry	Initial Enrichments[a] Wt-% ^{235}U, U.S. Origin		
					Enr.1	Enr.2	Enr.3
Reactors Possessing HEU Fuel							
1	Vienna	Austria	0.25	Rods	70	-	20
2	Salazar	Mexico	1	Rods	70	-	20
3	SSR	Romania	14	Rods	93	-	20
4	Ljubljana	Slovenia	0.25	Rods	70	-	20
5	Seoul #2	S. Korea	2	Rods	70	-	20
Reactors Possessing LEU Fuel							
6	Dhaka	Bangladesh	3	Rods	-	-	20
7	Belo Horiz.	Brazil	-	Rods	-	-	20
8	Helsinki	Finland	0.25	Rods	-	-	20
9	Hannover	Germany	-	Rods	-	-	20
10	Heidelberg	Germany	0.25	Rods	-	-	20
11	Mainz	Germany	0.1	Rods	-	-	20
12	Bandung	Indonesia	1	Rods	-	-	20
13	Yogyakarta	Indonesia	0.1	Rods	-	-	20
14	Pavia	Italy	0.25	Rods	-	-	20
15	Rome	Italy	1	Rods	-	-	20
16	Mushashi Inst	Japan	0.1	Rods	-	-	20
17	NSRR-Tokai	Japan	0.3	Rods	-	-	20
18	Rikkyo U.	Japan	0.1	Rods	-	-	20
19	Kuala Lumpur	Malaysia	1	Rods	-	-	20
20	ACPR	Romania	0.5	Rods	-	-	20
21	Seoul #1	S. Korea	0.25	Rods	-	-	20
22	Istanbul	Turkey	0.25	Rods	-	-	20
23	Imp Chem Ind.	U. K.	0.25	Rods	-	-	20
24	TRICO II	Zaire	1	Rods	-	-	20
Shutdown Reactors							
25	TRICO I	Zaire	-	Rods	-	-	20

[a] *Initial enrichments, in weight-% ^{235}U, of the fuels possessed by each reactor. Only fuels containing uranium of U.S.-origin are included.*

Table B-5 Foreign Research and Test Reactors that Possess Both Aluminum-Based and TRIGA Fuel Containing HEU and LEU of U.S.-Origin.

	Reactor	Country	Power, MW	Fuel Geometry	Initial Enrichments[a] Wt-% ^{235}U, U.S. Origin		
					Enr.1	Enr.2	Enr.3
1	PRR-1	Philippines	3	TRIGA Rods	-	-	20
			-	Plates	93	-	20
2	THOR	Taiwan	1	TRIGA Rods	-	-	20
			-	Plates	93	-	
3	TRR-1	Thailand	2	TRIGA Rods	-	-	20
			-	Plates	90	-	-

[a] *Initial enrichments, in weight-% ^{235}U, of the fuels possessed by each reactor. Only fuels containing uranium of U.S.-origin are included.*

Note:
All three of these reactors have been converted from plate-type, aluminum-based HEU fuel to TRIGA LEU fuel. The PRR-1 reactor in the Philippines possesses both HEU and LEU cores of plate-type aluminum-based fuel elements.

material that occurs in nature in a significant quantity is ^{235}U. Natural uranium consists of 0.711 weight percent (w/o) ^{235}U, 99.283 w/o ^{238}U; and 0.0055 w/o ^{234}U as a negligible trace constituent. Uranium-235 is the only fissile material used in foreign research reactors.

In a research reactor, the fuel matrix typically consists of enriched uranium metal in an alloy of aluminum or zirconium hydride. The enriched uranium may contain up to 93 weight percent ^{235}U. The fuel matrix form is either plates (flat or curved), tubes made of three curved plates, or pellets combined into rods. The cladding is the encapsulation (typically aluminum or stainless steel) that surrounds the fuel for confinement and protection. The structural part of a fuel element holds fuel plates or tubes in the proper configuration and directs coolant flow (light or heavy water) over the fuel. Structural parts are usually aluminum. The fuel rods do not require additional structural parts. The size of a fuel element ranges from approximately 1 kg (2.2 lb) to more than 100 kg (220 lb), and lengths range from 76 to 300 cm (2.5 to 9 ft).

As the fuel in a reactor is irradiated, it undergoes nuclear transmutations that cause its composition to change. In the reactor, the fissionable materials in the fuel undergo a process called "fission reaction." Fission reaction occurs when an atom of ^{235}U interacts with a free neutron causing the ^{235}U atom to split into two lighter nuclei which are referred to as "fission products." The fission reaction also results in the release of heat and additional free neutrons that are available to sustain the fission reaction or to maintain criticality. In addition to fission products, heavier elements such as plutonium and other isotopes of uranium are formed when uranium in the fuel absorbs free neutrons rather than undergoing the fission process. The changes in composition of the fuel bring about changes in the fission reaction rate of the fuel. As the reactor operation continues, the fission reaction rate decreases and eventually the reactor will no longer remain critical unless some spent nuclear fuels are replaced with fresh fuels. The discharged fuel is called "spent nuclear fuel." The extent of change in the composition of the fuel is expressed in terms of "burnup," in either percent (atom percent) of fissile material consumed, or the number of megawatt days of heat released per element (or per metric ton of uranium).

When initially discharged from the reactor, spent nuclear fuel is highly radioactive and generates a significant amount of heat. Therefore, the spent nuclear fuel must be stored in a wet pool that provides both shielding and cooling environments. The cooling is required in order to prevent the spent nuclear fuel from being damaged by the heat that fission products generate, and the shielding is needed to protect the workers who handle the fuel.

The quantity of radioactive material in spent nuclear fuel, and the resulting heat generation, decreases over time because of decay of fission products in the spent nuclear fuel. Radioactive decay refers to a process whereby the radioactive elements undergo nuclear transformations that ultimately convert them to stable (nonradioactive) elements. Many fission products formed during reactor operation have short half-lives (the time required for a quantity of radioactive material to decrease to one-half of its original amount) and others remain radioactive for tens to thousands of years. The high initial quantities of fission products in the spent nuclear fuel put the greatest requirements on providing shielding and cooling during the first few months after the spent nuclear fuel is discharged from the reactor. The rapid decay of short half-lived radioactive material leads to reduction of the amount of radioactive material in the spent nuclear fuel over time. This, in turn, reduces the need for continued storage of the spent nuclear fuel in a wet pool. After about 1 year, the heat generation rate in a spent nuclear fuel element decreases to about one percent of the level present at the time of its discharge from the reactor, and this heat generation rate would not damage the spent nuclear fuel if it is stored in a "dry" cask in preparation for transportation and dry storage.

B.1.3 Foreign Research Reactor Spent Nuclear Fuel Designs

Foreign research reactors use a number of different fuel designs. These designs can be organized into five categories: (1) plate-type design, (2) concentric tube-type design, (3) pin-type design, (4) special-type design, and (5) rod-type design. The first four designs are aluminum-based fuel while the fifth is a TRIGA type. The first two fuel types (plate-and tube-type fuels) are known as material test reactor (MTR) fuels. The following summarizes specific characteristics of the different types of fuel named above.

B.1.3.1 Plate-Type Design

This type of fuel design is used in the majority of foreign research reactors. The thermal power of these reactors ranges from 1 MW to 50 MW. Figures B-1 and B-2 show typical fuel elements using this type of fuel design. The number of fuel plates in an element varies between 6 and 23, and the initial ^{235}U content varies between 37 g (1.3 oz) and 420 g (14.8 oz) per element. Similarly, the average burnup of a discharged spent nuclear fuel varies between 15 and 76 percent (^{235}U atom percent). The uranium enrichment in this type of fuel varies from just below 20 to 93 percent.

The following provides additional information on a typical plate-type spent nuclear fuel element which was used in a 50 MW foreign research reactor, as shown in Figure B-2.

The fuel element is made of an alloy of 23 percent by weight of 93 percent enriched uranium in aluminum with a thin (0.38 mm) aluminum cladding. Each fuel element contains 19 fuel plates. The nominal dimensions and weights of each fuel plate and the fuel element are:

	Fuel Plate	*Element*	*Element (cut)*
Dimensions (mm):			
Length	778	1,200	800
Width	70.8	77.0	77.0
Height	1.27[a]	77.0	77.0
Weight (g):			
^{235}U	15	285	285
Total	202	--	5,500
Burnup:			
^{235}U (g)	~3	60	60

[a] *Thickness*

The cut element reflects that portion of the fuel element that contains fuel material. The aluminum nose cone and the aluminum top section of the fuel element are cut to reduce the size of the spent nuclear fuel prior to shipment. This action is usually performed at the foreign research reactor site if the site is equipped to do so. The cutting is necessary to pack more cut elements in a transportation cask, and also since some casks cannot accommodate the whole element length.

B.1.3.2 Concentric Tube Design

This type of fuel design is used in four foreign research reactors: Australian (HIFAR), Belgian (BR-2), Japanese (JRR-2) and Danish (DR-3). The Belgian reactor is a 125 MW reactor, and the other three are each 10 MW. Figure B-3 shows a typical fuel element using concentric tube (tubular) fuel type. The number of fuel tubes in an element varies between 4 and 6, and the initial ^{235}U content varies between

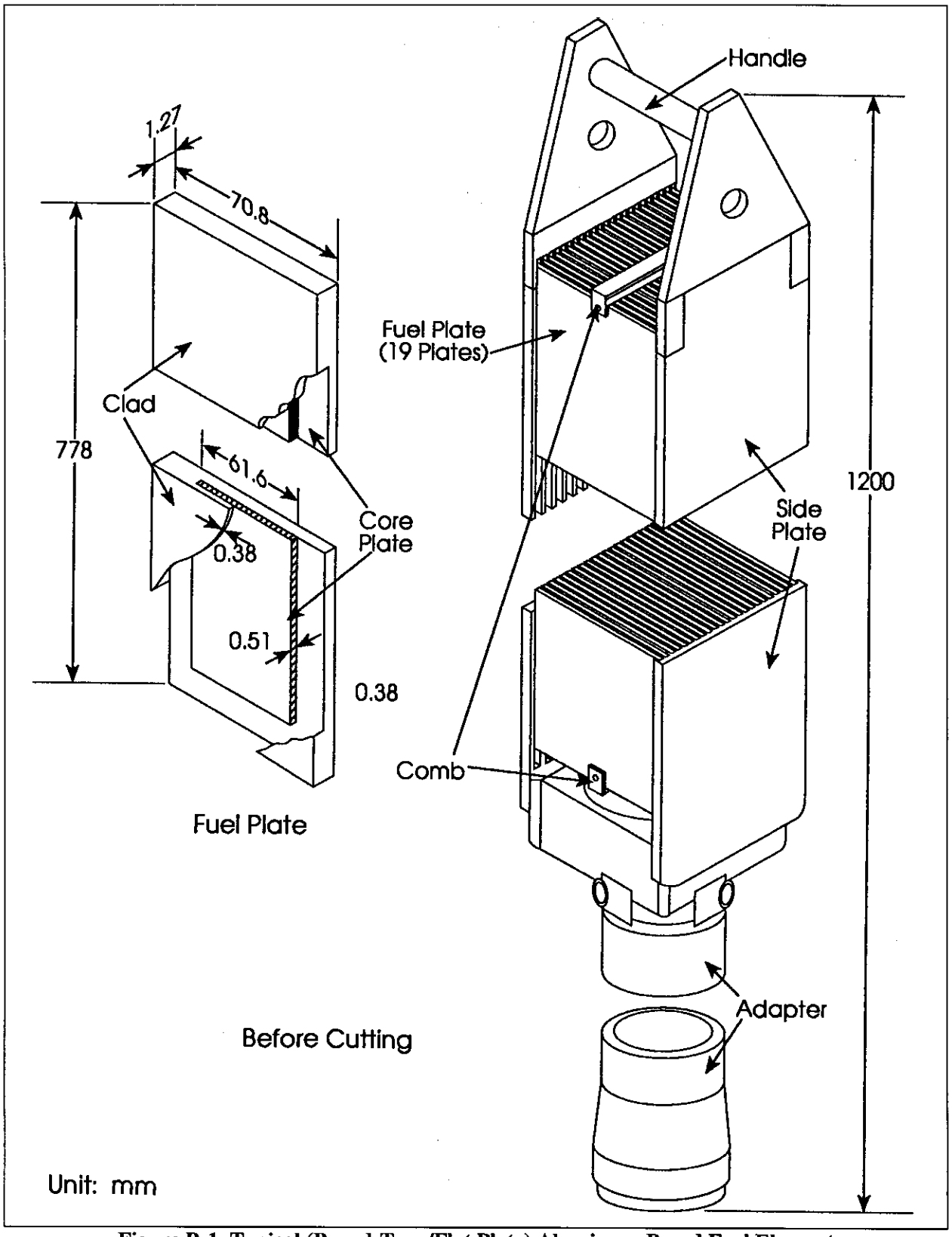


Figure B-1 Typical (Boxed-Type/Flat Plate) Aluminum-Based Fuel Element Schematic

Part No.	Name of Part	Material	Number
1	Bottom Adapter	Aluminum Alloy	1
2	Top Adapter	Aluminum Alloy	1
3R	Side Plate	Aluminum Alloy	1
3L	Side Plate	Aluminum Alloy	1
4	Outer Fuel Element	U-Al Alloy	2
5	Inner Fuel Element	U-Al Alloy	15
6	Comb	Aluminum Alloy	1
7	Comb Pin	Aluminum Alloy	1

Unit = mm

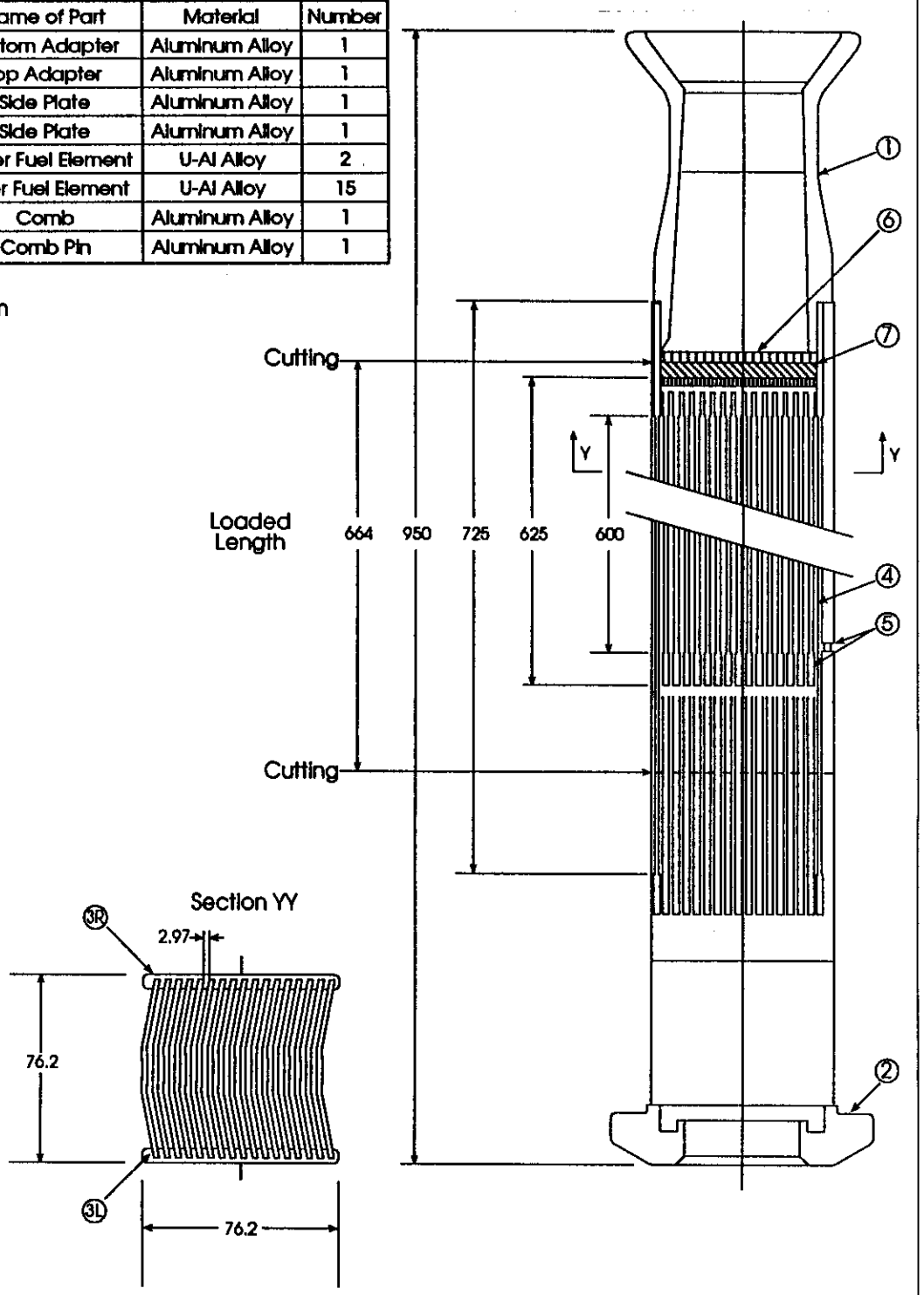
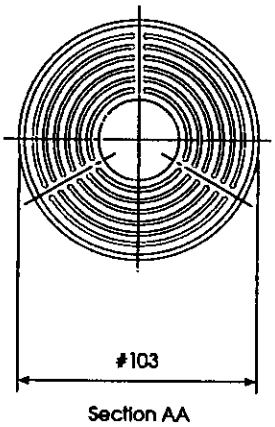


Figure B-2 Typical (Boxed-Type/Curved Plate) Aluminum-Based Fuel Element Schematic

Part No.	Name of Part	Material	Number
1	Bottom Adapter	Aluminum Alloy	1
2	Top Adapter	Aluminum Alloy	1
3	Outer Cylinder	Aluminum Alloy	1
4	Plate	Aluminum Alloy	3
5	Fuel Plate	———	5x3
6	Inner Layer Plate	Aluminum Alloy	3
7	Top Capsule Guide	Aluminum Alloy	1
8	Comb Plate	Aluminum Alloy	3
9	Capsule Guide	Aluminum Alloy	1
10	Comb	Aluminum Alloy	3

Unit = mm



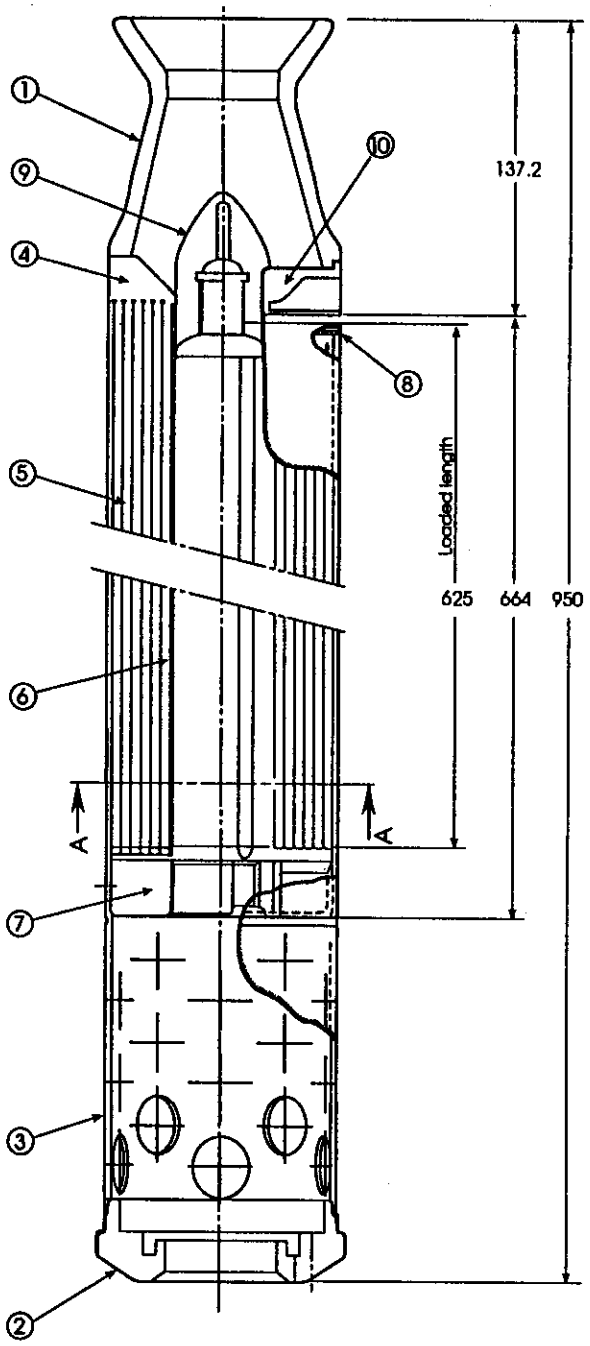


Figure B-3 Typical (Tube-Type) Aluminum-Based Fuel Element Schematic

150 g (5.3 oz) and 400 g (14.6 oz) per element. The average burnup of discharged spent nuclear fuels from these reactors ranges between 47 and 55 percent (^{235}U atom percent). The uranium enrichment used in this fuel varies from just below 20 to 93 percent.

The following provides additional information on a typical tubular type spent nuclear fuel element (shown in Figure B-3) that was used in a 10 MW reactor.

This fuel element initially contains 220 g (7.7 oz) ^{235}U, and consists of 5 concentric fuel tubes. Each tube is made of three curved fuel plates. The fuel is an alloy of uranium in aluminum with a thin (0.38 mm) aluminum cladding. Five different curved fuel plate width sizes with 1.27 mm (0.05 in) thickness and 625 mm (24.6 in) height are used. The overall outside diameter of the outermost tube is 103 mm (4 in). The plate width and the ^{235}U content for each plate size are:

Plate Number	*1*	*2*	*3*	*4*	*5*
Width (mm)	57.9	66.9	75.8	84.8	93.7
^{235}U (g)	10.70	12.70	14.60	16.60	18.60

The overall dimensions of a cut element, leaving the fuel portion intact, are 103 mm (4 in) outside diameter and 664 mm (25.4 in) in length, with an overall weight of approximately 6,000 g (13.2 lbs).

B.1.3.3 Pin-Type Design

Three types of foreign research reactors use pin-type design fuel. They are: the Canadian Safe LOW POwer critical [K] Experiment (SLOWPOKE) (20 kW power); the Canadian NRU (125 MW power) and South Korean KMRR (30 MW) reactors; and the Romanian TRIGA (14 MW) reactors. Among these reactors, the SLOWPOKE fuel pins are the smallest in size and uranium content. The NRU and KMRR reactor fuels are considered special type fuel, and the Romanian reactor fuels are TRIGA or rod-type fuel. Special-type and rod-type materials are discussed below.

The SLOWPOKE reactor fuel pins have an outside diameter of 4.73 mm (0.2 in), a length of 220 mm (8.7 in), and contain 93 percent enriched uranium fuels. The ^{235}U content of each pin is 2.8 g (0.1 oz). The maximum fuel burnup of discharged spent nuclear fuels is about 2 percent (^{235}U atom percent) in 10 to 20 years of reactor operation.

The SLOWPOKE spent nuclear fuel pins are usually bundled together in 10 to 15 pins per bundle. In the past, this fuel was shipped to Savannah River Site in 50.8-mm (2-in) outside diameter, 2.9-m- (9.6-ft-) long canisters containing between 150 to 160 pins per canister.

B.1.3.4 Special-Type Design

Special-type design fuels are used in the French RHF (57 MW power), Canadian NRU (125 MW power) and NRX (24 MW power), and the South Korean KMRR (30 MW) reactors. The fuel type in the Canadian research reactors consists of clusters of about 3-m- (9.84-ft-) long uranium aluminum alloy fuel pins clad in aluminum. The initial ^{235}U content of each fuel cluster varies between 491 g (17.3 oz) and 545 g (19.2 oz). The current operating reactor (NRU) uses a fuel element that consists of a cluster of 12 long pins containing 491 g (17.3 oz) of ^{235}U per cluster. Each fuel pin has an overall length of 296 cm (116.5 in), and the fuel portion is 274.3 cm (107.9 in) long. The fuel cluster including the flow tube is cut to a length of 292.6 cm (115.2 in) before shipment. The average burnup of discharged spent nuclear fuels from an NRU reactor is about 76 percent (^{235}U atom percent). Figure B-4 shows a 12-pin cluster NRU fuel element. The fuel in the South Korean research reactor consists of two types of fuel clusters; one is

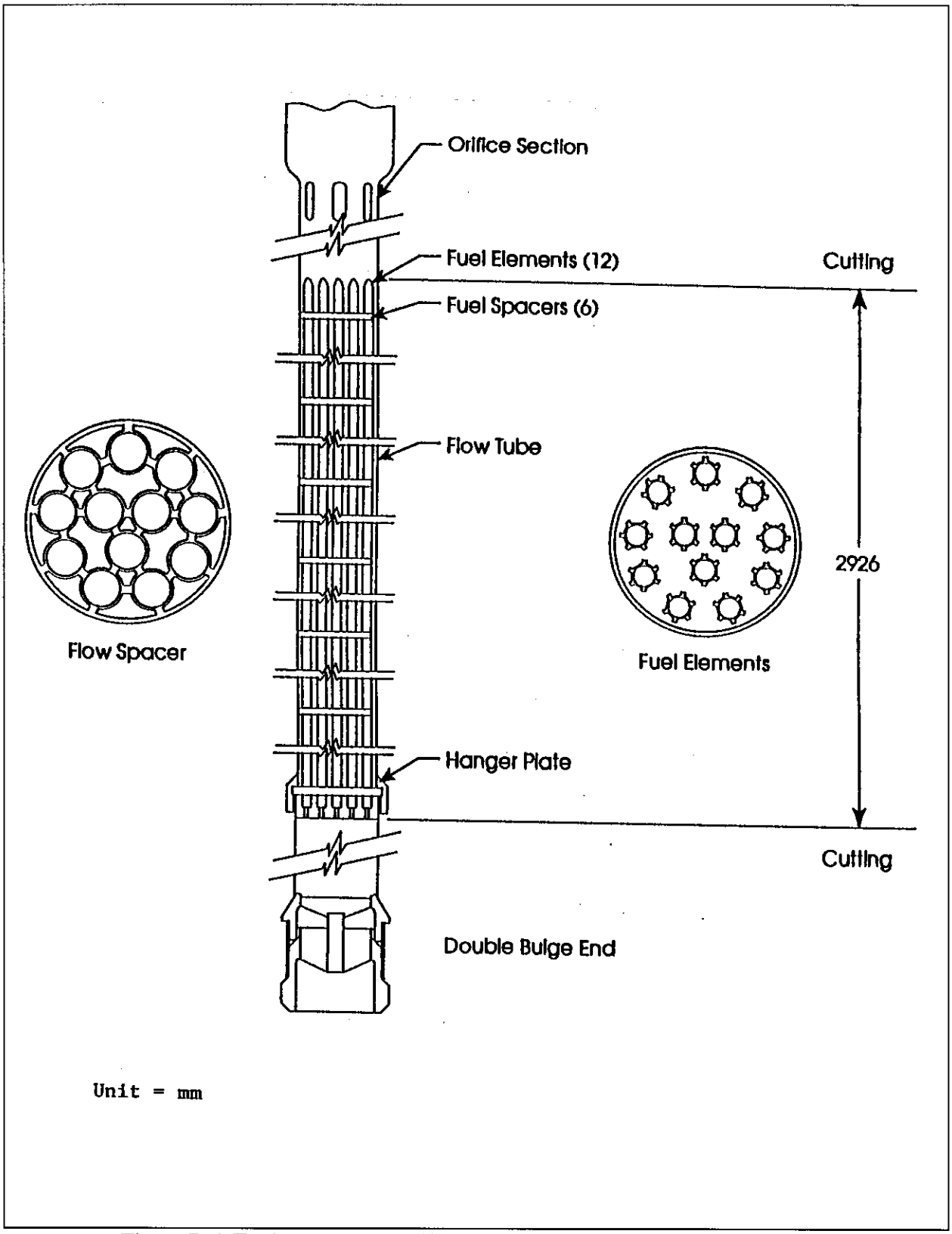


Figure B-4 Typical NRU Type (Aluminum-Based) Fuel Element Schematic

18 pins per cluster with an initial ^{235}U content of 248 g (8.7 oz). The second is 36 pins per cluster with an initial ^{235}U content of 435 g (1 lb). The expected burnup of a discharged spent nuclear fuel from this reactor is approximately 65 percent (^{235}U atom percent).

The fuel used in the RHF research reactor is an annular-type fuel element. The RHF research reactor uses only one fuel element at a time. The RHF fuel element contains 9.2 kg (20.3 lbs) of uranium, enriched to 93 percent, in ^{235}U in 280 involute fuel plates made of uranium aluminum alloy (UAl_3-Al) clad in aluminum. The weight of a cut element is about 100 kg (220 lbs). The fuel is in the annulus of two aluminum tubes: the inner tube has an outside diameter of 274 mm (10.8 in), and the outer tube has an outside diameter of 414 mm (16.3 in). The expected average burnup of a discharged spent nuclear fuel is 36 percent (^{235}U atom percent). Figure B-5 shows a schematic drawing of a configuration of annular fuel element similar to that of RHF fuels.

B.1.3.5 Rod-Type Design

This fuel type design is used in TRIGA research reactors. These research reactors have power ranging from 100 kW to 14 MW. The TRIGA fuel is mainly made up of three basic types of fuel elements: aluminum-clad elements, stainless steel-clad elements, and incoloy-clad elements. All aluminum-clad elements and stainless steel-clad elements are 38.1-mm (1.5-in) diameter by 762-mm- (30-in-) long rods including end fittings (see Figure B-6). The incoloy-clad elements are of the same length, but with a smaller diameter, ranging from 13.7 mm (0.54 in) to 30.7 mm (1.2 in). The 13.7-mm (0.54-in) fuel is currently being used in the Romanian TRIGA research reactor. The fuel is a solid, homogeneous mixture of uranium zirconium hydride alloy. A 6.35-mm (0.25-in) hole is drilled through the center of the active fuel section to facilitate hydriding; a zirconium rod is inserted in this hole after hydriding is complete.

The aluminum-clad elements are the original TRIGA fuel rods that are still in use at some foreign research reactors. The active part of the aluminum-clad fuel element contains about 8 percent by weight of uranium enriched to just below 20 percent ^{235}U. The hydrogen-to-zirconium atom ratio is approximately 1.0. The initial loading of ^{235}U is about 38 g (1.3 oz). The average burnup of this type of fuel is about 8 percent. Each rod weighs 3.2 kg (7.04 lbs) on the average.

The current standard TRIGA fuel rods are the stainless steel-clad elements. The fuel content of the stainless steel element can vary according to the type used. The fuel content of a standard rod consists of 8 to 9 percent by weight of 19.95 percent enriched uranium [about 39 g (1.4 oz) of ^{235}U] in zirconium hydride, with a hydrogen-to-zirconium atom ratio of 1.7. Another type, known as FLIP, contains 8.5 percent by weight of 70 percent enriched uranium [137 g (4.8 oz) of ^{235}U]. The annular core pulsed reactor fuel type contains 12 percent by weight of just below 20 percent enriched uranium [about 54 g (1.9 oz) of ^{235}U] in zirconium hydride with a hydrogen-to-zirconium ratio of 1.7. The expected average burnup of the discharged spent nuclear fuel is approximately 15 percent. Each rod weighs 3.6 kg (7.9 lbs) on the average.

The incoloy-clad element has a longer active fuel length [558.8 mm (22 in) compared to 381 mm (15 in) for standard stainless steel-clad]. The fuel section consists of four pellets, each 139.7-mm (5.5-in) long, and contains approximately 45 percent by weight of uranium enriched to 20 percent [approximately 54 g (1.9 oz) of ^{235}U] in zirconium hydride. There are no graphite reflectors within this element. Instead, a 76.2-mm (3-in) spring is inserted at the top and bottom of the element, and stainless steel end fixtures are attached to both ends of the can. The expected average burnup of this fuel in the Romanian TRIGA reactor is about 52 percent.

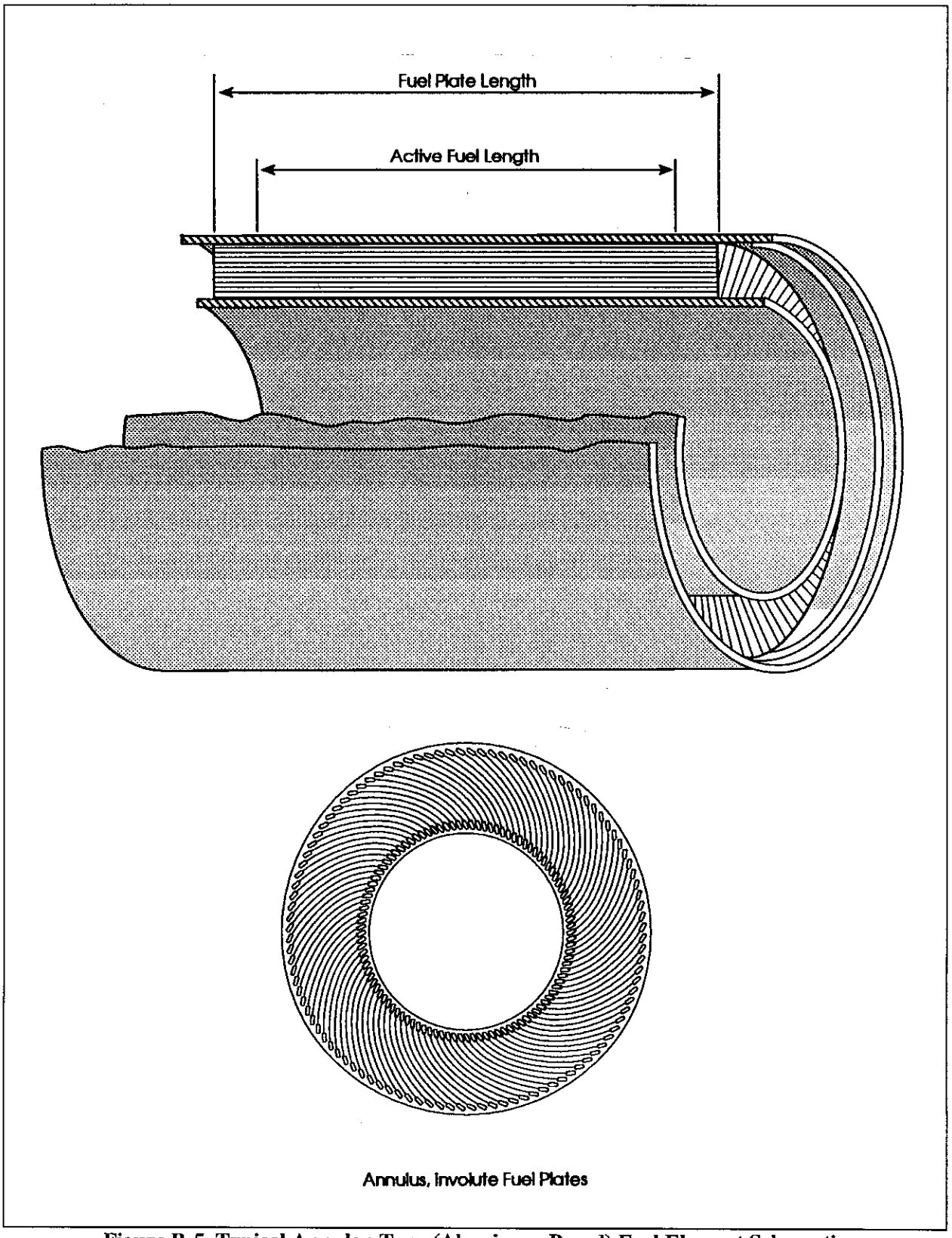


Figure B-5 Typical Annular-Type (Aluminum-Based) Fuel Element Schematic

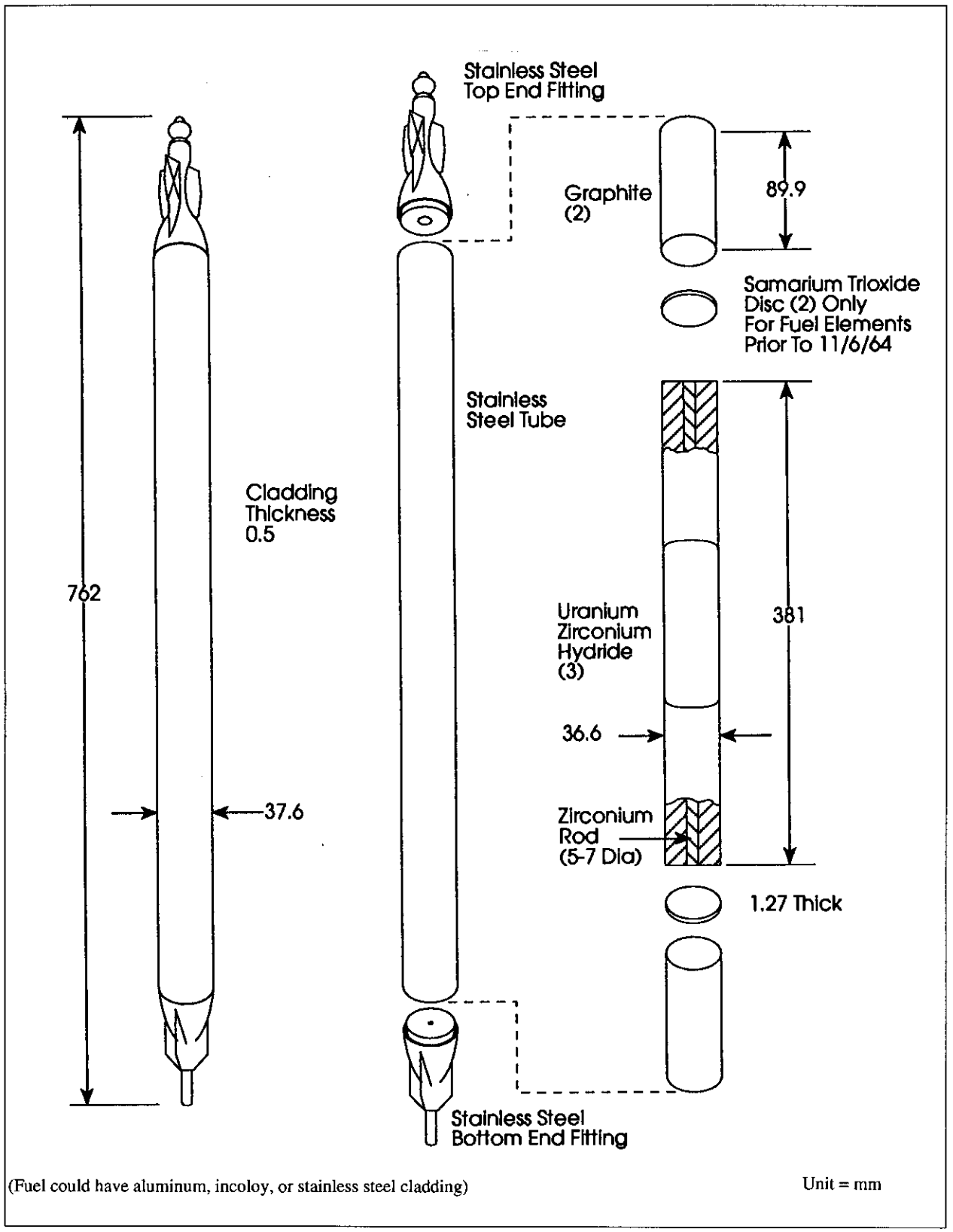


Figure B-6 Typical TRIGA Fuel Element Schematic

B.1.4 Description of the Bounding Radionuclide Inventory

The spent nuclear fuel radionuclide concentration (or inventory) is directly related to the initial mass of fuel (fissile and fertile), the level of burnup, and the cooling period (or decay period) following fuel discharge from the reactor. The fuel burnup is a function of the fuel position inside the reactor core resulting in some fuels burning more than others. A well-designed fuel management program, however, reduces burnup variations among fuel elements. The radionuclide generation in an irradiated fuel is a function of reactor power level and the duration of irradiation process. Research reactors have irregular irradiation profiles, and are typically operated at various power levels. For the calculation of the radionuclide inventory, each fuel element was assumed to have been burnt uniformly and continuously at full reactor power before its discharge. These assumptions maximize the radionuclide inventories of the spent nuclear fuel. As stated earlier, the cooling or decay time after fuel discharge from the reactor determines the amount of radionuclides that remains in the spent nuclear fuel.

Based on the discussion in Section B.1.3, the foreign research reactor spent nuclear fuels were grouped into three classes and four fuel categories for the determination of bounding radionuclide inventories. This subdivision was created in order to provide a better representation of potential radionuclide inventories associated with each type of fuel. This subdivision also provides a means for identifying the type of transportation casks needed, and estimating the number of spent nuclear fuel shipments. The radionuclide inventories per shipment are needed as input to marine and ground transportation and cask handling impact analyses.

The selected fuel types for the determination of bounding radionuclide inventories are:

1. ***Special.*** These are aluminum-based fuels that are neither TRIGA nor MTR. Special fuels are also different in size and geometry.

   1a. *Single Element Reactors.* Spent nuclear fuel from research reactors that operate with one element (e.g., RHF of France). These spent nuclear fuels contain several kg of ^{235}U and require special shipping baskets, casks, and transportation analyses.

   1b. *NRU Type Spent Nuclear Fuel.* Spent nuclear fuel from Canadian Research Laboratories' research reactors (e.g., NRU, and NRX) that require special transportation analysis. These spent nuclear fuels are geometrically different from an MTR-type and TRIGA spent nuclear fuel both in cross section and length, and require special shipping arrangements in addition to being transported overland by truck or rail.

2. ***MTR Spent Nuclear Fuel.*** This category covers all MTR-type spent nuclear fuels. These spent nuclear fuels have similar geometrical characteristics and use common type transportation casks.

3. ***TRIGA Spent Nuclear Fuel.*** Spent nuclear fuel from TRIGA reactors. These spent nuclear fuels also have almost similar geometrical characteristics and use common types of transportation casks.

In the case of special-type fuel, the bounding spent nuclear fuels are the RHF of France and the NRU of Canada. For the identification of a bounding spent nuclear fuel within the MTR and TRIGA fuel types, a series of ORIGEN2 (Croff, 1980) computer runs was made using different spent nuclear fuels within each fuel type. ORIGEN2 generates the radionuclide inventory in a spent nuclear fuel based on the fuel burnup, initial fissile and fertile inventory, and decay time. The radionuclide inventories of selected bounding

spent nuclear fuel within each fuel type were determined assuming that the spent nuclear fuel has been cooled for a specified period after its discharge from the reactor. In order to maximize the radionuclide inventory per transportation cask, a review of the potential casks was performed. It was determined that the use of IU-04 (Pegase) transportation casks, maximizes the radioactive inventory and requires the shortest cooling period (maximum of 1 year) (see Section B.1.6). Based on this review, the cooling period for each bounding spent nuclear fuel was determined. The bounding spent nuclear fuels for MTR and TRIGA type fuel are found to be BR-2 type spent nuclear fuel, and a spent nuclear fuel from a 3-MW TRIGA reactor burning 31g (1.1 oz) of ^{235}U in 3 years, respectively. The bounding TRIGA spent nuclear fuel identified here is a theoretical bounding fuel for this category.

Table B-6 provides a list of the radioactive isotopes and their inventories for selected bounding spent nuclear fuel types. The list of isotopes is generated from ORIGEN2 output based on the following criteria:

1. All isotopes (from a list of 270 elements) that could have a potential to contribute 1 mrem from inhalation and ingestion are considered. The estimates of dose associated with each isotope intake were based on the effective committed dose equivalent factors provided in DOE/EH-0071 (DOE, 1988).

2. Once all isotopes were selected, those that contribute to 99.9 percent of total health hazard were chosen.

3. Isotopes such as ^{85}Kr, ^{235}U, and ^{238}U were added to the list as historically significant isotopes, although they do not meet the above criteria.

It is important to note that the radionuclide inventories identified here are for calculational purposes only. The majority of the spent nuclear fuels would have lower radionuclide inventories than what is identified here, and the likelihood of a full cask containing maximum inventory during the acceptance policy period would be low. By the time the policy would become effective in late 1995, there could be about 10,000 spent nuclear fuel elements, of which 80 percent would have had more than 2 years of cooldown (decay). The number of spent nuclear fuels that receive maximum burnup used in the estimation of the radionuclide inventory is very small when compared to the total number of the spent nuclear fuel elements estimated in each fuel category.

B.1.5 Characteristics and Radionuclide Inventories of Target Materials

Under Implementation Alternative 1 to Management Alternative 1 of the proposed action, DOE would plan to manage target material. The total amount of target material is estimated to be about 0.56 MTHM having a volume of 6.5 m^3 (230 ft^3). Target materials are residual materials from target fuels that have been irradiated in a research reactor to produce ^{99}Mo, which decays to ^{99}Tc, a medical isotope. Four countries (Canada, Belgium, Argentina, and Indonesia) use target fuel containing U.S.-origin enriched uranium for the production of medical isotopes. Canada, Argentina, and Belgium currently use aluminum-based targets containing HEU, and Indonesia currently uses a target that consists of a layer of HEU oxide (UO_2) material plated on the interior surface of a stainless steel tube. The distribution of target materials from these countries includes: 0.525 MTHM from Canada; 0.029 MTHM from Belgium; 0.0014 MTHM from Indonesia; and 0.0011 MTHM from Argentina. A target fuel is irradiated to a burnup level of about 3 percent (^{235}U atoms percent) before being discharged from the reactor. Once the target fuel is removed from the reactor, within a short period the fuel is dissolved and ^{99}Mo is separated from the solution. The residual material is then decayed. Prior to shipment, the residual materials are transformed to an acceptable form.

Table B-6 Bounding Radionuclide Inventories per Element for Selected Fuel Categories (Curies)

Isotope	Fuel Category			
	BR-2	RHF	NRU	TRIGA
Tritium	2.40	37	3.95	0.328
Krypton 85	68.7	1,070	113	9.10
Strontium 89	1,133	17,600	405	68.8
Strontium 90	578	8,930	967	79
Yttrium 90	578	8,930	967	79
Yttrium 91	2030	31,400	842	115
Zirconium 95	2,972	46,300	1,410	163
Niobium 95	6,111	94,900	3,060	320
Ruthenium 103	247	3,770	60.0	21.1
Rhodium 103m	247	3,770	60.0	21.1
Ruthenium 106	597	9,160	767	63.5
Rhodium 106m	597	9,160	767	63.5
Tin 123	11.9	184	10.0	0.978
Antimony 125	24.7	381	38.0	2.98
Tellurium 125m	5.89	90.6	9.21	0.718
Tellurium 127m	24.6	382	18.4	1.40
Tellurium 129m	5.25	79.8	0.958	0.578
Cesium 134	456	4,000	1,480	29.0
Cesium-137	572	8,870	958	79.8
Cerium 141	159	2,440	277	175
Cerium 144	8,667	135,000	10,600	633
Praseodymium 144	8,667	135,000	10,600	633
Promethium 147	1,342	24,600	1,240	175
Promethium 148m	2.10	29.2	0.0583	1.17
Europium 154	17.2	163	56.3	1.05
Europium 155	3.61	45.6	10.2	0.565
Uranium 234	0.0000254	0.000374	0.0000654	0.00000453
Uranium 235	0.000383	0.0109	0.000253	0.000199
Uranium 238	0.00000947	0.000206	0.00000111	0.000163
Plutonium 238	1.78	10.3	11.3	0.0760
Plutonium 239	0.0511	0.0889	0.0138	0.0138
Plutonium 240	0.0333	0.421	0.0101	0.0523
Plutonium 241	7.89	67.7	2.95	5.33
Americium 241	0.0110	0.0967	0.00517	0.0102
Americium 242m	0.0000292	0.000155	0.0000250	0.000225
Americium 243	0.000120	0.00376	0.000146	0.0000110
Curium 242	0.0486	0.127	0.0429	0.131
Curium 244	0.0369	0.00926	0.0113	0.000178
Total (Curies)	35,129	546,000	34,700	2,740
Thermal (Watts)	147	2,250	150	10.4

There are currently two methods for preparing the residual materials containing aluminum for transport. The first method is calcining and canning the material with the existing aluminum, and the second is a method that first removes aluminum from the residual materials and then oxidizes the remains. The final products are then canned. A process similar to the latter is used for the Indonesian target materials. Since

the Indonesian target materials do not contain aluminum, no aluminum separation is needed. In this case, a precipitation process is used to separate the target materials from the solution. The precipitated materials are then dried and canned in preparation for transport.

The canned material from the first process contains 40 grams (1.4 oz) of ^{235}U per can. The second process allows a higher amount of ^{235}U, 200 g (7 oz), to be packed in a similar can. Can material could be aluminum or stainless steel. In the past, the target material was shipped to the Savannah River Site in aluminum cans 64 mm (2.5 in) in diameter and 280 mm (11 in) long. The use of the first process would result in a total 140 shipments of this material to the United States, and the second process would result in a total of 57 shipments. These number of shipments were estimated based on an assumption that the target material cans would be in transportation casks that would not contain other types of spent nuclear fuel. However, in all likelihood, with small amounts of target materials (such as Indonesia and Argentina), would not ship a partially filled transportation cask when other spent nuclear fuel could be added to fill the cask. Therefore, these estimates represent an upper bound on the total number of the target material shipments. The radionuclide inventory of a target material can containing from 40 to 200 g (1.4 to 7 oz) of ^{235}U, and that of a transportation cask containing this material, is given in Table B-7. This inventory is estimated based on 1 year decay time of the target material solution before the canning process.

B.1.6 Foreign Research Reactor Spent Nuclear Fuel Shipment Estimates

Tables B-1 and B-2 provide the estimated number of foreign research reactor spent nuclear fuel shipments from each country. These estimates were based on a set of assumptions that maximize the potential impacts from transportation. Review of the potential transportation casks identified eight casks with various capabilities (see Section B.2.2). These casks are certified to accommodate between 1 and 126 spent nuclear fuel elements per cask based on a variety of cask cavity configurations. Each transportation cask can be certified to ship different fuel types by using various baskets in the cask cavity. For example, a transportation cask like IU-04 has been certified to accommodate several different fuel types by using various baskets in the cask cavity. On the other hand, a cask like LHRL-120 is currently certified to accommodate only one specific fuel type (Australian HIFAR fuel). Based on this review, IU-04 was identified as the bounding cask (highest curies content for the number of elements shipped per cask) for the transportation accident analyses.

In an attempt to capture various types of spent nuclear fuel, maximize the amount of radionuclides per cask, and allow for potential partial cask shipments, for the purposes of the analyses in this EIS, the following assumptions were made to estimate the number of shipments for each type of fuel:

1. The number of shipments for MTR-type spent nuclear fuel elements was estimated based on 30 elements per cask. The radionuclide inventory per cask was estimated based on a full cask, that is, 36 spent nuclear fuel elements of the bounding MTR-type (BR-2 fuel) per cask. One exception: for the Australian spent nuclear fuel, cask LHRL-120 which was built specifically for this fuel was used for estimating the number of shipments. The allowed radionuclide inventory in this cask is the smallest of all casks identified. Nonetheless, each of the LHRL-120 casks was assumed to contain the same quantity of radionuclide inventories as that of a cask containing 36 elements of the bounding MTR-type spent nuclear fuel.

2. The number of shipments for NRU-type spent nuclear fuel was estimated based on 24 NRU elements per cask. The radionuclide inventories per cask were also based on 24 NRU elements per cask.

Table B-7 Radionuclide Inventories of Target Material per Can and per Transportation Cask (Curies)

Isotope	*Curies for 40 g ^{235}U per Can*	*Curies for 200 g ^{235}U per Can*	*Cask Curies with 40 g per Can*	*Cask Curies with 200 g per Can*
Strontium 89	4.06E+00	2.03E+01	1.95E+02	4.87E+02
Strontium 90	3.28E+00	1.64E+01	1.58E+02	3.94E+02
Yttrium 90	3.28E+00	1.64E+01	1.58E+02	3.94E+02
Yttrium	9.18E+02	3.84E+01	3.69E+02	9.22E+02
Zirconium 95	1.18E+01	5.90E+01	5.67E+02	1.42E+03
Niobium 95	2.53E+01	1.27E+02	1.21E+03	3.04E+03
Ruthenium 103	7.4E-01	3.72E+00	3.57E+01	8.93E+01
Rhodium 103m	7.5E-01	3.73E+00	3.58E+01	8.95E+01
Ruthenium 106	3.11E+00	1.55E+01	1.49E+02	3.73E+02
Rhodium 106m	3.11E+00	1.55E+01	1.49E+02	3.73E+02
Tin 123	6.0E-02	2.8E-01	2.70E+00	6.74E+00
Antimony 125	1.4E-01	6.8E-01	6.51E+00	1.63E+01
Tellurium 125m	3.0E-02	1.6E-01	1.56E+00	3.91E+00
Tellurium 127m	1.1E-01	5.6E-01	5.39E+00	1.35E+01
Tellurium 129m	1.0E-02	7.0E-02	6.7E-01	1.68E+00
Cesium 134	1.0E-02	6.0E-02	6.1E-01	1.53E+00
Cesium-137	3.26E+00	1.628E+01	1.56E+02	3.91E+02
Cerium 141	4.2E-01	2.11E+00	2.03E+01	5.07E+01
Cerium 144	4.53E+01	2.27E+02	2.18E+03	5.44E+03
Praseodymium 144	4.57E+01	2.29E+02	2.20E+03	5.49E+03
Promethium 147	1.07E+01	5.36E+01	5.14E+02	1.29E+03
Promethium 148m	5.06E-04	2.53E-03	2.43E-02	6.07E-02
Europium 154	1.65E-03	8.23E-03	7.90E-02	1.97E-01
Europium 155	6.97E-02	3.49E-01	3.35E+00	8.37E+00
Uranium 234	1.42E-07	7.09E-07	6.81E-06	1.70E-05
Uranium 235	8.29E-05	4.15E-04	3.98E-03	9.95E-03
Uranium 238	1.50E-06	7.52E-06	7.22E-05	1.80E-04
Plutonium 238	3.33E-06	1.67E-05	1.60E-04	4.00E-04
Plutonium 239	6.15E-04	3.08E-03	2.95E-02	7.38E-02
Plutonium 240	1.43E-05	7.13E-05	6.85E-04	1.71E-03
Plutonium 241	1.48E-04	7.38E-04	7.09E-03	1.77E-02
Americium 241	2.42E-07	1.21E-06	1.16E-05	2.91E-05
Americium 242m	4.43E-12	2.22E-11	2.13E-10	5.32E-10
Americium 243	3.07E-12	1.54E-11	1.47E-10	3.69E-10
Curium 242	1.43E-09	7.15E-09	6.86E-08	1.72E-07
Curium 244	3.40E-12	1.70E-11	1.63E-10	4.08E-10
Total (Curies)	1.69E+02	8.30E+02	7.97E+03	1.99E+04
Thermal (Watts)	6.8E-01	3.40E+00	3.26E+01	8.160E+01

3. The number of shipments for RHF type spent nuclear fuel was estimated based on one element per cask. The bounding cask can only accommodate one bounding spent nuclear fuel element per cask.

4. The number of shipments for TRIGA spent nuclear fuel was estimated based on 30 elements per cask. The radionuclide inventories per cask were based on 40 elements of bounding TRIGA spent nuclear fuel element per cask.

Table B-8 provides a list of radionuclide inventories per transportation cask for selected fuel categories.

Table B-8 Bounding Radionuclide Inventories per Transportation Cask for Selected Fuel Categories (Curies)

Isotope	*Fuel Category*			
	BR-2	*RHF*	*TRIGA*	*NRU*
Tritium	86.4	37.0	13.1	94.8
Krypton 85	2,470	1,070	364	2,710
Strontium 89	40,800	17,600	2,750	9,720
Strontium 90	20,800	8,930	3,160	23,200
Yttrium 90	20,800	8,930	3,160	23,200
Yttrium 91	73,000	31,400	4,580	20,200
Zirconium 95	107,000	46,300	6,500	33,800
Niobium 95	220,000	94,900	12,800	73,400
Ruthenium 103	8,900	3,770	844	1,440
Rhodium 103m	8,900	3,770	844	1,440
Ruthenium 106	21,500	9,160	2,540	18,400
Rhodium 106m	21,500	9,160	2,540	18,400
Tin 123	427	184	39.1	240
Antimony 125	890	381	119	912
Tellurium 125m	212	90.6	28.7	221
Tellurium 127m	887	382	55.8	442
Tellurium 129m	189	79.8	23.1	23.0
Cesium 134	16,400	4,000	1,160	35,400
Cesium 137	20,600	8,870	3,190	23,000
Cerium 141	5,740	2,440	7,000	6,650
Cerium 144	312,000	135,000	25,300	254,000
Praseodymium 144	312,000	135,000	25,300	254,000
Promethium 147	48,300	24,600	7,000	29,800
Promethium 148m	75.6	29.2	46.8	1.40
Europium 154	620	163	41.8	1,350
Europium 155	130	45.6	22.6	245
Uranium 234	0.000914	0.000374	0.000181	0.00157
Uranium 235	0.0138	0.0109	0.00794	0.00606
Uranium 238	0.000341	0.000206	0.00650	0.0000267
Plutonium 238	64.2	10.3	3.04	270
Plutonium 239	1.84	0.0889	0.551	0.332
Plutonium 240	1.20	0.421	2.09	0.242
Plutonium 241	284	67.7	213	70.9
Americium 241	0.396	0.0967	0.407	0.124
Americium 242m	0.00105	0.000155	0.00900	0.000600
Americium 243	0.00433	0.00376	0.000438	0.00351
Curium 242	1.75	0.127	5.25	1.03
Curium 244	1.33	0.00926	0.00713	0.270
Total (Curies)	1,260,000	546,000	110,000	833,000
Thermal (Watts)	5,290	2,250	416	3,600
Number of casks by January 2006	473	86	162	116

B.1.7 Amount of Foreign Research Reactor Spent Nuclear Fuel In Implementation Alternative 2a of Management Alternative 1

Under this implementation alternative (see Section 2.2.2.2), DOE would adopt an alternative policy duration of 5 years (1995-2000). The amount of spent nuclear fuel expected under this alternative is approximately 18,800 elements, containing approximately 13 MTHM, and having a volume of 87 m^3 (3,300 ft^3). Tables B-9 and B-10 provide an estimate of the total amount of spent nuclear fuel that would be available (i.e., currently stored or to be generated) in each country by January 2001 (Matos, 1994). These tables also provide the estimated number of shipments expected from each country. The breakdown of the number of shipments in terms of the four bounding fuel categories (as defined in Section B.1.4) are: 377 of BR-2, 56 of RHF, 154 TRIGA, and 91 of NRU type fuel shipments.

B.1.8 Distribution of Foreign Research Reactor Spent Nuclear Fuel by Fuel Type and Geography

This section summarizes the estimated amount of foreign research reactor spent nuclear fuel, in terms of fuel type and geography,[4] that could be received under different implementation alternatives of Management Alternative 1 to the proposed action. The estimated amount of spent nuclear fuel for the two policy durations (i.e., a 10-year and a 5-year spent nuclear fuel generation period) are provided in Tables B-1 and B-9, for aluminum-based fuels, and in Tables B-2 and B-10 for TRIGA fuels. These tables provide a breakdown of the estimated amount of spent nuclear fuel to be accepted from each country. Table B-11 summarizes the same information given in the above tables by fuel type and geography. The information provided in this table is the basis for the calculations of transportation (ground and marine) impacts under the basic implementation of Management Alternative 1, the proposed action to manage foreign research reactor spent nuclear fuel in the United States.

DOE is also considering a Management Alternative 3, which is a hybrid of Management Alternatives 1 and 2. Under this Management Alternative as described in Section 2.4, some of the foreign research reactor spent nuclear fuels would be reprocessed overseas, and the remaining spent nuclear fuels would be brought back to be managed in the United States. Overseas reprocessing is considered only for countries that currently have the technology and capability to store research reactor fission product high- or intermediate-level wastes. The countries that can accept research reactor fission product wastes, based on the historical evidence, are: Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom. Under this Management Alternative, DOE would encourage the reprocessing of aluminum-based spent nuclear fuels from the research reactors in the above countries at western European reprocessing facilities (i.e., at Dounreay Scotland, and/or other locations) and that the recovered ^{235}U be blended down and used as LEU fuel. Reprocessing spent nuclear fuels from the above countries overseas would reduce the amount of foreign research reactor spent nuclear fuels that would be managed in the United States. Table B-12 provides a distribution of the remaining foreign research reactor spent nuclear fuels by fuel type and geography that would be brought to the United States under this Hybrid Alternative. As indicated in this table, the reduction only affects spent nuclear fuels entering through the East Coast of the United States (compare Tables B-11 and B-12). It is important to note that the existing overseas reprocessing facilities have not separated ^{235}U from TRIGA fuels. This does not mean that these facilities will not be able to process TRIGA fuels in the near future. At least one facility has stated that it has a specialty plant that can reprocess small quantities of TRIGA spent nuclear fuels (UKAEA, 1994). If this

4 *Geography refers to that amount of spent nuclear fuel that is expected to arrive at an East Coast or a West Coast port of entry to the United States. Spent nuclear fuel shipments from foreign research reactors located in Europe, Africa, Middle East, and Eastern part of Central and South America are designated as East Coast shipments. All others are designated as West Coast shipments.*

Table B-9 Estimated Number of Aluminum-Based Spent Nuclear Fuel Elements of U.S.-Origin Generated by Foreign Research Reactor Operators by January 2001

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium, (kg)*[b]	*Estimated Number of Shipments*
Argentina[a]	283	71	9
Australia	795	247	7
Austria	130	147	4
Belgium	1,391	569	46
Brazil[a]	155	99	5
Canada	2,243	3,058	92
Chile[a]	58	12	2
Colombia[a]	16	2	1
Denmark	485	372	16
France	1,432	2,110	102
Germany	1,111	471	37
Greece[a]	199	73	6
Indonesia[a]	138	164	4
Iran[a]	29	6	1
Israel	153	34	5
Italy	150	43	5
Jamaica[a]	2	1	1
Japan	2,401	2,219	80
Korea (South)[a]	98	187	4
Netherlands	1,141	678	38
Pakistan[a]	82	16	3
Peru[a]	29	39	1
Philippines[a]	50	24	2
Portugal[a]	79	51	3
South Africa	50	10	2
Spain [c] (from Scotland)	40	16	1
Sweden	864	915	29
Switzerland	159	128	5
Taiwan	127	66	4
Thailand[a]	31	5	1
Turkey[a]	50	51	2
United Kingdom	12	4	1
Uruguay[a]	19	18	1
Venezuela[a]	120	82	4
Total	**14,122**	**11,988**	**524**

[a] *Countries other than high-income economies (World Bank, 1994). These are considered to be "developing" countries.*

[b] *To derive uranium mass in pounds, multiply the amount by 2.2*

[c] *40 spent nuclear fuel elements of Spain's JEN-1 reactor core are stored in Dounreay, Scotland.*

capability is acquired, then the amount of spent nuclear fuel to be managed in the United States would be lower than that indicated in Table B-12 by 834 TRIGA spent nuclear fuel elements containing 157 kg of LEU heavy metal resulting in 28 less shipments to the eastern coast of the United States by January 2005.

Table B-10 Estimated Number of TRIGA Reactor Spent Nuclear Fuel Elements of U.S.-Origin Generated by Foreign Research Reactor Operators by January 2001

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium (kg)*[b]	*Estimated Number of Shipments*
Austria	102	19	3
Bangladesh[a]	100	49	3
Brazil[a]	75	14	3
Finland	171	33	6
Germany	338	64	11
Indonesia[a]	233	44	7
Italy	343	64	11
Japan	321	61	11
Korea (South)[a]	320	61	11
Malaysia[a]	89	44	3
Mexico[a]	175	33	6
Philippines[a]	120	74	4
Romania [a]	1,451	189	48
Slovenia[a]	318	60	10
Taiwan	134	80	4
Thailand[a]	136	35	4
Turkey[a]	69	13	2
United Kingdom	89	17	3
Zaire[a]	132	25	4
Total	**4,716**	**979**	**154**

[a] *Countries other than high-income economies (World Bank, 1994). These are identified as "developing" countries.*

[b] *To derive uranium mass in pounds, multiply the amount by 2.2.*

If additional countries were to be able to accept research reactor fission product waste, additional spent nuclear fuels could be reprocessed overseas. This would reduce the amount of spent nuclear fuel to be managed in United States even further.

B.2 Transportation Casks

Spent nuclear fuel elements are transported in stainless steel packages called transportation casks, or just casks.

B.2.1 Transportation Cask Regulations

This section discusses the international and domestic regulations on transportation cask design, performance, certification, use, and transport.

B.2.1.1 International Regulations

To ensure public safety worldwide, the international community has adopted regulations for the transport of radioactive materials. The international authority for these regulations is the International Atomic Energy Agency. The emphasis of the International Atomic Energy Agency regulations for radioactive materials transport is package integrity. As promulgated in International Atomic Energy Agency Safety

Table B-11 Summary of the Distribution of Foreign Research Reactor Spent Nuclear Fuel by Fuel Type and Geography

	January 2001					January 2006				
	No. of Elements	No. of Shipments	Initial Kg U	HEU Kg U	LEU Kg U	No. of Elements	No. of Shipments	Initial Kg U	HEU Kg U	LEU Kg U
All Countries:										
1. Aluminum-Based	14,122	524	11,988	3,992	7,995	17,803	675	18,184	4,531	13,650
East[a]	10,395	419	9,024			13,186	544	13,919		
West[a]	3,727	105	2,963			4,617	131	4,263		
2. TRIGA	4,716	154	980	79	901	4,940	162	1,033	83	950
East	3,088	101	499			3,245	107	528		
West	1,628	53	481			1,695	55	505		
Developing Countries:										
1. Aluminum-Based	1,488	52	911	155	756	1,686	59	1,195	157	1,038
East	1,084	38	480			1,152	40	561		
West	404	14	431			534	19	634		
2. TRIGA	3,218	105	642	77	565	3,359	109	674	81	593
East	2,045	67	302			2,134	70	319		
West	1,173	38	340			1,225	39	355		

[a] *East refers to the eastern United States ports of entry. Spent nuclear fuel shipments from foreign research reactors located in Europe, Africa, Middle East, and eastern part of Central and South America are designated as East Coast shipments. All others are designated as West Coast shipments.*

Table B-12 Distribution of Foreign Research Reactor Spent Nuclear Fuel by Fuel Type and Geography for the Hybrid Alternative

	January 2001					January 2006				
	No. of Elements	No. of Shipments	Initial Kg U	HEU Kg U	LEU Kg U	No. of Elements	No. of Shipments	Initial Kg U	HEU Kg U	LEU Kg U
All Countries:										
1. Aluminum-Based	9,839	328	8,650	2,259	6,391	12,210	406	12,912	2,263	10,646
East[a]	6,112	223	5,687			7,593	275	8,645		
West[a]	3,727	105	2,963			4,617	131	4,263		
2. TRIGA	4,716	154	980	79	901	4,940	162	1,033	83	950
East	3,088	101	499			3,245	107	528		
West	1,628	53	481			1,695	55	505		
Developing Countries:										
1. Aluminum-Based	1,488	52	911	155	756	1,686	59	1,195	157	1,038
East	1,084	38	480			1,152	40	561		
West	404	14	431			534	19	634		
2. TRIGA	3,218	105	642	77	565	3,359	109	674	81	593
East	2,045	67	302			2,134	70	319		
West	1,173	38	340			1,225	39	355		

[a] *East Refers to the eastern United States ports of entry. Spent nuclear fuel shipments from foreign research reactors located in Europe, Africa, Middle East, and eastern part of Central and South America are designated as East Coast shipments. All others are designated as West Coast shipments.*

Series 6, radioactive materials must be transported in specially designed transportation casks that minimize the potential consequences of transportation accidents. Transportation cask designs must demonstrate their capability to ensure containment and to provide shielding by testing or analysis to the extent required by these regulations. Under International Atomic Energy Agency regulations, spent nuclear fuel transportation cask integrity must be demonstrated by successful performance during a sequence of tests that simulate accident conditions. These tests include being dropped onto an unyielding surface, dropped onto a steel post, subjected to extremely high temperatures of 800°C (1475°F) for 30 minutes, and submersed in water. Cask designs that meet these performance criteria are issued a "Certificate of Compliance" by a delegated national authority, referred to as the "Competent Authority." The Competent Authority is responsible for certifying casks that are designed or used within its "national boundary." The Competent Authority for the United States is the Department of Transportation.

To be used outside the country of origin, transportation casks must have a Certificate of Competent Authority from the country of intended use. As the Competent Authority, the Department of Transportation is responsible for granting a Certificate of Competent Authority to foreign-designed transportation casks intended for use in the United States.

B.2.1.2 Domestic Regulations

Regulations for the transport of radioactive materials in the United States are issued by the Department of Transportation, and are codified in Title 49 of the Code of Federal Regulations Parts 171-178 (49 CFR §171-178). These regulations reference accepted standards promulgated by organizations such as the International Atomic Energy Agency, the International Civil Aviation Organization, the International Air Transport Agency, the International Maritime Organization, and the U.S. Nuclear Regulatory Commission (NRC). Federal standards are updated periodically to reflect new information and to remain current with international standards, to minimize delays in international traffic, and avoid duplication of effort.

The regulation authority for radioactive materials transport is jointly shared by the Department of Transportation and NRC. As outlined in a 1979 Memorandum of Understanding with NRC, the Department of Transportation specifically regulates the carriers of spent nuclear fuel and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements. The Department of Transportation also regulates the labeling, classification, and marking of all spent nuclear fuel packages. NRC regulates the packaging and transport of spent nuclear fuel for its licensees, which include commercial shippers of spent nuclear fuel. In addition, NRC sets the standards for packages containing fissile materials and spent nuclear fuel. A detailed discussion of Federal design and performance regulations for transportation cask begins with Section B.2.1.3.

DOE policy requires compliance with applicable Federal regulations regarding domestic shipments of spent nuclear fuel. Accordingly, DOE has adopted the requirements of 10 CFR §71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions," and 49 CFR §171-179, "Hazardous Material Regulations." Foreign research reactor spent nuclear fuel shipments are subject to regulations set by the Department of Transportation and NRC.

B.2.1.3 Cask Design Regulations

Spent nuclear fuel is transported in robust "Type B" transportation casks that are certified for transporting radioactive materials. These transportation casks are subject to stringent design, fabrication and operating requirements imposed by the Competent Authority for the country of origin. Casks designed and certified for spent nuclear fuel transportation within the United States must meet the applicable requirements of

NRC for design, fabrication, operation, and maintenance as contained in 10 CFR §71. These regulations generally conform to International Atomic Energy Agency regulations that are presented in the International Atomic Energy Agency Safety Series 6 manual.

Cask design and fabrication can only be done by approved vendors with established quality assurance programs (10 CFR §71.101). Cask and component suppliers or vendors are required to obtain and maintain documents that prove the materials, processes, tests, instrumentation, measurements, final dimensions, and cask operating characteristics meet the design basis established in the Safety Analysis Report for Packaging for the cask, and that the cask will function as designed.

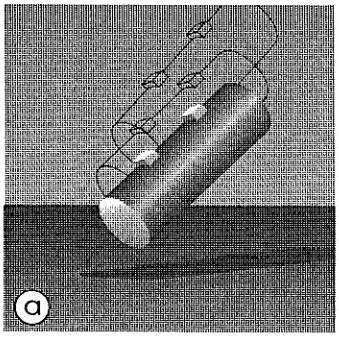
Regardless of where a transportation cask is designed, fabricated, or certified for use, it must meet certain minimum performance requirements (10 CFR §71.71-71.77). The primary function of a spent nuclear fuel transportation cask is to provide containment, criticality control, and shielding. Regulations require that casks must be operated, inspected, and maintained to high standards, ensuring their ability to contain their contents in the event of a transportation accident (10 CFR §71.87). There are no documented cases of a release of radioactive materials from spent nuclear fuel shipments even though thousands of shipments have been made by road, rail, and water transport modes. Further, a number of obsolete casks have been tested under severe accident conditions to demonstrate their adherence to design criteria without failure. Such tests have demonstrated that transportation casks are not only fabricated to a very high factor of safety; they are even sturdier than required.

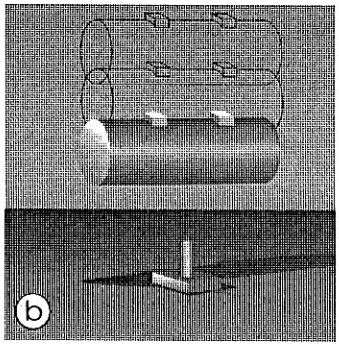
Transportation casks are built out of heavy, durable structural materials, such as stainless steel. These materials must ensure cask performance under a wide range of temperatures (10 CFR §71.43). In addition to the structural materials, shielding is provided to limit radiation levels at the surface and at prescribed distances from the surface of transportation casks (10 CFR §71.47). Shielding typically consists of dense material such as lead or depleted uranium. In some cases, additional materials are added to provide neutron shielding such as water-filled outer jackets, or highly hydrogenous materials such as polyethylene. The cask cavity is configured to hold various contents including spent nuclear fuel assemblies. The assemblies are supported by internal structures or baskets that provide shock and vibration resistance, establish minimum spacing and criticality control through the use of nuclear poison materials such as boron-impregnated metals, and heat transfer to maintain the temperature of the contents within the limits specified in the Safety Analysis Report for Packaging.

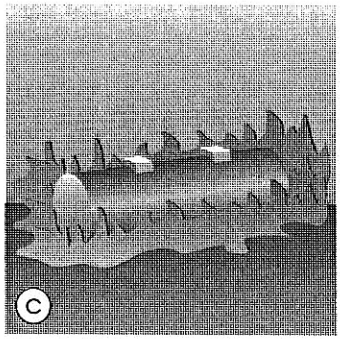
Finally, to limit impact forces and minimize damage to the structural components of a cask in the event of a transportation accident, impact-absorbing structures may be attached to the exterior of the cask. These are usually composed of balsa wood, foam, or aluminum honeycomb that is designed to readily deform upon impact to absorb impact energy. All of these components are designed to work together in order to satisfy the regulatory requirements for a cask to operate under normal conditions of transportation and maintain its integrity in an accident.

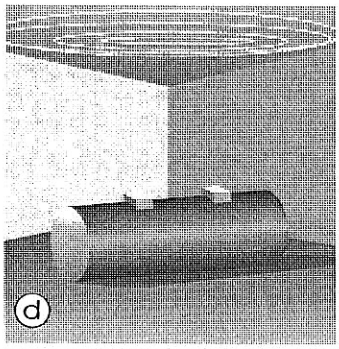
Design Certification

For certification, transportation cask must be shown by analysis and/or test to withstand a series of hypothetical accident conditions. These conditions have been internationally accepted as simulating damage to transportation casks that could occur in most reasonably foreseeable accidents. The impact, fire, and water-immersion tests are considered in sequence to determine their cumulative effects on one package. These accident conditions are described in Figure B-7. The NRC recently issued revised regulations, 10 CFR Part 71, governing the transportation of radioactive materials. These regulations become effective on April 1, 1996 (NRC, 1995). The revised regulations conform with those of the International Atomic Energy Agency and current legislative requirements. The revised regulations









Standards for Spent Fuel Casks

For certification by the NRC, a cask must be shown by test or analysis to withstand a series of accident conditions. These conditions have been internationally accepted as simulating damage to spent fuel casks that could occur in most severe credible accidents. The impact, fire, and water-immersion tests are considered in sequence to determine their cumulative effects on one package. A separate cask is subjected to a deep water-immersion test. The details of the tests are as follows:

Impact

Free Drop (a) – The cask drops 30 feet onto a flat, horizontal, unyielding surface so that it strikes at its weakest point.

Puncture (b) – The cask drops 40 inches onto a 6-inch-diameter steel bar at least 8 inches long: the bar strikes the cask at its most vulnerable spot.

Fire (c)

After the impact tests, the cask is totally engulfed in a 1475°F thermal environment for 30 minutes.

Water Immersion (d)

The cask is completely submerged under at least 3 feet of water for 8 hours. A separate cask is completely immersed under 50 feet of water for 8 hours.

Figure B-7 Standards for Transportation Casks (NRC, 1987)

affecting "Type B" casks require that a spent nuclear fuel transportation cask with activity greater than 10^6 curies be designed and constructed so that its undamaged containment system would withstand an external water pressure of 290 psi, or immersion in 200 m (656 ft) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask. Except for the addition of the deep water immersion test, the regulations applicable to the transportation of foreign research reactor spent nuclear fuel are unchanged.

Under the Federal certification program, a "Type B" packaging design must be supported by a Safety Analysis Report for Packaging, which demonstrates that the design meets Federal packaging standards. The Safety Analysis Report for Packaging must include a description of the proposed packaging in sufficient detail to identify the packaging accurately and provide the basis for evaluating its design. The Safety Analysis Report for Packaging must provide the evaluation of the structural design, materials properties, containment boundary, shielding capabilities, and criticality control, and present the operating procedures, acceptance testing, and maintenance program. Upon completion of a satisfactory review of the Safety Analysis Report for Packaging by NRC to verify compliance to the regulations, a Certificate of Compliance is issued.

B.2.1.4 Transportation Regulations

To assure that the transportation cask is properly prepared for transportation, trained technicians perform numerous inspections and tests (10 CFR §71.87). These tests are designed to ensure that the cask components are properly assembled and meet leak-tightness, thermal, radiation, and contamination limits. The tests and inspections are clearly identified in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks can only be operated by registered users who conduct operations in accordance with documented and approved quality assurance programs meeting the requirements of the regulatory authorities. Records must be maintained that document proper cask operations in accordance with the quality requirements of 10 CFR §71.91. Reports of defects or accidental mishandling must be submitted to NRC.

B.2.1.4.1 Communications

Proper communication assists in assuring safe preparation and handling of transportation casks. Communication is provided by labels, markings, placarding, and shipping papers or other documents. Labels (49 CFR §172.403) applied to the cask document the contents and the amount of radiation emanating from the cask exterior (transport index). The transport index lists the ionizing radiation level (in mrem/hr) at a distance of 1 m (3.3 ft) from the cask surface.

In addition to the label requirements, markings (49 CFR Subpart D and §173.471) should be placed on the exterior of the cask to show the proper shipping name and the consignor and consignee in case the cask is separated from its original shipping documents (40 CFR §172.203). Transportation casks are required to be permanently marked with the designation "Type B," the owner's (or fabricator's) name and address, the Certificate of Compliance number, and the gross weight (10 CFR §71.83).

Placards (49 CFR §172.500) are applied to the transport vehicle or freight container holding the transportation cask. The placards indicate the radioactive nature of the contents. In the United States, spent nuclear fuel is a Highway Route Controlled Quantity which must be placarded according to 49 CFR §172.507. Placards provide the first responders to a traffic or transportation accident with initial information about the nature of the contents.

Shipping papers should have entries identifying the following: the name of the shipper, emergency response telephone number, description of spent nuclear fuel, and the shipper's certificate as described in 49 CFR §172 Subpart C.

In addition, drivers of motor vehicles transporting spent nuclear fuel must have training in accordance with the requirements of 49 CFR §172.700. The training requirements include: familiarization with the regulations, emergency response information, and the spent nuclear fuel communication programs required by the Occupational Safety and Health Administration. Drivers are also required to have training on the procedures necessary for safe operation of the vehicle used to transport the spent nuclear fuel.

B.2.1.4.2 Marine Transport

Relevant regulations applying to transport of spent nuclear fuel by vessel are found in 10 CFR §71 and 73, and 49 CFR §176. The U.S. Coast Guard, part of the Department of Transportation, inspects vessels for compliance with applicable regulations and requires 24-hour prenotification (33 CFR §160.207, 211, and 213).

49 CFR §171.12 (d) states that: "Radioactive materials being imported into or exported from the United States, or passing through the United States in the course of being shipped between places outside the United States, may be offered and accepted for shipment in accordance with International Atomic Energy Agency *Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, 1985 Edition.*" Compliance with certain specified conditions of this section is required. For example, highway route controlled quantities of radioactive material must be shipped in accordance with appropriate provisions of the hazardous materials regulations and a Certificate of Competent Authority must be obtained, with any necessary revalidations. A Certificate of Competent Authority fulfills the International Atomic Energy Agency requirement for multilateral approval for a shipment of "Type B" packages in international commerce.

49 CFR §176.5 details the application of the regulations to vessels: "...this subchapter applies to each domestic or foreign vessel when in the navigable waters of the U.S., regardless of its character, tonnage, size, or service, and whether self-propelled or not, whether arriving or departing, underway, moored, anchored, aground, or while in drydock."

49 CFR §176.15 provides for enforcement of 49 CFR Subchapter C: "(a) An enforcement officer of the U.S. Coast Guard may at any time and at any place, within the jurisdiction of the U.S., board any vessel for the purpose of enforcement of this subchapter and inspect any shipment of hazardous materials as defined in this subchapter." Provision is also made in this section to detain a vessel which is in violation of the hazardous materials regulations.

The U.S. Coast Guard may accept a certificate of loading issued by the National Cargo Bureau, Inc., as evidence that the cargo is stowed in conformity with law and regulatory requirements. The National Cargo Bureau, Inc., is a nonprofit organization directed by Government and industry representatives (49 CFR §176.18). 49 CFR §176.18 authorizes inspectors of the National Cargo Bureau, Inc., to assist the Coast Guard in administering the hazardous materials regulations. Their functions are as follows:

"(1) Inspection of vessels for suitability for loading hazardous materials;

(2) Examination of stowage of hazardous materials;

(3) Making recommendations for stowage requirements of hazardous materials cargo; and,

(4) Issuance of certificates of loading setting forth that the stowage of hazardous materials is in accordance with the requirements of 46 U.S.C. 170 and its subchapter."

Detailed requirements for radioactive materials are located in 49 CFR §176 Subpart M of the Hazardous Materials Regulations. General radioactive material stowage requirements state that "(b) A package of radioactive materials which in still air has a surface temperature more than 5°C (9°F) above the ambient air may not be overstowed with any other cargo. If the package is stowed under the deck, the hold or compartment in which it is stowed must be ventilated," (49 CFR § 176.700).

Except for exclusive-use shipments, requirements relating to transport indexes state that:

". . . the number of freight containers with packages of radioactive materials contained therein must be limited so that the total sum of the transport indexes in the containers in any hold or defined deck area does not exceed 200, and:

(1) The sum of transport indexes for any individual freight container, or group of freight containers, does not exceed 50; and,

(2) Each freight container or group of freight containers is (are) handled and stowed in such a manner that groups are separated from each other by a distance of at least six m (20 ft)," [49 CFR § 176.704(c)].

Section 176.76(a) includes provision for freight containers with hazardous materials to be carried onboard a vessel in accordance with the following:

" (I) The material must be in proper condition for transportation according to the requirements of this subchapter;

(2) All packages in the transport vehicle or container must be secured to prevent movement in any direction. However, vertical restraint is not required if the shape of the packages and the stuffing pattern precludes shifting of the load;

(3) Bulkheads made of dunnage which extend to the level of the cargo must be provided unless the packages are stowed flush with the sides or ends;

(4) Dunnage must be secured to the floor when the cargo consists of dense materials or heavy packages."

Each freight container must be placarded as required by 49 CFR §172 Subpart F of the Hazardous Materials Regulations [176.76(f)].

Section 176.80 requires that radioactive materials be segregated from other hazardous materials so that they do not interact dangerously in an accident, or alternatively, requires that the radioactive material be in separate holds when stored under deck. In 49 CFR §176.83(b), a table is provided (Table II) that specifies the minimum separation distances for different classes of hazardous materials onboard a vessel. A minimum horizontal separation distance of 3 m (10 ft) projected vertically from the reference package is required. For specified hazardous materials, the "separate from" requirement means that the materials must be placed in separate holds when stowed under deck.

B.2.1.4.3 Ground Transport

Overland shipments (by railcar or by truck) are regulated by a variety of the Department of Transportation and NRC regulations dealing with packaging, notification, escorts, and communications. In addition, there are specific regulations for carriage by rail and carriage by truck.

When provisions are made to secure a package so that its position within the transport vehicle remains fixed during transport, with no loading or unloading between the beginning and end of transport, a package shipped overland in exclusive-use closed transport vehicles may not exceed the following radiation levels as provided in 49 CFR §173.441(b):

- 1,000 mrem/hr on the external package surface;
- 200 mrem/hr at any point on the outer surface of the vehicle;
- 10 mrem/hr at any point 2 m from the vertical planes projected from the outer edges of the vehicle;
- 2 mrem/hr in any normally occupied position in the vehicle, except that this provision does not apply to private motor carriers when the personnel are operating under a radiation protection program and wear radiation-exposure monitoring devices.

The shipper of record must comply with the requirements of 10 CFR §71.5 and §73.37. Section 71.5 provides that all overland shipments must be in compliance with the Department of Transportation and NRC regulations; these regulations provide for security of irradiated reactor fuel. General requirements include: providing notification to NRC in advance of each shipment, developing a shipping plan, providing escort instructions, establishing a communications center to be staffed 24 hours a day, making arrangements with local law enforcement agencies along the route for their response (if not using law enforcement personnel as escort), ensuring that the escorts are trained in accordance with Section 73.37 Appendix D, and ensuring that escorts make notification calls every 2 hours to the communications center. Additional requirements include having two armed escorts within heavily populated areas (when not in heavily populated areas, only one escort is needed) and the capability of communicating with the communications center and local law enforcement agencies through a radiotelephone or other NRC approved means of two-way voice communication.

The shipper of record, as required by 49 CFR §173.22, provides physical security measures for spent nuclear fuel shipments equivalent to those of NRC. The shipper, or the shipper's agent, provides notification for unclassified spent fuel shipments to State officials.

B.2.1.4.3.1 Rail Transport

Rail transportation requirements for radioactive materials are contained in 49 CFR §174. Briefly, for rail shipments of spent nuclear fuel the following additional requirements apply:

- railcars carrying radioactive materials must be segregated from other cars within a train, and cannot be next to other placarded hazardous materials (49 CFR §174.85) or occupied engines or cabooses; and
- hazardous materials shipments (including radioactive) must be expedited (49 CFR §174.14).

In addition, Association of American Railroad Interchange rules require that spent nuclear fuel be shipped only on railcars meeting certain construction and packaging retention requirements (AAR Rule 88A 1d). Rail routing has not been regulated by the Department of Transportation because the railroads are privately-owned companies. However, rail routes used for spent nuclear fuel shipping must be approved by NRC under a physical security plan (10 CFR §73.37).

B.2.1.4.3.2 Truck Transport

Truck transportation requirements for radioactive materials are contained within 49 CFR §177.800. In addition to requirements for securement and segregation by total transport index (50), there are road routing requirements as well. For carriage by truck, the carrier will use interstate highways or State-designated preferred routes for movement of radioactive materials in conformity with the Department of Transportation rulemaking, Docket HM-164. These regulations, found in 49 CFR Subpart D, establish routing and driver training requirements for highway carriers of packages containing "highway-route-controlled quantities" of radioactive materials. Spent nuclear fuel shipments constitute such quantities. The Department of Transportation also issues road operating requirements for radioactive materials shipments, including parking and operating rules. Primarily, these rules require trucks to stop and undergo visual inspection by the driver every 160 km (100 mi). Domestic road routing must also be approved by NRC under a physical security plan.

Many State and local governments have established their own rules, specifying such things as prenotification requirements, time-of-day restrictions, routes, and special equipment. State and local regulations that unnecessarily burden, delay, or ban shipments of radioactive materials will be preempted under the Hazardous Materials Transportation Act. The Department of Transportation rules make routing designation by appropriate State agencies enforceable by the Federal Government according to a determination by the Department of Transportation that such route designations are likely to result in further reduction of radiological risk.

B.2.2 Potential Transportation Casks

This section provides a description of the transportation casks that could be used for marine and ground transport of foreign research reactor spent nuclear fuel. The casks were identified from a review of the "Directory of National Competent Authorities' Approval Certificates for Package Design, Special Form Material, and Shipment of Radioactive Material, 1993 Edition," and the RAMPAC (radioactive material package) database for certified radioactive materials packaging (NRC, 1993). The review included only those transportation casks with current "Type B" designations for spent nuclear fuel.

B.2.2.1 Marine Transport

Table B-13 identifies the potential transportation casks for marine transport of foreign research reactor spent nuclear fuel. Each of these casks has both a certification from the country of origin and a certificate of competent authority from the Department of Transportation, which is designated as the Competent Authority for the United States. Except for the Unifetch, each of the casks has been previously used or accepted for use by DOE.

Table B-13 Proposed Transportation Casks for Marine Transport

Transportation Cask	*Certificate*	*DOE Experience*	*Country of Origin*
LHRL-120	USA/0389/B(U)F	Yes	Australia
GNS-11	USA/0381/B(U)F	Yes	Germany
TN-1	USA/0316/B(U)F	Yes	Germany
IU-04	USA/0100/B(U)F	Yes	France
TN-7 (TN-7/2)	USA/0130/B(U)F	Yes	Germany
NAC-LWT	USA/9225/B(U)F	Yes	United States
Unifetch	GB/1113/B(M)F	No	Great Britain
Goslar	USA/0094/B(M)F	Yes	Germany

Table B-14 summarizes the essential characteristics of the marine transportation casks for foreign research reactor spent nuclear fuel, such as physical dimensions, weight, type, and quantity of spent nuclear fuel elements each cask can accommodate, cooling time before shipment, maximum activity content in a cask, and the maximum initial ^{235}U content of each element. A summary of important characteristics of these casks is also provided after Table B-14.

Table B-14 Transportation Cask Design Characteristics for Marine Transport

Shipping Cask	*Weight (MT)*	*^{235}U g/Elements*	*Fuel Type*	*Number of Elements per Cask*	*Cooling Time (Days)/Activity per Cask (kCi)*	*Cask Dimensions (mm)*
LHRL-120	21.4	150-170	MTR tubular (HIFAR)	114	2,557/80	H: 3,400 D: 2,300
GNS-11	13.6	173 323	Tubular MTR Boxed-type MTR	21-28 33	180/41 180/27	H: 1,460 D: 1,185
TN-1	18.4	NA	Boxed-type MTR	126	NA	H: 2,910 D: 950
IU-04	18.9	Varied: 150-8600	Special Tubular Boxed-type MTR TRIGA Spent Nuclear Fuel	1 36-40 40-44	about one yr/1,250	H: 2,240 D: 1,880
TN-7 (TN-7/2)	25.5 (24.5)	290 8,500	Tubular-Boxed type MTR Special	60-64 2	250-1,780/2,000 310/2,000	H: 3,155 D: 1,030
NAC-LWT	23.2	17,575 Natural* 354	Commercial PWR Commercial BWR Metallic Rod Boxed-type MTR	1 - PWR 2 - BWR 15 - Met. Rods 42	730 - PWR 730 - BWR 365 - Met. Rods 1,095	H: 5,100 D: 1,120
Unifetch	16.9	405 170	Tubular Boxed-type MTR	24 40	90/45.4 90/123	H: 2,100 D: 1,800
Goslar	10.9	320	Boxed-type MTR	13	120/960	H: 1,460 D: 1,190

**Natural = Maximum initial U^{235} is 0.711 weight percent*
NA = Not Available
PWR = Pressurized Water Reactor
BWR = Boiling Water Reactor

LHRL-120

The LHRL-120 consists of a cylindrical cask surrounded by an impact limiter supported on cradles attached to a skid that is bolted to the base of a shipping container. The cask is a right circular cylinder with two concentric walls of steel for structural strength, with the annular area between the walls. The inner shell forms the containment.

The cask is built of inner and outer shells welded to the bottom end closure plate and top bolt ring, and secured by a bolted lid with a double o-ring seal. The annular space between the shells is filled with lead and supplementary lead shielding plates are provided on the bottom end closure plate and lid. The cask has two external lifting trunnions and, except for the high strength steel bolts, lead shielding, and synthetic rubber o-ring, is constructed of stainless steel plugs.

The impact limiter consists of a steel shell filled with dense polyurethane foam arranged to provide energy absorption and thermal insulation.

During transport, the cask body is completely enclosed by an impact limiter which provides both thermal and impact protection. The impact limiter is constructed in two pieces which bolt together and surround the cask body. LHRL-120 is designed for passive cooling by means of cooling tubes that penetrate the impact limiter. Tubes in the bottom half also transfer loads to the cradles. The cask and the impact limiter are secured to the skid by two tie-down straps and restraints. The skid is bolted to the base of an open conventional shipping container and is in turn enclosed by a steel weather cover fitting inside the end walls of the container and bolted to the container base. The container has standard International Standards Organization lifting arrangements and is approved under the international convention for safe containers.

The length and diameter of the cask with the impact limiter are 3.4 m (134 in) and 2.3 m (91 in), respectively. The total mass of the cask with contents, impact limiter, skid and tie-downs is 21.36 metric tons (47,080 lb) and the gross mass of the package including lift yoke, bolt tooling, tool box, weather cover and shipping container is approximately 24 metric tons (52,800 lb).

The cask was designed by Eggers, Ridelhalgh, and Partners of Columbus, Ohio for spent nuclear fuel from the High Flux Australian Research Reactor (HIFAR).

Permitted Contents:

Irradiated spent nuclear fuel elements with a minimum decay period of 7 years

Maximum number of fuel elements per package	120[a]
Maximum fuel mass	554 kg (1,200 lb)
Maximum decay heat	290 Watts
Maximum mass of both baskets (empty)	891 kg (1,965 lb)
Maximum activity of package	80,000 Ci ($3.0 \times 10^{+15}$ Bq)
Transport Index	50

[a] *The maximum number allowed is 114.*

Two identical baskets are authorized for the LHRL-120 cask in a 1 x 2 array, (i.e., in a stacked configuration). The baskets are constructed exclusively of aluminum alloys 6061 and 6063. Each basket contains 60 cells, each providing an 11-cm- (4.3-in-) diameter by 65-cm- (25.75-in-) high cylindrical

cavity for each fuel assembly. Only 57 of the cells are loaded with fuel elements; the 3 center-most positions are left unloaded. The nominal wall thickness of the cell is 6.4 mm (0.25 in). Maximum mass of both baskets tiers (empty) is 891 kg (1,965 lb).

GNS-11

The GNS-11 consists of a welded stainless steel/lead construction which is tightly closed with a primary lid. The cask body can be closed at the lid region with a protection plate. The spent nuclear fuel elements fit into a fuel basket which is inserted in the cask cavity. The cask is an upright circular cylinder with two concentric walls of steel for structural strength with the annular area between the walls filled with lead for radiation shielding. The inner steel shell forms the containment.

The containment system is formed by the cask body, the primary lid including elastomer seal rings, plugs, and boltings. During transport, hood shaped impact limiters consisting of steel plates with a soft wood filling are attached to the top and bottom of the cask. In the upper region, two trunnions are screwed to the cask body for handling. The cask has the following external dimensions:

Diameter (without impact limiters)	1,185 mm (46.7 in)
Diameter (with impact limiters)	1,355 mm (53.4 in)
Height (cask body)	1,460 mm (57.5 in)
Height (with impact limiters)	1,780 mm (70 in)

The cask body is protected during transport by top and bottom impact limiters while the cask is secured vertically on its low-boy transporter. The cask weighs about 13.6 metric tons (30,000 lb). The cask can be used to ship up to 28 tubular-type MTR elements and up to 33 box-type MTR elements with initial ^{235}U enrichment of up to 93 percent. In addition, the cask can also be used to transport other types of irradiated hardware. This cask is shown in Figure B-8.

Since the temperature on the outside of the package may exceed 50°C (122°F) and the transport index can be greater than 10, the package is to be transported as a full load or as a closed load. Therefore, a maximum of two casks could be fixed in a shipping container. The cask was designed and manufactured by the German company Gessellschaft fur Nuklear-Behalte GmbH. There are currently two GNS-11 casks available for use.

Permitted Contents:

Three different fuel baskets are authorized for use with this cask. These accommodate various types and amounts of fuel:

1. A maximum of 21 or 28 (depending on the type of fuel basket used) irradiated tubular MTR fuel elements consisting of 3 to 5 concentrically arranged fuel tubes, with the following further specifications per fuel element:

Maximum initial enrichment	80 percent
Chemical form	U-Al alloy
Maximum initial mass of ^{235}U	173.4 g (6 oz)
Maximum initial quantity of uranium	217.0 g (7.5 oz)
Maximum active length	61 cm (24 in)
Maximum diameter of outer tube	10.3 cm (4 in)
Minimum cooling time	180 days

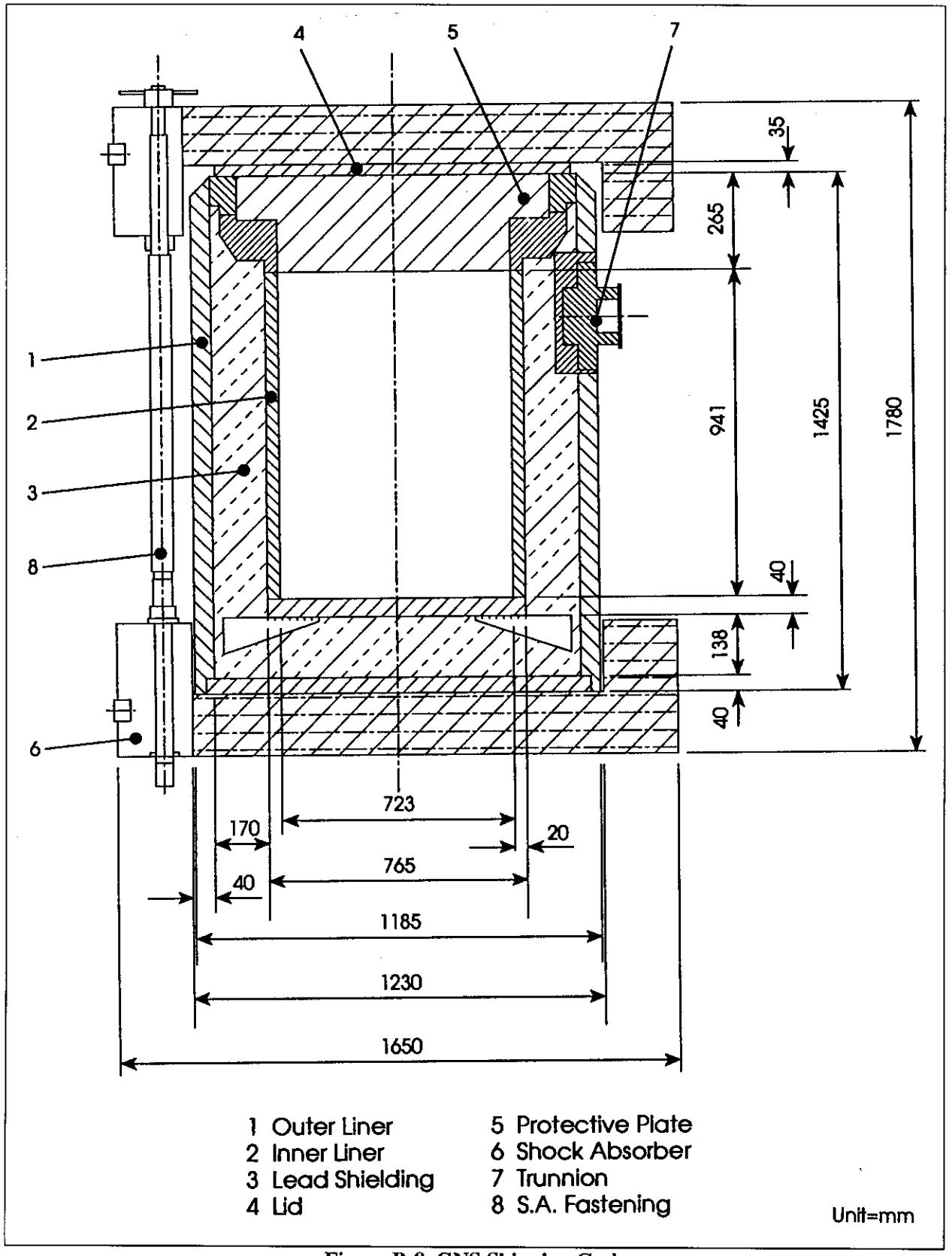


Figure B-8 GNS Shipping Cask

Thermal power (average)	maximum 76 Watts (for 21 fuel elements per cask)
Thermal power (average)	maximum 57 Watts (for 28 fuel elements per cask)
Maximum activity	40.5 kCi (1.5 PBq)

2. A maximum of 33 irradiated boxed-type MTR fuel elements, each containing a maximum of 23 aluminum-based fuel plates, with the following further specifications per fuel element:

Maximum initial enrichment	93 percent
Maximum initial mass of ^{235}U	268 g (9.3 oz)
Maximum initial mass of uranium	335 g (11.6 oz)
Active length	maximum 61 cm (24 in)
Cross-sectional area	approx. 81 x 76 mm (3.2 x 3.0 in)
Minimum cooling time	180 days
Thermal power (average)	maximum 48.5 Watts
Maximum activity	27 kCi (1 PBq)

3) A maximum of 33 irradiated boxed-type MTR-fuel elements each containing a maximum of 23 aluminum-based LEU fuel plates (containing dispersed U_3Si_2 or U_3O_8) with the following further specifications per fuel element:

Maximum initial enrichment	20 percent
Maximum initial mass of ^{235}U	323 g (11.2 oz)
Maximum initial mass of uranium	1,635 g (3.6 lbs)
Active length	maximum 61 cm (24 in)
Cross-sectional area	approx. 81 x 76 mm (3.2 x 3 in)
Minimum cooling time	360 days
Thermal power (average)	maximum 48.5 Watts
Activity	maximum 27 kCi (1 PBq)

TN-1

The TN-1 is a cylindrical double-walled steel container with lead and plaster for shielding. It is constructed of steel structural shells with the annulus between them filled with lead for gamma shielding and plaster as a heat shield. Additional heat insulation and impact resistance is provided by impact limiters. The cask, with the impact limiters, weighs 18.37 metric tons (40,500 lb). The internal cavity can accommodate three baskets, one on top of the other, each filled with up to 42 boxed-type MTR fuel elements of initial ^{235}U enrichment of up to 94 percent.

The containment system is formed by the cask body, lid with its "elastomer" seals and bolts, and three sealing plates with "elastomer" seal rings in the cask body via the quick connections.

Physical dimensions of TN-1 are as follows:

	Without Shock Absorber	*With Shock Absorber*
Width	950 mm (37.4 in)	1,284 mm (50.6 in)
Height	920 mm (36.2 in)	1,254 mm (49.4 in)
Length	2,910 mm (114.6 in)	3,075 mm (121 in)

TN-1 is designed by the French company Cogema. It is shipped in the horizontal position with the top and bottom impact limiters attached. The TN-1 cask is shown in Figure B-9.

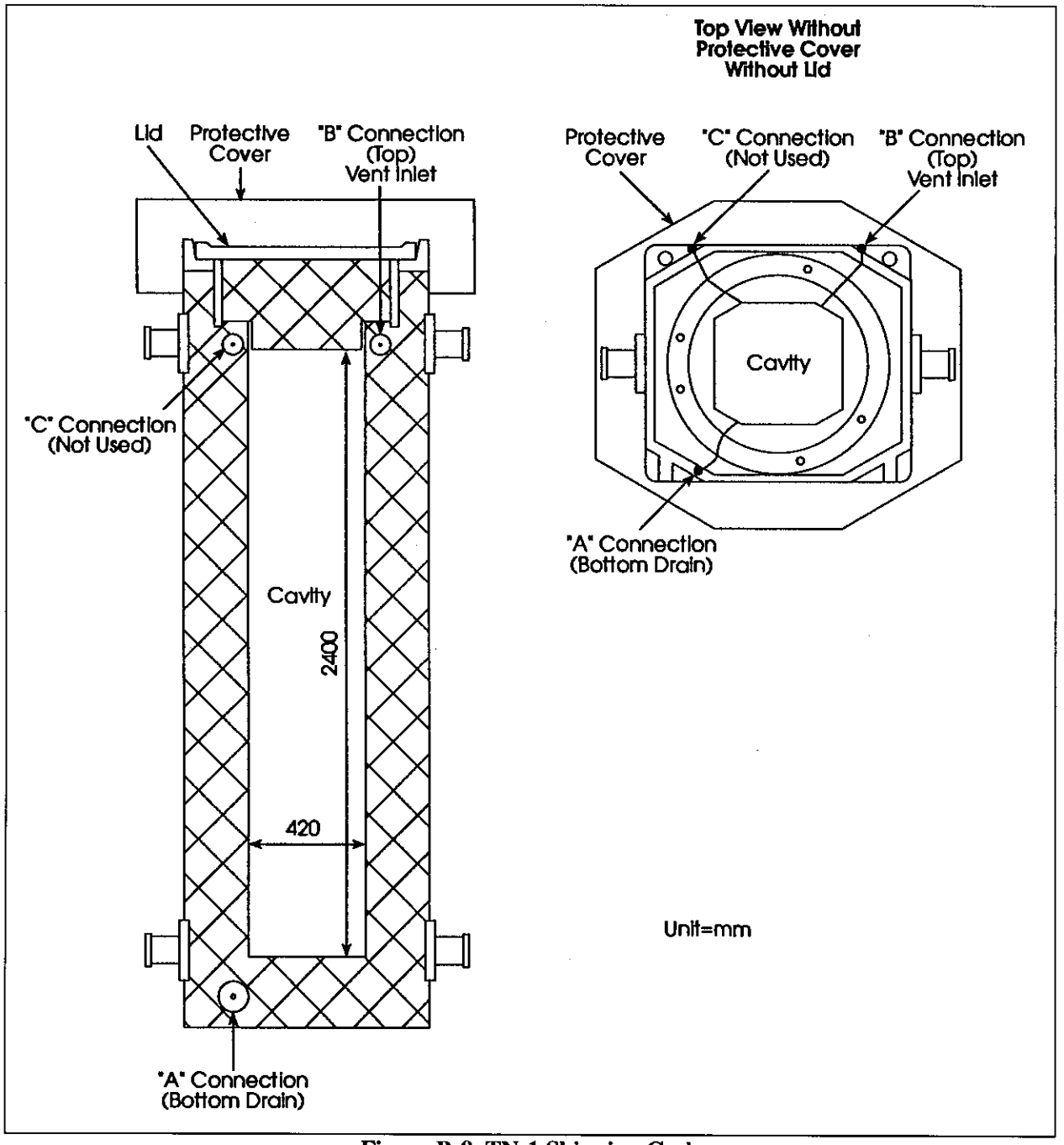


Figure B-9 TN-1 Shipping Cask

IU-04

The IU-04, also known as Pegase, consists of a body built of two stainless steel shells enclosing a lead shield. The inner confinement shell and the lead shield form a solid unit constituting the body. The outer shell is provided with a base plate filled with asbestos and is fitted with cooling fins. A layer of plaster is placed between the bottom of the outer shell and the lead. The steel lid is filled with lead and plaster. There are two pipe systems connecting the inner tank to the outside.

The cask is a right circular cylinder with two concentric walls of steel for structural strength with the annular area between the walls filled with lead for radiation shielding. The inner steel shell forms the containment. The IU-04 inner cavity can accommodate a variety of baskets which may be used to transport both MTR and TRIGA spent nuclear fuel. The cask weighs approximately 18.9 metric tons (41,670 lb). It is transported in the vertical position on the pallet, with top and bottom impact limiters to protect it in the event of an accident.

There is a main protective cover of stainless steel filled with balsa wood of two different densities. There are covers of mild steel protecting the pipe outlets which are filled with plaster. Like TN-1, this cask was also designed by Cogema. The IU-04 cask is shown in Figure B-10.

The cask is authorized to be used with various baskets designed for different types of spent nuclear fuel. The following summarizes a selected number of baskets designed for IU-04 casks:

1. Basket AA-267 - consists of cylindrical aluminum grid, 960 mm (37.8 in) high, containing 40 channels of square cross section, 84 x 84 mm (3.3 x 3.3 in), and 4 channels of cross section, 72 x 72 mm (2.83 x 2.83 in). The grid is surrounded by an aluminum belt with outside diameter of 795 mm (31.3 in). The aluminum contains two percent boron. The bottom end is covered by a 15-mm- (0.6-in-) thick aluminum plate welded to the cylindrical belt. It contains drain orifices.

Diameter	795 mm (31.3 in)
Total Height	1,030 mm (40.6 in)
Useful Height	960 mm (37.8 in)
Approx. Weight	360 kg (793.7 lbs)

   A total of 44 MTR boxed-type (72 x 72 mm cross section) fuel elements can be put in this basket. The maximum allowed residual thermal power per element is less than 80 Watts.

2. Basket TN-9083 - consists of a block of stainless steel containing five percent boron. The basket is 895 mm (35.2 in) long and has 36 lodgments of 81 x 87 mm (3.2 x 3.4 in) cross section, bored to a diameter of 98 mm (3.9 in). The bottom is covered by a plate, 12 mm (0.5 in) thick, fastened to the block by screws. The bottom plate contains drain orifices of 50 mm (2 in) diameter. The dimensions of the basket are as follows:

Base Height	90 mm (3.5 in)
Diameter	796 mm (31.3 in)
Total Height	907 mm (35.7 in)
Useful Height	895 mm (35.2 in)
Approx. Weight	1,410 kg (3,108 lbs)

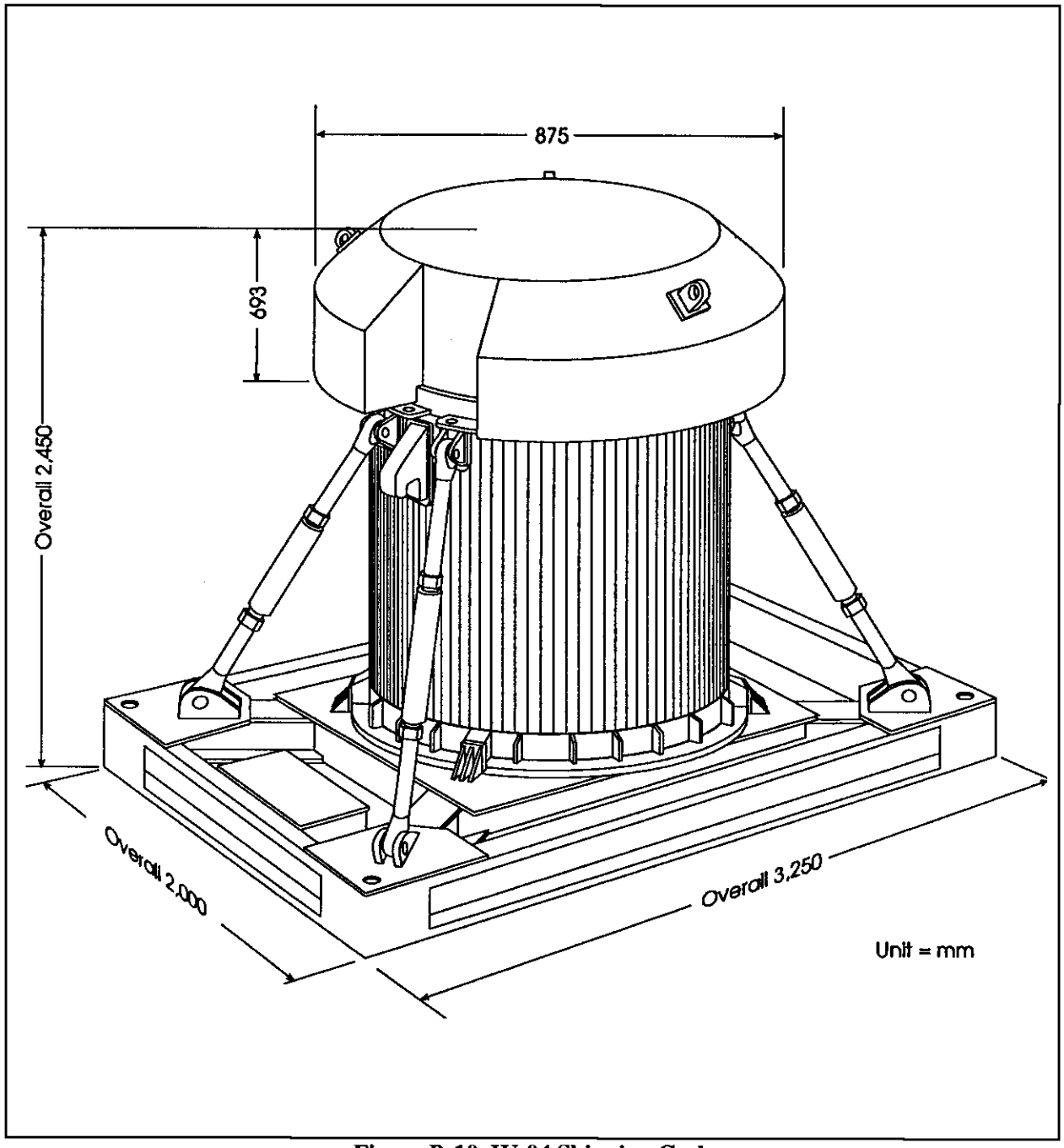


Figure B-10 IU-04 Shipping Cask

A maximum of 36 MTR tubular type fuel elements can be put in this basket. The maximum residual thermal power of each spent nuclear fuel must be less than 132 Watts.

Basket TN-9083 can also be used for TRIGA spent nuclear fuel. The maximum ^{235}U content of each TRIGA spent nuclear fuel element must be less than 40 g (1.4 oz).

3. Basket AA-49 - consists of 5 sectors of Copper-Cadmium alloy (at least 2 percent Cadmium by weight) with 5 square channels of 84 x 84 mm (3.3 x 3.3 in) and 1 channel of 71.5 x 71.5 mm (2.8 x 2.8 in), and a central core in stainless steel with a system for fastening the sectors.

Diameter	800 mm (31.5 in)
Total Height	1,030 mm (40.6 in)
Useful Height	970 mm (38.2 in)
Approx. Weight	2,500 kg (5,511 lbs)

Basket AA-49 accommodates 30 fuel elements of 93 percent enrichment from BR-2 with a maximum allowable residual power of 266 Watts per element.

4. Basket AA-50 - consists of 6 sectors of Copper-Cadmium alloy (at least 2 percent Cadmium by weight) with 6 channels of rectangular cross section 86 x 77.5 mm (3.4 x 3.1 in), and a central core in stainless steel with a system for fastening the sectors.

Diameter	800 mm (31.5 in)
Total Height	1,030 mm (40.6 in)
Useful Height	970 mm (38.2 in)
Approx. Weight	1,996 kg (4,400 lbs)

Basket AA-50 accommodates 36 boxed-type MTR fuel elements of up 93 percent enrichment. Maximum allowable residual power per each element is 200 Watts.

5. Basket AA-117 - is fabricated in Z2-CN-18-10 stainless steel with a base plate 10 mm (0.4 in) thick drilled with water drain holes in the center, 4 vertical posts 10 mm (0.4 in) thick bolted to the base plate and connected together by 3 circular spacers. Basket AA-117 accommodates 1 fuel element of 93.5 percent enrichment from RHF with a maximum allowable residual thermal power of less than 3,000 Watts.

Diameter	797 mm (31.4 in)
Total Height	1,030 mm (40.6 in)
Useful Height	420 mm (16.5 in)
Approx. Weight	165 kg (364 lbs)

TN-7 (TN-7/2)

The TN-7 consists of a cylindrical stainless steel exterior container with corresponding stainless steel lid with an integrated lead shielding; four trunnions; one bottom shock absorber; and a stainless steel, concentric cylindrical interior container, which together with its lead constitutes the "tight enclosure." Between the interior and the exterior container there is a lead shielding, 185 mm (7.3 in) thick at the sides, and 170 mm (6.7 in) thick at the lid. This shielding is surrounded by a humid cement thermal insulation. Within the interior container, up to four racks can be stacked upon each other for the admissible contents mentioned above.

The cask is a right circular cylinder with two concentric walls of steel for structural strength. The annular area between the steel walls is filled with lead for radiation shielding. The inner steel shell forms the containment.

The cask weighs about 25.5 metric tons (56,220 lb). The TN-7 was originally designed for the transportation of short light water reactor spent nuclear fuel but has the capability to accommodate the highly-enriched MTR spent nuclear fuel. In this capacity, the 4 baskets that fit in the inner cavity can accommodate up to 15 tubular or 16 box-type MTR fuel elements each.

The cask is transported in the horizontal position with top and bottom impact limiters providing protection in the event of an accident.

The TN-7/2 is very similar in design and dimension to the TN-7. The TN-7/2 is used to transport the same types and quantities of spent nuclear fuel as the TN-7. In addition, it can be used to transport up to 64 box-type MTR spent nuclear fuel elements or 2 RHF special spent nuclear fuel elements. The TN-7/2 is transported the same way as the TN-7. There is one TN-7 cask available for use at the present time. This cask has been designed by the German company Transnuklear GmbH.

Permitted contents:

1) Up to four insert racks, containing per rack:

   - maximum 15 irradiated tubular-type MTR fuel elements, each containing a maximum of 250 g (8.7 oz) of uranium enriched between 80 and 93 percent with a maximum of 200 g (6.9 oz) of ^{235}U in the form of a U-Al alloy, with a minimum cooling time of 250 days and a maximum activity of 40 kCi (1.48 PBq), or

   - maximum 16 irradiated Boxed-type MTR fuel elements, each containing a maximum of 363 g (12.6 oz) of uranium enriched between 80 and 93 percent, with a maximum of 290 g (10.1 oz) of ^{235}U in the form of a U-Al alloy, with a minimum cooling time of 1,780 days and a maximum activity of 20 kCi (740 TBq).

The racks can be combined within a cask, provided that the maximum thermal powers do not exceed 125 Watts per fuel element; 1,125 kW per rack; and 4.5 kW per cask.

OR

2) Up to two irradiated RHF type fuels, or a fuel containing a maximum number of 280 fuel plates each, with an active fuel length of about 900 mm (35.4 in), containing originally a maximum of 9.32 kg (20.6 lbs) of uranium enriched to 93 percent of ^{235}U with a maximum of 8.67 kg (19.1 lbs) of ^{235}U in the form of a U-Al alloy per element.

Maximum activity per fuel element	1,000 kCi (37 PBq)
Thermal output per fuel element	maximum 2.25 kW
Cooling time	310 days

TN-7 is authorized as Fissile Class II with a minimum Transport Index of 8.3 per package.

NAC-LWT

NAC-LWT is a steel encased lead shielded transportation cask. The cask body consists of a 19-mm- (0.75-in-) thick stainless steel inner shell, a 146-mm- (5.75-in-) thick lead gamma shield, a 30-mm- (1.2-in-) thick stainless steel outer shell, and a neutron shield tank. The inner and outer shells are welded

to a 101.6-mm- (4-in-) thick stainless steel bottom and forging. The cask bottom consists of a 76.2-mm- (3-in-) thick, 52.7-cm- (20.75-in-) diameter lead disk enclosed by a 88.9-mm- (3.5-in-) thick stainless steel plate and bottom end forging. The cask lid is a 287-mm- (11.3-in-) thick ring stainless steel stepped design, secured to a 362-mm- (14.25-in-) thick ring forging with twelve 25.4-mm- (1-in-) diameter bolts. The cask seal is a metallic O-ring. A second teflon O-ring and a test port are provided to leak test the seal. Other penetrations in the cask cavity include the fill and drain ports, which are sealed with port covers and teflon O-rings. The cask weighs about 22.4 metric tons (51,200 lb) including a maximum of 1.75 metric tons (4,000 lb) weight of fuel and basket.

The neutron shield tank consists of a 6.1-mm- (0.24-in-) thick stainless steel shell with 12.7-mm- (0.50-in-) thick end plates. The neutron shield region is 416.5 cm (164 in) long and 127 mm (5 in) thick. The neutron shield tank contains an ethylene glycol/water solution that is 1 percent boron by weight.

The overall dimensions of the package, with impact limiters, are 5.9 m (232 in) long by 165.1 cm (65 in) diameter. The cask cavity is 4.52 m (178 in) long and 340 mm (13.4 in) in diameter, having a volume of about 0.41 m^3 (14.5 ft^3). The cask is equipped with aluminum honeycomb impact limiters. The top impact limiter has an outside diameter of 165.7 cm (65.25 in) and a maximum thickness of 71.9 cm (28.3 in). Both impact limiters extend 30.5 cm (12 in) along the side of the cask body. The cask is transported in the horizontal position.

NAC-LWT is designed to transport one pressurized water reactor assembly, two boiling water reactor assemblies, up to 15 metallic fuel rods, or 42 boxed-type MTR foreign research reactor spent nuclear fuel with a proper basket design. There are several NAC-LWT casks available which could be used to transport foreign research reactor spent nuclear fuel. It is designed by the Nuclear Assurance Corporation in the United States.

Unifetch

Unifetch was originally designed for the transport of the spent nuclear fuel from the BR-2 (Belgium reactor). The cask weighs about 18.6 metric tons (41,000 lbs) and can accommodate either 24 or 40 spent nuclear fuel elements. The cask is transported in the vertical position. Unifetch is designed by Transport Technology in the United Kingdom.

Permitted Contents:

Two types of baskets are designed for Unifetch:

1) Baskets with maximum capacity of 24 fuel elements: Irradiated BR-2 nuclear fuel elements, assembled from plates, consisting of an inner core of natural or enriched uranium alloyed with aluminum contained within an aluminum cladding.

Fuel core thickness	0.51 mm (0.02 in)
Maximum pre-irradiation mass of ^{235}U	405 g (14.1 oz)
Maximum mass per unit length of ^{235}U/assembly	5.495 g/cm (0.5 oz/in)
Maximum decay heat per fuel element	10.7 Watts
Maximum decay heat per package	260 Watts
Minimum fuel active length	737 mm (29 in)
Maximum fuel active cross section	5,384.56 mm^2 (8.35 in^2)
Cladding thickness	0.38 mm (0.01 in)
Maximum activity of package	45.4 kCi (1.68 PBq)
Minimum cooling time	90 days

2) Baskets with maximum capacity of 40 fuel elements: Irradiated MTR boxed type nuclear fuel elements, assembled from plates, consisting of an inner core of natural or enriched uranium alloyed with aluminum contained within an aluminum cladding.

Maximum mass of ^{235}U per element	170 g (6.7 in)
Maximum mass of ^{235}U in the shield	1,265 g (2.8 lbs)
Maximum decay heat per fuel element	11.5 Watts
Maximum decay heat per package	460 Watts
Minimum fuel active length	58.42 cm (23 in)
Maximum activity of package	123.3 kCi (4.56 PBq)
Minimum cooling time	90 days

GOSLAR

The Goslar cask is a double-walled right circular cylindrical steel container that uses lead shielding in the annulus between the inner containment and outer structural container. The Goslar-Behatler was previously used to transport boxed-type MTR elements with ^{235}U enrichment between 20 percent and 93 percent from several foreign research reactors to the United States.

Goslar was designed and fabricated by Transnuklear GmbH. It weighs approximately 10.9 metric tons (24,000 lb) and has inner cavity dimensions of 483 mm (19 in) diameter x 960 mm (37.8 in) tall. Exterior dimensions, including impact limiters, are 1,185 mm (46.4 in) diameter and 1,460 mm (57.4 in) height.

Permitted Contents:

Three different fuel configurations are authorized to be used with this cask. These accommodate various types and amounts of fuel:

1) A maximum of 13 irradiated MTR fuel elements (consisting of flat or curved fuel plates) with the following further specifications per fuel element:

Maximum initial enrichment	93 percent
Chemical form	U-Al alloy
Maximum initial mass of ^{235}U	320 g (11.1 oz)
Minimum cooling time	120 days
Thermal power	maximum 300 Watts
Maximum activity	89.2 kCi (3.3 PBq)
Thermal power	maximum 3,200 Watts per cask
Maximum activity	960 kCi (35.5 PBq) per cask

2) A maximum of 13 irradiated boxed-type MTR fuel elements, with the following further specifications per fuel element:

Maximum initial enrichment	45 percent
Maximum initial mass of ^{235}U	323 g (11.2 oz)
Minimum cooling time	120 days
Thermal power	maximum 300 Watts
Maximum activity	89.2 kCi (3.3 PBq)
Thermal power	maximum 3,200 Watts per cask
Maximum activity	960 kCi (35.5 PBq) per cask

3) A maximum of 13 irradiated MTR fuel elements with a total of 10.4 kg (22.9 lbs) of uranium enriched between 17 to 80 percent with a maximum of 1.755 kg (3.9 lbs) of ^{235}U, with the following further specifications per fuel element:

Maximum initial enrichment	80 percent
Maximum initial mass of ^{235}U	135 g (4.7 oz)
Minimum cooling time	200 days
Thermal power	maximum 1 Watt
Activity	maximum 300 Ci (.0111 PBq)

B.2.2.2 Ground/Intersite Transport

Table B-15 identifies the transportation casks for ground/intersite transport of foreign research reactor spent nuclear fuel. Each of these casks has a valid certificate for use in the United States. Although some of these transportation casks are not currently certified for the shipment of research reactor spent nuclear fuel similar to that from foreign research reactors, it is anticipated that all of the casks could be recertified to accept such material.

Table B-15 Transportation Casks for Ground Transport

Transportation Cask	*Certificate Number*	*DOE Experience*	*Country of Origin*
NLI-10/24	USA/9023/B( )F	No	United States
IF-300	USA/9001/B( )F	Yes	United States
BMI-1	USA/5957/B(U)F	Yes	United States
GE-2000	USA/9228/B(U)F	No	United States
TN-8	USA/9015/B( )	Yes	Germany
NLI-1/2	USA/9010/B( )F	Yes	United States
NAC-LWT	USA/9225/B(U)F	Yes	United States

Design information for ground transportation casks is summarized in Table B-16. Additional narrative summary information on each of theses casks is also provided below. Although no numbers are given for each cask capacity in terms of number of foreign research reactor spent nuclear fuel elements, it has been estimated that the space for each pressurized water reactor element (assembly) can accommodate 12 to 16 foreign research reactor spent nuclear fuel elements.

NLI-10/24

The Nuclear Assurance Corporation NLI-10/24 is a railcar transported stainless steel transportation cask. The cask is 519.4 cm (204.5 in) long, 234.8 cm (96 in) diameter, and weighs 72.5 metric tons (159,000 lb) empty. Radioactive shielding is provided by lead, water, depleted uranium, and a high temperature polymer. The cask is authorized to contain either 10 pressurized water reactor or 24 BWR irradiated uranium-oxide fuel assemblies.

Table B-16 Transportation Cask Design Characteristics for Ground Transport

Transportation Cask	*Empty Weight (metric tons)*	*Fuel Type*	*Number of Elements /Cask*	*Decay Heat Generation (kW)*	*Cooling Time (Days)*	*Cask Dimensions (mm)*
NLI-10/24[a]	77.5	PWR or BWR	10 PWR/ 24 BWR	70	150	H: 5,995 D: 2,440
IF-300[a]	43.1	PWR or BWR	7 PWR/ 17 BWR	11.7	120	H: 5,335 D: 1,625
BMI-1	9.9	MTR boxed-type	24	1.5	90	H: 1,864 D: 856
GE-2000	12.7	HFIR[b] Irradiated fuel	1	0.6	120	H: 3,340 D: 1,829
TN-8[a] (TN-9)[a]	16.3 (16.3)	PWR (BWR)	3 (7)	35.5 (24.4)	150 (150)	H: 5,740 D: 1,700
NLI-1/2	21	PWR or BWR	1 PWR/ 2 BWR	10.6/ 10.6	150/ 120	H: 4,953 D: 1,200
NAC-LWT[a]	23.2	PWR or BWR MTR	1 PWR/ 2 BWR 15	2.5/ 1.1 1	730 365	H: 5,080 D: 1,120

[a] *Currently does not have proper certification for foreign research reactor spent nuclear fuel use.*

[b] *High Flux Isotope Reactor fuel is similar to that of RHF fuel.*

PWR = Pressurized Water Reactor
BWR = Boiling Water Reactor

IF-300

The General Electric IF-300 is a stainless steel encased, depleted uranium transportation cask. The cask is 533.4 cm (210 in) long, 162.6 cm (64 in) in diameter, and weighs 43.1 metric ton (95,000 lb) empty. Radioactive shielding is provided by depleted uranium, stainless steel, and a water-ethylene glycol mixture. The cask is permitted to ship 7 pressurized water reactor or 17 boiling water reactor irradiated uranium-oxide fuel assemblies. The IF-300 transportation cask is illustrated in Figure B-11.

BMI-I

The BMI-1 cask is a truck transported, steel-encased, lead shielded transportation cask. The basic body is a right circular cylinder measuring 1.86 m (73.37 in) high and 0.85 m (33.37 in) in diameter. The cask weighs about 9.9 metric tons (21,860 lb) empty. The cask is permitted to ship 24 MTR boxed-type irradiated fuel assemblies. DOE, the authorized user of the BMI-1, lends it almost exclusively for the domestic shipment of research reactor fuel. As such, its design includes eight licensed basket and canister combinations, including one for TRIGA fuel with an initial enrichment up to 93 percent. These fuels are very similar to those used by the foreign research reactors. The BMI-1 cask is illustrated in Figure B-12.

GE-2000

The GE-2000 is a truck transported, stainless steel transportation cask. It is constructed from stainless steel shells and uses lead as a shielding material. The cask is 3.34 m (131.5 in) long, 1.8 m (72 in) in diameter, and weighs about 12.7 metric tons (28,000 lb) fully loaded. Current authorized contents include irradiated fuel rods and by-product, source, or special nuclear material. The GE-2000 cask is used primarily for domestic shipments of research reactor spent nuclear fuel. It is currently being certified for

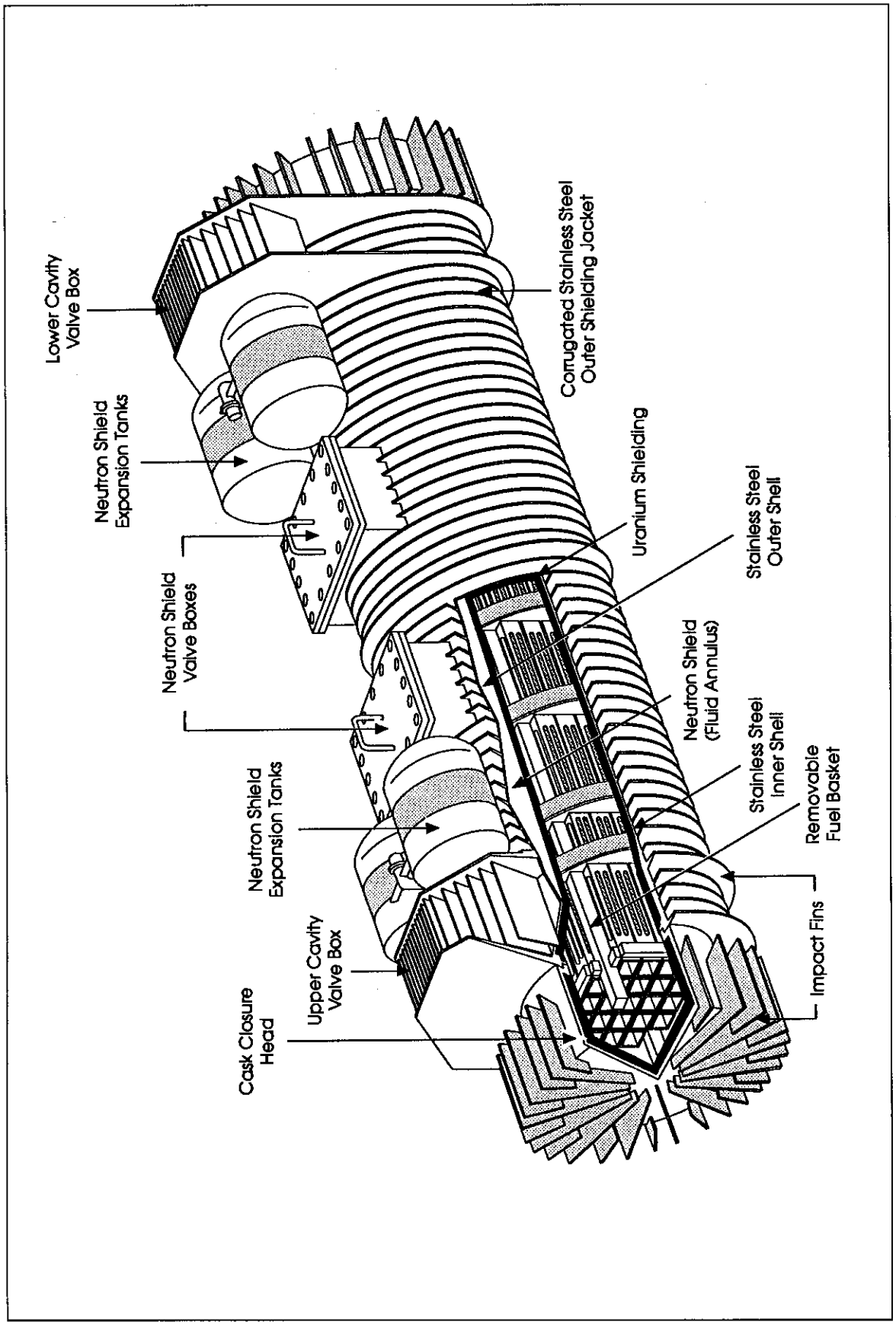


Figure B-11 IF-300 Shipping Cask

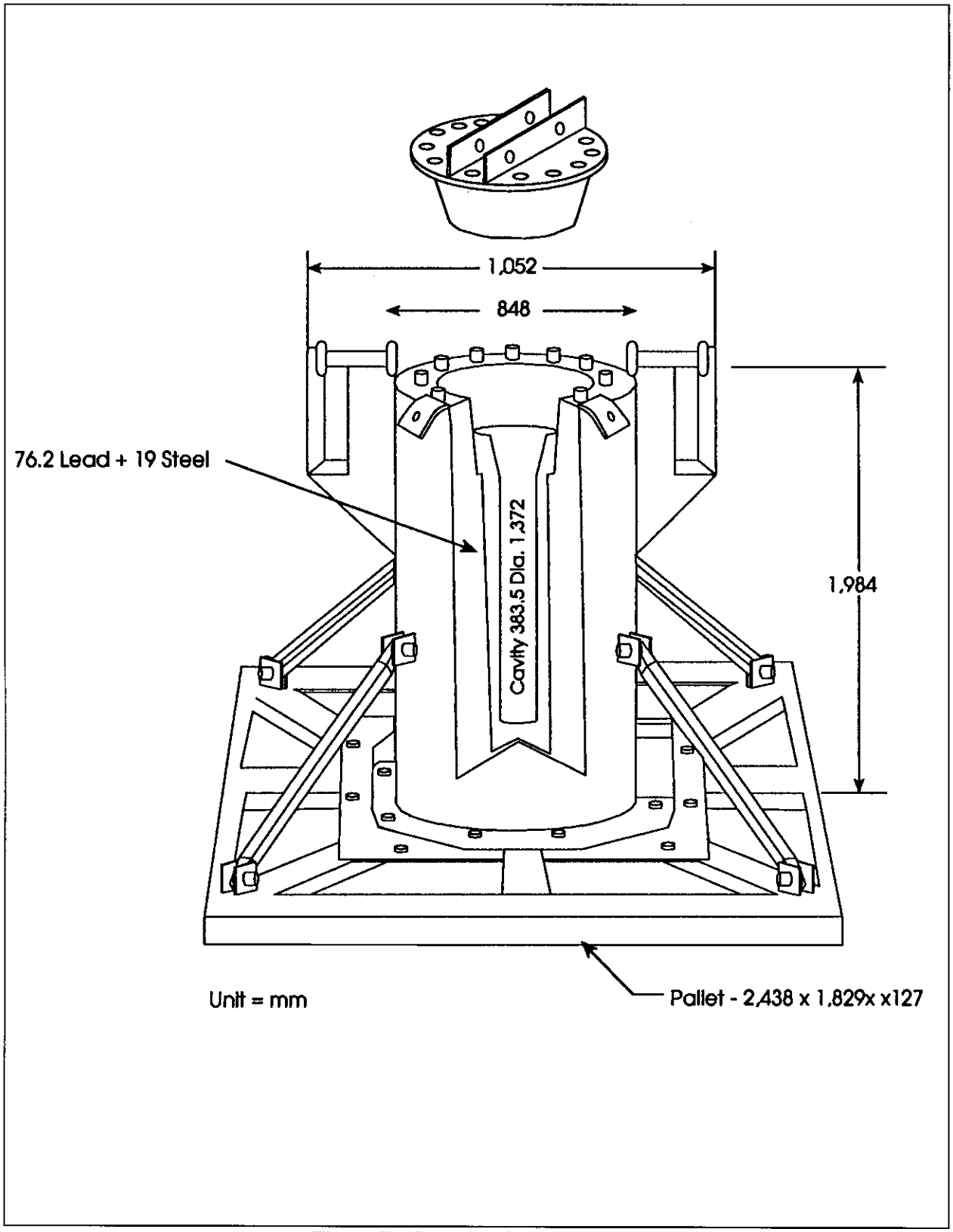


Figure B-12 BMI-1 Shipping Cask

use at the Oak Ridge National Laboratory for shipments of high flux isotope reactor fuel, which is almost similar in geometry to that used in RHF (see Section B.1.3) reactor but contains more ^{235}U fuel. The GE-2000 is illustrated in Figure B-13.

TN-8 (TN-9)

The Transnuclear TN-8 is a lead, steel, and resin shielded right cylinder, stainless steel transportation cask. The cask is 561.3 cm (221 in) long, 170 cm (67 in) in diameter, and weighs 16.3 metric tons (36,000 lb) empty. The TN-8 is permitted to ship three pressurized water reactor irradiated fuel assemblies. The TN-9 transportation cask is nearly identical to the TN-8, however, it is permitted to ship seven BWR irradiated fuel assemblies. These casks are classified as overweight truck casks in highway transport.

NLI-1/2

The Nuclear Assurance Corporation NL1-1/2 is a depleted uranium, water, and lead shielded transportation cask, encased in stainless steel. Shielding is provided by depleted uranium, lead, and a borated water-ethylene glycol mixture. The cask measures 495.3 cm (195 in) long, 120 cm (47.125 in) in diameter, and weighs 21 metric tons (49,250 lb) empty. It is permitted to ship either 1 pressurized water reactor or 2 boiling water reactor irradiated fuel assemblies. The NLI-1/2 is a legal weight truck cask that has been used at the Savannah River Site for the receipt of Taiwanese foreign research reactor spent nuclear fuel as recently as 1990. The NLI-1/2 is illustrated in Figure B-14.

NAC-LWT

The Nuclear Assurance Corporation NAC-LWT is a truck transported, steel-encased, lead shielded transportation cask. Radioactive shielding is provided by stainless steel and lead. The cask measures 508 cm (200 in) long, 165.1 cm (65 in) in diameter, and weighs 22.4 metric tons (51,200 lb) full. The cask is permitted to ship either one pressurized water reactor or two boiling water reactor irradiated fuel assemblies. This cask is also certified for the transport of Taiwanese foreign research reactor spent nuclear fuel. The NAC-LWT is nearly identical to the NLI-1/2.

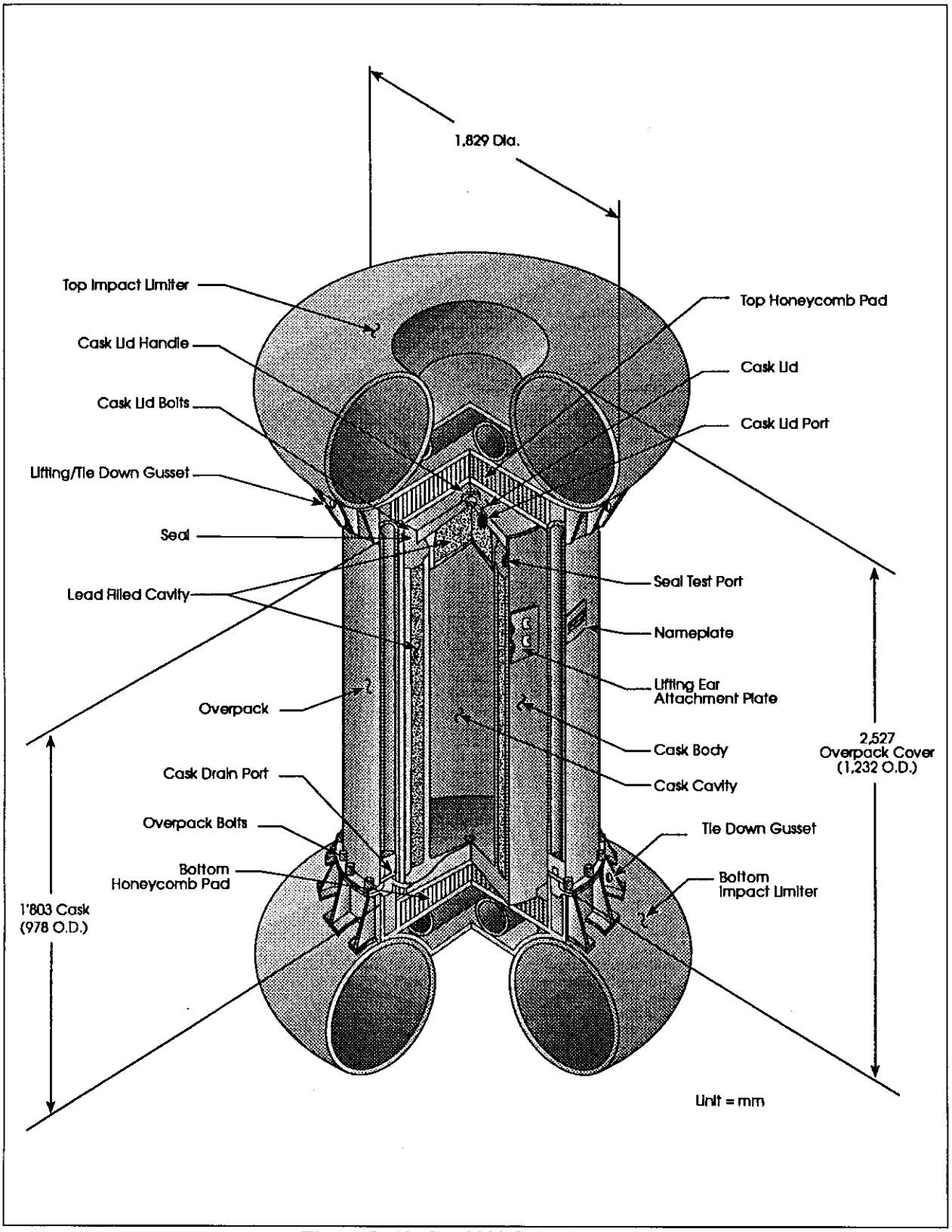


Figure B-13 GE-2000 Shipping Cask

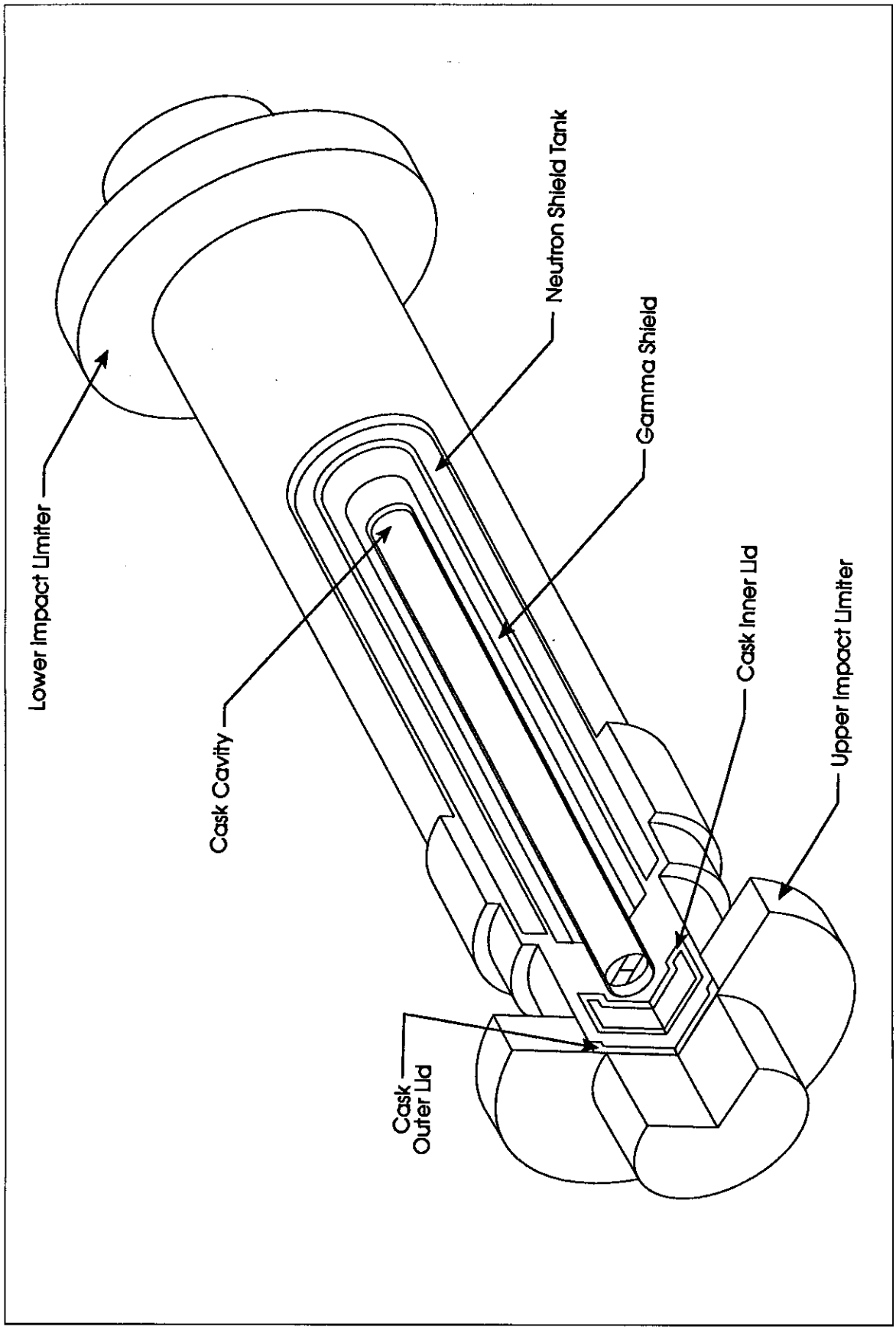


Figure B-14 NLI-1/2 Shipping Cask

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Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix C
Marine Transport and Associated Environmental Impacts

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Washington, DC 20585

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Appendix C
Marine Transport and Associated Environmental Impacts

C.1 Introduction

Shipment of any material via ocean transport entails risks to both the ship's crew and the environment. The risks result directly from transportation-related accidents and, in the case of radioactive or other hazardous materials, also include exposure to the effects of the material itself.

This appendix provides a description of the approach used to assess the risks associated with the transport of foreign research reactor spent nuclear fuel from a foreign port to a U.S. port(s) of entry. This appendix also includes a discussion of the shipping configuration of the foreign research reactor spent nuclear fuel, the possible types of vessels that could be used to make the shipments, the risk assessment methodology (addressing both incident-free and accident risks), and the results of the analyses. Analysis of activities in the port(s) is described in Appendix D.

The incident-free and accident risk assessment results are presented in terms of the per shipment risk and total risks associated with the basic implementation of Management Alternative 1 and other implementation alternatives. In addition, annual risks from incident-free transport are developed.

C.2 Scope

This appendix addresses the modes of marine transportation and the nonradiological and radiological risks associated with marine transportation.

Transportation Modes: Marine transport of foreign research reactor spent nuclear fuel could occur via a combination of four types of vessels: container ships, roll-on/roll-off vessels, general cargo (breakbulk) vessels, or purpose-built vessels. In the incident-free analysis, it was assumed that all shipments would be made on breakbulk vessels. Breakbulk cargo vessel speeds are typical of the four types of cargo vessels considered, which means that the breakbulk vessel time enroute, (i.e., from port of origin to port of entry) is representative of the four vessel types. The ship speed selected for the analysis, 15 knots or 17.3 mph, is at the lower end of the range of speeds for commercial cargo vessels. This, in turn, maximizes the radiation dose received by the ship's crew, which bounds the incident-free risk. No vessel type assumption is necessary for the analysis of the impacts associated with the accident conditions, since these impacts are essentially independent of the type of ship.

Nonradiological Impacts: These risks were assessed as resulting in a negligible impact on the health of the public and workers. The limited number of shipments (less than a thousand individual spent nuclear fuel containers) would not result in a significant change in the number of ocean crossings by transport vessels. Regardless of the ship selection – general cargo, container, roll-on/roll-off, or purpose-built vessel – a negligible increase in the exposure of the public to exhaust emissions or transportation-related accidents would occur.

More than 56,000 port calls of ships engaged in foreign trade are made at U.S. ports each year (DOC, 1994). The basic implementation of Management Alternative 1 would result in the addition of less than 50 round trip voyages by vessel per year; the actual number of voyages that might occur would be dependent on the manner in which the policy, if adopted, was implemented. On average, less than

60 foreign research reactor spent nuclear fuel casks would be required to be shipped each year to fulfill the basic implementation shipping needs. These shipments could be made on regularly scheduled commercial cargo vessels. Alternatively, these shipments could be made in a chartered vessel, where the transportation casks would be the only cargo onboard the vessel.

If commercial cargo vessels were used, the shipment of foreign research reactor spent nuclear fuel transportation casks would not result in additional voyages specifically for the transport of the foreign research reactor spent nuclear fuel. The approximately 60 transportation casks per year would be part of the general cargo carried by the ships. As discussed in Section C.3.1.2, container vessels typically have a capacity in the range of 800 to 1,000 containers, while some carry many more. General cargo vessels tend to be somewhat smaller, but still have capacities equivalent to several hundred containers. Each foreign research reactor spent nuclear fuel transportation cask is assumed to be shipped within a container. Therefore, for the tens of thousands of vessels received at U.S. ports each year, each carrying hundreds of containers, or their equivalent, the basic implementation alternative would add approximately 60 containers per year. This is equivalent to much less than the capacity of one cargo vessel.

If chartered vessels were to be used for the shipment of the foreign research reactor spent nuclear fuel, the number of shipments required per year would depend on the number of transportation casks loaded into each vessel. Many factors would affect this number, such as the size of the ship, the availability of the ship, originating point for the shipments, and the readiness of foreign research reactor operators to ship the spent nuclear fuel. Estimates of the number of transportation casks that could be shipped on a single vessel are in the range of two to eight. This range results in estimates of between 30 and less than 10 shipments per year. Thirty shipments involve less than 0.001 of the total number of port calls by vessels engaged in foreign trade received at U.S. ports each year.

A combination of the two means of shipping the foreign research reactor spent nuclear fuel, commercial cargo and charter vessels, would result in somewhat fewer additional voyages by cargo vessels than the use of dedicated vessels alone. The use of five chartered voyages (carrying eight casks each) in combination with commercial cargo vessels could result in more than half of the foreign research reactor spent nuclear fuel casks being transported on chartered vessels. These five chartered voyages would represent less than 0.0001 of the number of vessels received at U.S. ports.

Regardless of the types of ships selected, there would be negligible impact on the marine environment including endangered species or habitats because of the negligible increase in ship traffic.

Radiological Impacts: The risks resulting from the radioactive nature of the shipments are addressed for both incident-free and accident transportation conditions. The radiological risks associated with the incident-free shipping conditions would be the potential exposure of the members of the crew to external radiation in the vicinity of the packaged fuel. No other public exposure is considered, due to the relative isolation of the material from the general public during all phases of the marine transport of the spent nuclear fuel. The potential exposure to radiation due to accidents is assessed for the marine environment in the event of the loss of a cask at sea and the consequent release of the cask's inventory into the marine environment. Only the marine exposure pathway is considered in detail, as the relative isolation from land and populated areas of the material during almost all of the voyage would minimize direct exposure through air pathways. Additionally, since the damaged cask is assumed to be lost at sea (and if not lost at sea, any airborne release would be deposited on the ocean surface), the marine pathway is likely to have more severe consequences.

All radiologically-related impacts on humans are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent (EDE), which is the sum of the EDE from the external radiation exposure and the 50-year committed EDE from internal radiation exposure. The EDE is the sum of the tissue and organ-weighted dose equivalents for all irradiated tissues and organs. The committed EDE considers the initial exposure and the effects of radioactive decay and elimination of the radionuclide through ordinary metabolic processes over the 50-year period. Radiation doses are presented in units of person-rem for collective population and rem or mrem (equal to 0.001 rem) for individuals. The impacts are further expressed as health risks, primarily in terms of latent cancer fatalities (LCFs). The health risk conversion factors were derived from International Commission of Radiological Protection Publication 60 (ICRP, 1991). See Chapter 4 of this Environmental Impact Statement (EIS) for a more detailed discussion of radiation dose and risk.

C.3 Selection of Modes and Routes

C.3.1 Modes of Transportation

This section describes the possible shipping configurations of the cask and the types of vessels that could be used for ocean transport. In general, the shipping configuration of the cask conforms to the type of vessel to be used in ocean transport. The purpose of this section is to assist in understanding the specific operations or handling issues that arise in the various cask shipping configurations or in the use of specific vessel types.

Currently, the preferred method of commercial transport aboard ocean vessels is to mount casks in metal containers, sometimes called "International Standards Organization containers." Typically, containerized casks are transported on smaller general cargo vessels rather than on large vessels specifically designed for container transport.

As described in Section C.3.1.2, non-containerized transport is feasible, but is not generally used. An exception is the shipment of casks in purpose-built ships, which are specifically designed to accommodate radioactive material casks. Purpose-built ships for cask transport are described in Section C.3.1.2.

C.3.1.1 Cask Transport Configurations

This section describes the three configurations of casks for transport. The casks may be containerized, mounted on a wheeled trailer, or free-standing. Typically, containerized casks are mounted in a 6.1-m (20-ft) container, since casks rarely exceed 5.8 m (19 ft) in length. Wheeled cask trailers are usually dedicated trailers that have unique hardware used to secure the cask to the trailer frame. Free-standing casks are mounted on a skid, pallet, or cradle to facilitate handling the cask in intermodal transfer and in stowage.

Containerized Cask Configuration: Casks may be transported within International Standards Organization containers to take advantage of standardized port container lifting gear and vessel and transporter container tiedowns. The International Standards Organization container is a steel box that conforms to a set of standard dimensions, and has standard tiedown and lift points. The standard height and width is 2.4 m (8 ft). There are two standard lengths, 6.1 m and 12.2 m (20 ft and 40 ft). The four corners of the container are structural posts that have lifting points at the top and tiedown points at the bottom. These containers are commonly used to move all manner of goods transported by vessel and, because of the standardized dimensions and lifting points, can be rapidly transferred between the dock and the vessel.

Lifting, stowage, and transfer of containers is described in Appendix D.

Casks are mounted within the container using specially designed supports in the container floor. These supports mate with the tiedown structure of the cask to secure it to the container.

Figure C-1 shows a spent nuclear fuel cask being loaded into an International Standards Organization container. Containers may be either completely enclosed using a removable top, as shown in Figure C-1, or have open sides and top. Usually, an enclosed container is used with a cask that is certified for transport with a "personnel barrier." As its name implies, the personnel barrier is a structure that surrounds the cask in transport, to preclude inadvertent personnel contact with the cask surface. The barrier is a required feature if the cask surface can exceed about 52°C (125°F) in non-exclusive-use transport. The cask may become warm in transport due to the decay heat of the spent nuclear fuel within the cask. Usually, the barrier is constructed of expanded metal screen or other lightweight material. Casks that do not require a barrier may be mounted in open containers. In either case, the floor of the container is specially designed to support the weight of the cask, and to incorporate the tiedown fixtures of the cask. The tiedowns may be unique, as those shown in Figure C-1, or they may be bolts that secure the skid, pallet, or cradle to the floor of the container.

Since the introduction of International Standards Organization containers, shipment of spent nuclear fuel in casks mounted in containers has become the preferred configuration. Use of containers provides an improvement in the ease of securing the cask to the vessel. It also permits the use of standard container handling and transport equipment that is used at many ports.

Roll-On/Roll-Off Cask Configuration: Casks can be transported by vessel on a wheeled trailer that allows the cask to be rolled onto the vessel, and at the destination, rolled off. The cask (on its own unique, dedicated trailer) is moved on and off the vessel using a standard truck tractor or wheeled tug across a ramp extending between the vessel and the dock.

A few shipments have been made to the United States from Europe using casks mounted on their own dedicated trailers. However, current Federal regulations (49 CFR 176.76(b)) restrict trailered hazardous cargo (such as spent nuclear fuel) to transport on a trailership (roll-on/roll-off), trainship, ferry vessel, or car float. This regulation would preclude shipment of trailered casks containing spent nuclear fuel on general cargo, or other vessels. It has been assumed that the foreign research reactor spent nuclear fuel will be shipped as containerized cargo, not mounted on trailers. Use of containers will not limit the type of vessel that can be selected for transport.

Free-Standing Cask Configurations: Casks could be transported as a free-standing package. In this configuration, the cask would be mounted on a skid, pallet, or cradle to facilitate both lifting and tiedown. A pallet is usually required because casks have unique tiedowns and lift points that may not be readily accommodated by more common rigging and stowage bindings. The pallet is usually designed to provide a means of attaching the cask to the transport trailer or railcar. The cask is usually either attached to the pallet by bolting at the cask tiedown fixtures, or by the use of specially designed turn buckle cables.

Free-standing casks have previously been transported on general cargo vessels that carry cargo as "breakbulk." Breakbulk cargo is any cargo that is handled individually and may be containerized or otherwise unitized.

Shipments of free-standing casks are no longer routinely made, primarily because the securing of the cask to the vessel is considered to be somewhat less certain than that obtained with International Standards Organization containers, and because of the risk of damage to the cask in handling and stowage.

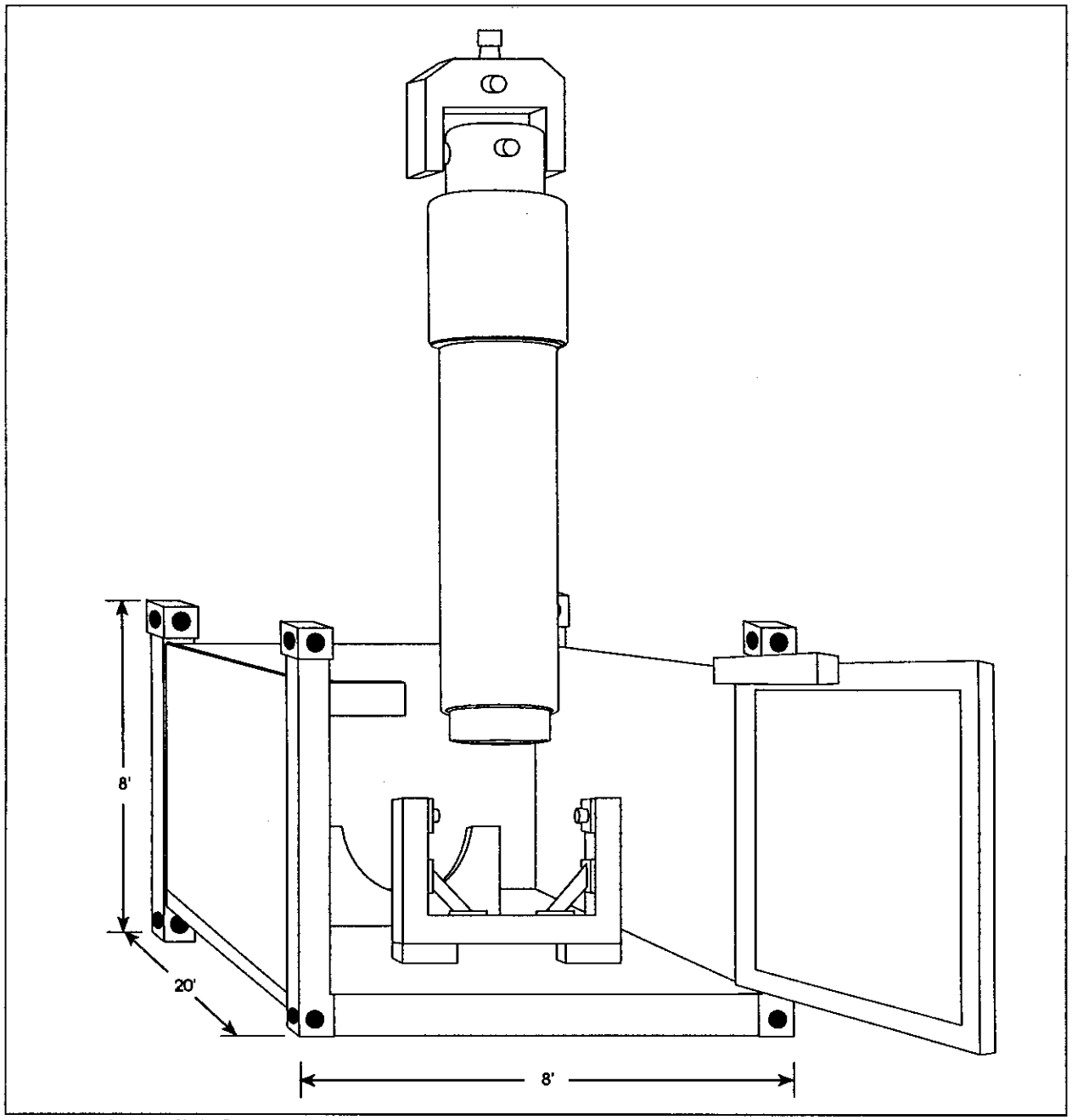

Figure C-1 Spent Nuclear Fuel Cask Being Loaded into an International Standards Organization Container

Recently, several purpose-built ships have been placed in service that transport casks in a free-standing (non-containerized) configuration. Purpose-built vessels are described in Section C.3.1.2. These dedicated vessels incorporate holds containing structural tiedowns designed to mate with the cask, and which provide additional shielding from radiation. The purpose-built vessels are operated by crews both trained in radiological safety and with a radiological control program in place.

C.3.1.2 Vessel Types, Cask Handling Requirements, and Methods of Service

This section describes the four principal types of vessels that could be used for the transport of casks. The vessel types include container, roll-on/roll-off, general cargo (also called breakbulk), and purpose-built vessels.

Each of these types of vessel have somewhat different handling requirements for the cargo they carry. Cask handling and equipment requirements are also described.

Individual shipments could be made by scheduled commercial vessel, or by charter vessel. Vessels on scheduled routes generally call on the more important ports. Scheduled vessels also typically call at intermediate ports between a given origin and destination.

Because of the general public aversion to nuclear materials, there has been a marked decrease in the number of steamship lines that will accept spent nuclear fuel cargoes in scheduled service. Also, many foreign ports and some U.S. ports do not currently permit docking or handling of spent nuclear fuel shipments, either en route or as a destination. This has led to an increased reliance on spent nuclear fuel ocean transport by chartered vessel. Vessels for charter are available from any number of steamship lines. Generally, smaller general cargo (breakbulk) vessels are used for charter shipments.

Container Vessels: Container vessels are typically large ships that are specifically intended for the transport of International Standards Organization containers (Figure C-2). Modern container ships can transport up to about 5,000 containers, although a more typical capacity is in the range of 800 to 1,000. A principal advantage of container vessels, because of standardization of containers, is that the vessel can be rapidly loaded or off loaded at those ports equipped with container gantry cranes. Containers can be removed from (or placed on) the vessel at an average rate of about 45 containers per hour. At well equipped container vessel ports, two cranes are used to move containers. Smaller container vessels may be equipped with an onboard crane allowing calls at ports that are less well equipped.

Because of cost, the only container ships generally used to transport spent nuclear fuel are in scheduled service. Smaller general cargo vessels are more suitable to chartered service, and these vessels accommodate containers.

Roll-On/Roll-Off Vessels: Roll-on/roll-off vessels are vehicle carriers (Figure C-3) used for the ocean transport of cars and trucks. The vessels are loaded and unloaded using a ramp between the vessel and dock. Ordinarily, the vessel carries its own ramp, which is deployed by an on-board crane, hydraulic cylinders, or chain drives. The ramp may extend from the stern of the vessel or from a hatch in the side hull of the vessel. At docks intended for roll-on/roll-off service, additional ramps may be deployed from the dock to expedite loading or unloading. For ocean transport, the trailers are lashed to the deck(s) of the vessel using ratchet or turnbuckle type bindings to fixed securement points in the deck. It is likely that a roll-on/roll-off capable vessel could be leased, should a roll-on/roll-off capability be required.

General Cargo (Breakbulk) Vessels: General cargo vessels (Figure C-4) are small-to-medium sized ships (compared to container vessels) that typically call on less well developed or equipped ports. They have on-board jib or boom type cranes that can be used to load or unload the ship. As the name implies, these vessels are intended to accommodate a wide variety of cargoes. Since the advent of the widespread use of containers, most of these ships are equipped with International Standards Organization lock fixtures to secure containers to the ship deck(s) and to each other. If necessary, containers can be lifted on and off these ships by using four-legged slings between the corners of the container and the hook of the crane. Because of the versatility of these vessels, casks configured for containerized or free-standing transport can be accommodated.

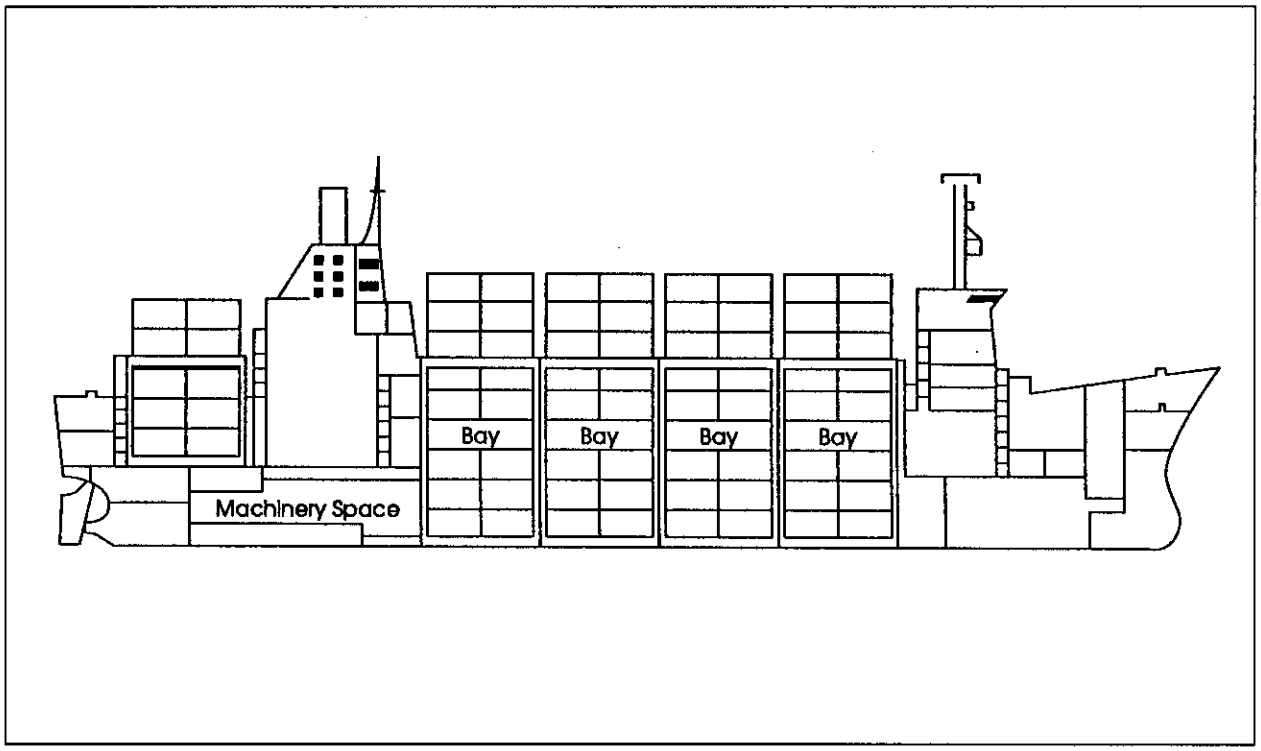

Figure C-2 Container Vessel

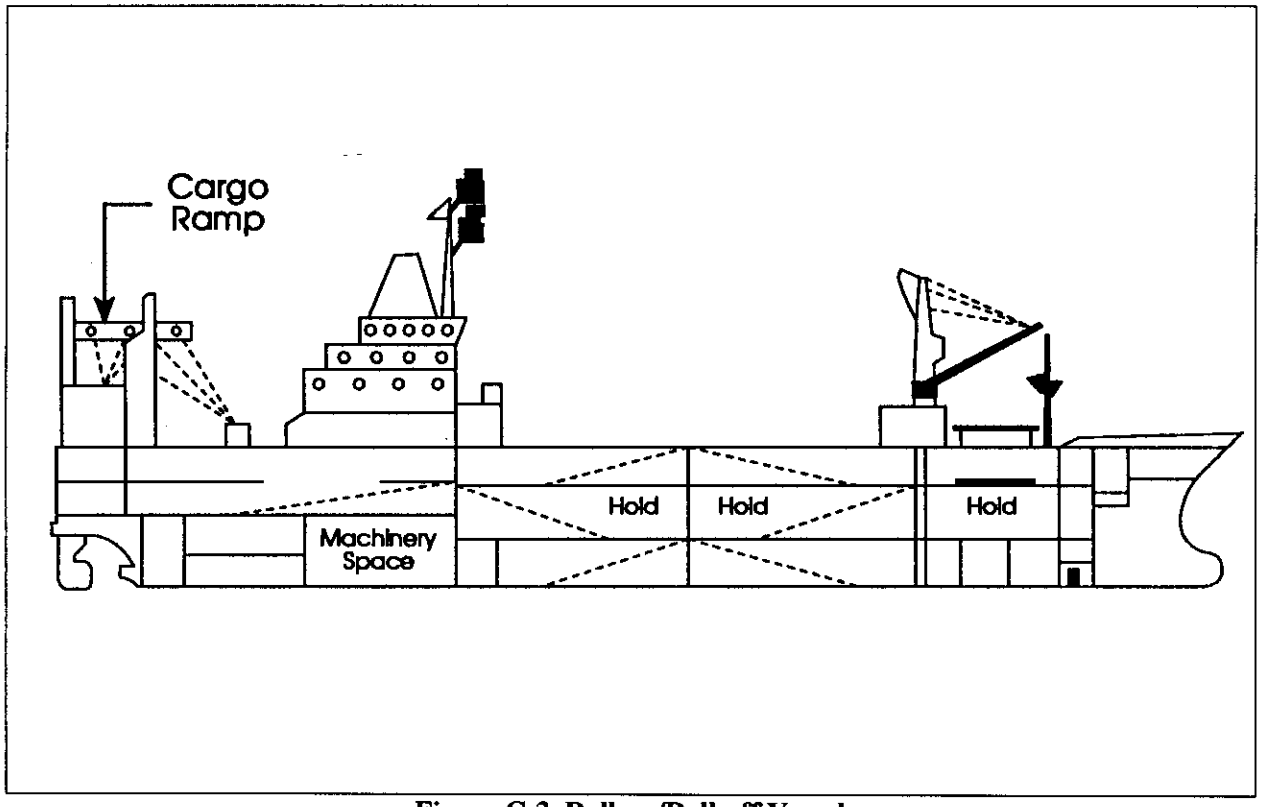

Figure C-3 Roll-on/Roll-off Vessel

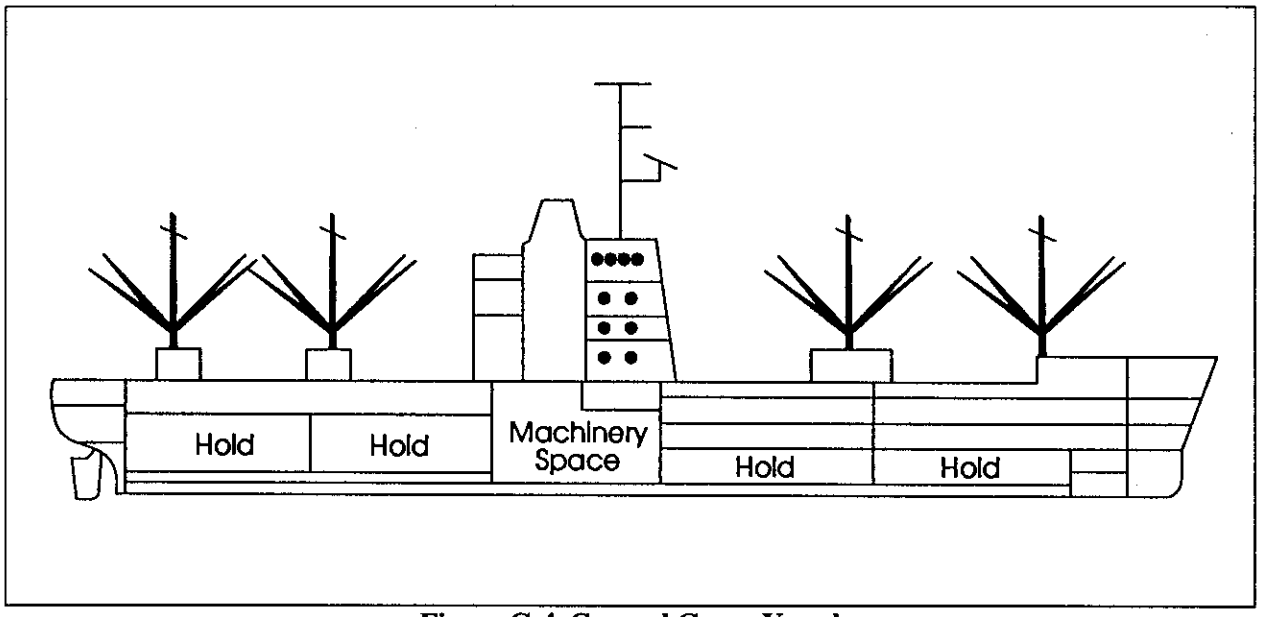

Figure C-4 General Cargo Vessel

Free-standing casks would be palletized for transport on a general cargo vessel. For stowage, the pallet would be lashed to the vessel hold or deck using conventional chains or binders. Pallets do not have standard tiedown fixtures, so there is wide variability in the specific tiedown requirements for each pallet design. Also, there is variability in provisions for lifting the pallet. The standard tiedown configuration of containers eliminates much of this variability. Consequently, containerized cask handling has resulted in an increase in the use of this configuration for the shipment of casks, and there has been a significant reduction in the number of casks shipped in the free-standing configuration.

General cargo ships have been routinely available for chartered shipment of containerized casks containing spent nuclear fuel from any number of U.S. or foreign ship lines. Because there are a comparatively small number of casks that are available for use, chartered small general cargo vessels are an option to scheduled service.

Purpose-Built Vessels: Purpose-built vessels, as used here, are those vessels specifically designed to transport spent nuclear fuel casks (Figure C-5). These vessels are not used for the transport of any other cargo and they operate as dedicated vessels. Casks are loaded directly into the holds of the vessel because the cargo compartments contain the hardware needed to mate with the tiedown fixtures of the cask. If the vessel has no crane, dockside cranes are used for loading and unloading. The cargo compartments are typically intended to handle a specific cask, and other casks cannot be used without modification to the tiedown mechanisms. For the relatively efficient transport of spent nuclear fuel, the casks normally used are very large. They are intended for the transport of power reactor spent nuclear fuel, and have a loaded weight on the order of 90 to 115 metric tons (99 to 126.5 tons). Commercial docks are not normally used, but most could be without significant problems.

The vessels have double bottoms and hulls, watertight compartments, and collision damage resisting structures within the hull. The vessel crew is trained in the handling of the cargo and in emergency response. These vessels also incorporate security features and satellite tracking systems.

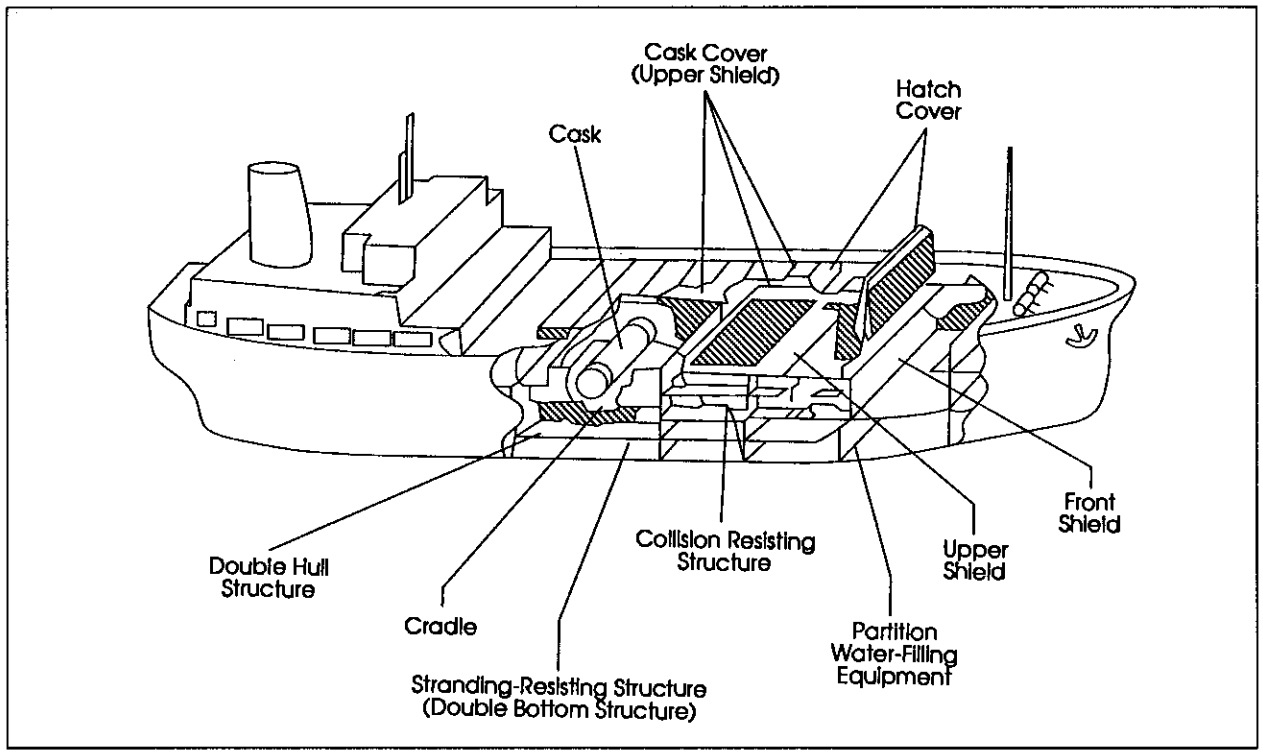

Figure C-5 Purpose-Built Ship

At present, purpose-built vessels are operated by Nuclear Transport Services of Japan, by the Swedish Nuclear Fuel and Waste Management Company, and by British Nuclear Fuels, Limited. They are used to move spent nuclear fuel from operating nuclear power plants to spent nuclear fuel reprocessing facilities operated by Cogema and British Nuclear Fuels, Limited; or, in the case of Sweden, to the repository in Forsmark. There are no U.S.-owned purpose-built vessels for spent nuclear fuel transport.

C.3.2 Identification of Routes

The foreign research reactor spent nuclear fuel that might be transported by sea under the proposed action could originate from 40 different countries. For calculation of shipping distances to the United States, shipping routes were selected to represent the transport of the fuel from a convenient port in the country of origin (for land-locked nations a port near the country of origin was selected) to both an East Coast and a West Coast U.S. port. Norfolk, VA, and Los Angeles, CA, were selected as the two port cities for use in determining a representative distance from the country of origin to the East and West Coasts of the United States. These distances were then combined to generate an average shipping distance between the country of origin and the United States. By using a city on both coasts of the United States to determine an average distance between ports, the analysis considers the possibility that shipments of foreign research reactor spent nuclear fuel would not necessarily be made to the closest U.S. port and, in fact, may be shipped to the "opposite" coast.

Table C-1 is a compilation of the distances for shipments from each of the countries that may participate in this program (except Canada) to the ports on both U.S. coasts. All route distances were obtained by using normal shipping lanes (DMA, 1991). For some of the shipments that might be received at the "opposite" U.S. coast port, the use of the Panama Canal was assumed. Other than the shipping requirements

Table C-1 Voyage Data

Country of Origin	Distance East (nautical miles)	Distance West (nautical miles)	Average Distance (nautical miles)	Voyage Duration (days)	Number of Casks	Number of Voyages
Argentina	5,824	7,265	6,545	21.2	9	5
Australia	12,728	6,511	9,620	29.7	9	5
Austria	5,026	8,955	6,991	22.9	8	4
Bangladesh	10,017	9,384	9,701	31.0	3	2
Belgium	3,582	7,782	5,682	19.3	59	30
Brazil	4,723	8,109	6,416	20.8	8	4
Chile	4,438	4,808	4,623	16.3	2	1
Colombia	2,174	3,265	2,720	11.1	1	1
Denmark	3,990	8,190	6,090	20.4	22	11
France	3,181	7,287	5,234	18.0	149	75
Finland	4,453	8,653	6,553	21.7	6	3
Germany	3,919	8,119	6,019	20.2	61	31
Greece	4,685	8,614	6,650	22.0	8	4
Indonesia	10,566	8,392	9,479	30.3	14	7
Iran	12,013	11,783	11,898	36.6	1	1
Israel	5,366	9,295	7,331	23.9	6	3
Italy	4,336	8,265	6,301	21.0	18	9
Jamaica	1,279	3,507	2,393	10.2	1	1
Japan	9,504	4,839	7,172	23.4	110	55
Korea (South)	10,480	5,229	7,855	25.3	18	9
Malaysia	10,417	7,867	9,142	28.9	3	2
Mexico	1,772	1,501	1,637	7.6	6	3
The Netherlands	3,582	7,782	5,682	19.3	49	25
Pakistan	11,460	10,749	11,105	34.4	3	2
Peru	3,172	3,655	3,414	13.0	1	1
Philippines	11,169	6,530	8,850	28.1	6	3
Portugal	3,129	7,550	5,340	18.3	3	2
Romania	5,353	9,282	7,318	23.8	48	24
Slovenia	4,172	8,372	6,272	20.9	13	7
South Africa	6,790	9,385	8,088	26.0	2	1
Spain	3,303	7,564	5,434	18.6	1	1
Sweden	4,331	8,531	6,431	21.4	37	19
Switzerland	5,026	8,955	6,991	22.9	5	3
Taiwan	11,732	7,093	9,413	29.7	9	5
Thailand	13,169	7,775	10,472	33.1	5	3
Turkey	5,002	8,931	6,967	22.9	4	2
United Kingdom	3,101	7,301	5,201	18.5	4	2
Uruguay	5,710	7,171	6,441	20.9	1	1
Venezuela	1,687	3,757	2,722	11.1	4	2
Zaire	5,864	8,583	7,224	23.6	4	2
Totals					**721**	**371**
Average				**21.3**		

Distance East - Distance in nautical miles from country of origin to Norfolk, Virginia
Distance West - Distance in nautical miles from country of origin to Los Angeles, California
Average Distance - Distance in nautical miles from country of origin to both U.S. ports
Voyage Duration - Average distance divided by 15 knots per hour plus additional days for busy way points (i.e., Panama Canal) and three days for additional stops
Number of Casks - Total casks from country of origin
Number of Voyages - Number of trips required assuming two casks per voyage

applicable to the entire journey, there are no known restrictions for spent nuclear fuel passing through either the Suez or Panama Canals. Figure C-6 provides a representation of the shipping routes selected for these shipments, although other normal shipping routes may be used.

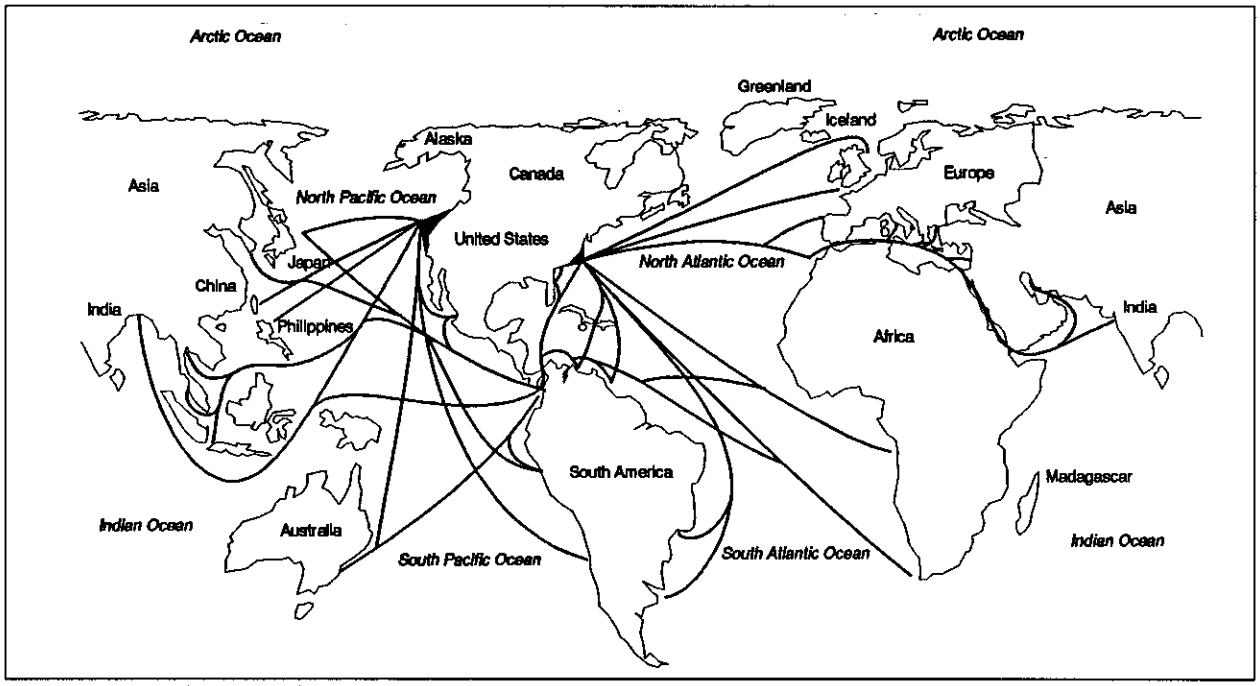

Figure C-6 Representative Shipping Routes for Foreign Research Reactor Spent Nuclear Fuel

C.4 Incident-Free Impacts: Methods and Results

C.4.1 Incident-Free Risk Assessment Methodology

External radiation from an intact transportation cask must be below specified limits that control the exposure of the handling personnel and general public. The U.S. limits are set forth in 49 CFR 173. The limit of interest established therein is 10 mrem per hour at any point 2 m (6.6 ft) from the vertical planes projected by the outer lateral surfaces of the transport vehicle. This limit is associated with an "exclusive-use" shipment, which is a shipment in which no other cargo is loaded in the container used for the foreign research reactor spent nuclear fuel transportation cask and the container is not off-loaded and restowed in transit, except as directed by the shipper. This does not mean that the vessel is used exclusively for foreign research reactor spent nuclear fuel. All shipments within this program are expected to fall within this category.

In general, much of the foreign research reactor spent nuclear fuel to be received will have been out of the reactor for a significant amount of time prior to shipment, resulting in external dose rates much less than the regulatory limit. Past shipments of research reactor fuel have not approached the 49 CFR 173 limit (many, in fact, had dose rates of much less than 1 mrem per hour at 1 m). Due to the scope of this program and the possibility that some of the spent nuclear fuel may be shipped with shorter "cooldown" times than previous shipments, an analysis using typical historical dose rates may not be fully representative of all shipments. Therefore, the analysis has been performed assuming a dose rate (1) at the above-cited regulatory limit, and (2) derived from measurements taken during earlier foreign research

reactor spent nuclear fuel shipments. Appendix F, Section F.5, provides a discussion of the development of the exposure dose rate versus distance relationship for a transportation cask having a dose rate at the selected exclusive-use regulatory limit.

The application of the 10 mrem per hour at 2 m (6.6 ft) exclusive-use regulatory dose limit and the "historical" dose rates provide two significant estimates for the assumed external dose rates. The exposure derived from the use of the selected regulatory limit for the dose rate is an estimate of the maximum exposure that could result from the shipments. The estimate derived from the "historical" data is closer to an expected value for the incident-free impacts. Therefore, the results of these two analyses provide an estimate of the range of incident-free impacts from the shipment of the foreign research reactor spent nuclear fuel.

The primary impact of incident-free marine transport of spent nuclear fuel is on the crews of the ships used to carry the casks. Members of the general public and marine life would not receive any measurable dose from the spent nuclear fuel during marine transport. In addition to the protection provided by the transportation casks, further protection for the general public and marine life is provided by the location of the cask in the ship (that is, the distance from the cask to the outer surface of the ship) and the ship's structure. From the outside of the ship, the foreign research reactor spent nuclear fuel shipments would be indistinguishable from any other commercial shipment. Under incident-free conditions of transport, public exposure would be limited to the ship's crew exposure, and the ship's crew exposure is limited to crew members exposed during the loading and offloading of the casks and to crew members who, on a daily basis, inspect cargo (to ensure secure stowage) and the vessel.

The type of vessel assumed to be used for transport of the spent nuclear fuel is a U.S. crewed breakbulk vessel with services not obtained on a charter basis. Breakbulk vessels typically have a number of holds, decks within each hold for carrying cargo, and their own cargo handling equipment that could be used for loading spent nuclear fuel casks. The flexibility of these vessels may be required to pick up spent nuclear fuel at some countries, since container vessel facilities may not be available.

The spent nuclear fuel cask is assumed to be in a container for ease of handling. With this assumption, the vessel with the longest cargo handling times for containerized cargo would be a breakbulk vessel. Differences in cask handling time is the key factor contributing to the differences between the incident-free impact of shipments of foreign research reactor spent nuclear fuel on different types of vessels. (See Appendix D, Section 4, for details of handling times). Therefore, the selection of this type of vessel results in a conservative estimation of the dose to the crew during transit and will bound the estimate of crew dose for any ship type selected for transport of the spent nuclear fuel.

Two different sets of assumptions have been made to assess the incident-free impacts of the shipment of foreign research reactor spent nuclear fuel. The first set of assumptions addresses the use of regularly scheduled commercial cargo vessels for the shipments. When using regularly scheduled commercial vessels, the assumption is made that two casks per vessel will be carried on each freighter, except in cases where the number of casks from a country of origin is an odd number, which would result in one shipment of only one cask. While it is likely that in some cases more than two casks per shipment could be coordinated at the same time, it is expected that the assumption of two casks per vessel should bound the incident-free analysis. The analysis assumes that both spent nuclear fuel casks are loaded into the same hold, resulting in a dose to the crew from the first cask loaded while the second cask is being loaded. This results in the crew being exposed from two sources at the same time for loading or unloading one of the two casks. Should more than two spent nuclear fuel casks be shipped on the same vessel, it has been assumed that the cargo loading would be limited to two spent nuclear fuel casks per hold. The crew would not receive any additional dose from the third, fourth, etc., cask while engaged in activities in the hold with

the first two spent nuclear fuel casks. The radiological exposure to the crew for a shipment of many casks would be equivalent to the radiological exposure due to multiple shipments of fewer casks. For example, if four casks are shipped on a single vessel, the crew dose for that single shipment would be equal to the crew dose from two shipments of two casks each.

The second set of assumptions addresses the use of a chartered cargo vessel for the shipment of the foreign research reactor spent nuclear fuel. Use of a chartered vessel (either a chartered commercial freighter or a purpose-built vessel) could result in the shipment of more than two casks per voyage. Economic considerations would suggest that a larger number of casks be shipped per voyage. For this analysis it has been assumed that eight transportation casks would be shipped on a chartered vessel. Consistent with the assumption made for the regularly scheduled commercial vessel, it has been assumed that the transportation casks would be loaded two to a hold. Again, this results in doses to the crew from the first cask loaded during activities associated with the loading of the second cask in the hold.

During loading operations, both on the regularly scheduled commercial and chartered vessels, it is assumed that five members of the ship's crew (Chief Mate, Mate on Watch, Bosun, and two Seamen) will be present during loading and securing of the spent nuclear fuel casks. While longshoremen will most likely be used for the cargo handling activity, ship's crew will be present, and therefore the crew dose resulting from this activity has been included in the analysis. Table C-2 shows the crew member distances from the spent nuclear fuel shipping cask and the duration of the crew members' exposure for each crew member during the time leading up to the stowage of the cask prior to setting sail for the ocean voyage. The distances and times are based on vessel loading activities for a two-cask-per-hold shipment. The total dose (based on the selected exclusive-use regulatory limit external dose rate of 10 mrem per hour at 2 m or 6.6 ft from the surface of the container) for each individual is calculated for each shipment. Since two casks are assumed to be shipped in each hold, when quantities allow, the condition exists for loading and securing of a cask to take place in the vicinity of another cask. The additional dose received by working around a cask already in the hold are accounted for in Table C-2. This was accomplished by increasing the exposure rate by a factor of 1.5 for the activities associated with securing the second cask. As listed, the estimated exposure represents the crew exposure for the regularly scheduled commercial vessel, which has been assumed to be limited to a total of two transportation casks. The exposure for each listed crew member in a chartered vessel would be four times these values, since the eight casks are assumed to be loaded into four holds.

Table C-2 Ship Crew Exposure During Loading of a Hold Containing Two Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Regulatory Dose Limits)

	Tasks						
	Rail to Hold			*Secure Cask*			*Total Dose per Loading[c] (person-rem)*
Crew Member	*Distance[a] (m)*	*Exposure Rate[b] (mrem/hr)*	*Time (min)*	*Distance[a] (m)*	*Exposure Rate[b] (mrem/hr)*	*Time (min)*	
Chief Mate	20	1	5	5.5	3.5	60	0.009
Mate on Watch	20	1	5	8	2.1	60	0.005
Bosun	20	1	5	5.5	3.5	60	0.009
Seaman (2)	20	1	5	5.5	3.5	60	0.018

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Exposure rate is calculated based on 10 mrem/hr at 2 m (6.6 ft) from the shipping container surface.*

[c] *Includes the exposure from the first loaded casks for activities associated with securing the second cask. The exposure rate for securing the second cask is 1.5 times the listed rate.*

While at sea, the crew dose is limited to those individuals who enter the ship's hold during transit. At all other times, the crew is shielded from the spent nuclear fuel cask by the decking and other structures of the vessel. The number of entries and inspections is a function of the voyage distance from the port of loading to the port of offloading (the U.S. port of entry for the foreign research reactor spent nuclear fuel). Since the port of offloading is unknown at this time, voyage distances were determined for each country of origin to a West and East Coast port of the United States. The average of these two distances was then calculated. Table C-1 shows the countries of origin, the number of casks, the distances to the East and West Coast ports, the average voyage distance, the days of travel, and the estimated number of casks and shipments for the basic implementation of Management Alternative 1. Because the actual shipping schedule is unknown, the average annual number of shipments was estimated. The length of a voyage was determined by assuming that the vessel would have an average speed of 15 knots for the entire duration of the voyage. In addition, intermediate port stops would be made, and additional travel time was added to account for portions of the voyage during which the vessel would not be expected to have a speed of 15 knots, (i.e., passage through busy locations, such as the Panama Canal).

Once a day while at sea or in port, the Chief Mate, the Bosun, and an Engineer are assumed to enter each cargo hold to inspect the bilges and verify the lashings for the containers. Table C-3 describes the times required for these activities, the distances from the casks during the activity, and doses received from the casks during the activity (based on the selected exclusive-use limit of external dose rate of 10 mrem per hour at 2 m or 6.6 ft from the surface of the container) for each of these individuals. The total dose due to inspection activities is a function of the voyage duration and the number of holds that contain foreign research reactor spent nuclear fuel casks.

Table C-3 Ship Crew Exposure Per Hold During At-Sea Inspections (Based on Regulatory Dose Limits)

Crew Member	*Distance[a] (m)*	*Dose Rate[b] (mrem/hr)*	*Time (min)*	*Dose per Daily Inspection[c] (mrem)*
Chief Mate	5.5	7.0	20	2.3
Bosun	5.5	7.0	20	2.3
Engineer	5.5	7.0	20	2.3

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Dose rate includes the sum of the effect of two casks in hold.*

[c] *For a ship carrying two casks on a voyage duration of 21 days, the daily inspection dose to a crew member would total 48.3 mrem.*

In the analysis, two possible routes for the shipment of the spent nuclear fuel are considered. In the first, when a regularly scheduled commercial vessel is used, two intermediate port stops are assumed to add three additional days to the voyage, and therefore three additional hold inspections. The possibility of the ship having intermediate port stops must be considered in the event that a regularly scheduled commercial vessel is used for the shipment of the foreign research reactor spent nuclear fuel since the shipment is being made as part of a commercial cargo shipment. Such shipments are not limited to a single port of call. Based on the information provided in Table C-1, the average duration of a voyage would be 21 days, which includes three days for intermediate port calls. The second route accounts for using chartered ships or regularly scheduled commercial ships for which the first port of call is the port of entry for the foreign research reactor spent nuclear fuel. For this route, no intermediate port stops are included, so the travel times listed in Table C-1 were reduced by three days, making the average duration of the voyage approximately 18 days. If a ship carrying foreign research reactor spent nuclear fuel were to encounter

mechanical problems or extreme weather and was forced to make an unscheduled port call, the incident-free radiation exposure to the ship's inspection crew would slightly increase as a result of the additional duration of the voyage. People in the refuge port would not receive any exposure because the foreign research reactor spent nuclear fuel would remain on the ship and would not be handled.

Once at the port of entry, all casks of the spent nuclear fuel would be off loaded. Table C-4 describes the estimated dose (based on the selected exclusive-use limit of an external dose rate of 10 mrem per hour at 2 m or 6.6 ft from the surface of the container) received by crew members involved in the offloading activities associated with the offloading of a single hold, that is, two casks. These doses are the same as those received during the loading phase of the transport activity. Once the spent nuclear fuel cask is over the rail of the ship, the ship's crew would not be in close proximity to it. As a result, no ship crew personnel are assumed to be involved with any of the activities associated with disengaging the spent nuclear fuel container from the handling gear or in securing the container to any transport vehicle used to move the container off the pier.

Table C-4 Ship Crew Exposure During Offloading of a Hold Containing Two Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Regulatory Dose Limits)

	Tasks						*Total Dose per Loading[c] (person-rem)*
	Rail to Hold			*Secure Cask*			
Crew Member	*Distance[a] (m)*	*Exposure Rate[b] (mrem/hr)*	*Time (min)*	*Distance[a] (m)*	*Exposure Rate[b] (mrem/hr)*	*Time (min)*	
Chief Mate	20	1	5	5.5	3.5	60	0.009
Mate on Watch	20	1	5	8	2.1	60	0.005
Bosun	20	1	5	5.5	3.5	60	0.009
Seaman (2)	20	1	5	5.5	3.5	60	0.018

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Exposure rate is calculated based on 10 mrem/hr at 2 m (6.6 ft) from the shipping container surface.*

[c] *Includes the exposure from the first loaded casks for activities associated with the first cask. The exposure rate for securing the first cask is 1.5 times the listed rate.*

Tables C-5 and C-6 summarize the total crew doses for the shipment activities on a per shipment basis, annually, and for all of the shipments in the program. The maximum individual and total population doses are based on the selected exclusive-use regulatory limit external dose rate of 10 mrem per hour at 2 m or 6.6 ft from the surface of the container. Table C-5 summarizes the crew doses if regularly scheduled commercial vessels were used for all foreign research reactor spent nuclear fuel shipments. Table C-6 summarizes the crew doses if chartered vessels were used for all foreign research reactor spent nuclear fuel shipments. The reduction in the program crew-doses for the dedicated vessels is a result of the reduced transit time associated with the chartered vessels due to the fact that they do not make intermediate port calls. In situations where the services of a ship are obtained on a non-exclusive-use basis, the maximum allowable annual dose to a member of the ship's crew would be 100 mrem per year [based on Nuclear Regulatory Commission (NRC) and U.S. DOE limits on the exposure of members of the public].

As shown in Table C-5, the maximum individual dose per shipment on a regularly scheduled commercial vessel is 66 mrem to the Chief Mate and Bosun, a dose well below the 100 mrem per year limit. If the assumption was made that the same vessel and crew was used for as many shipments as possible in one year, the maximum individual dose to a crew member would be approximately 600 mrem. This assumes

Table C-5 Total Regularly Scheduled Commercial Ship's Crew Exposure for Marine Transport of Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Regulatory Dose Limits and Assuming Intermediate Port Stops)

Crew Member	*Maximum Individual Exposure per Trip (mrem)*	*Maximum Individual Exposure per Year[a] (mrem)*	*Program Dose Total (All Ships' Crews) (person-rem)*
Chief Mate	66	599	24.8
Mate on Watch	11	98	4.0
Bosun	66	599	24.8
Seaman (2)	18	158	13.1
Engineer	49	441	18.2
Total			**84.9**

[a] *Exposure per year based on nine voyages per year, two casks per voyage.*

Table C-6 Total Chartered Ship's Crew Exposure for Marine Transport of Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Regulatory Dose Limits and Assuming No Intermediate Port Stops)

Crew Member	*Maximum Individual Exposure per Trip (mrem)*	*Maximum Individual Exposure per Year[a] (mrem)*	*Program Dose Total (All Ships' Crew) (person-rem)*
Chief Mate	238	1,668	21.7
Mate on Watch	43	303	3.9
Bosun	238	1,668	21.7
Seaman (2)	70	492	12.8
Engineer	168	1,176	15.3
Total			**75.4**

[a] *Exposure per year based on seven trips per year, eight casks per voyage (two casks per hold).*

nine trips per year based on the average voyage length of all shipments and results in the ships' crew being exposed to the foreign research reactor spent nuclear fuel shipments for 189 days a year. Since travel time to a port of loading would be required, and most ship crews are rotated on a three or six month basis, the assumption of nine trips should bound the dose for any individual members of dedicated crews, even when trips are shorter than the assumed average of 21 days. The annual dose of approximately 600 mrem exceeds the 100 mrem annual limit for a member of the general public, and would therefore require mitigation. See the end of this section for a discussion of mitigation.

Due to the larger number of casks on a chartered vessel, the largest annual dose to a crew member is estimated to be approximately 1,668 mrem (approximately 1.7 rem). This is based on an estimated exposure of 238 mrem per voyage and seven voyages per year. Seven voyages per year using a chartered vessel is sufficient to ship all transportation casks to be shipped in an average year. It has been assumed that the 721 shipments would be made over a 13-year period. The exposure total for the marine transport portion of the program can be expressed as the number of LCFs that are calculated to result from doses received during the policy period, if the basic implementation of Management Alternative 1 of the proposed action were adopted. For a regularly scheduled commercial vessel, the exposure of approximately 84.9 person-rem translates to 0.034 LCFs. The total exposure associated with the shipment of the foreign research reactor spent nuclear fuel on a chartered vessel, approximately 76.4 person-rem, translates into 0.031 LCFs.

Use of a chartered vessel results in a reduction of approximately ten percent in the total population exposure and corresponding risk to the ships' crews under the basic implementation of Management Alternative 1. This difference is due to the shorter voyage duration when a chartered vessel is used. From Tables C-2 through C-4, it is apparent that the largest doses to the ship's crew are a result of the daily

inspection of the cargo holds. The three day reduction in the voyage duration (gained when a chartered vessel is used) reduces the dose received from the daily inspections and results in the ten percent difference between the use of regularly scheduled commercial and chartered vessels.

Tables C-7 through C-11 present the results of the above analysis with one change. The exposure and crew doses are calculated based on the "historical" external dose rate data developed from measurements taken during earlier shipments of research reactor spent nuclear fuel (a dose rate of 2.25 mrem per hour at 1 m or 3.3 ft from the surface of the shipping cask, which is equivalent to 1 mrem per hour at 2 m or 6.6 ft from the cask surface). See Appendix F, Section F.5 for the data used to derive this historical dose rate. Although this "historical" data are based on distance from the surface of the cask, it has conservatively been assumed in this analysis that this dose rate represents the dose at distances from the surface of the container in which the cask is shipped. This set of calculations was performed in order to provide additional perspective about the risks associated with the foreign research reactor spent nuclear fuel program. Use of the exclusive-use regulatory limit for the external dose rate ensures that the estimates discussed previously are upper bounds on the potential risks to the ship's crew from incident-free transport of the spent nuclear fuel. Use of the historical data provides an estimate that is closer to the expected risks associated with the shipment of all of the foreign research reactor spent nuclear fuel. Although the exact external dose rates cannot be determined in advance for all shipments, most should be similar to those for shipments made in the past. Therefore, the "historical" external dose rates should be a more accurate prediction of the risks resulting from the shipment of all 721 casks.

In this analysis, all other assumptions regarding voyage length, crew activity (time and distance from the spent nuclear fuel cask), number of shipments, and the assumptions made to estimate annual doses remained the same as in the analysis performed using the external dose rates derived from the exclusive-use regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the surface of the shipping container.

Using the historic dose rates, the maximum dose to an individual per regularly scheduled commercial vessel shipment would be 6.6 mrem, and the annual maximum individual dose would be 60 mrem (this dose is calculated assuming that the same crew member is involved in nine separate voyages transporting two spent nuclear fuel casks each during a single year). These doses are an order of magnitude lower than the corresponding doses calculated using the exclusive-use regulatory external dose rates. The calculated maximum individual dose is well below the maximum allowable annual dose to a member of the public of 100 mrem.

Use of a chartered vessel for the shipments, versus the use of a regularly scheduled commercial vessel, would result in a ten percent reduction in the total ships' crews doses. The use of a chartered vessel would result in annual exposure at slightly less than twice the public dose limits for exposure to radiation established by both DOE and NRC (100 mrem per year).

The dose total for the marine transport portion of the entire program can be expressed as the number of LCFs that are calculated to result from exposures of that size. For a regularly scheduled commercial vessel a total exposure of approximately 8.5 person-rem translates to 0.0034 LCFs. The total calculated exposure associated with the shipment of the foreign research reactor spent nuclear fuel on a chartered vessel, approximately 7.6 person-rem, translates into 0.0030 LCFs.

The results of these analyses indicate that, in some circumstances, some individual crew members could receive doses that exceed the limit established by DOE and the NRC for exposure of a member of the public, especially when the dose rate from the casks are assumed to be at the regulatory limit. It is anticipated that for most shipments, the external dose rate for the loaded transportation case would be near

Table C-7 Ship Crew Exposure During Loading of a Hold Containing Two Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Historical Cask Dose Rates)

Crew Member	Tasks						Total Dose per Loading[c] (person-rem)
	Rail to Hold			Secure Cask			
	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	
Chief Mate	20	0.1	5	5.5	0.35	60	0.0009
Mate on Watch	20	0.1	5	8	0.21	60	0.0005
Bosun	20	0.1	5	5.5	0.35	60	0.0009
Seaman (2)	20	0.1	5	5.5	0.35	60	0.0018

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Exposure rate is calculated based on 2.25 mrem/hr at 1 m (3.3 ft) from the shipping container surface.*

[c] *Includes the exposure from the first loaded cask for activities associated with securing the second cask. The exposure rate for securing the second cask is 1.5 times the listed number.*

Table C-8 Ship Crew Exposure Per Hold During At-Sea Inspections (Based on Historical Cask Dose Rates)

Crew Member	Distance[a] (m)	Dose Rate[b] (mrem/hr)	Time (min)	Dose per Daily Inspection[c] (mrem)
Chief Mate	5.5	0.7	20	0.23
Bosun	5.5	0.7	20	0.23
Engineer	5.5	0.7	20	0.23

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Includes the effect of two casks in the hold.*

[c] *For a ship carrying two casks on a voyage duration of 21 days, the total dose to a crew member conducting daily inspections would be estimated at 4.8 mrem.*

Table C-9 Ship Crew Exposure During Offloading of a Hold Containing Two Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Historical Cask Dose Rates)

Crew Member	Tasks						Total Dose per Loading[c] (person-rem)
	Rail to Hold			Secure Cask			
	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	
Chief Mate	20	0.1	5	5.5	0.35	60	0.0009
Mate on Watch	20	0.1	5	8	0.21	60	0.0005
Bosun	20	0.1	5	5.5	0.35	60	0.0009
Seaman (2)	20	0.1	5	5.5	0.35	60	0.0018

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Exposure rate is calculated based on 2.25 mrem/hr at 1 m (3.3 ft) from the shipping container surface.*

[c] *Includes the exposure from the last off loaded cask during activities associated with the first off loaded cask. The exposure rate for securing the first cask is 1.5 times the listed rate.*

Table C-10 Total Regularly Scheduled Commercial Ships Crew Exposure for Marine Transport of Foreign Research Reactor Spent Nuclear Fuel Casks Assuming Intermediate Port Stops (Based on Historical Cask Dose Rates)

Crew Member	*Maximum Individual Exposure per Trip (mrem)*	*Maximum Individual Exposure per Year[a] (mrem)*	*Dose Total (All Ships' Crew) (person-rem)*
Chief Mate	6.6	60	2.5
Mate on Watch	1.1	10	0.4
Bosun	6.6	60	2.5
Seaman (2)	1.8	16	1.3
Engineer	4.9	44	1.8
Total			**8.5**

[a] *Exposure per year based on nine trips per year.*

Table C-11 Total Chartered Ships Crew Exposure for Marine Transport of Foreign Research Reactor Spent Fuel Casks Assuming No Intermediate Port Stops (Based on Historical Cask Dose Rates)

Crew Member	*Maximum Individual Exposure per Trip (mrem)*	*Maximum Individual Exposure per Year[a] (mrem)*	*Dose Total (All Ships' Crew) (person-rem)*
Chief Mate	24	167	2.2
Mate on Watch	4.3	30	0.4
Bosun	24	167	2.2
Seaman (2)	7.0	49	1.3
Engineer	17	118	1.5
Total			**7.6**

[a] *Exposure per year based on seven trips per year.*

the historic dose rates, which would not cause any personnel to exceed radiation exposure limits for the public. However, the existence of some shipments with external dose rates closer to the exclusive-use regulatory limit suggests that DOE should provide a means to assure that individual crew members do not receive doses in excess of the public dose limits. As a minimum, the program should establish administrative procedures that will maintain records of the dose rates associated with each shipment and the ports of departure and entry for the shipment. The measurement of interest for the record keeping would be the external dose rates outside the container, which houses the transportation cask, since the crew does not enter the container. (It should be noted that the analysis using the historical data did not consider the reduction in external dose rate due to the distance from the cask to the container surrounding the transportation cask.) These measurements can be used to identify shipments that would result in crew exposures above those calculated based on the historical spent nuclear fuel transportation external dose rate. By tracking this information, DOE would be able to identify if and when additional precautions to reduce individual exposures should be taken (i.e., restricting the use of crew members who are near the annual dose limit from further shipments that year). DOE would also include a clause in the contract for shipment of the foreign research reactor spent nuclear fuel requiring other crew members be used if any crew member approaches a 100 mrem dose in any year.

C.4.2 Incident-Free Marine Impacts of Policy Alternatives

Two implementation subalternatives to Management Alternative 1 and one subalternative under Management Alternative 2 of the proposed action were identified that could impact the incident-free marine risk calculations that were performed for the basic implementation (Chapter 2 describes the alternatives and subalternatives of Management Alternative 1 and the subalternatives of Management

Alternative 2). The implementation subalternative of accepting spent nuclear fuel only from developing countries would result in a reduction in the amount of spent nuclear fuel transported by ship. Table C-12 lists the countries that are considered developing countries and the number of shipments that would be required to transport their spent nuclear fuel to the United States.

Table C-12 Voyage Data for Acceptance of Foreign Research Reactor Spent Nuclear Fuel from Developing Nations Only

Port	*Voyage Duration (days)*	*Number of Casks*	*Number of Trips*
Argentina	21.2	9	5
Bangladesh	31.0	3	2
Brazil	20.8	8	4
Chile	16.3	2	1
Colombia	11.1	1	1
Greece	22.0	8	4
Indonesia	30.3	14	7
Iran	36.6	1	1
Jamaica	10.2	1	1
Korea (South)	25.3	18	9
Malaysia	28.9	3	2
Mexico	7.6	6	3
Pakistan	34.4	3	2
Peru	13.0	1	1
Philippines	28.1	6	3
Portugal	18.3	3	2
Romania	23.8	48	24
Slovenia	20.9	13	7
South Africa	26.0	2	1
Thailand	33.1	5	3
Turkey	22.9	4	2
Uruguay	20.9	1	1
Venezuela	11.1	4	2
Zaire	23.6	4	2
Totals		**168**	**90**
Average	**23**		

Under the implementation subalternative of using a policy duration of five years for the acceptance of foreign research reactor spent nuclear fuel, the number of transportation casks of foreign research reactor spent nuclear fuel requiring ocean transport would be reduced to 586. Appendix B presents the derivation of the total number of shipments (ocean transport plus land transport from Canada) estimated in this alternative.

Subalternative 1b (overseas reprocessing) under Management Alternative 2 also has the capability to impact the results of the incident-free marine risk analysis since it involves shipment of the vitrified waste to a storage facility in the United States. Under this subalternative to Management Alternative 2, eight transportation cask shipments of vitrified waste would be made to the United States.

In addition, a Hybrid Alternative was analyzed. In the Hybrid Alternative, those countries (for this option, assumed to be Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom) that have the capability to store high-level waste would be encouraged to reprocess the aluminum-based research reactor spent nuclear fuel and to accept for management the resulting high-level waste. The United States

would accept for management the research reactor spent nuclear fuel from those countries deemed not to have the high-level waste storage capability, and all TRIGA fuel. This Hybrid Alternative includes all countries identified in Table C-1 except for those seven nations just listed. Under this Hybrid Alternative, 452 shipments of spent nuclear fuel are assumed to be sent to the United States, excluding shipments of Canadian origin.

The incident-free marine risks associated with the two implementation subalternatives of Management Alternative 1 and the subalternative of Management Alternative 2 are discussed in the following sections.

Management Alternative 1, Implementation Subalternative Ia — Acceptance of Foreign Research Reactor Spent Nuclear Fuel Only from Developing Countries: This implementation subalternative of Management Alternative 1 would result in the shipment of 168 casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the incident-free marine impact of the basic implementation of Management Alternative 1 have been used in the analysis of this implementation subalternative. This implementation subalternative has been analyzed using the "exclusive-use" shipment regulatory transportation cask external dose rates. To compare this implementation subalternative to the basic implementation, it is only necessary to perform the analysis using one estimate of the external dose rate of the transportation cask. The relationship between the calculated impact of the two implementation subalternatives using the regulatory external dose rate would be the same as that calculated using the "historical" data. Therefore, the use of the one dose rate provides a sufficient point of comparison between the two alternatives.

The assumptions that have not changed between the analysis for the basic implementation and this implementation subalternative include the following:

- The same types of vessels should be available for use, so, the option for using chartered or regularly scheduled commercial vessels was examined, and
- The activities associated with the loading of the foreign research reactor spent nuclear fuel, the daily inspections of the cargo during the voyage, and the offloading of the foreign research reactor spent nuclear fuel do not change simply because there is a reduction in the number of shipments to be made.

The average duration of the voyages from these developing countries to the United States is slightly longer than the average for the voyages associated with the basic implementation. As shown in Table C-12, the average duration is 23 days (for a regularly scheduled commercial vessel) versus the 21 days in the basic implementation. For a chartered vessel, the voyage duration is three days less (i.e., 20 days). The longer average voyage duration results in an increase in the total of the daily inspection-related crew doses of approximately 4.6 mrem per crew member involved in the inspection. The inspection dose for a 23-day voyage would be 52.9 mrem (2.3 mrem times 23 days) per inspector.

The population dose to the ship's crew, per voyage, can be derived from the data contained in Tables C-5 and C-6. Incorporating the increase in the inspection dose into the data from Table C-5, the individual doses on a regularly scheduled commercial vessel would be 71 mrem to the Chief Mate and the Bosun, 11 mrem to the Mate on Watch, 18 mrem to each of two Seamen, and 54 mrem to the Engineer. The population (ship's crew) dose per shipment would be 242 mrem. If a chartered vessel is used (carrying eight transportation casks instead of two for the regularly scheduled commercial vessel), the corresponding doses are 257 mrem to the Chief Mate and the Bosun, 43 mrem to the Mate on Watch, 70 mrem to each of two Seamen, and 187 mrem to the Engineer. The population (ship's crew) dose per shipment would be 885 mrem.

The 168 cask-shipments, requiring 90 ocean voyages using regularly scheduled commercial cargo vessels (up to 23 voyages using chartered vessels), represent approximately 24 percent of the total number of shipments in the basic implementation. The total population (ship's crew) exposure resulting from this implementation subalternative would be approximately 27 percent of the exposure calculated for the basic implementation. The difference in these two percentages is a direct result of the longer average duration of ocean crossings. The total population exposure for the implementation subalternative, assuming that regularly scheduled commercial vessels are used, would be approximately 22.0 person-rem, and would be approximately 20.3 person-rem if chartered vessels are used. These population exposures translate into a risk to the ship's crew of 0.0091 LCF and 0.0081 LCF, respectively. As discussed in Section 4.1, the relationship between a dose and LCFs for workers (ship's crew) is that a 1 rem dose equates to 0.0004 LCFs.

Management Alternative 1, Implementation Subalternative 2a — Acceptance of Foreign Research Reactor Spent Nuclear Fuel for Five-Year Policy Duration: As stated above, this implementation subalternative results in the shipment of 586 casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the incident-free marine impact of the basic implementation have been used in the analysis of this implementation subalternative. This implementation subalternative has been analyzed using the "exclusive-use" shipment regulatory transportation cask external dose rates. To compare this implementation subalternative to the basic implementation it is only necessary to perform the analysis using one external dose rate. The relationship between the calculated impact of the implementation subalternative and the basic implementation using the regulatory external dose rate would be the same as that calculated using the "historical" data. Therefore, the use of the one dose rate provides a sufficient point of comparison.

The assumptions that have not changed between the analysis for the basic implementation and this implementation subalternative include the following:

- The same types of vessels should be available for use, and the option for using chartered or regularly scheduled commercial vessels was examined;
- The average voyage duration that was used in the analysis of the incident-free marine risk for the basic implementation was used for this implementation subalternative. The 586 shipments represent approximately 81 percent of the shipments made under the basic implementation and the distribution of shipments from the different countries of origin is similar to that modeled for the basic implementation; and
- The activities associated with the loading of the foreign research reactor spent nuclear fuel transportation casks, the daily inspections of the cargo during the voyage, and the offloading of the foreign research reactor spent nuclear fuel transportation casks do not change simply because there is a reduction in the number of shipments to be made.

Because there are no differences between the per-shipment activities in this implementation subalternative and the basic implementation, the per-voyage crew exposures will not differ from those presented in Tables C-5 and C-6 for the basic implementation. In addition, the maximum annual exposures to individual crew members will not change. The analysis has assumed a maximum number of voyages that a single crew would be involved in during a single year. Although the total number of shipments per year must increase in this alternative (an average of 73 casks must be shipped per year for eight years), no single ship's crew will be involved in more shipments than had been assumed in the analysis of the basic implementation. The annual doses presented in Tables C-5 and C-6 are applicable to this alternative as well as to the basic implementation.

The total population (ship's crew) exposure resulting from this implementation subalternative would be approximately 81 percent of exposure calculated for the basic implementation. The total population exposure for the implementation subalternative, assuming that regularly scheduled commercial vessels are used, would be approximately 69 person-rem, and would be approximately 61 person-rem if chartered vessels were to be used. These population exposures translate into a risk to the ships' crew of 0.028 LCF and 0.025 LCF, respectively. As discussed in Section 4.1, the relationship between a dose and LCFs for workers (ship's crew) is that a 1 rem dose equates to 0.0004 LCF.

Management Alternative 2, Subalternative 1b – Overseas Processing with Shipment of Waste to a U.S. Storage Facility: In this subalternative, the foreign research reactor spent nuclear fuel would be reprocessed overseas (most probably in Great Britain or France) and the waste products would be contained within a small number of vitrified waste logs. This high-level waste might be brought to the United States for storage at one of the management site facilities evaluated under the basic implementation of Management Alternative 1. Under these conditions, up to eight transportation casks containing 16 European-size canisters of vitrified waste would be shipped from Europe to the United States (see Section 4.4.2.2 for more information on the vitrification of the waste material). This analysis addresses the incident-free marine risks associated with transporting these eight casks of vitrified waste from Europe to the United States.

As with the shipment of unprocessed spent nuclear fuel, the primary impact of incident-free marine shipping of the vitrified waste is upon the crews of the ships used to carry the casks. Most of the assumptions used in the analysis of the crew exposure to the spent nuclear fuel (see Section C.4.1 of this appendix) have been used to analyze the impact of the shipment of vitrified waste. The crew exposure due to loading and offloading activities have been considered, but the primary contribution to the crew dose comes from the daily cargo inspection activities. The inspection activities on the ship carrying the vitrified waste have been modeled in the same manner as the inspections aboard the vessels carrying the spent nuclear fuel. Three crew members have been modeled as performing the inspections, and the same three crew members are assumed to perform this task for the entire voyage. For the purposes of this analysis, it has been assumed that the vitrified waste will be transported on a chartered vessel, there will be no intermediate port calls, and the shipment will originate in Europe. Because there are no intermediate port calls and the shipments originate in Europe, the voyage duration is estimated to be 15 days. This estimate is based on the average of the voyage durations for one trip from the United Kingdom to the East Coast of the United States, one to the West Coast of the United States, and the average of a trip from France to both U.S. coasts. The assumption that there are no intermediate port calls reduces the average duration of each of these trips by three days from the estimates presented in Table C-1.

Little information is available on the casks that might be used to transport the vitrified waste. Therefore, the assumption has been made that the exposure to the crew will be limited to the exclusive-use regulatory limit (10 CFR 71) of 10 mrem per hour at 2 m (6.6 ft) from the surface of the container. No attempt was made to extrapolate limited historical data to determine crew incident-free impacts from any other exposure rate other than the limit set forth in NRC and DOE regulations.

It has been assumed that two casks are being transported as part of a single shipment. This assumption results in additional exposure to the crew members due to exposure to two radiation fields during all activities which bring crew members into the vicinity of the transportation casks. Should all of the casks be shipped at once, this assumption is equivalent to assuming that this single shipment is made with two casks per hold on the vessel. The crew risk would be the same for this single (eight cask) shipment as for the four shipments with two casks per vessel.

Based on the assumptions outlined above, the incident-free impact of the shipment of vitrified waste on the ship's crew would be slightly less per shipment than that calculated for the shipment of foreign research reactor spent nuclear fuel. The trip duration of only 15 days, versus the average duration of 18 days, for a chartered vessel in the basic implementation of Management Alternative 1 results in a reduction of the dose to each inspector, the Chief Mate, the Bosun, and the Engineer, of approximately 6.9 mrem per journey (three fewer inspections, each of which would have resulted in a dose of 2.3 mrem). The population dose to the ship's crew, per voyage, can be derived from the data contained in Table C-6. Incorporating the reduction in the inspection dose into the data from this table, the individual doses would be: 210 mrem to the Chief Mate and the Bosun, 43 mrem to the Mate on Watch, 70 mrem to each of two Seamen, and 140 mrem to the ships Engineer. Per voyage, the total population dose to the ship's crew would be 0.74 person-rem.

With only eight casks to be shipped, the subalternative action could be achieved with a single shipment (the crew dose would be the same as that calculated if four shipments of two casks each were made). The population exposure results in a risk to the crew of 0.00030 LCF. Due to the reduced number of shipments, compared to the 721 shipments of spent nuclear fuel in the basic implementation of Management Alternative 1, the marine incident-free risk to the crew is approximately two orders of magnitude lower than that calculated for the basic implementation.

Management Alternative 3 – Combination of Components of Management Alternative 1 and 2 (Hybrid Alternative): Under the Hybrid Alternative, the United States would accept foreign research reactor spent nuclear fuel from countries without high-level waste storage capability. This Hybrid Alternative could result in the shipment of 452 casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the incident-free marine impact for the basic implementation of Management Alternative 1 have been used in the analysis of this Hybrid Alternative. This alternative has been analyzed using the selected "exclusive-use" regulatory dose limit for the shipment of spent nuclear fuel casks.

Included in the assumptions that have not changed between the analysis for the basic implementation and this alternative are the following:

- The same types of vessels should be available for use under this Hybrid Alternative, the option for using chartered or regularly scheduled commercial vessels was examined, and
- The activities associated with the loading of the foreign research reactor spent nuclear fuel, the daily inspection of the cargo during the voyage, and the offloading of the foreign research reactor spent nuclear fuel do not change simply because there is a reduction in the number of shipments to be made.

The average duration of the voyages from the countries without high-level waste storage capability to the United States is slightly longer than the average for the voyages associated with the basic implementation. Using the data in Table C-12, and eliminating the aluminum-based spent fuel shipments from Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom, the average voyage duration is almost 23 days (for a regularly scheduled commercial vessel) versus the 21 days for the basic implementation. For a chartered vessel, the voyage duration is three days less (i.e., almost 20 days). The longer average voyage duration results in an increase in the total of the daily inspection-related crew doses of approximately 4.6 mrem per crew member involved in the inspection. The inspection dose for a 23-day voyage would be 52.9 mrem (2.3 mrem times 23 days) per inspector.

The population dose to the ship's crew, per voyage, can be derived from the data contained in Tables C-5 and C-6. Incorporating the increase in the inspection dose into the data from Table C-5, the individual doses on a regularly scheduled commercial vessel would be 71 mrem to the Chief Mate and the Bosun, 11 mrem to the Mate on Watch, 18 mrem to each of two Seamen, and 54 mrem to the ship's Engineer. The population (ship's crew) dose per shipment would be 243 mrem. If a chartered vessel is used (carrying eight transportation casks instead of two for the regularly scheduled commercial vessel), the corresponding doses are 257 mrem to the Chief Mate and the Bosun, 43 mrem to the Mate on Watch, 70 mrem to each of two Seamen, and 187 mrem to the ship's Engineer. The population (ship's crew) dose per shipment would be 884 mrem.

The 452 cask shipments, requiring 236 ocean voyages using commercial regularly scheduled commercial cargo vessels, represent approximately 63 percent of the total number of shipments for the basic implementation. The total population (ships' crew) exposure resulting from this Hybrid Alternative would be approximately 69 percent of the exposure calculated for the basic implementation. The differences in these two percentages is a direct result of the longer average duration of ocean crossings. The total population exposure for the Hybrid Alternative, assuming that regularly scheduled commercial vessels are used, would be approximately 57.2 rem and would be approximately 52.2 rem if chartered vessels were used. These population exposures translate into a risk to the ships' crew, in terms of LCFs, of 0.024 LCF and 0.021 LCF, respectively. As discussed in Section 4.1, the relationship between a dose and LCFs is that a 1 rem dose equates to 0.0004 LCFs.

C.5 Accident Impacts: Methods and Results

C.5.1 Introduction

If the cask sinks anywhere in U.S. coastal waters, it will be recovered, regardless of depth. U.S. coastal waters in this case refers to waters within the 12-mile territorial limit. Recovery would be accomplished, even in the deepest parts of U.S. coastal waters, such as in Puget Sound, which reaches 305 meters or 1,000 feet (Encyclopedia Americana, 1991). Elsewhere in the world, if the cask sinks in coastal water (i.e., in water up to 200 m or 660 ft), every effort would be made to recover it. In deeper waters, the recovery is more problematic. As recovery, even in coastal waters, cannot be guaranteed, two scenarios need to be evaluated:

> ***Scenario A***: As the result of a maritime casualty (e.g., collision, foundering, fire), the vessel sinks in coastal waters, resulting in the submersion of the cask on the ocean floor. The cask is not retrieved. Analyses are done for two cases, (1) damaged cask, and (2) undamaged cask.
>
> ***Scenario B***: As the result of a maritime casualty (e.g., collision, foundering, fire), the vessel sinks in deep ocean waters, resulting in the submersion of the cask on the ocean floor. The cask is not retrieved. Analyses are done for one case only, a damaged cask, as it has been assumed that submersion in the deep ocean will damage the cask.

In 1988, the Nuclear Energy Agency of the Organization for Economic Cooperation and Development published a radiological assessment as part of a feasibility study for disposal of high-level radioactive waste into the seabed (NEA, 1988). As part of the radiological assessment, several accident scenarios were examined. In particular, a scenario involving a transportation accident at sea was examined. The results of calculations performed for the Nuclear Energy Agency radiological assessment are used here, with modification. The Nuclear Energy Agency results are based on vitrified high-level waste, which behaves differently in salt water than the metal foreign research reactor spent nuclear fuel. Also, the

inventory of radioactive material in the foreign research reactor spent nuclear fuel is considerably different than the vitrified high-level waste inventory. With modifications to compensate for these differences, the Nuclear Energy Agency results were used to predict the peak individual dose and biota dose for Scenario A and Scenario B.

C.5.2 Assumptions

1. The spent nuclear fuel and cask modeled are the BR-2 fuel and the Pegase cask. Based on the information provided in Appendix B, the loaded Pegase cask contains 0.0155 metric tons of heavy metal (MTHM) (15.5 kg) of fuel (assuming the cask is loaded with BR-2 type fuel). This fuel type was selected because BR-2 fuel has the highest isotope content per unit mass of heavy metal of the three fuel types considered in this analysis. Use of the highest inventory of radionuclides establishes a conservative upper bound on the estimated dose rates from the leaching of radionuclides into the sea. This is because the dose rates are a function of the corrosion rate of spent nuclear fuel, expressed in terms of mass per unit of time, and the specific activity of the spent nuclear fuel, expressed in terms of radioactivity per unit of mass.

2. The fuel rods contain aluminum-clad metallic spent nuclear fuel elements.

3. The deep ocean model is for the South Nares Abyssal Plain.

4. Corrosion of spent nuclear fuel inside a damaged cask begins immediately; corrosion of spent nuclear fuel inside an undamaged cask begins at the time the cask fails and allows seawater to come in contact with the spent nuclear fuel.

5. Once free of the fuel matrix through corrosion, the fission products exit the failed cask without delay.

6. The corrosion rate for spent nuclear fuel elements is constant. Radionuclides are leached from the spent nuclear fuel elements at a rate proportional to the corrosion rate depending on their relative concentrations.

Data from the Nuclear Energy Agency vitrified high-level waste model and on spent nuclear fuel corrosion rates are summarized in Table C-13.

Table C-13 Data For Estimating Spent Nuclear Fuel Dose Rates From the Nuclear Energy Agency Assessments for Vitrified High-Level Waste

Parameter Description	*Value*[a]	*Source*
Corrosion Rate for Glass (α_0)	0.000036 kg/m^2 day	NEA 1988
Corrosion Rate for Aluminum-Clad Fuel (α_1)	0.0086 kg/m^2 day	Rechard 1994
Sensitivity Coefficient for Corrosion Rate (a)	0.99	NEA 1988
Undamaged Cask Peak Individual Dose	9 rem/yr	NEA 1988
Damaged Cask Peak Individual Dose	650 rem/yr	NEA 1988
Undamaged Cask Peak Biota Dose (Fish)	3.6 rad/yr	NEA 1988
Undamaged Cask Peak Biota Dose (Crustaceans)	3.8 rad/yr	NEA 1988
Undamaged Cask Peak Biota Dose (Mollusks)	10.0 rad/yr	NEA 1988
Damaged Cask Peak Biota Dose (Fish)	29.0 rad/yr	NEA 1988
Damaged Cask Peak Biota Dose (Crustaceans)	31 rad/yr	NEA 1988
Damaged Cask Peak Biota Dose (Mollusks)	660 rad/yr	NEA 1988

[a] *Dose rates are based on a total Nuclear Energy Agency program mass of 100,000 MTHM*

C.5.3 Calculational Method For Dose Rate Estimates

The calculations presented here are designed to account for two differences between the Nuclear Energy Agency radiological assessment and the radiological assessment required for this EIS. First, in the radiological assessment performed for the Nuclear Energy Agency, a vitrified glass waste form was assumed. For this EIS, aluminum-clad metal matrix fuel elements are assumed. Thus, the corrosion rate of the matrix containing the radionuclides will be different in the two cases. Second, the radiological assessment for the Nuclear Energy Agency was performed assuming reprocessed fuel equivalent to 100,000 MTHM containing a total of 10 billion curies, for a specific activity of 100,000 Ci per MTHM. For this EIS, it is assumed that one Pegase cask contains 0.0155 MTHM (15.5 kg) of spent nuclear fuel and 930,000 Ci, for a specific activity of 60 million Ci per MTHM. Table C-14 contains a detailed list of the inventory of radionuclides for both the Nuclear Energy Agency vitrified high-level waste and the foreign research reactor spent nuclear fuel. The specific activity for the vitrified high-level waste is significantly lower than that of the foreign research reactor spent nuclear fuel because the Nuclear Energy Agency study uses data assuming a 100-year decay time for the waste, while the foreign research reactor spent nuclear fuel is assumed to only have been out of the reactor less than a year. The Nuclear Energy Agency study used 100-year decay time because in their study the spent nuclear fuel was not vitrified until it was 50 years out of the reactor, and it was assumed to take 50 years for their cask to fail once it was in the ocean.

The dose estimates from the Nuclear Energy Agency analysis are scaled for this EIS to reflect (1) the fact that spent nuclear fuel corrodes faster than vitrified glass, (2) there is significantly less mass of heavy metal in a spent nuclear fuel cask than was used in the Nuclear Energy Agency dose risk models, and (3) the specific activity of the foreign research reactor spent nuclear fuel is higher than the specific activity of the Nuclear Energy Agency vitrified high-level waste.

To account for differences in the waste matrix corrosion rate, the sensitivity of the calculated dose to the corrosion rate was used. In its radiological assessment, the Nuclear Energy Agency published sensitivity studies. For the accident analyses, an adjoin method was used to determine the sensitivity of the peak individual dose and the collective dose to key parameters in their performance assessment model, including the waste matrix corrosion rate.

The adjoin method employs a mathematical algorithm for calculating directly in one run the sensitivity of a performance assessment model to the model parameters. It gives as output the first derivative of the response of the performance assessment model (here, peak individual dose and collective dose) with respect to each of the model parameters (in particular, corrosion rate). Explicitly, the sensitivity coefficient is defined as:

$$a = \frac{\partial D/D}{\partial \alpha/\alpha} \qquad (1)$$

where a is the sensitivity coefficient, D is the dose (peak or cumulative), and α is a given parameter (leach rate). This expression can be used to determine the change in the dose for a change in the parameter value by integrating as follows in equation 2.

Total

$$\int_{D_0}^{D_1} \frac{\partial D}{D} = a \int_{\alpha_0}^{\alpha_1} \frac{\partial \alpha}{\alpha} \qquad (2)$$

or

Table C-14 Comparison of Radionuclide Inventories for Nuclear Energy Agency High-Level Waste Sub-Seabed Disposal Studies and BR-2 Foreign Research Reactor Spent Nuclear Fuel

Radionuclide	*Nuclear Energy Agency Vitrified High-Level Waste Inventory[a] (Ci)*	*Foreign Research Reactor Spent Nuclear Fuel Inventory[b] (Ci)*	*Radionuclide*	*Nuclear Energy Agency Vitrified High-Level Waste Inventory[a] (Ci)*	*Foreign Research Reactor Spent Nuclear Fuel Inventory[b] (Ci)*
Hydrogen-3	0.0	86.4	Cerium-141	0.0	5,700
Selenium-79	33,000	0.0	Cerium-144	0.0	310,000
Krypton-85	0.0	2,500	Promethium-147	11,000	48,000
Strontium-89	0.0	41,000	Promethium-148m	0.0	75.6
Strontium-90	2,000,000,000	21,000	Samarium-151	27,000,000	0.0
Yttrium-90	2,000,000,000	0.0	Europium-154	8,600,000	620
Yttrium-91	0.0	73,000	Europium-155	480,000	130
Niobium-95	0.0	220,000	Uranium-233	178	0.0
Zirconium-93	180,000	0.0	Uranium-234	300	0.0091
Zirconium-95	0.0	110,000	Uranium-235	0.0	0.014
Technicium-99	1,400,000	0.0	Uranium-236	47	0.0
Ruthenium-103	0.0	8,900	Uranium-238	0.0	0.00034
Ruthenium-106	0.0	22,000	Neptunium-237	32,000	0.0
Palladium-107	10,000	0.0	Plutonium-238	0.0	64.2
Tin-123	0.0	430	Plutonium-239	120,000	1.8
Tin-126	58,000	0.0	Plutonium-240	620,000	1.2
Antimony-125	990	890	Plutonium-241	3,500,000	280
Antimony-126m	58,000	0.0	Plutonium-242	600	0.0
Tellurium-125m	0.0	210	Americium-241	6,900,000	0.4
Tellurium-127m	0.0	890	Americium-242m	0.0	0.0011
Tellurium-129m	0.0	200	Americium-243	2,000,000	0.0043
Iodine-129	3.0	0.0	Curium-242	0.0	1.8
Cesium-134	108	16,000	Curium-244	0.0	1.3
Cesium-135	150,000	0.0	Curium-245	21,000	0.0
Cesium-137	3,000,000,000	21,000	Curium-246	5,500	0.0
Barium-137m	2,900,000,000	0.0			
			Total	**10,000,000,000**	**930,000**

[a] *Nuclear Energy Agency vitrified high-level waste radionuclide inventories are based on 100,000 MTHM that represent spent nuclear fuel radionuclide inventories for 100 years out of reactor. The Nuclear Energy Agency analysis based its dose rate estimate calculations on vitrified high-level waste that was produced from commercial light water reactor spent nuclear fuel at 50 years out of reactor, then the Nuclear Energy Agency analysis models the release of the vitrified high-level waste inventory into the ocean only after an additional 50 years of submersion.*

[b] *Foreign research reactor spent nuclear fuel radionuclide inventories are based on a Pegase cask filled with 36 elements of BR-2 spent nuclear fuel, 300 days out of reactor.*

$$\ln (D_1/D_0) = a \ \ln (\alpha_1/\alpha_0) \quad (3)$$

Using the data provided in Table C-13,

$$\ln (D_1/D_0) = 0.99 \ln (8.6 \times 10^{-3}/3.6 \times 10^{-5}) \quad (4)$$

or

$$D_1 = 227 \, D_0 \quad (5)$$

Where D_1 is the dose by foreign research reactor spent nuclear fuel, adjusted only for the difference in leach rate, and D_0 is the Nuclear Energy Agency dose.

Since the derivative in Equation (1) is evaluated at a particular value of each model parameter, it is by definition the sensitivity coefficient of the dose to small variations in each parameter around their assigned value. As a result, the calculation of dose using the sensitivity coefficient is valid only when changes in the leach rate remain "sufficiently small" compared to the leach rate. However, the Nuclear Energy Agency assessment states that many of the models in their assessment are linear, and it is possible to estimate changes in the dose even for large variations in the leach rate.

To account for differences in the waste inventory, the dose was scaled linearly according to the ratio of the specific activity of the BR-2 spent nuclear fuel to the specific activity of the vitrified high-level waste as shown in Equation (6).

$$D = D_1 \frac{\beta_{EIS}}{\beta_{NEA}} = D_1 \frac{0.0155}{1.0\text{E+05}} \frac{6.0\text{E+07}}{1.0\text{E+05}} = 9.3\text{E–05} \ D_1 \quad (6)$$

Finally,

$$D = 0.021 \ D_0 \quad (7)$$

C.5.4 Results

Dose rates were calculated in the Nuclear Energy Agency study for two types of ocean environments, coastal waters and deep ocean floors. The results of scaling the Nuclear Energy Agency dose rate estimates for the scenario of losing a cask of foreign research reactor spent nuclear fuel in coastal waters are shown in Table C-15, with the comparable Nuclear Energy Agency results. In Table C-16, the results of losing a cask containing foreign research reactor spent nuclear fuel in deep ocean waters are shown. Table C-15 presents results for both an undamaged and a damaged cask, however Table C-16 provides the estimated dose for a damaged cask only because it is assumed that the pressure from the deep ocean will damage the cask seals.

The doses associated with the foreign research reactor spent nuclear fuel in Table C-16 are, in the case of the mollusks, very high. However, to properly interpret this result, several factors must be considered. First, the calculation that produced these results is very conservative for two reasons. The radioactive material, once corroded, was assumed to immediately be released into the open ocean water. In fact, the cask is expected to provide a significant "hold-up" time. This is because only the seal is expected to fail, which means that, due to the small area of the seal, only a very limited amount of water movement through the cask will be experienced. Over time, this small flow would carry out all of the soluble fission products, but insoluble precipitates would remain in the cask. Also, no account was taken for the possibility that the cask would likely become buried in silt, greatly slowing the fission product's entry into

Table C-15 Coastal Waters Dose Rate Estimates for 100,000 MTHM Vitrified High-Level Waste and a Pegase Cask Loaded With BR-2 Foreign Research Reactor Spent Nuclear Fuel

Dose Category	*D_0 (NEA)*	*D (BR-2)*
Undamaged Cask Peak Individual Dose	9.0 rem/yr	0.19 rem/yr
Damaged Cask Peak Individual Dose	650 rem/yr	14 rem/yr
Undamaged Cask Peak Biota Dose (Fish)	3.6 rad/yr	0.077 rad/yr
Undamaged Cask Peak Biota Dose (Crustaceans)	3.8 rad/yr	0.081 rad/yr
Undamaged Cask Peak Biota Dose (Mollusks)	10 rad/yr	0.21 rad/yr
Damaged Cask Peak Biota Dose (Fish)	29 rad/yr	0.62 rad/yr
Damaged Cask Peak Biota Dose (Crustaceans)	31 rad/yr	0.66 rad/yr
Damaged Cask Peak Biota Dose (Mollusks)	660 rad/yr	14 rad/yr

Table C-16 Deep Ocean Dose Rate Estimates for 100,000 MTHM Vitrified High-Level Waste and a Pegase Cask Loaded with BR-2 Foreign Research Reactor Spent Nuclear Fuel

Dose Category	*D_0 (NEA)*	*D (BR-2)*
Damaged Cask Peak Individual Dose	0.00053 rem/yr	0.114 rem/yr
Damaged Cask Peak Biota Dose (Fish)	30,000 rad/yr	640 rad/yr
Damaged Cask Peak Biota Dose (Crustaceans)	41,000 rad/yr	880 rad/yr
Damaged Cask Peak Biota Dose (Mollusks)	1,400,000 rad/yr	30,000 rad/yr

the open water. Also, no account was taken of the reduction in corrosion rate in the deep ocean due to lower oxygen levels or the reduced temperatures. These factors indicate that if a rigorous calculation were possible, the resultant dose would be lower, and likely significantly lower.

Once out of the cask, the fission products are unlikely to be transported very far in the very slow current typical in the deep ocean. While this would concentrate the dose to those organisms in the area of the cask, especially the mollusks, it also means that the population affected would be relatively small, since only a small area would be contaminated.

Additionally, as explained in Chapter 3, the density of organisms in the deep ocean is around one percent that in coastal waters. This further reduces the affected population of organisms.

The risks associated with the dose estimated for the mollusk are very low, due to the low frequency of the event, as explained in the following section.

C.5.5 Risks Associated With Submersion of a Foreign Research Reactor Spent Nuclear Fuel Cask

Risks associated with submersion of foreign research reactor spent nuclear fuel casks were calculated for a single cask, even though more than one cask may be carried on some voyages. The risk (consequences multiplied by probability) is essentially independent of the number of casks carried per voyages. That is, the risk associated with eight voyages of one cask each are essentially the same as one voyage carrying eight casks.

The consequence estimates in Tables C-15 and C-16 are indicative of what could happen in the event that a foreign research reactor spent nuclear fuel cask were to become submerged in coastal waters or in the deep ocean and is not recovered. By combining an estimate of the frequency at which such a situation is

expected to occur with the consequence estimates, an estimate of the risk associated with ocean transportation can be developed. The frequency of a cask becoming submerged is: the mathematical product of the annual frequency of foreign research reactor spent nuclear fuel shipments, the probability that a shipment is involved in an accident, the probability that a ship sinks (given that an accident occurs), and the probability that a submerged cask is not recovered. Additionally, the frequency of a damaged cask becoming submerged in coastal waters includes the probability that a cask is damaged given that an accident occurs. The data for these events were taken from two sources, the Nuclear Energy Agency study (NEA, 1988) and the Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994). These data are summarized in Table C-17.

Table C-17 At Sea Risk Assessment Data

Parameter	*Coastal Data*	*Deep Ocean Data*
Shipment Accident Rate	0.00032/Shipment (DOE, 1994)	0.000046/Shipment (NEA, 1988)
Probability that Cask is Damaged, Given an Accident	0.002 (DOE, 1994)[a]	1.0[c]
Probability that a Ship Sinks Given an Accident	0.001 (Wheeler, 1994)	0.001 (Wheeler, 1994)
Probability that a Submerged Cask is not Recovered	0.0001 (NEA, 1988)[b]	0.05 (NEA, 1988)
Number of Shipments	721	721
Probability - Submerged Cask, Damaged, Unrecovered	4.6×10^{-11}	0.0000017
Probability - Submerged Cask, Undamaged, Unrecovered	2.3×10^{-8}	0.0c

[a] *This value represents the conditional probability that the severity of an accident is greater than Category II, as shown in Appendix E, Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994).*

[b] *Derivation of this probability is based in a fault tree analysis using data from the Nuclear Energy Agency analysis.*

[c] *The cask is assumed to fail at deep ocean depths.*

The risk estimate results for the basic implementation of Management Alternative 1 are shown in Table C-18. The risk for a peak dose to an individual is 6.4×10^{-7} mrem per year for a damaged cask in coastal water and 0.0000043 mrem per year for an undamaged cask. Risk associated with a submerged, unrecovered cask in the deep ocean is 0.00019 mrem per year for a damaged cask.

Table C-18 Radiological Risk Estimates for At Sea Accidents

Dose Category	*Damaged Cask*	*Undamaged Cask*
Coastal Dose Rate Risk Estimates		
Peak Individual Dose	6.4×10^{-7} mrem/yr	0.0000043 mrem/yr
Peak Biota Dose (Fish)	2.8×10^{-8} mrad/yr	0.0000018 mrad/yr
Peak Biota Dose (Crustaceans)	3.0×10^{-8} mrad/yr	0.0000019 mrad/yr
Peak Biota Dose (Mollusks)	6.4×10^{-7} mrad/yr	0.0000048 mrad/yr
Deep Ocean Risk Estimates		
Peak Individual Dose	0.00019 mrem/yr	Cask is assumed to fail at deep ocean depths
Peak Biota Dose (Fish)	1.1 mrad/yr	Cask is assumed to fail at deep ocean depths
Peak Biota Dose (Crustaceans)	1.4 mrad/yr	Cask is assumed to fail at deep ocean depths
Peak Biota Dose (Mollusks)	49 mrad/yr	Cask is assumed to fail at deep ocean depths

C.5.6 Marine Accident Impacts of Policy Alternatives

In Section C.4.2, two implementation subalternatives to Management Alternative 1 and one implementation subalternative to Management Alternative 2 of the proposed action that could impact the risk calculations were identified: accepting spent nuclear fuel from developing countries only, a 5-year

acceptance program, and overseas reprocessing of the foreign research reactor spent nuclear fuel. Implementation of any of these has the potential to impact the marine accident risks calculated for the basic implementation of Management Alternative 1 calculated above.

For the implementation subalternatives involving the shipment of different quantities of foreign research reactor spent nuclear fuel, the consequences of an accident are the same for the implementation subalternatives as they are for the basic implementation. In these subalternatives, the same type of spent nuclear fuel is being shipped in the same types of transportation casks and is subject to the same accidents as for the basic implementation. These are the variables between subalternatives that could have affected the consequences of a marine accident. Since none changed, the consequences do not change. Two of the implementation subalternatives fall into this category: the developing countries implementation subalternative and the five-year policy duration implementation subalternative. For these two alternatives, the marine accident risks are directly proportional to the number of foreign research reactor spent nuclear fuel shipments required to implement each implementation subalternative. It is therefore possible to scale the results presented in the previous section by the ratio of the number of cask shipments in implementation subalternative to the number of cask shipments in the basic implementation.

Subalternative 1b to Management Alternative 2 requires the shipment of the foreign research reactor spent nuclear fuel wastes in a different form than the basic implementation. With overseas reprocessing of the foreign research reactor spent nuclear fuel, any material that would be returned to the United States would be in the form of vitrified high-level waste. As discussed earlier in Section C.5, the high-level waste behaves differently when exposed to seawater than does spent nuclear fuel. The vitrified waste dissolves at a much slower rate than the foreign research reactor spent nuclear fuel. A second major difference is the amount of radioactivity present in each of the shipping casks carrying vitrified waste and spent nuclear fuel. As shown in Table C-14, the total curie content of a transportation cask carrying foreign research reactor spent nuclear fuel is approximately a million curies. Each vitrified waste transportation cask could contain approximately a hundred times this amount. The contents of 837 spent nuclear fuel transportation casks (all foreign research reactor spent nuclear fuel could be processed, including that from Canada, which was not included in the marine risk analyses for the basic implementation) are expected to be reduced to fit into eight transportation casks.

In addition, a Hybrid Alternative has been analyzed to assess the impact of encouraging overseas reprocessing of the foreign research reactor spent nuclear fuel for those countries capable of storing the resultant high-level waste. The United States would accept for management the research reactor spent fuel from countries that are unable to accept and store the high-level waste resulting from fuel processing. Under the Hybrid Alternative analyzed, Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom are assumed to process their aluminum-based spent nuclear fuel and accept the high-level waste. All other countries identified in Table C-1 would be allowed to ship spent nuclear fuel to the United States. The resulting 452 shipments of spent nuclear fuel (excluding the overland transport of fuel of Canadian origin) are the basis for the marine impact analysis for this Hybrid Alternative.

The marine accident risks associated with each of these management alternatives is presented in the following paragraphs.

Management Alternative 1, Implementation Subalternative 1a — Acceptance of Foreign Research Reactor Spent Nuclear Fuel Only from Developing Countries: This implementation subalternative would result in the shipment of 168 transportation casks of foreign research reactor spent nuclear fuel. This is 23 percent of the shipments required for the basic implementation. Using this relationship, the risks presented in Table C-18 can be scaled to produce the following results. The maximally exposed individual (MEI) would be exposed to a risk (in terms of a peak individual dose rate) of 0.000044 mrem per year as a result

of an accident causing the loss of a cask in the deep ocean. The consequences of this accident do not change; the peak individual dose remains at 0.114 rem per year. The loss of a damaged cask in coastal waters results in the lowest risk to man, 1.5×10^{-7} mrem per year. The risks to marine biota are reduced by the same ratio and will range from a high of 11 mrad per year to a mollusk from the loss of a cask in the deep ocean, to a low of 6×10^{-9} mrad per year to fish from the loss of a damaged cask in coastal waters.

Management Alternative 1, Implementation Subalternative 2a — Acceptance of Foreign Research Reactor Spent Nuclear Fuel for 5-Year Policy Duration: This implementation subalternative results in the shipment of 586 transportation casks of foreign research reactor spent nuclear fuel. This is 81 percent of the shipments required for the basic implementation. Using this relationship, the risks presented in Table C-18 can be scaled to produce the following results. The MEI will be exposed to a risk (in terms of a peak individual dose rate) of 0.00015 mrem per year as a result of the accident causing the loss of a cask in the deep ocean. The loss of a damaged cask in coastal waters results in the lowest risk to man, 5×10^{-7} mrem per year. The risks to marine biota are reduced by the same ratio and will range from a high of 40 mrad per year to a mollusk (deep sea accident) to a low of 2×10^{-8} mrad per year to fish (coastal water, damaged cask accident).

Management Alternative 2, Subalternative 1b — Overseas Processing with Shipment of Waste to a U.S. Storage Facility: In this subalternative, all of the foreign research reactor spent nuclear fuel (including that generated in Canada) is sent to either Great Britain or France for processing and the vitrified high-level waste generated in the process would be shipped to the United States. Based on the processing of approximately 23 metric tons (25.3 tons) of spent nuclear fuel, enough vitrified high-level waste would be generated to require up to eight transportation casks of vitrified high-level waste being shipped to the United States. Only the impact of the marine shipments from the processing facility to the United States was calculated.

The consequences of an accident at sea that results in the loss of a transportation cask filled with vitrified high-level waste can be derived from the information used to develop the marine accident consequences for a foreign research reactor spent nuclear fuel cask. The consequences listed in Tables C-15 and C-16 for D_0 represent the consequences associated with the loss of 100,000 MTHM equivalent of vitrified high-level waste. Based on eight shipments for the approximately 23 metric tons (25.3 tons) of spent nuclear fuel, each shipment in this subalternative will contain approximately 2.9 metric tons (3.2 tons) equivalent of vitrified high-level waste. Table C-19 presents the consequences from Tables C-15 and C-16 scaled to represent the consequences for an accident resulting in the loss of a transportation cask containing 2.9 metric tons (3.2 tons) equivalent.

Table C-19 Consequences Resulting from the Loss of a Transportation Cask Containing Vitrified High-Level Waste[a]

	Coastal Waters		*Deep Ocean*
	Undamaged Cask	*Damaged Cask*	*Damaged Cask*
Peak Individual Dose (Man) rem/yr	0.0003	0.019	1.5×10^{-8}
Peak Biota Dose (Fish) rad/yr	0.0001	0.0008	0.9
Peak Biota Dose (Crustaceans) rad/yr	0.0001	0.0009	1.2
Peak Biota Dose (Mollusks) rad/yr	0.0003	0.019	41

[a] *These estimates are based on the best estimate values presented in the Nuclear Energy Agency report (NEA, 1988)*

From the accident frequency data in Table C-17, a per-shipment accident frequency can be developed for all three accidents of interest: 1) the loss of an undamaged cask in coastal waters, 2) the loss of a damaged cask in coastal waters, and 3) the loss of a damaged cask in the deep ocean. These frequencies are the product of the shipment accident rate, the probability of the vessel sinking after an accident, the probability that a submerged cask is not recovered, and where applicable (for the damaged cask in coastal waters only), the probability that the cask is damaged in the accident. The resulting per shipment accident probabilities are 3.2×10^{-11} for the loss of an unrecovered, undamaged cask in coastal waters, 6.4×10^{-14} for the loss of an unrecovered damaged cask in coastal waters, and 2.3×10^{-9} for the unrecovered loss of a damaged cask in the deep ocean.

With the assumption that there are only up to eight shipments of vitrified high-level waste, the risks associated with the marine transport of this material are almost non-existent. The risks in terms of rem per year peak public dose and rad per year peak dose to marine biota, of an unrecovered cask in coastal waters are essentially zero, less than 1.0×10^{-10}. The risks calculated for the deep ocean accidents are: much less than 1×10^{-10} rem per year peak dose to man, 2×10^{-8} rad per year peak dose to fish and crustaceans, and 7×10^{-7} rad per year peak dose to mollusks.

Management Alternative 3 — Combination of Components of Management Alternatives 1 and 3 (Hybrid Alternative): Under the Hybrid Alternative, the United States would accept foreign research reactor spent nuclear fuel from countries unable to store high-level waste. This Hybrid Alternative could result in the shipment of 452 transportation casks of foreign research reactor spent nuclear fuel to the United States. This is approximately 63 percent of the shipments required in the basic alternative. Using this relationship, the risks presented in Table C-18 can be scaled to produce the following results. The MEI will be exposed to a risk (in terms of a peak individual dose rate) of 0.00012 mrem per year as a result of an accident causing the loss of a cask in the deep ocean. The consequences of this accident do not change from the basic implementation; the peak individual dose remains at 0.114 mrem per year. The loss of a damaged cask in coastal waters results in the lowest risks to man, 4×10^{-7} mrem per year. The risks to marine biota are reduced by the same ratio and will range from a high of 31 mrad per year to a mollusk (deep sea accident) to a low of 1.8×10^{-8} mrad per year to fish (coastal water, damaged cask accident).

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FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix D
Selection and Evaluation of Potential Ports of Entry

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Appendix D
Selection and Evaluation of Potential Ports of Entry

This appendix describes the process used by the Department of Energy (DOE) in selecting the potential ports of entry analyzed in this Environmental Impact Statement (EIS). In addition, the appendix provides the basic information required to evaluate ports and port activities, and the potential environmental impacts (incident-free and accidents) associated with the receipt and handling of foreign research reactor spent nuclear fuel from vessels to intermodal transport in ports.

D.1 Ports of Entry Selection Process

The adopted port selection process was based on a set of criteria developed by DOE to identify those ports that would be most capable of providing for the safe receipt, handling, and transhipment of foreign research reactor spent nuclear fuel. This appendix first describes the process through which DOE developed the port selection criteria, and then describes the application of the criteria, resulting in the identification of the specific ports available for consideration.

Because the basic implementation of Management Alternative 1 of the proposed action would involve shipments from many foreign countries to several potential foreign research reactor spent nuclear fuel management sites in addition to those identified in the Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994d), it was necessary to expand on the port analysis approach used in the Urgent Relief Environmental Assessment. The Urgent Relief Environmental Assessment was concerned with fewer shipments within a short timeframe, with the shipments going only to the Savannah River Site. Also, as stated in the Urgent Relief Environmental Assessment, this EIS considers future population trends and future port developments.

Independent maritime safety experts consulted during the preparation of this EIS informed DOE that any modern breakbulk or container terminal can accommodate the safe receipt, handling, and transhipment of foreign research reactor spent nuclear fuel in approved shipping casks (USMMA, 1994). This view is supported by the well-documented excellent safety record for shipping foreign research reactor spent nuclear fuel in the United States (NRC, 1993). In addition, the port selection criteria discussed in this appendix, taken collectively, provide a reasonable additional basis for identifying those candidate ports suitable for the safe receipt and handling of foreign research reactor spent nuclear fuel.

D.1.1 Background

Since 1979, when the U.S. Nuclear Regulatory Commission (NRC) first began approving spent nuclear fuel shipments in the United States, 317 spent nuclear fuel shipments in Type B casks have been transported safely into several U.S. ports of entry. These include Newport News, Norfolk, and Portsmouth, VA; Portland, OR; Savannah, GA; and Oakland, CA (NRC, 1993). However, prior to the fall of 1993, DOE did not have any generally applicable criteria for identifying ports of entry. For this EIS, as well as for the Urgent Relief Environmental Assessment, DOE developed criteria to identify candidate ports of entry. The criteria used in this evaluation to identify potential ports for the receipt of foreign research reactor spent nuclear fuel are based on consideration of several independent factors, each described in more detail in the following sections.

D.1.2 Information on Past Spent Nuclear Fuel Shipments

The NRC has the authority under the Atomic Energy Act of 1954, as amended, to regulate certain aspects of spent nuclear fuel transportation. Of the thousands of shipments completed over the last 30 years in the United States and abroad, none has resulted in an injury due to the radioactive nature of the cargo (NRC, 1993). For the same period, about 1,200 (924 domestic and 293 foreign) overland shipments of spent nuclear fuel took place without any injury attributable to accidents or incident-free radiation doses during transport. Table D-1 provides the number of NRC and Department of Transportation regulated international and domestic overland shipments since 1979 (excluding DOE shipments). The casks that would be used in this program are robust Type B containers. The safety, safeguards, and precautions used for such shipments have historically been very successful (NRC, 1993).

Table D-1 Number of NRC/Department of Transportation Regulated Overland and International Spent Nuclear Fuel Shipments

Year	Overland		International[a]		
	Highway	Railway	Export	Import	Transient
1979	2	11	0	14	0
1980	73	5	2	55	0
1981	30	2	3	48	0
1982	80	0	1	43	0
1983	92	0	2	23	0
1984	209	3	2	34	0
1985	114	18	0	21	0
1986	88	15	0	17	0
1987	85	15	3	19	0
1988	10	7	0	15	0
1989	11	6	1	4	0
1990	0	8	2	0	3
1991	7	10	4	0	1
1992	17	6	0	0	0
Total	**818**	**106**	**20**	**293**	**4**

[a]*Ports included Newport News, Norfolk, and Portsmouth, VA; Savannah, GA; Portland, OR; and Oakland, CA.*

D.1.3 Federal Court Ruling

In the December 1991 decision of the U.S. District Court, District of Columbia Circuit, on the return of spent nuclear fuel from Taiwan, the court ruled that DOE must consider a reasonable range of alternative ports, including (at least) two low population density ports near DOE's Savannah River Site (U.S. District Court, 1991). In this appendix, DOE has identified a reasonable range of alternative ports on the East, Gulf, and West Coasts (including several low population ports) for the receipt, handling, and transhipment of foreign research reactor spent nuclear fuel to five potential DOE management sites (including the Savannah River Site) being considered in this EIS.

D.1.4 Notice of Intent Port Criteria

The Notice of Intent for this EIS (DOE, 1993) listed a series of preliminary criteria which might be applied to a potential list of candidate seaports to identify ports which would be acceptable for receipt, handling, and transhipment of foreign research reactor spent nuclear fuel. These proposed criteria included: "(a) adequacy of harbor and dock characteristics to satisfy the cask carrying ship requirements;

(b) availability of safe and secure lag storage; (c) adequacy of overland transportation systems from ports to the storage site(s); (d) experience in safe and secure handling of hazardous cargo; (e) emergency preparedness status at the port and nearby communities; and (f) proximity to the proposed storage sites." Either implicitly or explicitly, these criteria were considered in the port screening, as discussed in the following sections.

D.1.5 The U.S. Merchant Marine Academy Workshop Recommendations

A DOE-sponsored workshop on port selection criteria for spent nuclear fuel was held at the U.S. Merchant Marine Academy at Kings Point, New York, on November 15-16, 1993 (USMMA, 1994). Participants at the workshop included experts from the maritime industry in the areas of marine transportation, intermodal systems, marine insurance, admiralty law, U.S. Coast Guard Operations, U.S. Navy Operations, Military Sealift Command Operations, and national cargo, pilotage, and ships operations.

A series of panel discussions focused on issues such as economics and transportation safety, advantages of shipping spent nuclear fuel on various types of vessels, and shipping spent nuclear fuel through large versus small ports. The purpose of such discussions, in part, was to enable DOE to identify port criteria that would minimize both the actual and perceived risk involved in spent nuclear fuel shipments. The workshop participants agreed that any port capable of handling an ocean-going vessel is capable of receiving spent nuclear fuel. While some of these ports might have features that would make them more desirable than others (e.g., easy access from the open sea, modern facilities, etc.), no port would have such limitations as to preclude safe receipt of the spent nuclear fuel. While individual ports might not satisfy all the criteria recommended at the workshop, the workshop participants concluded that the criteria would provide a means of evaluating the relative merits of ports.

The three criteria recommended as necessary for safe shipment were: short distance from the open ocean to the port, adequate port cargo facilities, and intermodal access (i.e., for truck or rail shipments from the port to the management site).

A second set of recommendations that were listed as "important but not necessary" included: an experienced risk management staff, emergency preparedness and response capabilities, a skilled labor force aboard ship and in port, good port security, no local restrictions or regulations on movement of hazardous cargo, and no significant environmental considerations for the port.

Finally, the workshop also provided a list of "desirable" attributes for ports, including: distance of the port from a population center, proximity of the port to a spent nuclear fuel management location, "local economic issues" (e.g., areas that receive a significant fraction of their revenues from maritime and shipping activities), and personnel with training and experience in radioactive shipments and incident response.

D.1.6 Provisions of the National Defense Authorization Act for Fiscal Year 1994

On November 30, 1993, the National Defense Authorization Act for Fiscal Year 1994 was signed into law (NDAA, 1993). Section 3151 stipulates specific criteria that must be used "if economically feasible" and "to the maximum extent practicable" in selecting U.S. ports for both emergency and nonemergency receipt of foreign research reactor spent nuclear fuel at the Savannah River Site. Although the National Defense Authorization Act does not specifically address other potential DOE management sites, DOE assumed that the guidance provided for foreign research reactor spent nuclear fuel shipments to Savannah

River Site should be considered for the other four potential sites being considered in this EIS (Idaho National Engineering Laboratory, Hanford Site, Oak Ridge Reservation, and the Nevada Test Site), to the extent feasibility and practicability permitted.

Specifically, the National Defense Authorization Act requires that DOE may not receive foreign research reactor spent nuclear fuel if it "cannot be transferred in an expeditious manner from its port of entry in the United States to a storage facility that is located at a Department of Energy facility and is capable of receiving and storing the spent nuclear fuel." Further, it requires that the "Secretary of Energy shall, if economically feasible and to the maximum extent practicable, provide for the receipt of spent nuclear fuel....at a port of entry in the United States which...compared to each other port of entry....that is capable of receiving the spent nuclear fuel - (1) has the lowest human population in the area surrounding the port of entry; (2) is closest in proximity to the facility which will store the spent nuclear fuel; and (3) has the most appropriate facilities for, and experience in, receiving nuclear fuel (NDAA, 1993)."

D.1.7 Comments Received During the EIS Scoping Meetings and on the Urgent Relief Environmental Assessment

Nine public scoping meetings were held in November and December, 1993, at six cities being considered as potential ports for the receipt of spent nuclear fuel from foreign research reactors, and four cities near the potential spent nuclear fuel management sites discussed in this EIS. As a result of these meetings, DOE received several groups of similar comments, which have been incorporated into the development of the criteria (DOE, 1994a).

The largest number of comments (44) received on any general port-related issue dealt with avoiding ports in high population areas. Reasons ranged from concerns about accident consequences and possible terrorist attacks, to concerns about the ability to adequately respond to emergencies and possible evacuation of populations.

The second largest number of comments (32) suggested that alternative ports in low-population areas or ports operated by the military be seriously considered, and that ports that are closest to the storage sites and/or have the most direct transportation routes between the ports and management sites be considered.

Other comments that fall within the jurisdiction of DOE and within the scope of this EIS include: suggestions that selected ports should have experience handling spent nuclear fuel (9 comments); the safest marine terminals should be used at the port selected (3 comments); and that DOE should allow case-by-case designation of ports based on the most sensible options at the time each individual shipment occurs, considering the vessel, country of origin, time, cost, and overall experience of the ports (2 comments).

In addition to comments presented at the EIS Scoping Meetings, DOE has also considered individual comments and a list of suggestions from the Sierra Club on the draft Urgent Relief Environmental Assessment (DOE, 1994d).

D.1.8 Key Assumptions and Methodology for Port Identification

A number of possible maritime shipment modes are potentially available for shipping the foreign research reactor spent nuclear fuel over the next 10 or more years. The various transport modes generally determine which port facilities are adequate at each specific port [e.g., container cranes are required for container vessels, a pier for roll-on/roll-off vessels, and breakbulk cranes for breakbulk vessels]. While regularly scheduled cargo ships servicing commercial ports could be an important mode selected by

owners of the foreign research reactor spent nuclear fuel for their shipments, smaller unscheduled vessels would also be a common mode of transport for multiple cask shipments (e.g., the first shipment of foreign research reactor spent nuclear fuel under the Urgent Relief Environmental Assessment in September 1994). This means that there will be a somewhat greater number of potential ports of entry to consider than if only larger, regularly scheduled commercial container vessels were to be used (details on potential vessel types that might be used are provided in Appendix C).

In addition to the types of vessels that could be used, the way foreign research reactor spent nuclear fuel casks are "packaged" for shipment is also a determinant in the selection of potential ports. For the Urgent Relief Environmental Assessment shipments to Savannah River Site, the Terms and Conditions for Financial Settlement for Receipt and Disposition of Foreign Research Reactor Spent Fuels (DOE, 1994c) required that spent nuclear fuel casks be containerized in 20 ft International Standards Organization containers (nominally, 2.4 m x 2.4 m x 6.1 m, or 8 ft x 8 ft x 20 ft), also called 20-ft equivalent units. Therefore, it was assumed that spent nuclear fuel casks would only be shipped containerized. This eliminates consideration of receipt and handling of foreign research reactor spent nuclear fuel casks in a "palletized" mode. Thus, the EIS focuses primarily on reasonable options for ports qualified for the receipt, handling, and transhipment of containerized spent nuclear fuel on any viable vessel type.

Among the ports that routinely handle containerized freight, two groups of ports - those along the in-land Mississippi River (above New Orleans) and those around the Great Lakes - are not considered in this evaluation. Access to these ports requires a long inland transit from open ocean. The U.S. Merchant Marine Academy recommendations discouraged such transits.

Finally, since the National Defense Authorization Act did not establish numerical distance or transport time limits for spent nuclear fuel transport, DOE concluded that, consistent with current and past Federal practice for transport of spent nuclear fuel in the contiguous United States, all overland shipments should be managed such that the spent nuclear fuel is kept moving as expeditiously as possible from the time it is placed on the transportation vehicle at the port of entry until it reaches the DOE management site, to the maximum extent practicable. For example, truck shipments (which typically involve two drivers in a tractor with a sleeping area) are assumed to be basically nonstop in order to deliver the spent nuclear fuel promptly, stopping only for fuel and food. This has been, and is expected to remain, DOE practice for such shipments.

NRC recently reported that for the period 1979-1992, rail transport only accounted for 8.6 percent of the total spent nuclear fuel shipments in the United States, but these shipments accounted for about 66 percent of the total quantity of spent nuclear fuel shipped (NRC, 1993). Rail travel (freight) typically takes much longer than truck transport when moving spent nuclear fuel on a dedicated railcar, where even short trips may require movement through additional intermodal terminals (e.g., transfer from rail to truck for site delivery), or intermediate points of dedicated railcar transfers to other train systems (e.g., from a local freight handler to one or more long distance freight lines). However, in the case of dedicated trains where entire multiple cask shipments (such as those used for the Urgent Relief shipment from the Military Ocean Terminal at Sunny Point (MOTSU) to the Savannah River Site) go directly from the port to the management site, rail travel times are expected to be somewhat longer than those for truck transport. Generally, rail distances are also typically somewhat longer than those for trucks using interstates, and rail transport generally costs more, and potentially exposes larger numbers of people since transits typically pass through major railyards in inner cities (see Appendix E for comparative travel distances for truck and rail).

In both cases, DOE concludes that by proper planning and compliance with current Department of Transportation and NRC shipment requirements (including use of pre-approved routes), each shipment of foreign research reactor spent nuclear fuel could be moved expeditiously from each port to each management site, and specific distance and time considerations do not serve to usefully discriminate against ports in the contiguous 48 States.

D.1.9 Methodology for Port Selection

The methodology for identifying acceptable ports of entry began with a list of 153 commercial ports throughout the contiguous United States. These ports included the 151 ports that were originally considered in the Urgent Relief Environmental Assessment (DOE, 1994d). The two additional commercial ports are Eddystone, PA, and Fernandina Beach, FL. Also, eight additional military ports in the contiguous United States suggested by the Military Traffic Management Command (MTMC, 1994a) were evaluated. The eight candidate military ports were those believed to routinely handle dry containerized cargoes (largely munitions), on breakbulk, container, and/or roll-on/roll-off vessels. Military ports are subject to extreme fluctuations in port activities as a function of national need. By using the criteria described below, ports that did not meet each DOE mandatory criterion in the sequence were eliminated. Those ports not eliminated at each step of the screening process were then evaluated in the same fashion against the remaining required criteria.

The required screening criteria DOE used to identify potential ports of entry are:

- The ports must have appropriate (routine) experience handling containerized cargo (Criterion 1);
- The ports must offer favorable transits from the open ocean to the selected terminals (Criterion 2);
- The ports must have appropriate facilities for safe receipt, handling, and transhipment of foreign research reactor spent nuclear fuel (Criterion 3);
- The ports must have ready access for intermodal transport (i.e., truck and rail facilities at or close to the selected terminal) (Criterion 4); and
- The human population of the ports and along transportation routes must be low to the extent economically feasible and maximum extent practicable (Criterion 5).

In selecting the final list of seaports from those found acceptable under Criterion 5, DOE applied several desirable port attributes. The potentially most useful of these ports for receipt, handling, and transhipment of foreign research reactor spent nuclear fuel to any of the five DOE management sites, and which also had the highest number of other desirable attributes, were selected for consideration and detailed analysis in the EIS.

D.1.9.1 Criterion 1: Appropriate Port Experience

The first criterion selected is one of the National Defense Authorization Act requirements for using ports with appropriate experience. The criterion is used first because if a port does not currently have appropriate container handling experience, or is unlikely to have this experience during the time period analyzed in this EIS, there is no reason to consider it further. For this screening, commercial ports that handle on the order of at least 20,000 20-ft equivalent units of containerized cargo per year [i.e., any mix

of breakbulk, combination breakbulk/container ships, or self-contained ships that are equivalent to unloading (or loading) a small container vessel every week or two] were selected for further detailed analysis under the remaining criteria.

Because containerized spent nuclear fuel requires no special port experience or facilities specific to the handling of radioactive material, ports were not eliminated from consideration because of lack of such experience or facilities.

This criterion excludes experience in handling bulk liquid cargoes (e.g., oil or petrochemicals) or other bulk cargoes (e.g., grain, coal, etc.) unloaded using special cargo equipment not of the type used for receipt and handling of containerized foreign research reactor spent nuclear fuel shipments. It also excludes ports used primarily by fishing fleets or cruise ship liners.

Ports meeting the appropriate experience requirement would be those where port terminal(s) and operators routinely load and/or unload all types of containerized dry cargoes requiring the same type of handling as containerized spent nuclear fuel (e.g., everything from television sets and machine parts to toxic materials, flammable or explosive cargoes, etc.), or are likely to acquire such experience during the time period analyzed in this EIS (i.e., large cargo or container port expansions or improvements are planned within the next several years). DOE found that the status of commercial port facilities is very dynamic and subject to rapid and unpredictable changes. For example, the Port of San Francisco, CA lost four of its five major container lines to the Port of Oakland, CA early in 1994, and the Port of Morehead City, NC, has gone from on the order of 10,000 containers per year a few years ago to essentially no container service at the present time (DOE, 1994d). Similarly, the Port of Richmond, CA (while it still has two container cranes available and acceptable facilities) no longer receives significant numbers of containers (AAPA, 1994), although that could change in the near future.

This criterion also effectively eliminated ports that have infrequent container/breakbulk ship calls, marginal equipment or facilities, and were less likely to have well-trained and experienced personnel than busier ports during the period analyzed in this EIS (adequacy of ports and facilities for receiving, handling, and transhipping such cargoes will be addressed in Section D.1.9.3).

Out of the original list of 153 commercial candidate ports in the contiguous United States that were discussed earlier (excluding the 29 Great Lakes and upper Mississippi River ports), this screening resulted in the identification of 31 candidate seaports (see Table D-2 and Figure D-1). Many of the rejected ports were associated with oil or other bulk shipments, and were not viable for either breakbulk or container operations. These 31 commercial ports are considered to be reasonably representative of the total population of viable commercial seaports in the contiguous United States. Three of the eight military ports evaluated were found to generally satisfy this criterion, allowing for the cyclical nature of military activities at these ports (see Figures D-1 and D-2 and Table D-3). The acceptable military ports included the Military Ocean Terminal Bay Area in Oakland, CA, and the Naval Weapons Station (NWS) in Concord, CA, as potential West Coast ports of entry, and MOTSU for a potential East Coast port of entry. This criterion screened out all naval bases and shipyards in the contiguous United States because they do not regularly handle containerized cargo from ocean-going vessels in any significant quantity.

There is great uncertainty associated with attempts to project the future of port activities and possible availability for receipt, handling, and transhipment of foreign research reactor spent nuclear fuel. Many of the features and facilities of ports addressed in Criterion 3 are inextricably related to the likelihood that any given port will meet the minimum requirements for "appropriate" experience in the future. Thus, for example, if a specific port lacks adequate facilities and equipment at present, and there is no identifiable intention of improving the port in the future, it is unlikely that the port will develop the appropriate

Table D-2 Commercial Ports with Appropriate Experience Receiving, Handling, and Transhipping Containerized Dry Cargoes[a]

U.S. Seaport	Appropriate Experience	U.S. Seaport	Appropriate Experience	U.S. Seaport	Appropriate Experience
Alameda, CA	No	Gloucester City, NJ	No	Port Angeles, WA	No
Albany, NY	No	Gramercy, LA	No	Port Arthur, TX	No
Alexandria, VA	No	Grays Harbor, WA[c]	No	Port Canaveral, FL	No
Anacortes, WA	No	Green Bay, WI	NA	Port Costa, CA	No
Antioch, CA	No	Gulfport, MS	Yes	Port Everglades, FL	Yes
Ashtabula, OH	NA	Hopewell, VA	No	Port Hueneme, CA	No
Astoria, OR	No	Houston, TX	Yes	Port Manatee, FL[c]	No
Baltimore, MD	Yes	Huntington Beach, CA	No	Port Neches, TX	No
Baton Rouge, LA	NA	Huron, OH	NA	Port Royal, SC	No
Bay City, MI	NA	Jacksonville, FL	Yes	Port San Luis, CA	No
Beaumont, TX[c]	No	Kalama, OR	No	Port Sulfur, LA	No
Bellingham, WA	No	Kenosha, WI	NA	Port St. Joe, FL	No
Benicia, CA	No	La Place, LA	No	Port Townsend, WA	No
Boston, MA	Yes	Lake Charles, LA	Yes	Portland, OR	Yes
Bridgeport, CT	No	Long Beach, CA	Yes	Portland, ME[c]	No
Brownsville, TX	No	Longview, WA[c]	No	Portsmouth, NH[c]	No
Brunswick, GA	No	Lorain, OH	NA	Portsmouth, VA	Yes
Buffalo, NY	NA	Los Angeles, CA	Yes	Providence, RI	No
Burns Harbor, IN	NA	Mandalay Beach, CA	No	Raymond, WA	No
Cambridge, MD	No	Manitowoc, WI	NA	Redwood City, CA	No
Camden, NJ[c]	No	Marcus Hook, PA	No	Reedsport, OR	No
Carlsbad, CA	No	Marine City, MI	NA	Reserve, LA	No
Carpinteria, CA	No	Miami, FL	Yes	Richmond, VA	Yes
Charleston, SC	Yes	Milwaukee, WI	NA	Richmond, CA[c]	No
Chicago, IL	NA	Mobile, AL[c]	No	Rochester, NY	NA
Cleveland, OH	NA	Morehead City, NC	No	Sacramento, CA	No
Conneaut, OH	NA	Moss Beach, CA	No	Saginaw, MI	NA
Coos Bay, OR	No	Muskegon, MI	NA	San Diego, CA[c]	No
Corpus Christi, TX[c]	No	New Bedford, MA	No	San Francisco, CA	Yes
Crescent City, CA	No	New Haven, CT	No	Sandusky, OH	NA
Crockett, CA	No	New London, CT	No	Savannah, GA	Yes
Delaware City, DE	No	New Orleans, LA	Yes	Searsport, ME	No
Detroit, MI	NA	New York, NY[b]	Yes	Seattle, WA	Yes
Duluth, MN	NA	Newport News, VA	Yes	Sheboygan, WI	NA
Eddystone, PA	Yes	Newport, OR	No	Stockton, CA	No
Edmonds, WA	No	Norfolk, VA	Yes	Superior, WI	NA
El Segundo, CA	No	Oakland,CA	Yes	Tacoma, WA	Yes
Erie, PA	NA	Ogdensburg, NY	NA	Taft, LA	No
Essexville, MI	NA	Olympia, WA	No	Tampa, FL[c]	No
Estero Point, CA	No	Orange, TX	No	Texas City, TX	No
Eureka, CA	No	Ostrica, LA	No	Toledo, OH	NA
Everett, WA	No	Oswego, NY	NA	Uncle Sam, LA	No
Fairport Harbor, OH	NA	Palm Beach, FL	Yes	Vallejo, CA	No
Fall River, MA	No	Panama City, FL	No	Vancouver, WA[c]	No
Ferndale, WA	No	Pascagoula, MS	No	Venice, LA	No
Fernandina Beach, FL	Yes	Paulsboro, NJ	No	Ventura, CA	No
Freeport, TX	Yes	Pensacola, FL	No	Willapa Bay, WA	No
Friday Harbor, WA	No	Philadelphia, PA	Yes	Wilmington, DE	Yes
Galveston, TX	Yes	Pilottown, LA	No	Wilmington, NC	Yes
Gaviota, CA	No	Pittsburgh, CA	No	Winslow, WA	No
Georgetown, SC	No	Point Wells, WA	No (Closing)		

[a] *For possible use by breakbulk, container, Roll-on/Roll-off or combination vessels. No Great Lakes ports or ports far up the Mississippi River were evaluated in detail because of the unacceptably long transits on crowded inland waterways or need for additional intermodal transfers (listed as "not applicable" or NA).*

[b] *Includes the preferred terminal at Elizabeth, NJ.*

[c] *Does have limited dry cargo facility; could acquire appropriate experience during 10 to 15-years.*

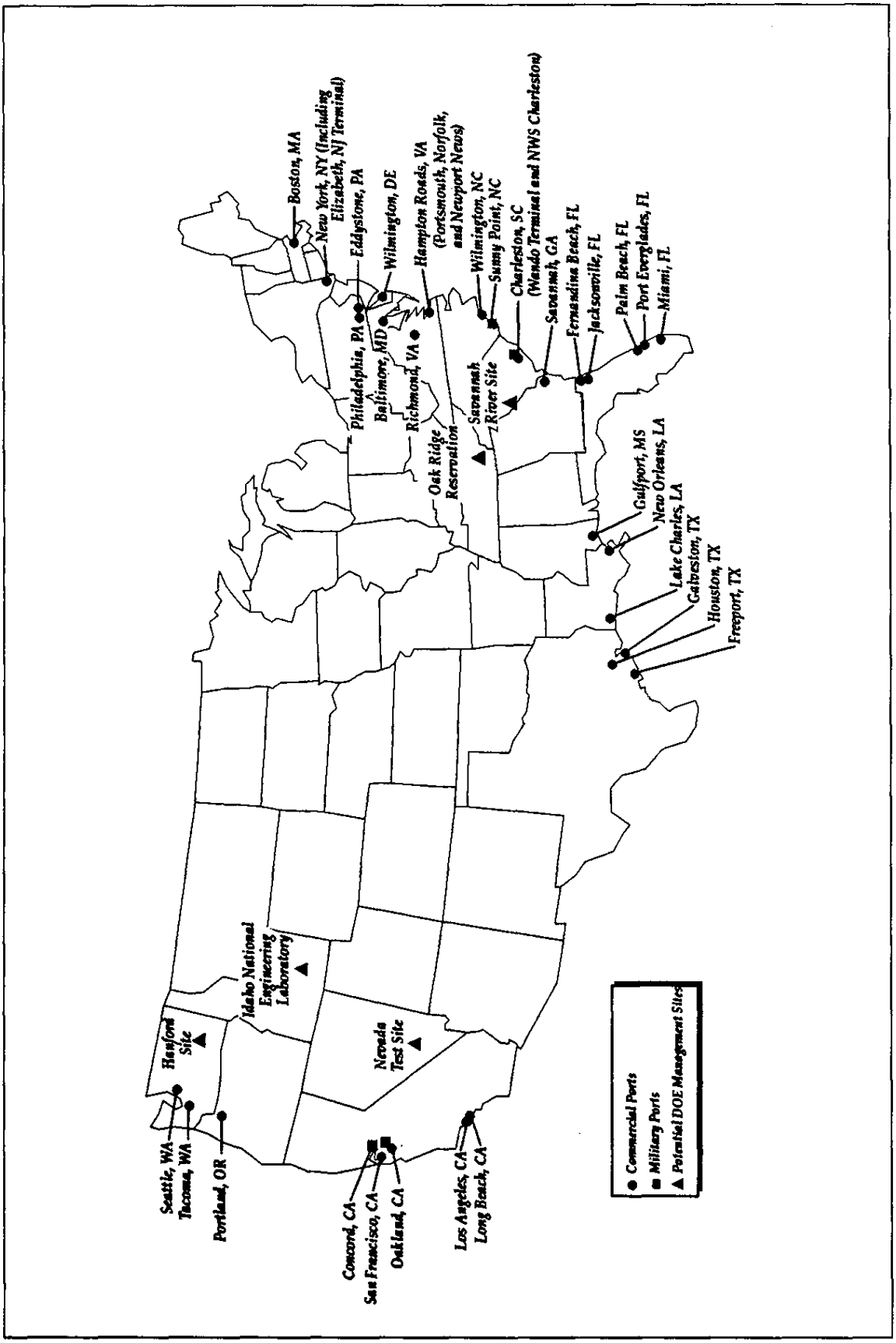


Figure D-1 Locations of Candidate Ports of Entry and DOE Management Sites

1 *Appropriate Port Experience:*

- Ports routinely handle containerized cargo (at least 20,000 TEUs/yr)

a. Accepted 31 Commercial Ports:

Baltimore, MD*
Boston, MA
Charleston, SC
Eddystone, PA*
Elizabeth, NJ
Fernandina Beach, FL
Freeport, TX
Galveston, TX
Gulfport, MS
Houston, TX
Jacksonville, FL
Lake Charles, LA
Long Beach, CA
Los Angeles, CA
Miami, FL
Newport News, VA*
New Orleans, LA
Norfolk, VA*
Oakland, CA*
Palm Beach, FL
Philadelphia, PA
Port Everglades, FL
Portland, OR*
Portsmouth, VA*
Richmond, VA
San Francisco, CA
Savannah, GA*
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

*b. Accepted 3 Military Ports**:*

Military Ocean Terminal
Sunny Point, NC*

Military Ocean Terminal
Oakland, CA

Naval Weapons Station
Concord, CA

** Database indicates Port has handled SNF or other Type B cask shipments*

*** Military ports meet 20,000 TEU requirement on a periodic basis, but cycle between high and low work loads based on military demands*

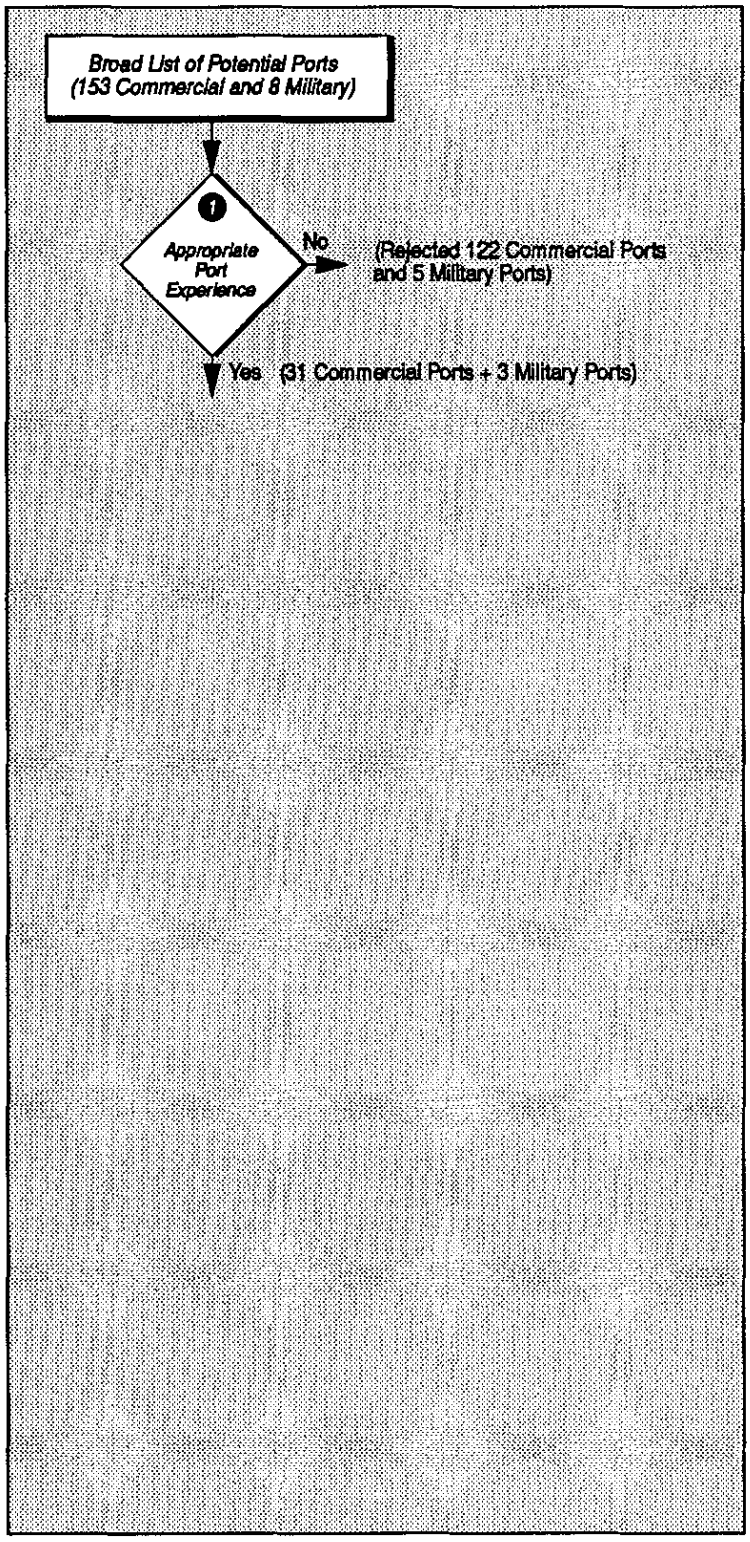


Figure D-2 Screening Ports with Appropriate Experience Criterion

Table D-3 Military Ports with Appropriate Experience Receiving, Handling, and Transhipping Containerized Dry Cargoes

U.S. Military Port	*Appropriate Experience*[a]
Bayonne, NJ (Military Ocean Terminal)	No
Cheatham Annex, VA (Navy)	No
Concord, CA (Naval Weapons Station)	Yes
Kings Bay, GA (Submarine Base)	No
Oakland, CA (Military Ocean Terminal)	Yes
Port Hueneme, CA (Naval Construction Battalion Center)	No
Sunny Point, NC (Military Ocean Terminal, MOTSU)	Yes
Yorktown, VA (Naval Weapons Station)	No

[a]*Military ports meet 20,000 twenty-foot equivalent units/yr requirement on a periodic basis and retain a corp of experienced port workers.*

Sources of information include: MTMCTEA, 1992 (Bayonne, NJ); MTMC, 1994a (Cheatham Annex, VA); Yocum, 1994a and 1994b (Concord, CA); FHI, 1993b (Kings Bay, GA); MTMC, 1994b; MTMCTEA, 1990 (Port Hueneme, CA); DOE, 1994d (Sunny Point, NC); and FHI, 1994a (Yorktown, VA).

experience required during the period analyzed in this EIS. As a result, DOE searched the available literature and scanned port-specific information from a number of sources (including direct discussions with numerous port officials) to identify planned future port improvements to see if that information could be used to tentatively increase the number of potential ports in subsequent port screenings for other necessary criteria. The results of this review are addressed further in Attachment D1 to this appendix. The basic finding was that all identifiable future port improvements were generally being made in ports that already meet the appropriate experience criterion (also see Attachment D1 to this appendix for projected port improvements that have been identified). As a result, that test for potential future port utilization did not yield any additional ports for subsequent screening.

On the other hand, DOE also found that some ports already have adequate facilities, but for one reason or another have been unable to attract enough container shipments to ensure a reasonable core of experienced workers (e.g., Port of Richmond, CA). In such cases, future projections are extremely uncertain, since there is no way of knowing whether ports will be successful in marketing new business opportunities in the future. As a result, DOE concluded that there was no useful purpose served in keeping these ports in the list for subsequent screening.

D.1.9.2 Criterion 2: Favorable Transit From Open Ocean

This criterion was based on recommendations from the U.S. Merchant Marine Academy Workshop participants who found that a short transit from the open ocean to port was necessary to maximize the safety of shipments of spent nuclear fuel (USMMA, 1994). However, it is clear that since this criterion focuses on ship safety, it is essentially synonymous with the requirement for a favorable ship transit from the open ocean to the port. Thus, while a port might be within a few miles of the open sea, if there were numerous hard shoals, ship wrecks, or reefs along the ship channel, this port might be less desirable than other ports with longer but less risky transits. On the other hand, ports that can only be reached by transporting spent nuclear fuel through long, narrow, winding, or crowded ship channels present additional risks that can be avoided by using ports that are easier to reach.

As a result, DOE concluded that ports meeting the intent of this criterion would have relatively short trips to port from large, deep bodies of water that were either oceans, seas, or notable extensions thereof, such as large bays or sounds (e.g., Chesapeake Bay, San Francisco Bay, or Puget Sound), and which present no

special navigational hazards to ships (including adequate width and depth of water in ship channels). A minimum channel depth (mean low water) of 7.6 m (25 ft) was selected to permit use by at least small to intermediate size vessels.

Less desirable were potential ports that could only be reached by traversing long, narrow and/or winding, or crowded ship channels [e.g., the St. Lawrence seaway to a Great Lakes port or the long passage up the Galveston/Houston ship channel to Houston (which is crowded by oil tankers in the channel and numerous petroleum and petrochemical plants along the channel that could impact on ship safety in the event of a plant or pipeline accident)].

Reliable data on risks associated with transits are difficult to find. In 1991, the U.S. Coast Guard established a national database on ship accidents. The 46 Code of Federal Regulations (CFR) §4.05-1 defines reportable accidents as those events that (1) leave a vessel damaged and presenting a navigational hazard (e.g., loss of propulsion or steering) or affect seaworthiness, (2) cause damage in excess of $25,000, or (3) result in serious injury or loss of life. Included in the database are allisions (single ship collisions with fixed structures such as buoys, docks, or bridges), collisions (between two vessels while under power), hard groundings (where a vessel cannot free itself), and fires onboard cargo vessels due to other accidents (USCG, 1994a and 1994b). However, since these accident statistics are not comprehensive and include barge accidents in addition to those involving ocean-going vessels, it is difficult to provide sound and reliable estimates of accident frequencies and types per transit to port.

Using all of the information currently available that pertains to analysis of this criterion, DOE found that the Ports of Richmond VA, New Orleans and Lake Charles, LA, and Houston, TX do not meet the criteria for receipt of foreign research reactor spent nuclear fuel at this time.

As shown in Table D-4 and Figure D-3, application of this criterion resulted in the retention of 27 commercial seaports and three military ports for further analysis.

D.1.9.3 Criterion 3: Appropriate Port Facilities

The National Defense Authorization Act requires the use of ports with "appropriate port facilities" that allow safe handling of foreign research reactor spent nuclear fuel. The U.S. Merchant Marine Academy Workshop recommended as "necessary for safe shipment" that an acceptable port have "adequate port cargo facilities," which included (1) berthing options (e.g., so that conflicting activities at an adjacent berth or onshore could be avoided if necessary), and (2) onsite cranes with trained operators (while it was recognized that ports without cranes could use other means to offload a vessel, the panel preferred ports with cranes).

Thus, port facilities must possess the following minimum physical attributes: (1) adequate water depths alongside piers [at least 7.6 m (25 ft) was selected for this screening] for docking at least small to intermediate-sized vessels, (2) adequate wharfs and quays, with berthing options (in case a potential for conflicting operations exists near the berth of choice), for securing vessels and safely offloading, and carrying the necessary spent nuclear fuel loads, and (3) at least one adequate crane for offloading containerized spent nuclear fuel onto ground transport [at least a 30 metric ton (33 ton) capacity crane was selected for this screening].

While many small ports have cranes with large lift capacities [100 metric tons (110 tons) or more], they are not purpose-built container cranes and must use special container spreaders for use with containers. Although the U.S. Merchant Marine Academy Workshop found it "desirable" for a port to have an adequate purpose-built container crane available, participants determined it was unnecessary to have one.

Table D-4 Required Maritime Transit Criterion for Selection of Seaports for Foreign Research Reactor Spent Nuclear Fuel Shipments

Seaports	*Distance from Open Sea (km)*[a]	*Favorable Transit*
	Commercial	
Baltimore, MD	240	Yes
Boston, MA	12	Yes
Charleston, SC	11	Yes
Eddystone, PA	120	Yes
Elizabeth, NJ	18	Yes
Fernandina Beach, FL	15	Yes
Freeport, TX	6	Yes
Galveston, TX	16	Yes
Gulfport, MS	30	Yes
Houston, TX	71	No
Jacksonville, FL	11	Yes
Lake Charles, LA	52	No
Long Beach, CA	4	Yes
Los Angeles, CA	5	Yes
Miami, FL	5	Yes
Newport News, VA	40	Yes
New Orleans, LA	160	No
Norfolk, VA	35	Yes
Oakland, CA	15	Yes
Palm Beach, FL	6	Yes
Philadelphia, PA	130	Yes
Port Everglades, FL	2	Yes
Portland, OR	140	Yes
Portsmouth, VA	40	Yes
Richmond, VA	190	No
San Francisco, CA	19	Yes
Savannah, GA	24	Yes
Seattle, WA	5	Yes
Tacoma, WA	5	Yes
Wilmington, DE	100	Yes
Wilmington, NC	38	Yes
	Military	
NWS Concord, CA	60	Yes
MOTBA, CA	15	Yes
MOTSU, NC	16	Yes

[a]*To convert distance to miles, divide by 1.6.*

Thus, while it is preferable to avoid any additional risks associated with the use of general purpose cranes (even though small) by using terminals with equipment designed to handle containerized cargo (an alternative to port container cranes might be the use of combination breakbulk/container vessels with shipboard container cranes that are generally operated by trained and experienced port stevedores), a purpose-built container crane was not determined to be necessary to satisfy this criterion. Military ports also represent a special case, since most do not have such purpose-built container cranes, and use a container spreader attachment when necessary.

2 *Favorable Transit to Port:*

- Port within reasonable distance from the open sea, with favorable transit

a. Accepted 27 Commercial Ports:

Baltimore, MD
Boston, MA
Charleston, SC
Eddystone, PA
Elizabeth, NJ
Fernandina Beach, FL
Freeport, TX
Galveston, TX
Gulfport, MS
Jacksonville, FL
Long Beach, CA
Los Angeles, CA
Miami, FL
Newport News, VA
Norfolk, VA
Oakland, CA
Palm Beach, FL
Philadelphia, PA
Port Everglades, FL
Portland, OR
Portsmouth, VA
San Francisco, CA
Savannah, GA
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. Accepted 3 Military Ports:

Military Ocean Terminal
Sunny Point, NC

Military Ocean Terminal
Oakland, CA

Naval Weapons Station
Concord, CA

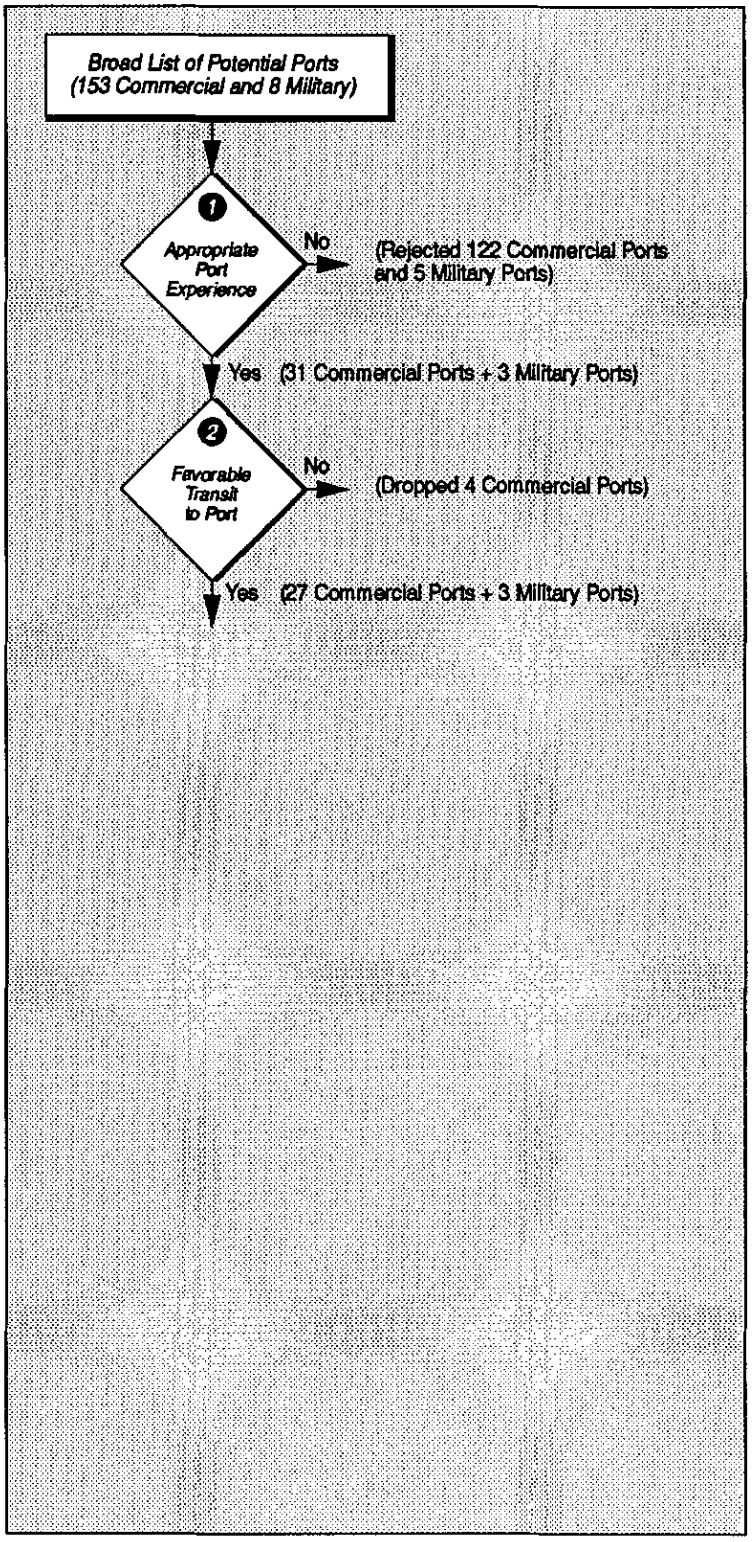


Figure D-3 Screening Ports for Favorable Transit Criterion from Sea to Port

Because containerized spent nuclear fuel requires no special port experience or facilities specific to the handling of radioactive material, ports were not eliminated from consideration because of lack of such experience or facilities.

As noted earlier, there is no reasonable way of determining the future likelihood that currently marginal ports that already have adequate facilities (but simply lack "appropriate experience") will acquire such experience. It depends totally on whether the ports will be able to induce shipping lines to use their facilities. In the area of appropriate facilities, however, there is much less uncertainty in making such determinations, since the planning process for port improvements must be made years in advance in order to allow time for land acquisition, funding, and other approvals before such improvements can be made. Therefore, those ports that have current plans for improvements that might permit their consideration for purposes of this EIS are much easier to identify. As a result, available information relating to future port improvements was studied carefully. Ports with substantial identified improvements or developments during the period analyzed in this EIS include: Baltimore, MD; Boston, MA; Charleston, SC; Fernandina Beach, FL; Gulfport, MS; Jacksonville, FL; San Francisco, CA; Oakland, CA; Long Beach, CA; Naval Weapons Station Concord, CA; Los Angeles, CA; Miami, FL; Mobile, AL; New Orleans, LA; Norfolk, VA; Philadelphia, PA; Port Everglades, FL; New York, NY; Portland, OR; Savannah, GA; Seattle and Tacoma, WA; and Wilmington, DE. Details on these improvements are shown in Attachment D1 to this appendix. All of these ports (except Mobile, AL) currently have both adequate experience and facilities without further improvements. Therefore, no additional ports were identified for foreign research reactor spent nuclear fuel receipt in the future. (Mobile, AL will meet the requirement for experience if it approximately doubles its current container business in the future, but that is too speculative to be useful at this time).

In addition to physical attributes of port facilities, public safety also depends on the reliability of the personnel operating the facilities. In addition to port accidents related to a failure of the container-handling equipment, human error can also increase risks of accidents. The U.S. Merchant Marine Academy Workshop identified the skill of the labor force at a port as an important (but not mandatory) criterion.

While the U.S. Merchant Marine Academy Workshop preferred ports with cranes, it also considered the use of roll-on/roll-off vessels, but preferred the use of other vessels for a number of reasons. Although roll-on/roll-off is not as likely to be used for spent nuclear fuel shipments as conventional cargo vessels due to costs, availability, and other factors, such vessels require only adequate water depths and appropriate piers to receive foreign research reactor spent nuclear fuel. As a result, the presence of roll-on/roll-off facilities is noted in Section D.2 in the detailed discussions of potential ports, but was not considered an adequate sole basis for port selection.

Several related "desirable" attributes for port facilities were recommended by the U.S. Merchant Marine Academy Workshop or identified during the port analyses. These attributes which would contribute to a port having "appropriate facilities," but were not required for safe receipt, handling, and transhipment of spent nuclear fuel include: (1) secure short-term storage areas (in the event of unexpected events such as snow or icing of roads), (2) the existence of emergency planning and training, (3) the absence of environmentally sensitive areas in port or local restrictions on movement of spent nuclear fuel, (4) the absence of conflicting uses (e.g., explosives, petroleum, tourism), and (5) minimal likelihood of severe natural phenomena impacting port activities (such as high winds from hurricanes, floods, earthquakes, volcanoes). These desirable port attributes are used in final port selection in Section D.1.8.5, and are discussed for each port in Section D.2.

In applying the DOE criteria, it became evident that the majority of the ports that met the first required screening criterion (Appropriate Port Experience) also met these requirements. Application of the Appropriate Facility Criterion retained 25 commercial ports and three military seaports for further analysis. Two commercial ports, Freeport, TX and Palm Beach, FL were dropped due to the application of this criterion. The results of this screening are summarized in Figure D-4.

D.1.9.4 Criterion 4: Ready Access to Intermodal Transportation

A U.S. Merchant Marine Academy Workshop criterion determined to be necessary for safe shipment of spent nuclear fuel was "intermodal access", which means "ready access from a port" to truck and rail routes. It is becoming common practice for ports with intermodal transfer facilities to carry off-loaded containers on special port-owned container handling equipment to a marshalling yard adjacent the terminal, where the containers are loaded onto trucks or rail for shipment to the consignee. Such transfers tend to minimize traffic congestion at shipside by using experienced port personnel and specialized port equipment. These intermodal transfers are increasingly accomplished with purpose-built container handling equipment (straddle carriers, sidelifts, front-end loaders, stackers, and container forklifts) that require no additional workmen for container handling. Given that the handling is done very rapidly and securely by a single operator, the opportunities for additional worker exposures or serious accidents are minimized. Moving a container from a pier to a marshalling area a few kilometers more distant than one at a terminal has little significance with regard to either worker exposure or public risk (see, for example, container handling equipment in Jane's, 1992).

DOE found that most of the potential ports accepted under the three preceding required criteria also had good access to interstate highways and rail transport. However, smaller ports and most military ports with more limited facilities could also accept containerized spent nuclear fuel from combination breakbulk/container vessels or roll-on/roll-off vessels. These ports often have limited intermodal capabilities for rail in the immediate vicinity of a pier, but the spent nuclear fuel could be trucked to a rail area (often a few miles or less) for loading on a railcar. While these arrangements could involve an additional intermodal transfer, such transfers are typically also done rapidly using special container handling equipment. Therefore, they do not involve significant additional opportunities for worker exposure or accidents than would be the case for movement of foreign research reactor spent nuclear fuel from a pier to an intermodal yard at a large port.

DOE concluded that the lack of an intermodal rail facility immediately at a terminal should not eliminate an otherwise desirable port from further evaluation, if rail access was reasonably close to the port (e.g., container cargo from the Wando Terminal in Charleston, SC, must be trucked a few miles to an intermodal facility in North Charleston for transfer to rail). All ports evaluated have acceptable intermodal access for trucks, although smaller ports typically do not have dedicated truck routes for access to interstates, and may require short transports through sometimes congested local traffic to reach the interstate highway system. This apparent conflict between requirements for ready intermodal access at ports and the National Defense Authorization Act requirement for using ports with the "lowest human populations" has been balanced to permit some small ports with more limited intermodal capabilities to be considered for further screening, since the additional public impacts associated with a few miles' transport through urban populations would be small compared to public impacts associated with transport over hundreds or thousands of miles of the country's Interstate highway system.

Application of the intermodal access criterion resulted in acceptance of 25 commercial seaports and three military ports (i.e., no additional ports were rejected) for further analysis using the remaining DOE criteria. The results of applying this criterion to commercial and military ports are shown in Figure D-5. Details regarding intermodal access are addressed in each port description in Section D.2.

❸ ***Appropriate Facilities:***

- Adequate crane(s), piers, depth of water alongside, etc.

a. Accepted 25 Commercial Ports:

Baltimore, MD
Boston, MA
Charleston, SC
Eddystone, PA
Elizabeth, NJ
Fernandina Beach, FL
Galveston, TX
Gulfport, MS
Jacksonville, FL
Long Beach, CA
Los Angeles, CA
Miami, FL
Newport News, VA
Norfolk, VA
Oakland, CA
Philadelphia, PA
Port Everglades, FL
Portland, OR
Portsmouth, VA
San Francisco, CA
Savannah, GA
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. Accepted 3 Military Ports:

Military Ocean Terminal
Sunny Point, NC

Military Ocean Terminal
Oakland, CA

Naval Weapons Station
Concord, CA

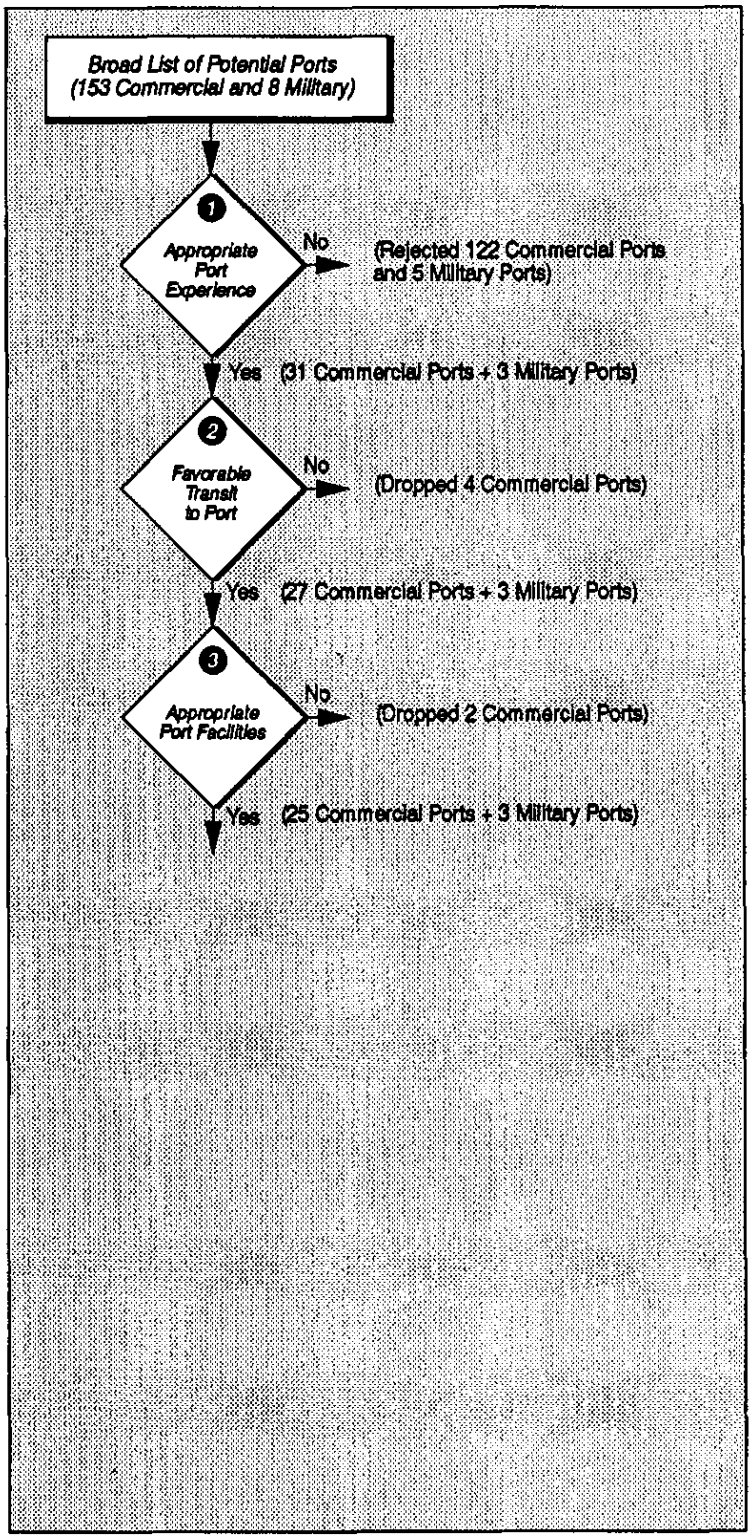


Figure D-4 Screening Ports with Appropriate Facilities Criterion

4 *Intermodal Access:*

a. Accepted 25 Commercial Ports (SRS and ORNL unless otherwise specified):

Baltimore, MD
Boston, MA
Charleston, SC
Eddystone, PA
Elizabeth, NJ
Fernandina Beach, FL
Galveston, TX
Gulfport, MS
Jacksonville, FL
Long Beach, CA
Los Angeles, CA
Miami, FL
Newport News, VA
Norfolk, VA
Oakland, CA
Philadelphia, PA
Port Everglades, FL
Portland, OR
Portsmouth, VA
San Francisco, CA
Savannah, GA
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. Accepted 3 Military Ports:

Military Ocean Terminal
Sunny Point, NC

Military Ocean Terminal
Oakland, CA

Naval Weapons Station
Concord, CA

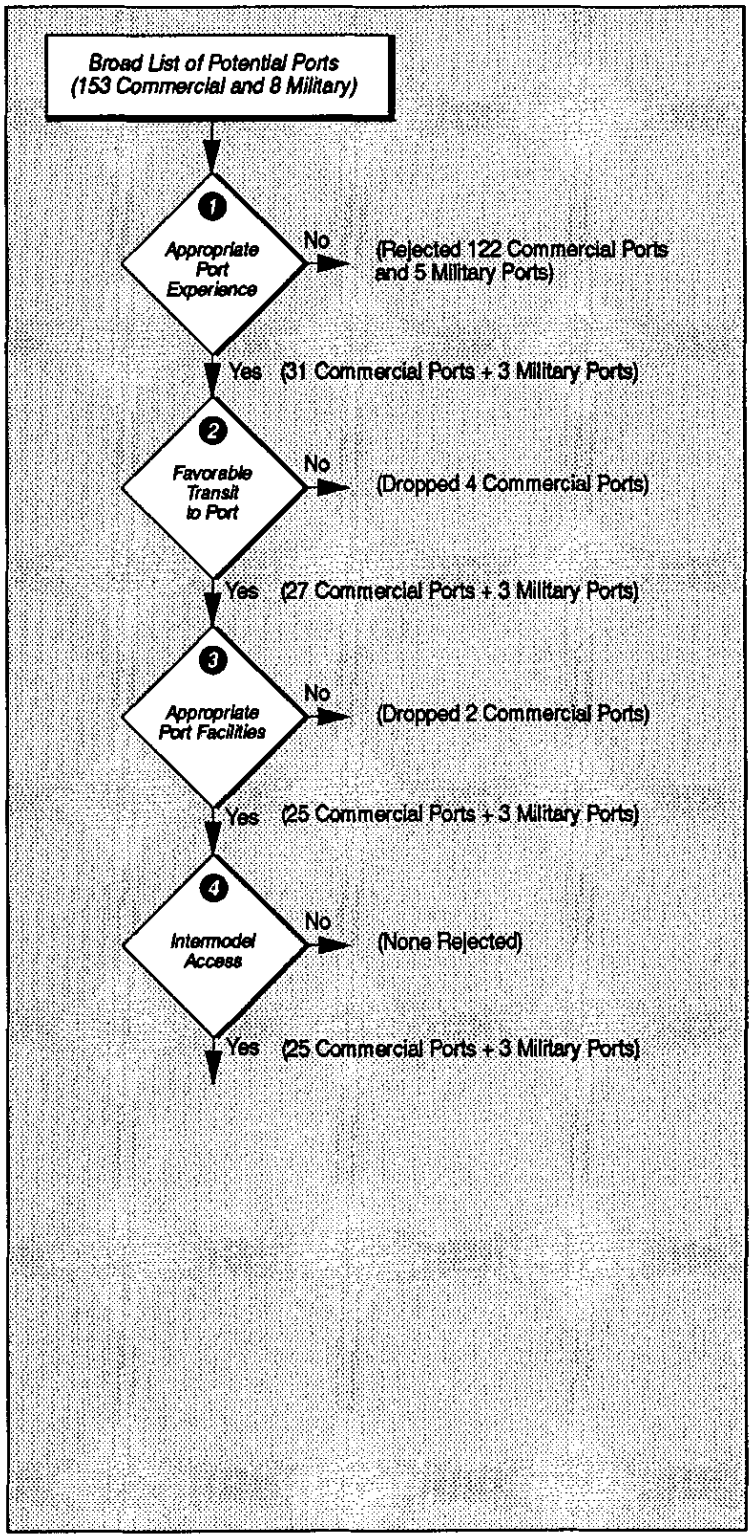


Figure D-5 Screening Ports for Ready Access to Intermodal Transportation

D.1.9.5 Criterion 5: Human Populations

While only dealing with foreign research reactor spent nuclear fuel shipments bound for the Savannah River Site, the Federal court ruling discussed in Section D.1.3 indicates that the courts consider port populations to be an important ingredient in the National Environmental Policy Act (NEPA) process for assessing the range of reasonable port alternatives.

NEPA requires that DOE consider a range of reasonable alternatives for potential ports of entry. On the other hand, the National Defense Authorization Act port selection factors required that, if economically feasible and to the maximum extent practicable, ports of entry for foreign research reactor spent nuclear fuel bound for the Savannah River Site have the lowest human populations in the area surrounding the port. While the National Defense Authorization Act was written specifically to regulate the receipt and storage of foreign research reactor spent nuclear fuel at the DOE's Savannah River Site, DOE elected to apply this criterion in identifying ports of entry for all five potential sites, to the maximum extent practicable.

DOE has considered a number of potential definitions of "lowest human populations" and resulting models that might be used to satisfy the National Defense Authorization Act lowest population factor (NDAA, 1993). These include using the same approach used in the Urgent Relief Environmental Assessment (DOE, 1994d), and variations that might be useful for identifying ports for inclusion in this EIS. A description of the various approaches that were considered are provided in Attachment D3 to this appendix.

As shown in the Urgent Relief Environmental Assessment (DOE, 1994d) and this EIS (Chapter 4), public risk is driven not only by port populations, but by the populations within the immediate proximity of truck and rail shipments from each port to each management site. For each selected port and each selected mode of overland transport (truck or rail), the total "affected" population represents a unique population surrounding the port plus those along the transport route to each of the five potential DOE management sites. DOE considered the affected populations outside the immediate port vicinity along the routes to the management sites to be as important for protection of public health and safety as those within the vicinity of the port terminals, for both incident-free transport and a range of potential accidents.

DOE evaluated port populations within the radii of three distances: 1.6 kilometers (km) [1 mile (mi)], 8.0 km (5 mi), and 16 km (10 mi). These populations are shown in Table D-5. DOE expects that the 1.6 km (1 mi) radius population would include resident members of the public immediately outside the port who would be the most likely to be affected by severe accidents and incident-free impacts. In addition, the radioactivity, which is hypothesized to be released from a very severe accident (long-term fire leading to severe cask damage), would be lofted high into the air and would not normally produce peak ground-level air concentrations until well outside the 1.6 km (1 mi) radius. Therefore, the 1.6 km (1 mi) population was not considered adequate to reflect the population criterion.

The population within a 16 km (10 mi) radius was selected to be consistent with the results of analyses of severe hypothetical accidents described in Section D.5 of this appendix. For severe accidents in ports, the maximum radiation dose to an individual located 16 km (10 mi) from the port is typically much lower than the dose to the maximally exposed individual. However, analyses of the potential impacts of severe accidents in a range of port populations show that the average dose to members of the public within a 16 km (10 mi) radius of the port is higher than the average per capita dose for any of the larger radii around the port for typical (i.e., 50th percentile) meteorology and typical dry deposition and fallout patterns. Further, as discussed in section D.5.4, most of the population dose for even severe accidents occurs within the 16 km (10 mi) radius. Less energetic accident scenarios would cause less dispersion and

Table D-5 Total Populations within Three Distances of Selected U.S. Ports

Name of the Port	Within 1.6 km (1 mi)	Within 8 km (5 mi)	Within 16 km (10 mi)
NWS Concord, CA	14	71,152	381,070
MOTSU, NC	21	960	7,995
Tacoma, WA	94	172,124	511,575
Portland, OR	280	69,039	356,064
Elizabeth, NJ	378	596,076	3,223,038
MOTBA, CA	419	312,133	1,288,699
Jacksonville, FL	523	72,313	334,212
Seattle, WA	557	270,145	753,296
Wilmington, DE	753	166,165	381,502
Gulfport, MS	761	50,218	113,153
Baltimore, MD	818	352,730	1,182,024
Savannah, GA	860	30,845	155,166
Long Beach, CA	1,025	270,336	1,014,418
Charleston, SC	1,550	81,874	233,424
Oakland, CA	1,901	296,661	1,387,611
Miami, FL	2,043	251,551	833,057
Fernandina Beach, FL	2,086	11,787	32,952
Portsmouth, VA	2,554	269,314	665,700
Newport News, VA	2,637	86,993	430,757
Wilmington, NC	2,690	60,308	115,057
Los Angeles, CA	2,918	362,397	1,124,493
Norfolk, VA	2,982	227,290	681,864
Boston, MA	3,084	495,679	1,466,233
Port Everglades, FL	3,927	175,320	714,176
Philadelphia, PA	5,878	50,687	1,915,775
Eddystone, PA	6,179	204,969	827,564
Galveston, TX	8,115	49,175	73,322
San Francisco, CA	9,671	592,869	1,265,529

even smaller doses beyond 16 km (10 mi). Therefore, DOE selected the 16 km (10 mi) radius population to represent the port populations most likely to be impacted by both incident-free transport and the entire range of potential port accidents.

It should be noted that while the populations within the 16 km (10 mi) radius include the populations within 0.8 km (0.5 mi) of the transportation route out to 16 km (10 mi) from the port and result in some double-counting of populations, the results provide only somewhat conservative estimates of the total affected population for each port/management site combination considered.

The populations along truck and rail routes are those computed in Appendix E for the transportation analysis impacts for incident-free transportation.

In summary, this evaluation considered the following population factors:

- Total 1990 Census population within a 16 km (10 mi) radius of the port facilities, and
- Total 1990 Census population within 0.8 km (0.5 mi) of the transport route that would be exposed during transport (from each port to each of the potential DOE management sites).

The statistical distribution of these combined populations for truck transport is shown in Figure D-6. The distribution exhibits some skewing due to a few very large port/site populations, such as around Boston, MA and Elizabeth, NJ. The statistical distribution of combined populations for rail transport is shown in Figure D-7, and again exhibits some skewing due to a few very high population ports. These port/site populations are not clearly normal and are better fit by a Poisson (so-called rare event) distribution, which is often the case for small sample sizes. However, for purposes of developing a systematic and fair method (i.e., one with minimal subjectivity) for evaluating port/site populations, DOE assumed, given the large uncertainty and variances for the small sample sizes for each port/site combined population, that the combined populations for truck transit and the combined populations for rail transit are approximately normal. The port/site population distributions for each of the five management sites (truck and rail routes) are shown in Figures D-8 through D-17, with the bounds associated with the mean plus and minus one standard deviation marked for reference.

For purposes of identifying an acceptable range of ports of entry for the receipt of foreign research reactor spent nuclear fuel, DOE assumed that port/site population combinations greater than approximately one standard deviation above the mean would not be desirable (i.e., about 84 percent of the port/site populations would exhibit statistically lower populations). Thus, the range of ports would include most of the 28 ports being considered, but avoid the extremely large populations around Boston, MA, Elizabeth, NJ, and Philadelphia, PA.

From the remaining 25 ports, DOE assumed that population combinations below the mean combined population would meet the low population criterion while combined populations above the mean would not. As seen in Figures D-8 through D-17, some unique port/site populations would be acceptable for several potential management sites, while other populations would have very limited utility. The potential usefulness of low population ports in relation to this EIS is addressed in Section D.1.9.6. This screening would result in the elimination of an additional five commercial ports and one military port from the list. These commercial ports are Baltimore, MD, and Long Beach, Los Angeles, Oakland, and San Francisco, CA. The military port is Military Ocean Terminal Bay Area in Oakland, CA. The results of the population screening are summarized in Figure D-18.

As previously discussed, the position of maritime experts (USMMA, 1994) is that all of the ports evaluated under the DOE-developed criteria for populations could safely receive and tranship foreign research reactor spent nuclear fuel to all five of the potential DOE management sites. Further, the EIS analyses show that the conservatively calculated impacts would be extremely low. The identification of a smaller number of preferred ports of entry is driven by the requirements of the National Defense Authorization Act, not by any significant safety issues.

As promised in the Urgent Relief Environment Assessment, DOE has also considered future population growth near potential port facilities over the time period considered in this EIS. Year 2010 estimates of projected growth from the 1990 census populations were provided by the states hosting the selected ports and other sources where necessary. Population growth patterns in port cities are continuously changing in ways that cannot be accurately forecast 10 or more years into the future. Nevertheless, the projected port populations based on these growth factors were scrutinized to be sure that no unacceptably large growth would occur around the list of ports selected under the DOE "lowest human population" criterion. The port growth factors used for projecting potential future impacts of port accidents are summarized in Attachment D2 to this appendix, and were used to make final port selections, where appropriate, as discussed in the next section.

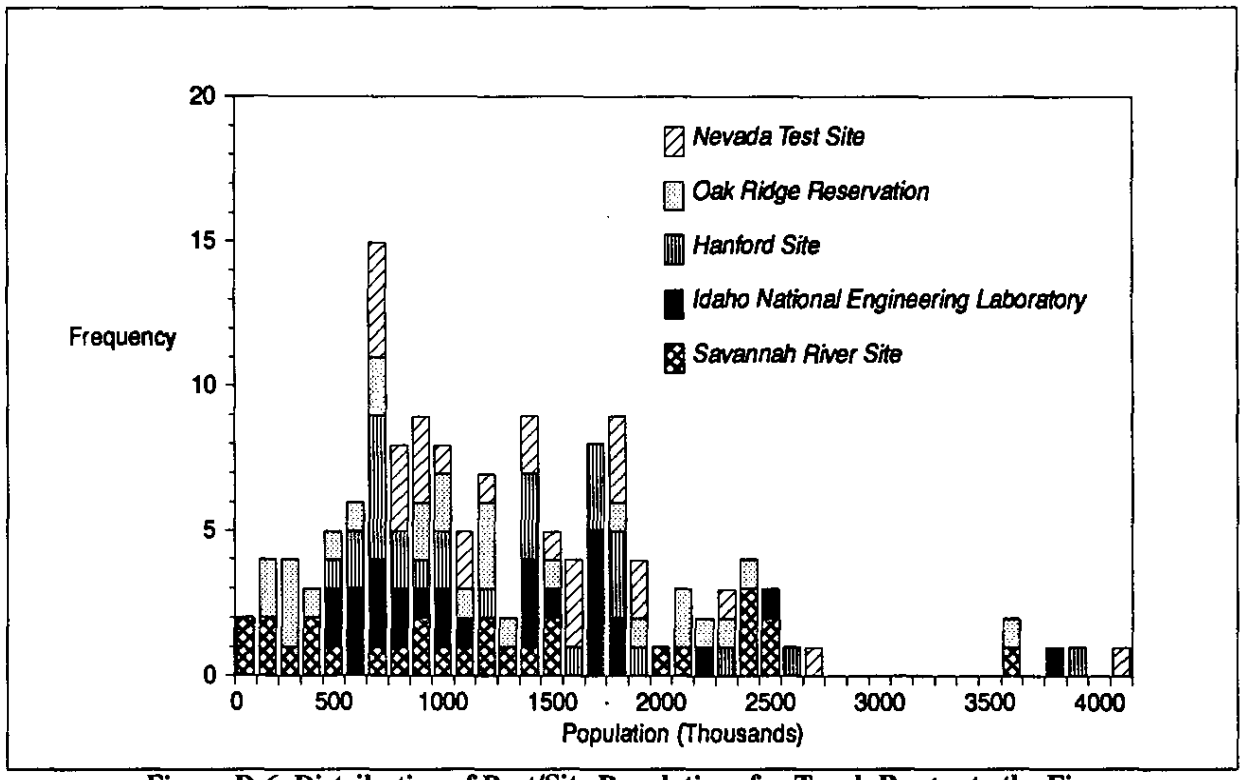


Figure D-6 Distribution of Port/Site Populations for Truck Routes to the Five Management Sites

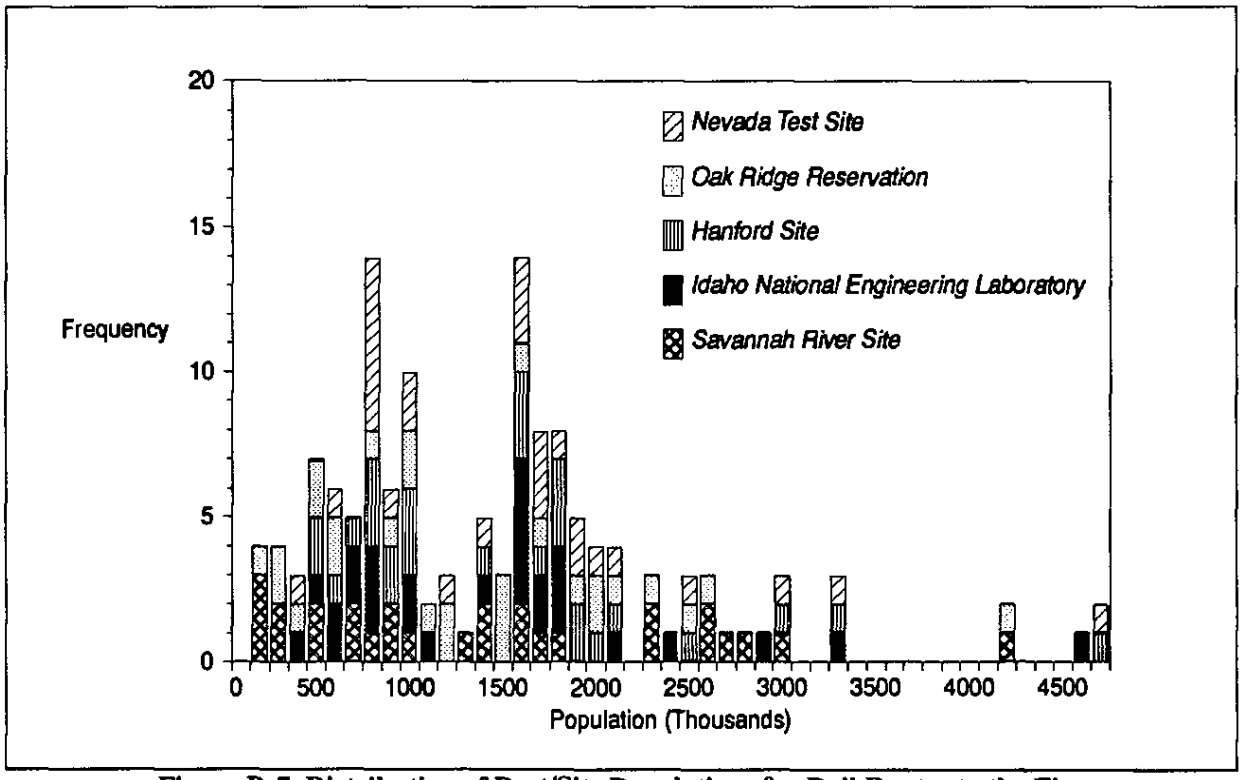


Figure D-7 Distribution of Port/Site Populations for Rail Routes to the Five Management Sites

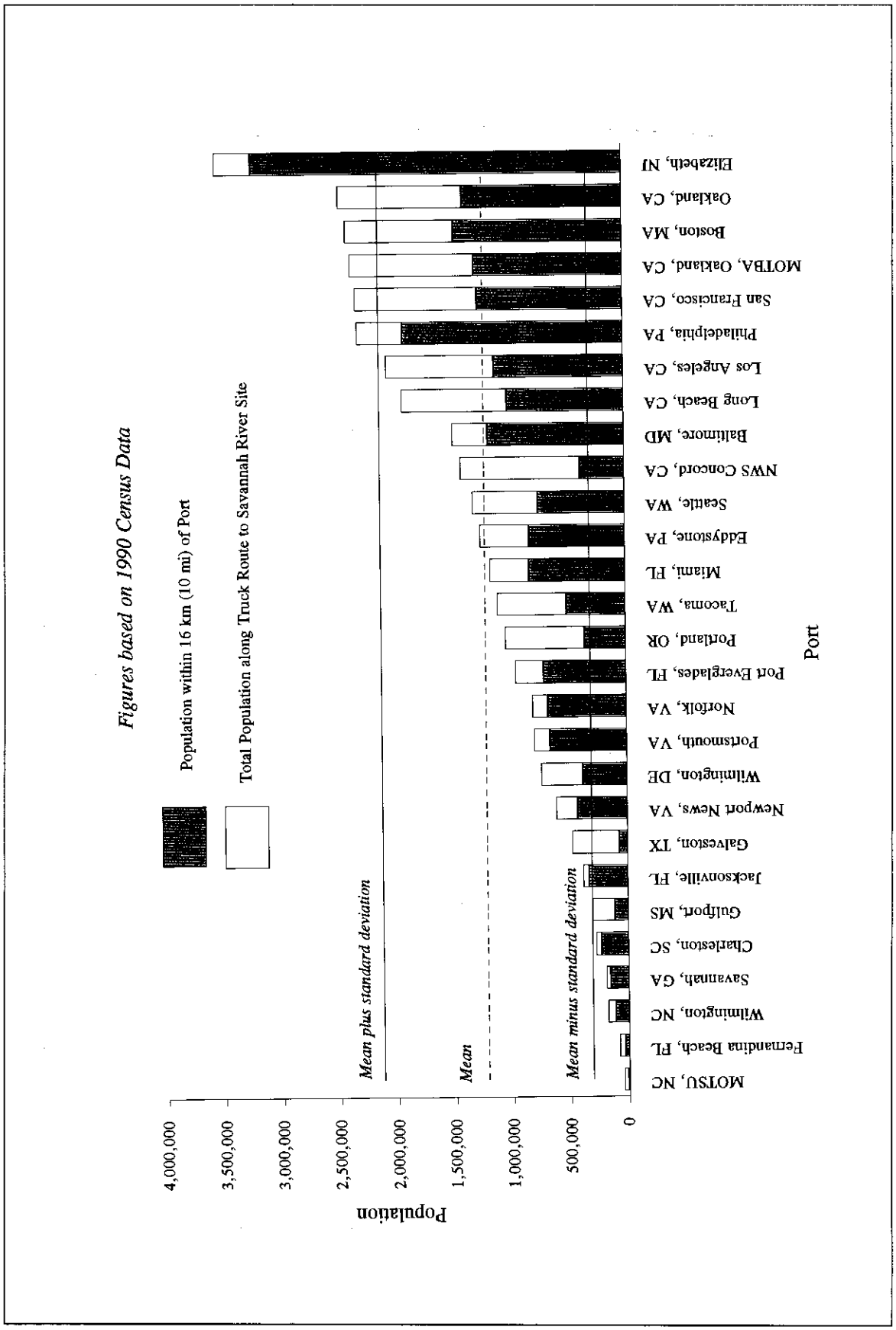


Figure D-8 Population Distribution for Savannah River Site by Truck

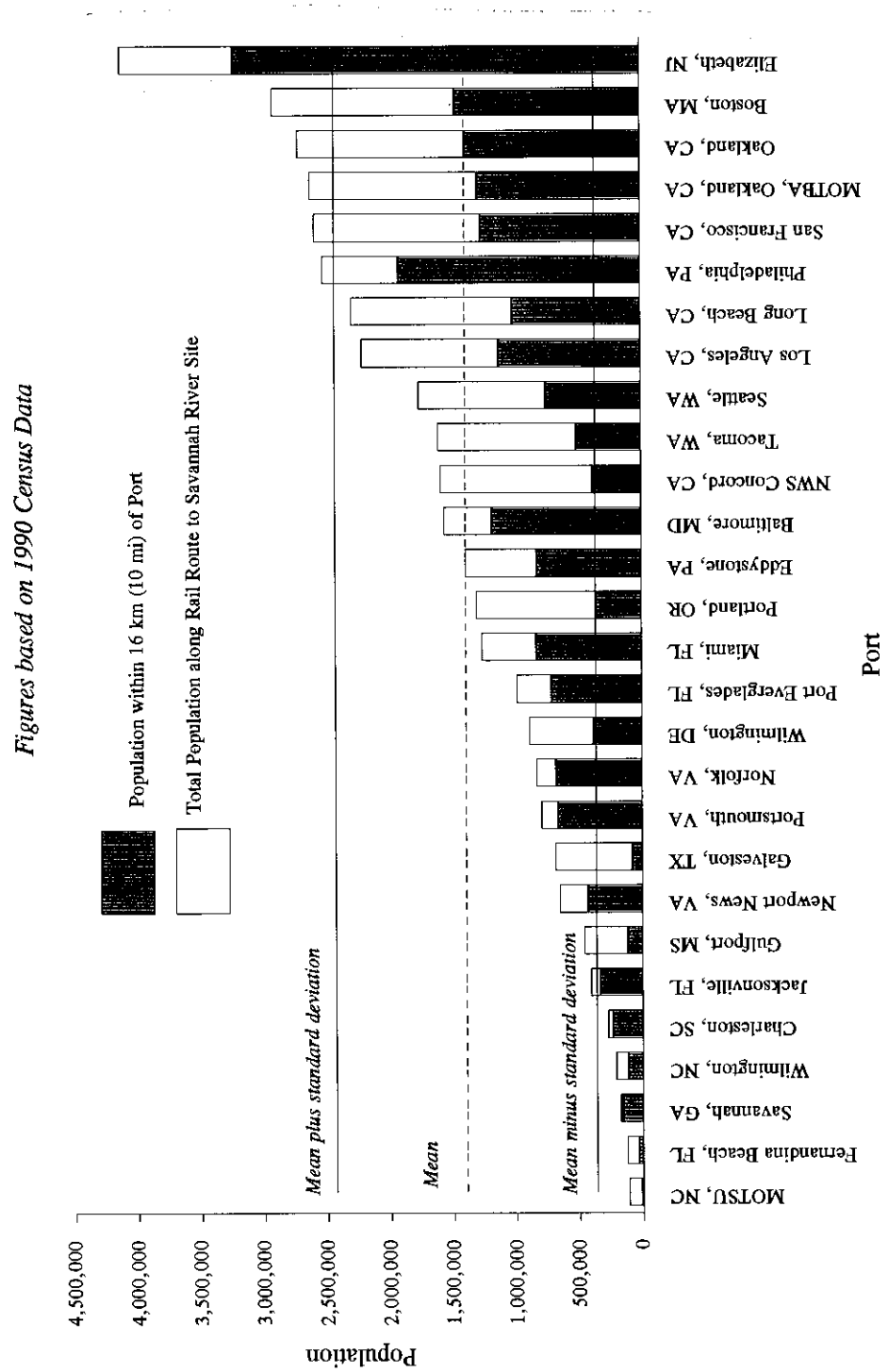


Figure D-9 Population Distribution for Savannah River Site by Rail

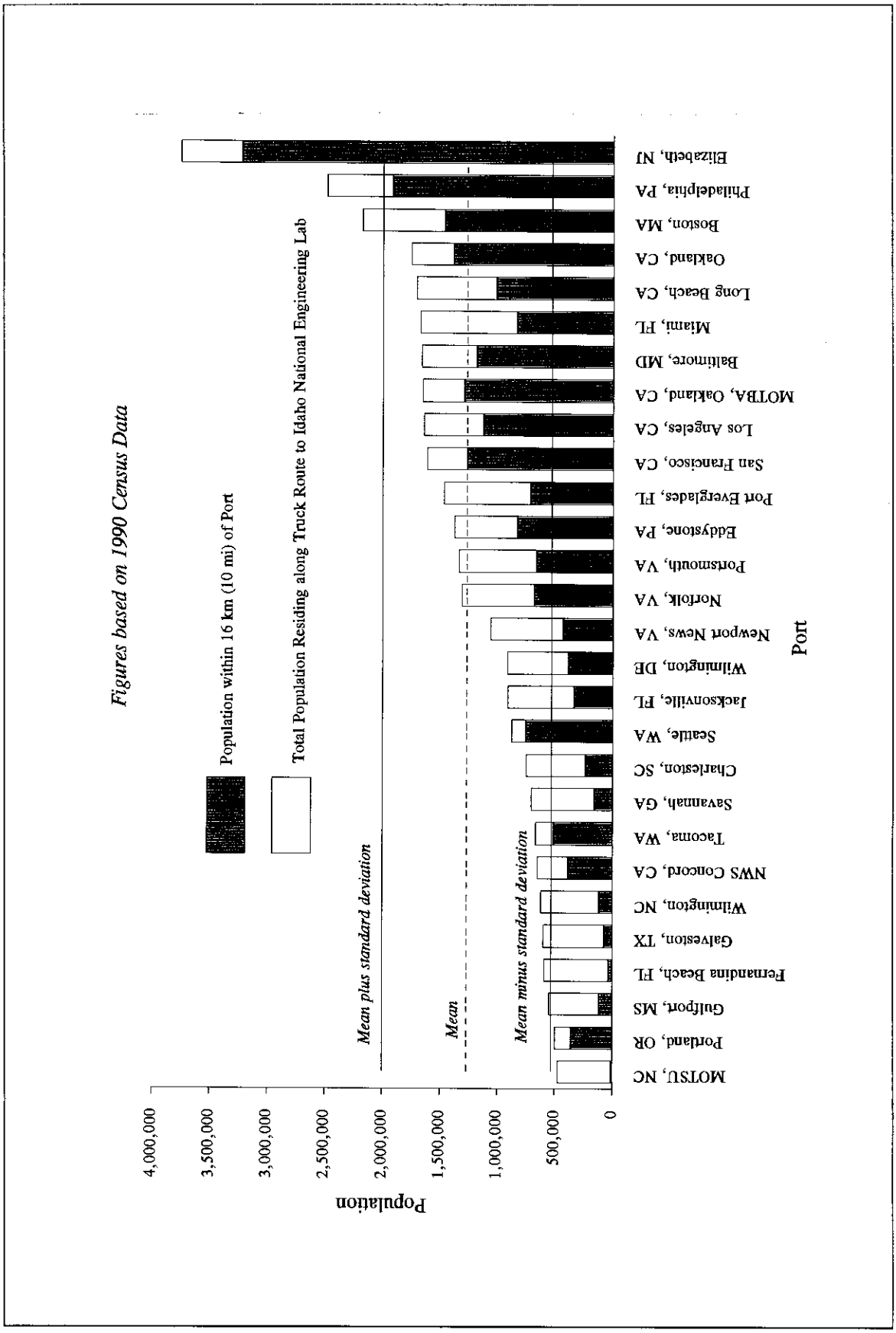


Figure D-10 Population Distribution for Idaho National Engineering Laboratory by Truck

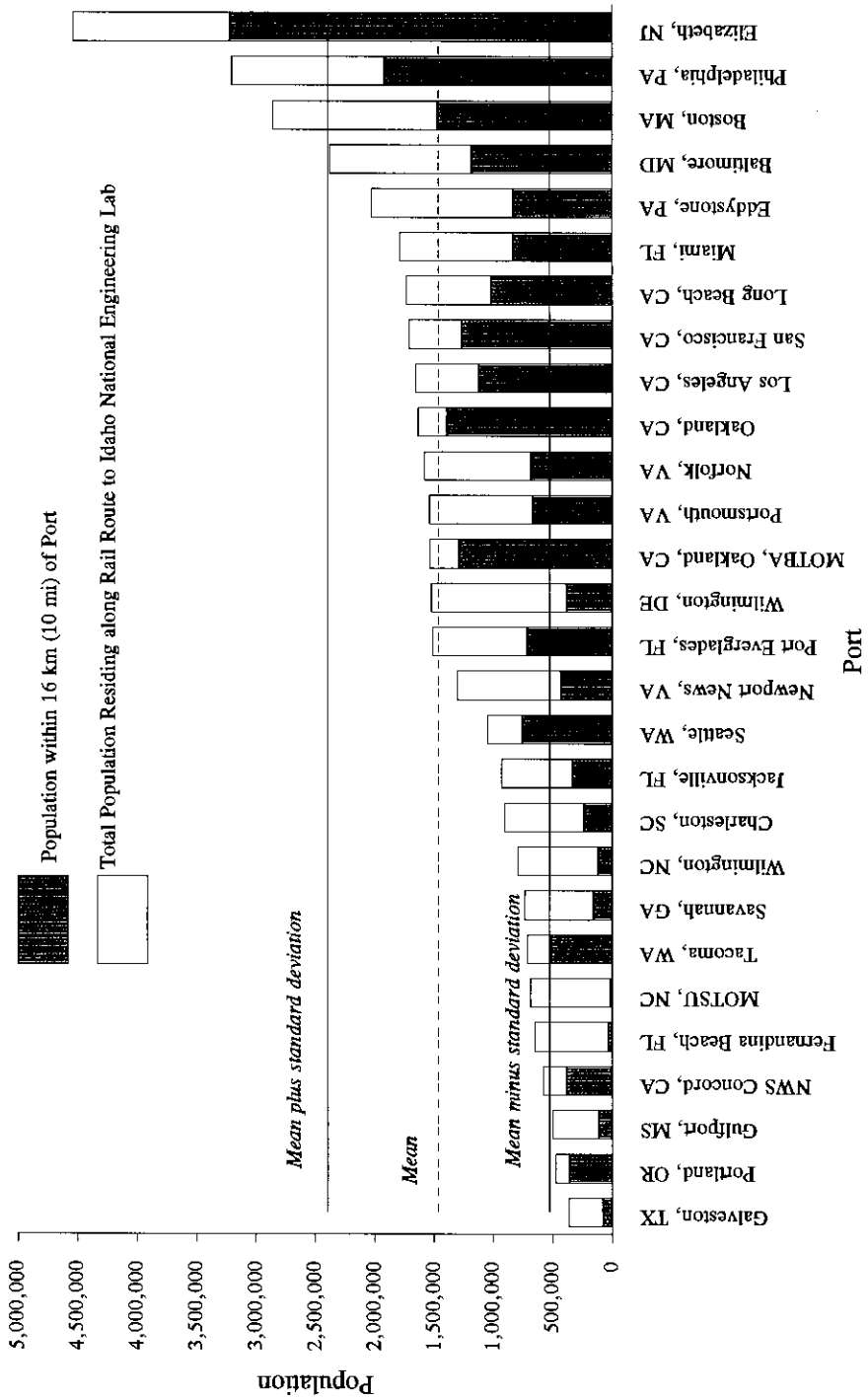


Figure D-11 Population Distribution for Idaho National Engineering Laboratory by Rail

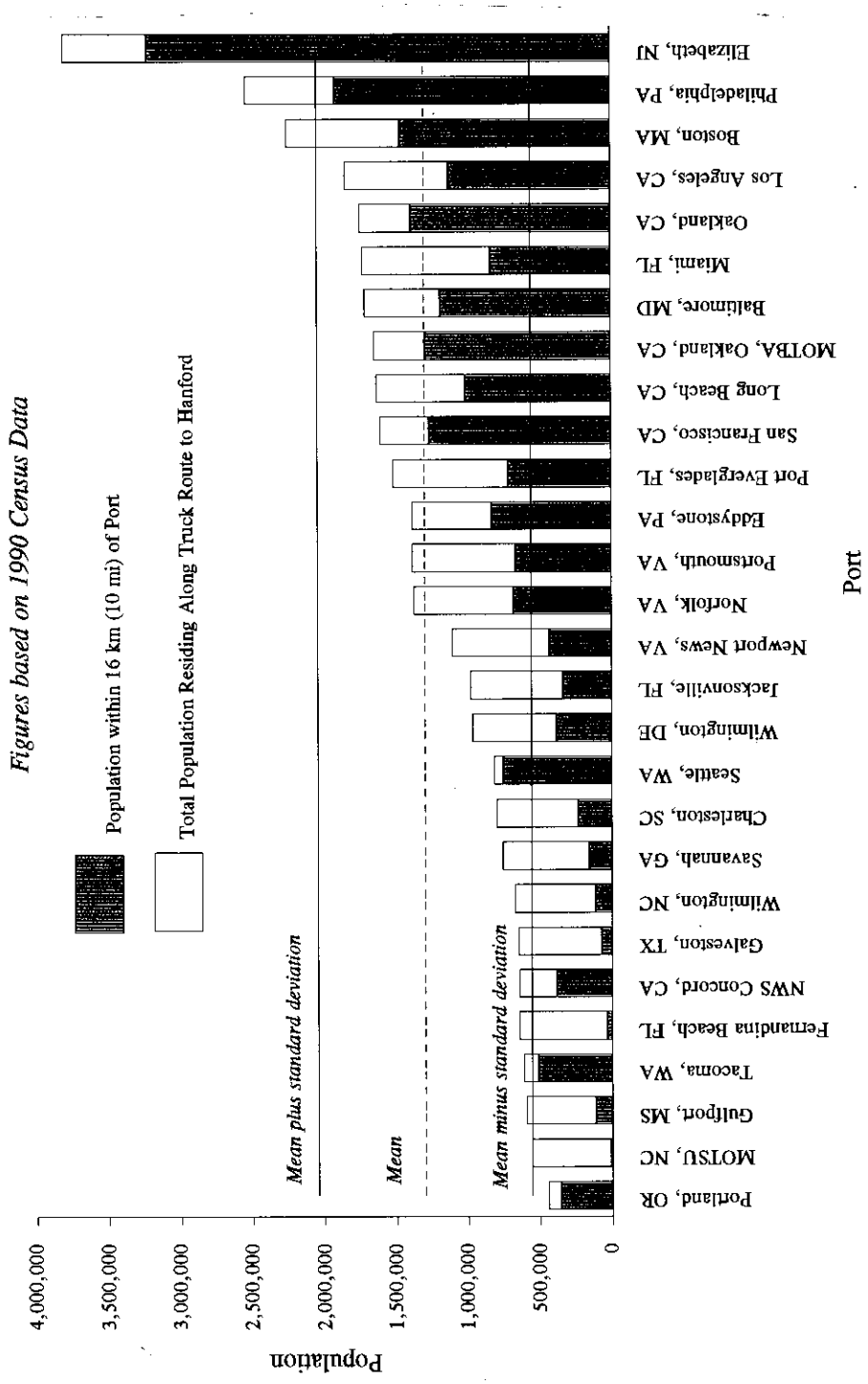


Figure D-12 Population Distribution for Hanford Site by Truck

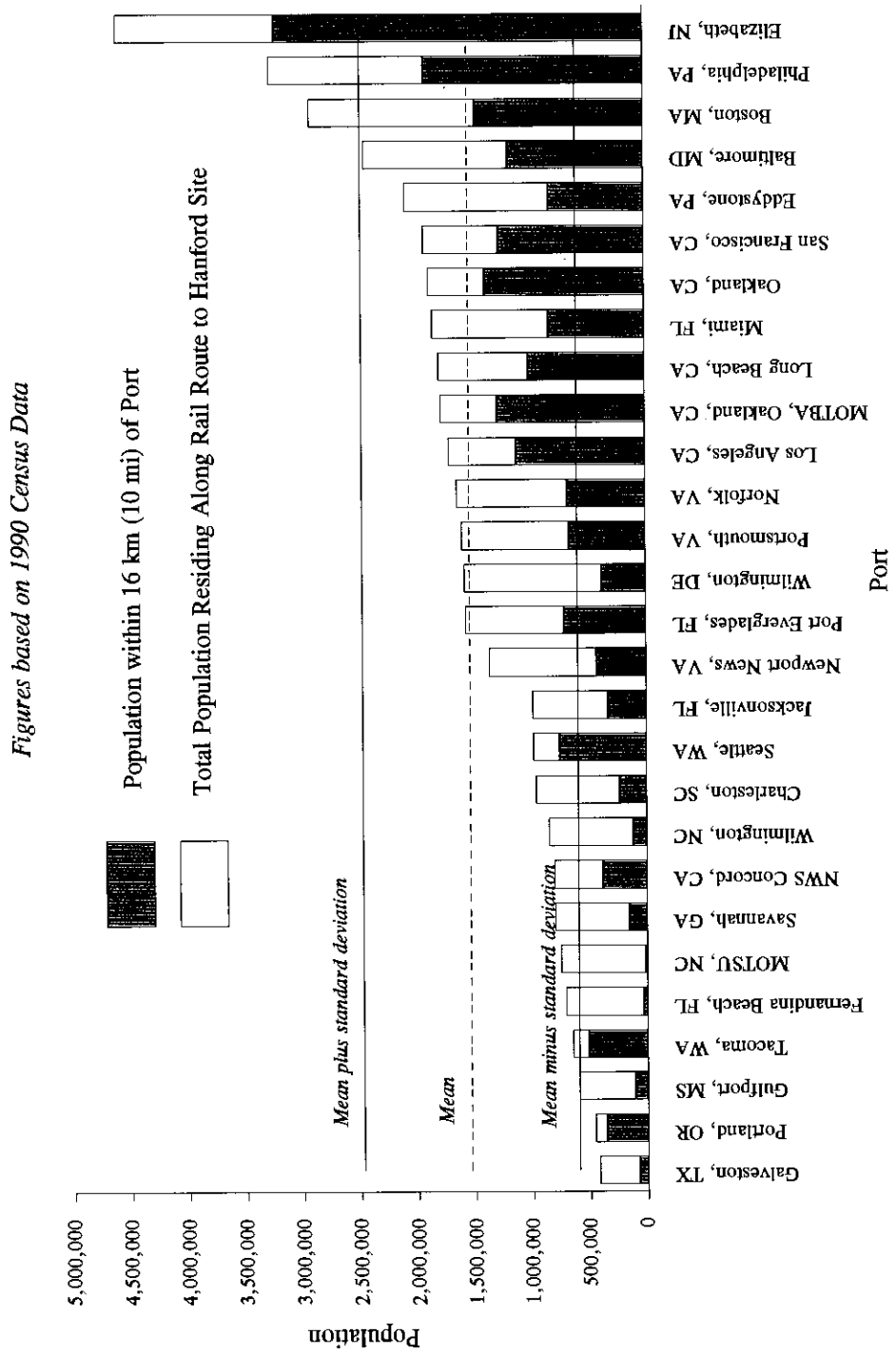


Figure D-13 Population Distribution for Hanford Site by Rail

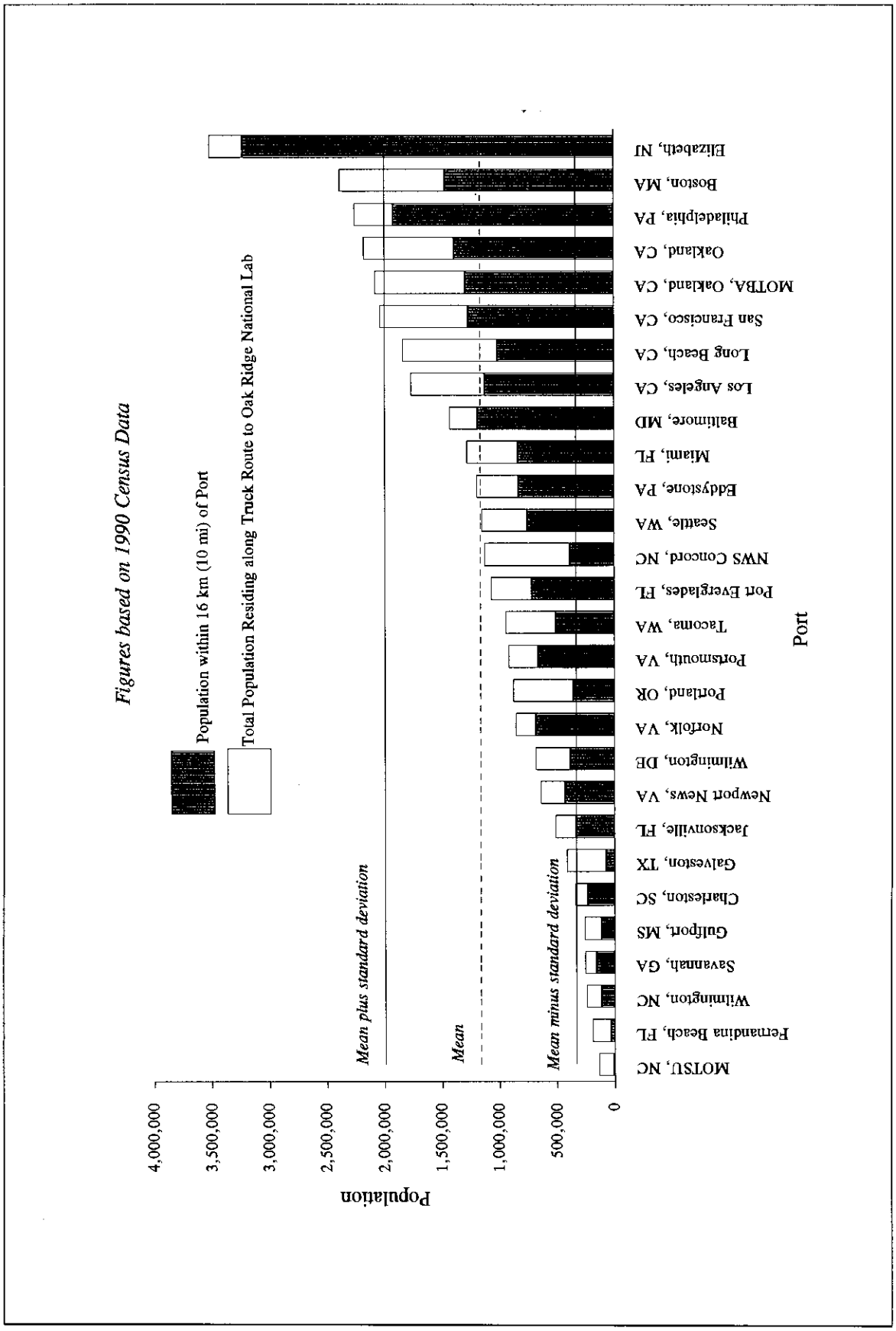


Figure D-14 Population Distribution for Oak Ridge Reservation by Truck

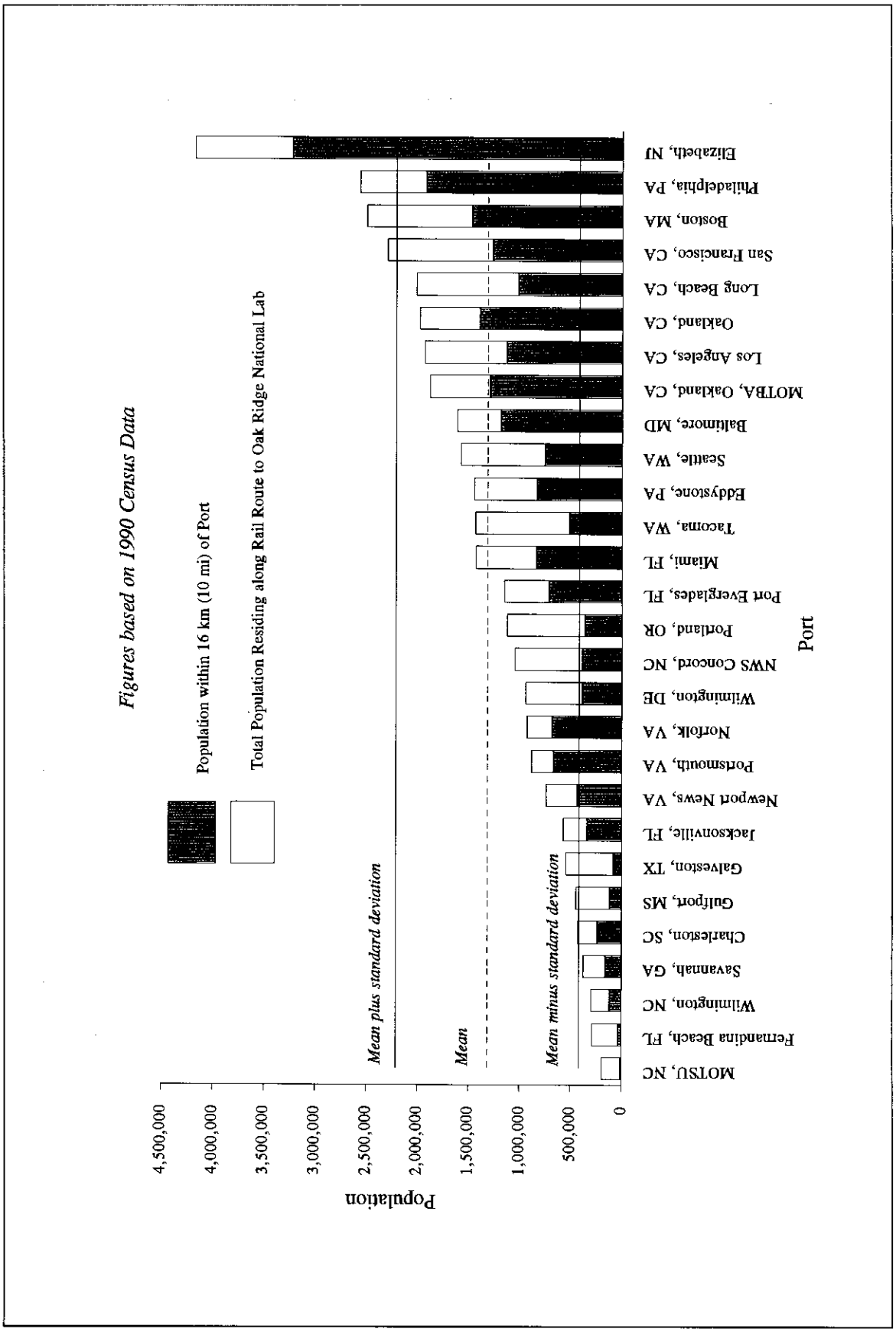


Figure D-15 Population Distribution for Oak Ridge Reservation by Rail

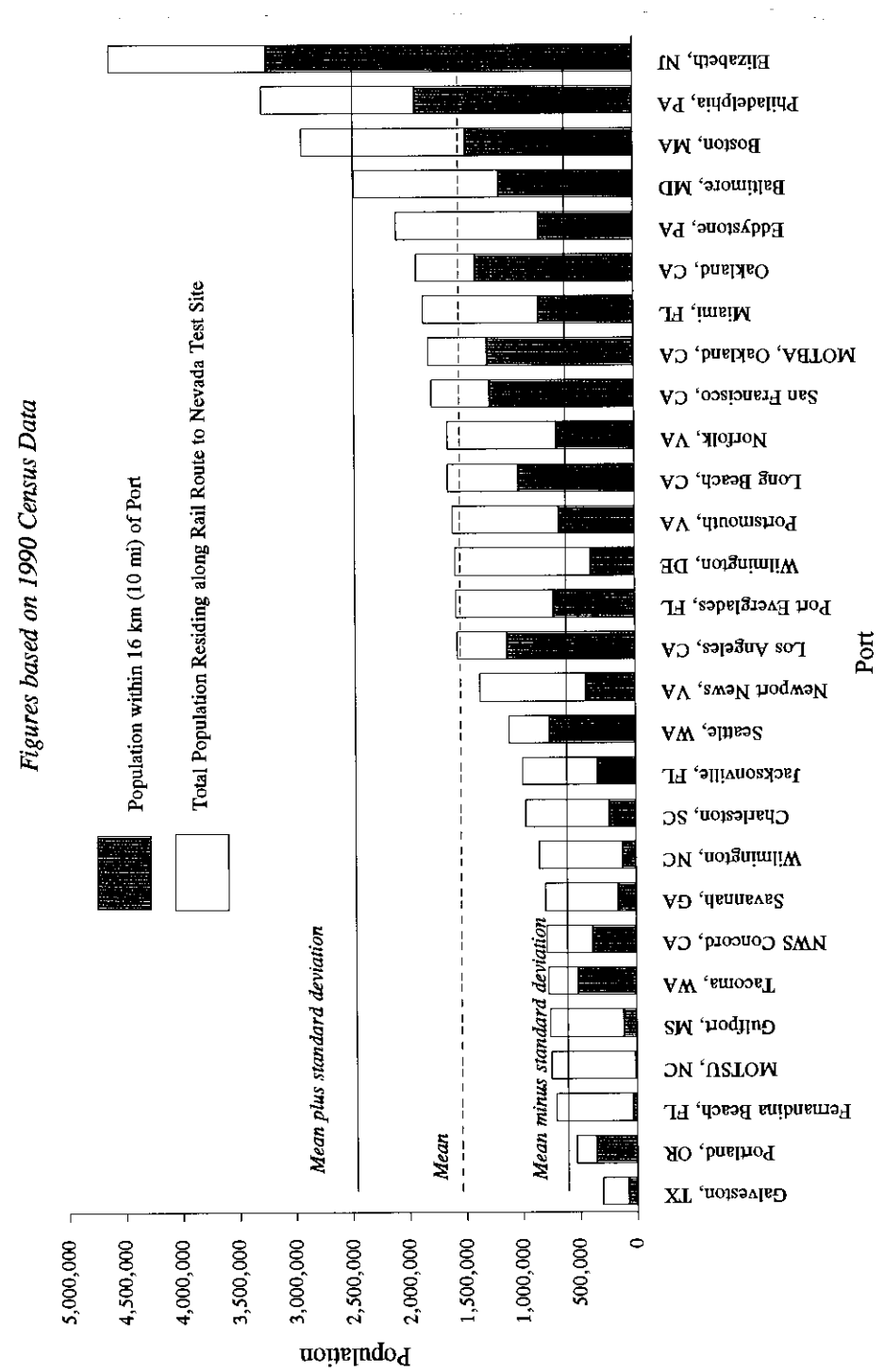


Figure D-17 Population Distribution for Nevada Test Site by Rail

5 Low Population Criteria:

a. *Accepted 17 Commercial Ports:*

Charleston, SC
Eddystone, PA
Fernandina Beach, FL
Galveston, TX
Gulfport, MS
Jacksonville, FL
Miami, FL
Newport News, VA
Norfolk, VA
Port Everglades, FL
Portland, OR
Portsmouth, VA
Savannah, GA
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. *Accepted 2 Military Ports:*

Military Ocean Terminal
Sunny Point, NC
Naval Weapons Station
Concord, CA

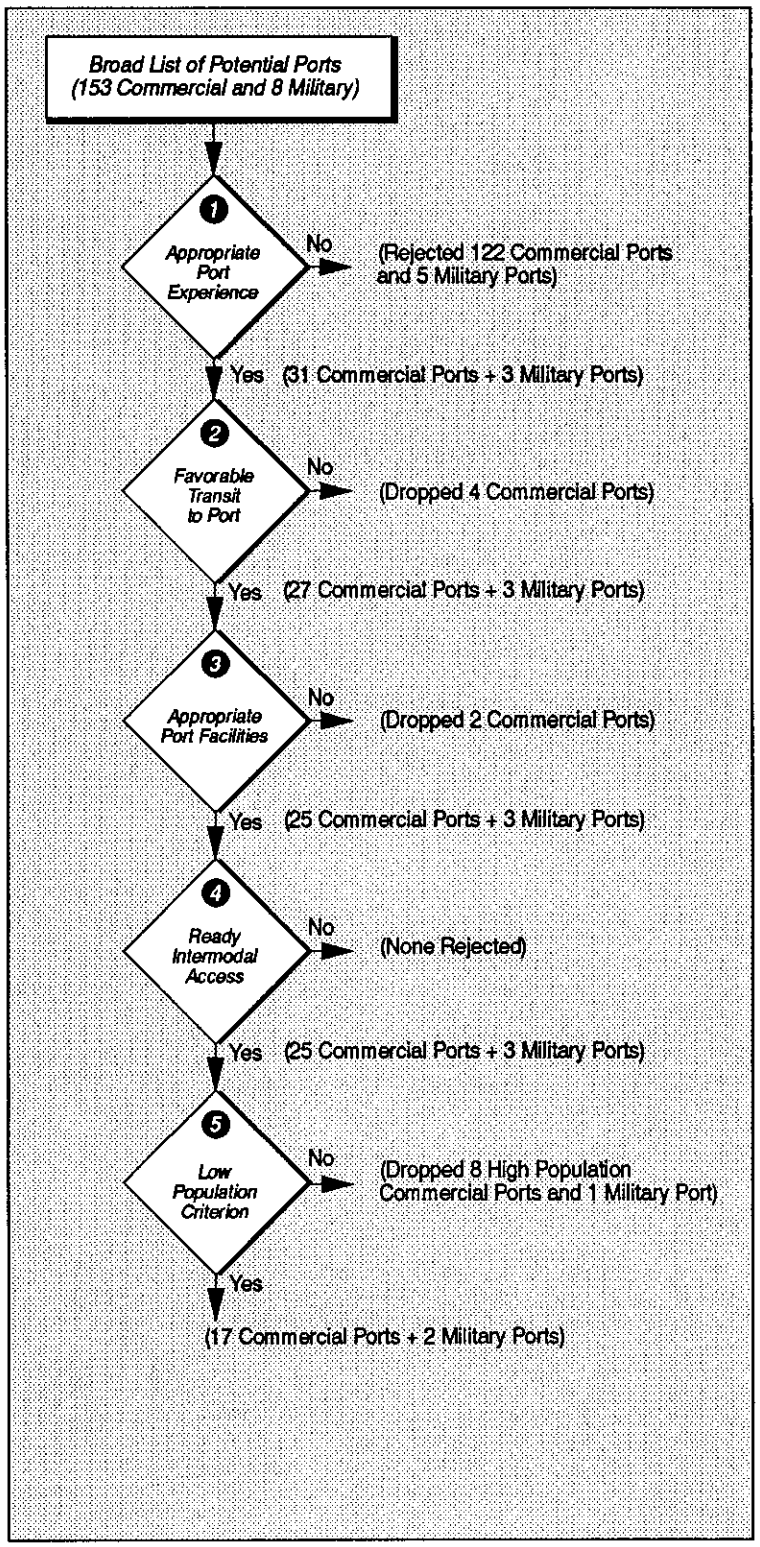


Figure D-18 Screening Ports for DOE "Lowest Human Population" Criteria

D.1.9.6 Desirable Port Attributes

As discussed in Section D.1.9, there are a number of desirable attributes that were not important enough individually to reject an otherwise acceptable port, but have been collectively used to select proposed ports from the list of ports found acceptable under the other DOE criteria. As an element of desirable attributes, DOE examined the likely usefulness of ports for foreign research reactor spent nuclear fuel shipments to any of the five DOE management sites.

The term "usefulness," as used here, is a relative term wherein the relative numbers of scheduled shipping lines and the types of ships that service each port and the countries served by those lines, are compared for two or more otherwise acceptable ports for purposes of selecting the best of that group. This helped to select the ports most likely to be useful in relation to this EIS. This information is shown in Table D-6.

In using these factors, the Ports of Newport News, Norfolk, and Portsmouth, VA, are examined as a single port: Hampton Roads, VA. Table D-6 shows the results of the evaluation of the low population ports for usefulness. The limited usefulness of a port for truck or rail access and service to the potential foreign research reactor spent nuclear fuel management sites eliminated the Ports of Eddystone, PA, Miami and Port Everglades, FL, Wilmington, DE, and Seattle, WA from further consideration.

DOE also identified the most desirable attributes of the remaining ports, such as terminals that do not have conflicting activities nearby (e.g., cruise ship lines, large tourist populations, large petroleum or petrochemical facilities, etc.), and are well-separated from high density populations, have secure short-term storage for contingencies, and have adequate emergency preparedness.

Absence of Conflicting Activities in Port Facilities

While it is the long-held position of DOE, Department of Transportation, and the NRC (NRC, 1977) that spent nuclear fuel shipped in certified Type B casks is well-protected from possible damage due to accidental cask drops, transportation fires, or immersion in water, DOE also concluded that the small public risks associated with such activities could be reduced further if the port in question also had no potentially dangerous, unavoidable conflicting activities, such as regularly scheduled transport of explosive or flammable cargoes, no petroleum carriers or storage facilities in the immediate vicinity of foreign research reactor spent nuclear fuel carrying vessels, and no large numbers of tourists in the immediate area of the terminal who could be unacceptably impacted by a severe accident (good planning and scheduling for arrival of foreign research reactor spent nuclear fuel carrying vessels could mitigate many potential conflicts). The absence of conflicting activities at potential commercial ports was considered in the final port selection to the maximum extent practicable.

A similar important factor for military ports is whether there is adequate separation of the potential port facilities to be used for receipt of the spent nuclear fuel from other hazardous activities (e.g., loading munitions). An exception would be military facilities that were designed and constructed to mitigate the potential impacts of explosions or fires at other piers. Two examples of such facilities which were accepted under the other DOE criteria are MOTSU, NC and the NWS Concord, CA, where such activities are routinely carried out with a high degree of safety. In addition, such conflicts can be avoided by scheduling foreign research reactor spent nuclear fuel shipments at times when no explosives are present at piers.

Table D-6 Relative Usefulness of Low Population Ports for Foreign Research Reactor Spent Nuclear Fuel Shipments

Ports-of-Entry	*Relative Usefulness by Storage Site*					*Relative Usefulness by Foreign Research Reactor Spent Nuclear Fuel Shippers*	
	SRS	*ORR*	*INEL*	*HS*	*NTS*	*Conventional Carriers*	*Charters*
Commercial							
East Coast							
Charleston, SC (Wando Terminal)	T&R	T&R	T&R	T&R	T&R	Europe, Far East, Japan, Australia	Yes
Eddystone, PA	R	No	No	No	No	Central/South America	Yes
Fernandina Beach, FL	T&R	T&R	T&R	T&R	T&R	South/Central America, Mediterranean (monthly)	Yes
Hampton Roads, VA	T&R	T&R	T&R	T&R	T&R	Most of the world	Yes
Jacksonville, FL	T&R	T&R	T&R	T&R	T&R	Most of the world	Yes
Miami, FL	T&R	No	No	No	No	Central/South America, Mediterranean, Mexico, Far East	Yes
Port Everglades, FL	T&R	T&R	No	No	No	South American, Northern Europe, Mediterranean, Mideast, Scandinavia	Yes
Savannah, GA	T&R	T&R	T&R	T&R	T&R	Most of the world	Yes
Wilmington, DE	T&R	T&R	T	T	T	Central/South America	Yes
Wilmington, NC	T&R	T&R	T&R	T&R	T&R	Northern Europe, Mediterranean, Mideast, East and South Africa, South America, Far East, Australia	Yes
Gulf Coast							
Galveston, TX	T&R	T&R	T&R	T&R	T&R	Northern Europe, Mediterranean, Mexico, South America, Central America	Yes
Gulfport, MS	T&R	T&R	T&R	T&R	T&R	Northern Europe, Central/South America	Yes
West Coast							
Portland, OR	T&R	T&R	T&R	T&R	T&R	Most of the Pacific Rim, Mediterranean	Yes
Seattle, WA	No	T	T&R	T&R	T&R	Most of the Pacific Rim, Mediterranean	Yes
Tacoma, WA	T	T	T&R	T&R	T&R	Most of the Pacific Rim, Mediterranean	Yes
Military							
MOTSU (NC)	T&R	T&R	T&R	T&R	T&R	None	Yes
NWS Concord (CA)	No	T&R	T&R	T&R	T&R	None	Yes

SRS = Savannah River Site, ORR = Oak Ridge Reservation, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, NTS = Nevada Test Site; T = truck, R = rail

Emergency Response Capabilities

The U.S. Merchant Marine Academy Workshop identified the importance of a risk management staff and emergency response capabilities (including response plans and training of operating personnel) in determining the acceptability of ports for receipt and handling of foreign research reactor spent nuclear fuel. DOE focused on identification of ports that have current emergency response plans and personnel appropriately trained to respond to a port emergency to protect workers and the public from avoidable risks (however small). Since few ports have detailed response plans for radiological emergencies involving spent nuclear fuel, DOE determined that such shortcomings do not prevent consideration of such ports provided the ports have in place appropriate response plans and training for hazardous cargo

accidents, since many of the features are the same (e.g., identification of decisionmakers, first responders, and support personnel to mitigate impacts of fires, etc.). In addition, for ports that have no specific response plans for spent nuclear fuel accidents in port, DOE could provide assistance in the development of radiological emergency response plans (in addition to existing hazardous cargo emergency response capability) and training at such ports in the event they were ultimately selected for foreign research reactor spent nuclear fuel shipments. Thus, appropriate plans and training would likely be in place prior to actual receipt of any such shipments. Ports having current emergency response capabilities were considered more desirable than those that do not.

Spent Nuclear Fuel Handling Experience

The National Defense Authorization Act would also require, "to the maximum extent practicable," that the ports selected for receipt of foreign research reactor spent nuclear fuel have spent nuclear fuel handling experience. At the present time, there are only a few ports in the United States with relatively recent experience handling either spent nuclear fuel or high-level radioactivity in Type B casks. As a result, this criterion, while desirable, unnecessarily restricts considerations to an unacceptably small group of potential ports, and strictly applied, could preclude shipments of spent nuclear fuel from some of the countries being considered under this EIS except by chartered ship. However, because all containerized cargoes are handled in the same manner as the containerized spent nuclear fuel would be handled, DOE concluded that current experience (especially any involving routine handling of potentially hazardous cargoes, or other radioactive cargoes in Type B casks) is much more important for public safety than foreign research reactor spent nuclear fuel handling experience in years past. This is especially true since the trained longshoremen are likely to have changed jobs, ports, or retired during the several years between the last shipments of spent nuclear fuel and the potential onset of future shipments under this EIS.

In addition, ports that have satisfied the "appropriate experience" and "port facilities" criteria are expected to be fully capable of currently handling spent nuclear fuel containers, and would gain experience as the program progressed.

Environmental Concerns Near Ports

Marine areas, immediately surrounding most of the ports considered in this selection process, tend to be severely impacted as a result of necessary periodic dredging or construction of new port facilities, including turning basins, high volumes of marine traffic, and routine port activities. As a result, ports generally are no longer environmentally sensitive areas within the context of NEPA. However, consistent with U.S. Merchant Marine Academy Workshop recommendations and in response to public comments, DOE decided that when special protected or sensitive areas were identified nearby the terminal(s) being considered, these areas would be identified in the EIS and used for final port identification as appropriate. No serious issues have been identified in the immediate vicinity of any ports selected under the DOE low population criterion review, with the possible exception of the NWS Concord, CA and Fernandina Beach, FL.

Environmental Concerns from Severe Natural Phenomena

Other factors that were considered desirable attributes for ports include average or lower risks from severe weather (e.g., extremely high winds, hurricanes, etc.) or other natural phenomena (e.g., seiches, earthquakes, volcanism, etc.). These attributes are not expected to be of great significance in practice, since the time involved with potential receipt and transhipment of containerized spent nuclear fuel represents such an extremely short period of risk (typically less than 24 hours), that the probability of

severe natural phenomena impacting foreign research reactor spent nuclear fuel shipments is vanishingly small. Further, some natural events, such as hurricanes, can often be avoided. However, these characteristics were examined in conducting the port evaluation.

Separation of Port Facilities from Urban Populations

The following desirable characteristics are examined:

- Terminals used for spent nuclear fuel shipments should be physically separated from densely populated city centers (by several kilometers if possible) to help ensure that the general public would be unlikely to be exposed to significant radiation doses from either incident-free transport or accidents within the port (e.g., cask drops, fires, or truck or rail accidents, etc.).

- Transport of spent nuclear fuel through large, densely-populated, congested areas around the port should be avoided where practical.

These geographic/demographic characteristics, while not explicitly addressed in the evaluation of "lowest human populations" for ports, are implicitly included in the 16 km (10 mi) radius populations used for screening ports. While absence of these characteristics would not necessarily eliminate the use of such ports under this EIS, DOE reviewed these ports to determine if there were terminals or piers within the port that provided these characteristics. In many cases, development of new port facilities in recent years has resulted in specific terminals and/or piers that meet all of the required criteria (USMMA, 1994, and NDAA, 1993), and that also have most or all of the additional desirable characteristics (e.g., the Wando Terminal in Charleston, SC, the Blount Island Terminal in Jacksonville, FL, or Terminal T6 in Portland, OR).

Absence of Local Restrictions on Receipt and Handling of Spent Nuclear Fuel

Another desirable port factor recommended by the U.S. Merchant Marine Academy Workshop is the absence of local regulatory restrictions on receipt and handling of spent nuclear fuel. It is well established that local restrictions on international or interstate commerce are void under the U.S. Constitution, and similar challenges have been rejected by the Federal courts. For example, the Port of Oakland, CA indicated that a citizen's legislative initiative in 1987 led to a ban on the handling and transport of foreign research reactor spent nuclear fuel through the port. Although Oakland's ban was invalidated by the Federal District Court, the Port Authority has maintained some control over radioactive shipments through the port through its permitting system (Adams, 1993). Nevertheless, although claiming to be a "nuclear free zone," the port continues to allow permitted shipments of certain radioactive materials, handling approximately 500 metric tons (551 tons) of radioactive shipments between January and June 1994 (Adams, 1994).

Further, if DOE were to avoid selection of ports with restrictions by local ordinances, every port wishing to close its doors to receipt of spent nuclear fuel (or any other type of cargo) would simply promulgate an ordinance. Therefore, the EIS will only identify existing local restrictions (formal or informal) in section D.2 for consideration by decisionmakers, and this criterion will have no immediate impact on determination of the acceptability of ports within this EIS.

Secure Short-Term Storage

Although the National Defense Authorization Act requires, to the extent practicable, expeditious movement of casks from a port, the presence of regular guards, fences, and lighted areas that provide security at all times is a desirable attribute. Such additional features provide assurance of safe segregation and short-term storage of foreign research reactor spent nuclear fuel shipments away from workers and the public in the event of unexpected local occurrences, such as snow or ice storms, traffic congestion, and other events beyond the control of spent nuclear fuel shippers.

To best comply with this attribute, the storage area should be one designated for the storage of hazardous materials (referred to as a facility of particular hazard). Such designations are normally simple processes which result in U.S. Coast Guard approval following a request by the terminal operator. While all the military ports are designated as "facilities of particular hazard," some commercial facilities may only request periodic designations for specific incoming or outgoing cargoes (e.g., the Port of Tacoma, WA periodically designates Terminal 7B for occasional shipments of potentially explosive ammonium nitrate). Table D-7 shows which commercial ports have traditionally had secure storage areas for hazardous cargoes, and DOE has assumed such storage would be available in the future for receipt and short term storage of foreign research reactor spent nuclear fuel. (More detailed information on "facilities of particular hazard" may be found in section D.4.3).

D.1.10 Application of the Desirable Port Attributes in Port Selection

As a result of the evaluation of desirable attributes, two additional ports, Fernandina Beach, FL, and Gulfport, MS, were removed from the potential ports of entry list (Table D-7). The port of Fernandina Beach, FL, is not well-separated from the urban population surrounding the port, and the population is expected to substantially grow by about 82 percent by the year 2010 (see Attachment D2). Also, entry to the port requires ship passage through a State sea manatee (an endangered species) preserve. The Port of Gulfport, MS, does not currently have a well-secured area designated for the storage of foreign research reactor spent nuclear fuel, and it is unlikely it ever will due to casino operations. There is a former cruise ship terminal at the East Pier, which is slated for new casino development, a floating casino located in the port and two new casinos on the West Pier. In addition, the port is not well-separated from surrounding urban population.

Conclusion

As a result of the evaluation, ten ports remained as the final list of ports acceptable for the potential receipt, handling, and transhipment of foreign research reactor spent nuclear fuel. These ten ports [Charleston, SC; Galveston, TX; Hampton Roads (includes terminals in Newport News, Norfolk, and Portsmouth), VA; Jacksonville, FL; MOTSU, NC; NWS Concord, CA; Portland, OR; Savannah, GA; Tacoma, WA, and Wilmington, NC] represent the final list of ports considered for the receipt of foreign research reactor spent nuclear fuel.

D.2 Detailed Information on Potential Ports of Entry

This section of Appendix D provides detailed information that served as the bases for identifying the candidate ports addressed in Section D.1. For convenience, the port details are divided into two categories: (1) the DOE candidate ports of entry that met the criteria developed for port identification in Section D.1, and (2) the remainder of the ports that fully or marginally satisfied the first criterion for appropriate port experience. Within each of the categories, the ports are arranged in alphabetical order. The location of the ports is shown in Figure D-1.

Table D-7 Use of Desirable Attributes for Selecting Final "Low Population" Ports for Foreign Research Reactor Spent Nuclear Fuel Shipments

Ports-of-Entry	*Free of Conflicting Uses at Port Facilities*	*Emergency Preparedness*	*Short-Term Secure Storage*	*Free of Environmental Concerns*	*Free of Severe Natural Phenomena*	*Terminal Well-Separated from High Density Populations*
Commercial						
East Coast						
Charleston, SC	Yes (Wando)	Yes	Yes	Yes	E, H	Yes (Wando)
Fernandina Beach, FL	T	Yes	No	Some (Manatee)	H	No
Hampton Roads, VA	Yes	Yes	Yes[b]	Yes	Yes	Yes (Newport News, VA)
Jacksonville, FL	Yes (Blount Island)	Yes	Yes	Yes	H	Yes (Blount Island)
Savannah, GA	P	Yes	Yes[b]	Yes	E, H	Yes (Container Port)
Wilmington, NC	P, Some Ex[a]	Yes	Yes[b]	Yes	H	Yes
Gulf Coast						
Galveston, TX	Some P, T, Ex[a]	Yes	Yes[b]	Yes	H	No
Gulfport, MS	T[c]	Yes	No	Yes	H	No
West Coast						
Portland, OR	Yes	Yes	Yes[b]	Yes	E, V	Yes (T6)
Tacoma, WA	Yes, some Ex[a]	Yes	Yes	Yes	E, V	Yes
Military						
MOTSU (NC)	Ex[a]	Yes	Yes	Yes	H	Yes
NWS Concord (CA)	Ex[a]	Yes	Yes	Some (wetlands and Tule elk)	E	Yes

Ex= explosives, T = tourism, P = petroleum handling/storage facilities, H = hurricanes/tropical storms, V = volcanoes, E = earthquakes

[a]*Separation of piers and scheduling of spent nuclear fuel and explosive shipments on different days makes consideration of these ports appropriate*

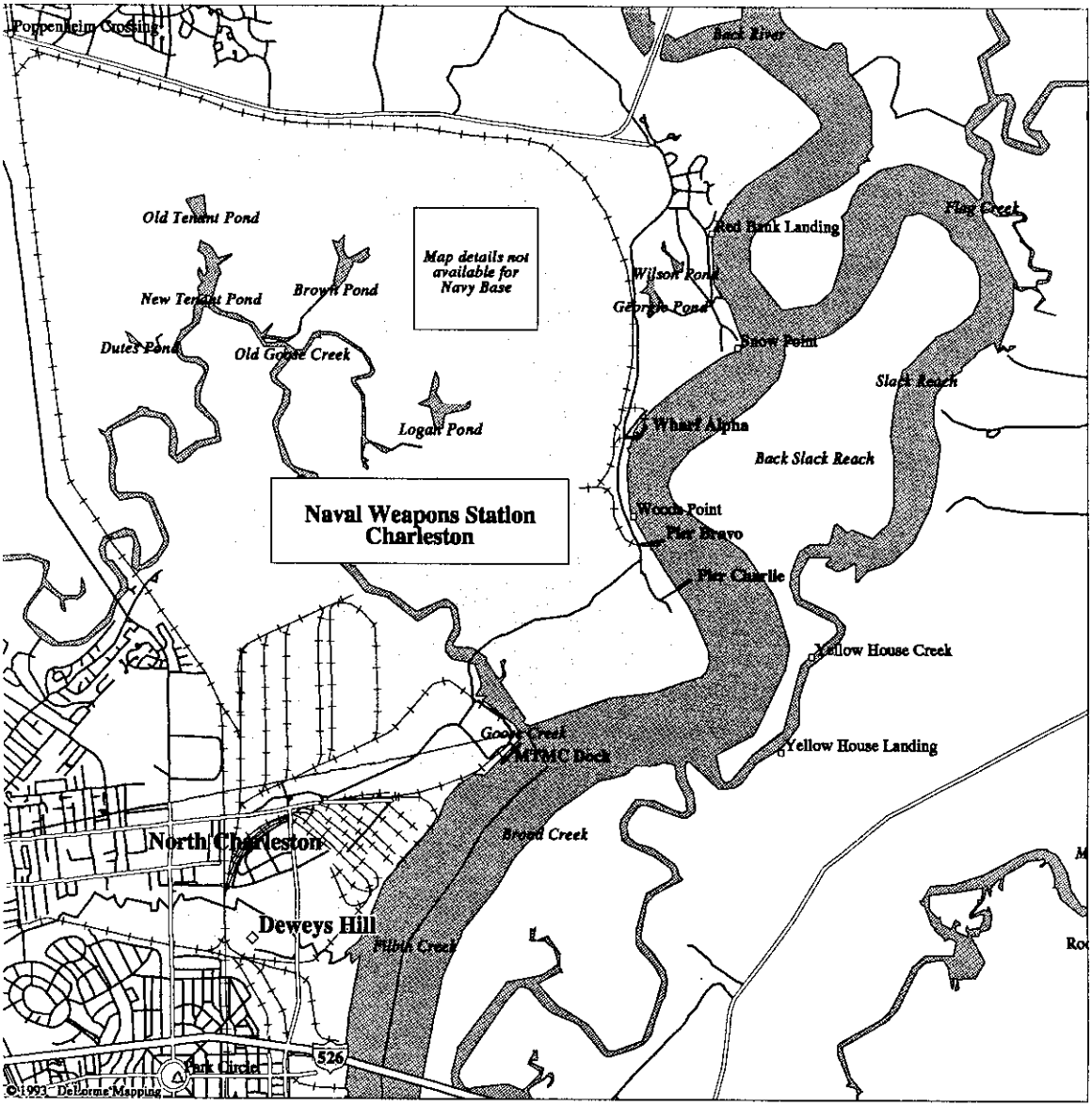
[b]*No currently designated facilities of particular hazard at preferred terminal(s)*

[c]*Extensive casino development within 1,000 feet*

D.2.1 Detailed Information on Candidate Ports of Entry

D.2.1.1 Charleston, SC (Includes the Naval Weapons Station Terminal and the Wando Terminal)

Charleston is the largest port city in South Carolina, and the greater Charleston area is one of the major seaports on the East Coast of the United States. The city of Charleston itself is located at the confluence of the Cooper and Ashley Rivers, approximately 11 km (7 mi) from the entrance from the sea. The principal wharves are along the west bank of the Cooper River except for the Wando Terminal which is along the east bank of the Wando River near Mount Pleasant, about 20 km (11 mi) from the Atlantic Ocean. The city is the center of a rich agricultural district for which it is the distribution point. The entrance to the harbor is maintained by a Federal project providing a channel depth of 10.7 m (35 ft) over the bar, through the entrance and into the major reaches of the Cooper River. The harbor is easy to access in day or night in clear weather, and is one of the best harbors of refuge on the South Atlantic coast (DOC, 1993d). The maps of the port are shown in Figures D-19 (Naval Weapons Station, Charleston) and D-20 (Wando Terminal).



LEGEND

- Population Center
- Geo Feature
- County Boundary
- Street, Road
- Major Street/Road
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)

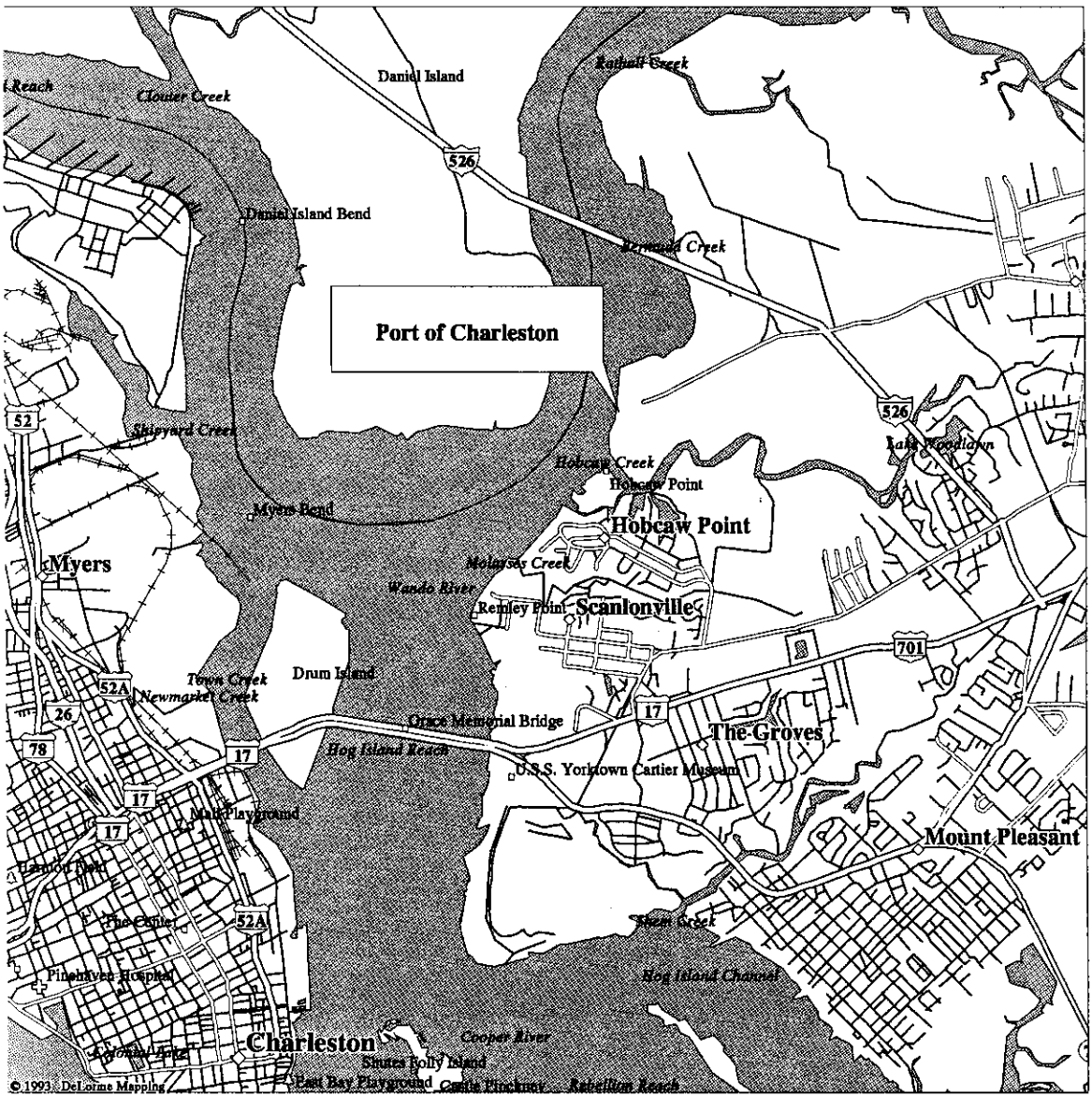
5000 Feet

1000 Meters

Mag 13.00
Tue Dec 19 11:12:29 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-19 Map of the Naval Weapons Station, Charleston, SC



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- Park
- Interstate, Turnpike
- US Highway
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)
5000 Feet
1000 Meters

Mag 13.00
Mon Mar 27 17:25:11 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-20 Map of the Wando Terminal, Charleston, SC

However, the areas to the east and southeast of the port entrance are used extensively by the U.S. Navy and other military services for training exercises which may result in occasional restrictions. Under unfavorable weather conditions current velocities in some areas have been reported as high as 2.1 meters-per-sec (4 knots) (DOC, 1993d). All of the port terminals have 12.2 m (40 ft) of water alongside at mean low water. The port is serviced by many of the world's largest container shipping lines (a total of 56), that handled 807,106 standard 20-ft container equivalents in 1991 (AAPA, 1993; FHI, 1993a). These lines provide service between Europe, the Far East, Japan, Australia and other countries (Jane's, 1992).

The South Carolina State Ports Authority owns and operates four large general cargo and container terminals within the greater Charleston area. The City of Charleston hosts two facilities (Union Pier Terminal and Columbus Street Intermodal Terminal) that were eliminated from consideration because they are not well separated from dense urban populations, and are within the city limits and subject to potential restrictions on receipt and handling of spent nuclear fuel (Jane's, 1992; AAPA, 1993).

The North Charleston Terminal is a container terminal located about 16 km (10 mi) upstream from the city of Charleston. This facility was considered to be inferior to the Wando Terminal because it requires additional transport up a heavily trafficked and more confining channel (only about 120 m (or 400 ft) wide in many reaches) on the upper Cooper River, with ships required to pass below an additional bridge (I-526) over the river (in comparison to Wando Terminal). Further, superior facilities and better separation from populated areas are found at the Wando Terminal discussed below.

In the Draft version of this EIS, only the Wando Terminal was addressed in detail. Public commentors from the Charleston area and other candidate port areas suggested that DOE further consider military ports. Since the Draft EIS was published, the Record of Decision for the SNF&INEL Final EIS (DOE, 1995) directs all aluminum-based spent nuclear fuel to the Savannah River Site. Because of the public requests and the relative proximity of the Savannah River Site to the greater Charleston area, the NWS Charleston has been added as a candidate port of entry, and detailed information is provided in the following section.

Other Pertinent Information: The City of Charleston has a city ordinance restricting the transport of spent nuclear fuel through the city. According to information gathered, the ordinance does not preclude shipment, but requires a permit and approval from the city. The Sandia National Laboratories Radioactive Materials Postnotification Database indicates that the port has not received any spent nuclear fuel since the database was initiated in October 1984 (SNL, 1994), and the NRC has no record of foreign research reactor spent nuclear fuel shipments since 1979, when NRC began approving spent nuclear fuel shipments (NRC, 1993). From discussions with senior port officials, it was determined that Wando Terminal would handle spent nuclear fuel shipments provided they had the approval of the U.S. Coast Guard Captain of the port and the Charleston Fire Department (Moise et al., 1993). Use of City terminals, rather than the Wando Terminal, has the potential for delays in the receipt and transhipment of foreign research reactor spent nuclear fuel, which could result in failing to move the foreign research reactor spent nuclear fuel from the port of entry to the management site "expeditiously." The NWS Charleston is capable of handling spent nuclear fuel shipments provided that the NWS Charleston receives appropriate program "Waivers". A program waiver would have to be issued by the Chief of Naval Operations to allow NWS Charleston facilities to be used to handle spent nuclear fuel shipments. Event waivers would have to be issued by the NWS Charleston Commanding Officer to allow each shipment to be handled. Event waivers are routine procedures used by the NWS Charleston Commanding Officer to place restrictions on conflicting activities, such as ammunition handling (Stark, 1995).

The South Carolina State Ports Authority Port Police are part of an emergency response team comprised of the local fire departments, Coast Guard, and private hazardous materials response organizations. The Ports Authority provides operating personnel basic hazardous materials training. Dock workers are trained in hazardous materials placard recognition and other basic information by the port's stevedores. Security is provided by perimeter fencing with controlled access and the South Carolina State Ports Authority Police Force, which maintains 24-hour manned access booths, patrols, and surveillance. All container terminals have secure, open and/or covered storage space for temporary storage of spent nuclear fuel if necessary (Moise et al., 1993).

The Wando Terminal is located several kilometers northeast of downtown Charleston in a relatively low population area with good access to interstate highways. Aside from general environmental concern for the wetlands around the port, there are no known special sanctuaries or habitats of concern although the port is subject to severe hurricanes (with high water) and tropical storms. It was the site of the largest earthquake (Modified Mercalli Intensity X) in the Eastern United States in recorded history, on August 31, 1886 (Bolt, 1978). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Charleston, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a moderate seismic zone with an acceleration of 0.15 g.

There are several tanker terminals and petroleum storage depots along the west bank of the Cooper River downstream of the North Charleston Container Terminal (which is also immediately adjacent the Naval Weapons Station off Goose Creek). However, there do not appear to be any conflicting cargoes or activities at the Wando, Columbus, or Union Pier Terminals. The port officials contacted indicated that they believe that radioactive shipments have been made through the port in the past, but they were not sure if spent nuclear fuel had been handled (Moise et al., 1993).

Environmental Conditions

The State of South Carolina has given the lower portion of the Wando River two different water quality classifications. The water is classified as SFH or SA. SFH waters are shellfish harvesting waters and SA waters are suitable for primary and secondary recreation and for other water uses requiring lower quality. According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for James Island, SC, the Wando Terminal and the NWS Charleston are located in a mid-salinity estuarine habitat (generally 5 to 16.5 parts per thousand). The Charleston Harbor, which is traversed on the way to either terminal, is located in a high-salinity estuarine habitat (generally 16.5 to 30 parts per thousand) (FWS, 1980b).

The State of South Carolina has also classified the water quality of the portion of the Cooper River above the confluence with the Ashley River as SB (SB waters are tidal saltwaters suitable for secondary contact recreation, crabbing, and fishing, except the harvesting of clams, mussels, or oysters for market purposes and human consumption). The waters of Goose Creek, upstream of the confluence with the Cooper River to the dam at the Charleston Waterworks, are also Class SB (Department of the Navy, 1994).

The lower Wando River, the Charleston Harbor, and the NWS area support a large number of aquatic and terrestrial species. According to the South Carolina Heritage Trust, no rare, threatened, or endangered species or communities have been recorded in the area near the Wando Terminal (McBee, 1994). State or Federally protected endangered or threatened aquatic species in the vicinity of the Charleston Harbor include the Shortnose sturgeon, Atlantic sturgeon, and the American shad. Bachman's warbler is a Federally protected bird species also found in the vicinity (FWS, 1980b).

In addition, the U.S. Fish and Wildlife Service reports that several protected marine species are known to occur in Charleston County (Banks, 1994). These are the west indian manatee (endangered), Kemp's ridley sea turtle (endangered), leatherback sea turtle (endangered), loggerhead sea turtle (threatened), and the green sea turtle (threatened). Protected bird species include the arctic peregrine falcon (threatened), bald eagle (endangered), wood stork (endangered), red-cockaded woodpecker (endangered), and the piping plover (threatened).

In recent years, two pairs of bald eagles (*Haliaeetus leucocephalus*) nested on the NWS Charleston. One nest was located north of Foster Creek near the POMFLANT storage and activity area, over four miles north of Wharf Alpha. The other was located on the golf course west of Pier X. Their nest tree was destroyed by Hurricane Hugo and the pair have relocated to Big Island, located north of Foster Creek (Department of the Navy, 1994).

Prior to Hurricane Hugo, 12 colonies of red-cockaded woodpeckers, *Picoides borealis*, were known at the NWS Charleston. The red-cockaded woodpecker requires mature pines old enough to be susceptible to red heart disease, which makes the wood soft enough for these small woodpeckers to create a cavity. Such trees are generally at least 70 to 75 years old. Nearly all trees this age were destroyed by Hurricane Hugo, so it is unlikely that the colony could reestablish at this site in the near future (Department of the Navy, 1994).

Wetlands are plentiful on and adjacent to the NWS Charleston. Three basic habitat types found within the area's wetland ecosystem are forested wetlands, nonforested wetlands, and open water. The station's wetland habitats had previously been identified according to the National Wetland Inventory classification system. Within this classification system, the station's wetlands had been placed in four major categories: estuarine, palustrine, lacustrine, and riverine. Based upon that classification, the station contained 1,356 acres of estuarine, 1,730 acres of palustrine and 131 acres of lacustrine wetlands (Department of the Navy, 1994). Field investigations have been performed at the NWS, and no rare or endangered plants or animals were observed (Department of the Navy, 1990 and 1994).

Wetlands at the station contain potential habitat for the flatwoods salamander (*Ambystoma cingulatum*), which is pending placement on the threatened or endangered species list. However, two spot checks of the area by the NWS Charleston environmental personnel have failed to locate either adults or larvae. It has been indicated that a detailed study may be performed of the area in the future (Department of the Navy, 1994).

Climatic Conditions

In general, the elevation of the area ranges from sea level to approximately 6 m (20 ft) on the peninsula. The climate of this region is temperate, primarily due to its close proximity to the Atlantic Ocean. The prevailing winds are generally northerly in the fall and winter months, becoming more southerly during the summer months. This type circulation serves to "warm" the region during winter and "cool" it during the summer. Summer is the rainy season in Charleston, with the city receiving 41 percent of the annual total rainfall during these months. Except for the occasional tropical storm or hurricane, the majority of this rain occurs during afternoon and evening thunderstorms. The late summer and early fall brings the highest probability of tropical storm activity to the Charleston, SC area. The fall season is a transitional period, where temperature extremes are rare and sunshine is abundant. The winters in this area are mild with periods of rain. However, in contrast to the summer, the winter rains tend to be steady and uniform, and last for several days. The most unstable period in this region is spring when the confluence of warm moist

tropical air and cool dry continental air increase the occurrence of severe weather in this region. The average earliest freeze in this area is in early December and the average last frost is in late February (NOAA, 1992c).

D.2.1.1.1 Naval Weapons Station - Charleston

The NWS Charleston is located on the west bank of the Cooper River, north of the city of North Charleston in southeastern Berkeley County, South Carolina. The station occupies about 7080 hectares (17,500 acres) along a 14-km (9-mi) stretch of the Cooper River, starting about 30 km (19 mi) from the Atlantic Ocean. The primary missions of the NWS Charleston are to provide material support for assigned weapons and weapon systems, to provide housing and community support facilities for personnel assigned to the Charleston area, and to do additional tasks such as home porting and logistics support for ammunition ships, and other fleet and shore activities dealing with weapons. Major tenant activities on the station include the new Army Strategic Mobility Logistics Base, a Propulsion Training Facility and the Military Traffic Management Command, an Army organization (Department of the Navy, 1990 and 1994). The Army Strategic Mobility Logistics Base is being constructed on the formerly Polaris Missile Facility Atlantic site (Lewis, 1995).

In selecting a port this far from the open ocean, DOE considered the navigation safety through the Charleston Harbor and up the Cooper River. As previously described, the harbor experiences a significant amount of deep draft traffic, and is accustomed to managing ship traffic in several 10.7 m (35 ft) deep channels. The Navy maintains a 10.7-m (35-ft) deep channel up the Cooper River to all relevant piers and wharfs of the NWS. The channel is as narrow as 120 m (400 ft) in some areas and extends to the edge of piers that may be in use for handling ammunition, petroleum products or other hazardous cargo (DOC, 1993d). However, with proper management of the harbor by the U.S. Coast Guard and pilots, and the proper planning by the U.S. Navy, additional assurance of a safe transit can be provided.

The Charleston Harbor Navigational Guidelines (DOC, 1993d) identifies areas of particular concern in the Cooper River area and provides guidelines for navigation, overtaking and passing in these areas. The Charleston Branch Pilots Association procedures (Commissioners of the Pilotage, 1995) require strict adherence to these guidelines for deep draft vessels and vessels carrying hazardous materials. The Coast Guard Captain of the port has broad regulatory authority over all port activities, and procedurally delegates control of vessel movements to the Charleston Branch Pilots Association (Bennett, 1995). This authority includes activities ranging from minor additional traffic restrictions on passing to establishing and enforcing a moving safety zone around a ship traversing the harbor area. A moving safety zone requires advance notification in the *Federal Register*, but is routinely done in areas in which hazardous materials are carried (USCG, 1994c). The Coast Guard and the South Carolina State Police have conducted exercises on moving a ship up the Cooper River under the threat of terrorist activities (Millar, 1995).

The U.S. Navy is the only significant user of the Cooper River north of the North Charleston facilities, and is a major user of the Cooper River. Cooper River transits can be planned by the Navy so that they would not conflict with other Naval activities and ship movements. The Commanding Officer of the NWS Charleston would determine which facility is most appropriate for each shipment based on the characteristics of the vessel carrying the fuel, the planned mode of overland transportation and any conflicting activities at the NWS Charleston. The NWS Charleston has four facilities that can handle spent nuclear fuel. The northern facility is Wharf Alpha, which is more than 12 m (40 ft) wide and has about 300 m (1000 ft) of useful berthing area dredged to 12.3 m (40 ft) deep. Wharf Alpha was previously used to service Polaris missile-carrying nuclear submarines, including removal and replacement of nuclear missiles. This function is no longer necessary, and submarines do not regularly visit the NWS Charleston.

Wharf Alpha is currently being used to service and load U.S. Navy ammunition ships. With the construction of the Army Strategic Mobility Logistics Base, the Army plans to upgrade in 1998 the rail lines serving Wharf Alpha as well as expanding the wharf itself. The wharf has a mobile crane, and is served by truck and rail loops, meaning that either trucks or trains can drive directly onto the wharf, load, and exit without turning around or reversing direction. Pier Bravo, located about 1 km (3281 ft) south of Wharf Alpha, is also used for ammunition handling. Pier Bravo protrudes 300 m (983 ft) into the Cooper River, with 214 m (703 ft) on the south side and 166 m (545 ft) on the north side of useful berthing in 11.9 m (39 ft) deep waters. The pier has a mobile crane that can off-load spent fuel casks directly onto trucks or trains parked on the pier.

Pier Charlie, located about 1000 m (3281 ft) south of Pier Bravo, has been used for berthing nuclear submarines and a tender. It is not in regular use, but is maintained as a backup for Wharf Alpha and Pier Bravo. A portable crane, which is capable under ideal conditions of lifting 36,280 kg (40 tons), can be moved to Pier Charlie and used to off-load casks onto trucks. The Military Traffic Management Command dock is located about 3 km (9840 ft) down the river from Pier Charlie. This facility is in regular use for roll on/roll off military cargo. The Military Traffic Management Command dock is a safe distance from any weapons handling operations at Wharf Alpha or Pier Bravo. However, the same portable crane that could be used at Pier Charlie would be used at the Military Traffic Management Command dock.

The portable cranes available on the NWS Charleston may not be able to off load some larger casks, especially if they have to extend horizontally over the ship. The DOE and the NWS Charleston would plan to use shipboard cranes or rented cranes, or schedule these certain shipments to dock at Wharf Alpha or Pier Bravo. Commercial 82 metric ton (90 ton) and 118 metric ton (130 ton) cranes are available in the greater Charleston, SC area (Silver, 1995). Additionally, Pier Charlie and the Military Traffic Management Command dock are not directly served by rail. However, several rail heads on the NWS are in secure and isolated locations that can be used to load the fuel from trucks to trains.

The NWS Charleston is a fenced facility with several guarded gates and a 24 hour security force. Additional guard facilities and temporary barricades are used to keep unnecessary personnel away from ammunition handling, and could be used for this program. The on-base emergency facilities are appropriate for fire and rescue response to ammunition handling and other potential accidents. Additionally, the Propulsion Training Facility includes two operating nuclear reactors. The staff is adequately trained and equipped to make initial response and assessment of any accident with the potential for radioactive release or radiation exposure.

The NWS Charleston facility is capable of supporting the implementation of a policy to accept foreign research reactor spent nuclear fuel. The facility is experienced in handling nuclear and hazardous cargo, can be safely reached from open ocean, has adequate facilities, ready access to truck and rail transportation, and low human populations. Desirable attributes include excellent emergency response capabilities, acceptable environmental concerns, moderate concerns from severe natural phenomena, separation from urban population, no local restrictions and secure short term storage. The risk associated with the conflicting uses can be mitigated by rigorous compliance with Naval operation procedures.

The 1990 population within 16 km (10 mi) of the Wharf Alpha was 209,188. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 46,200; Oak Ridge Reservation, 108,000; Idaho National Engineering Laboratory, 498,000; Hanford Site, 550,000; and Nevada Test Site, 540,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: Savannah River Site,

301 km (188 mi); Oak Ridge Reservation, 644 km (402 mi); Idaho National Engineering Laboratory, 3,910 km (2,441 mi); Hanford Site, 4,580 km (2,858 mi); and Nevada Test Site, 3,930 km (2,543 mi). Distances along rail routes are slightly longer.

D.2.1.1.2 Wando Terminal

The South Carolina State Port Authority Wando Terminal is an ultra modern marine facility that is the designated hazardous materials terminal for the port ("Facility of Particular Hazard"), and is a superior terminal for receipt of spent nuclear fuel. In addition to being outside the city limits of Charleston and not subject to any potential restrictions on receipt and handling of spent nuclear fuel, it is closest to the Atlantic Ocean, and has outstanding facilities. The terminal has 3 container berths and 67.7 ha (167 acres) of paved container storage yard. It has a 428 m (1,400 ft) by 427 m (1,400 ft) turning basin. It currently has 740 m (2,430 ft) of lineal berthing space, but a fourth berth and 35.2 ha (87 acres) of additional paved storage area is currently under construction. The terminal is 8.1 km (5 mi) from the Mark Clark Expressway (I-526), which by-passes most of the city of Charleston and joins Interstate 26 at North Charleston. Of the four terminals in the Port of Charleston, Wando is the only one without direct rail service, requiring trucking of containers about 15 km (9 mi) to intermodal rail yards serviced by the CSX and Norfolk Southern Railroads. This was not considered a serious problem, since most shipments are anticipated to be carried overland by trucks.

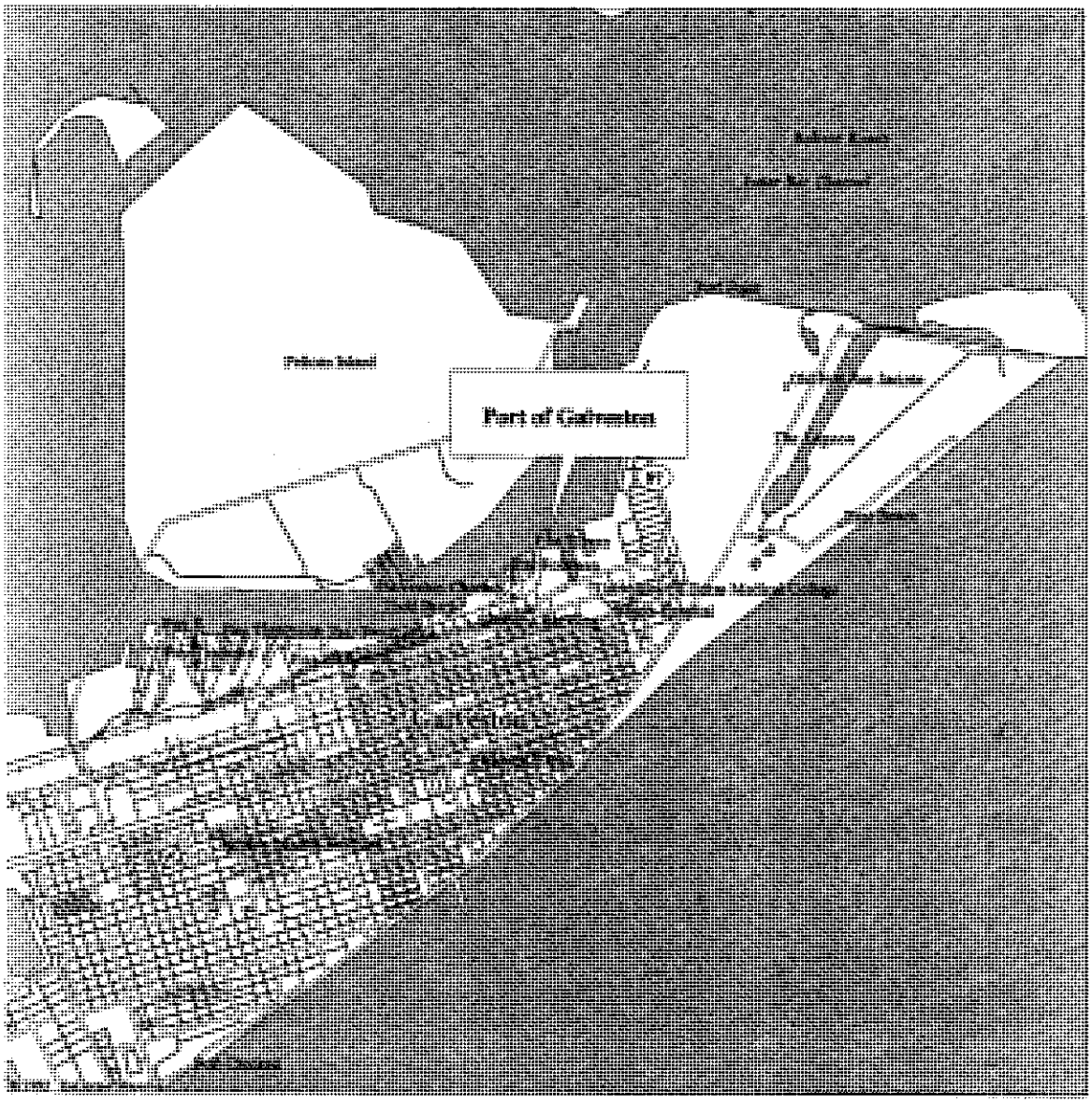
The 1990 population within 16 km (10 mi) of the Wando Terminal was 233,434. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 65,700; Oak Ridge Reservation, 127,000; Idaho National Engineering Laboratory, 518,000; Hanford Site, 569,000; and Nevada Test Site, 559,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: Savannah River Site, 325 km (203 mi); Oak Ridge Reservation, 668 km (417 mi); Idaho National Engineering Laboratory, 3,935 km (2,456 mi); Hanford Site, 4,600 km (2,873 mi); and Nevada Test Site, 4,098 km (2,558 mi). Distances along rail routes are slightly longer.

D.2.1.2 Galveston, TX

The Port of Galveston is about 16 km (10 mi) from the Gulf of Mexico via the Galveston channel. The City of Galveston, TX, occupies the entire width of the east end of Galveston Island. The shipping wharves are on the north side of the island and the Gulf of Mexico is on the south. The Port of Galveston is located in the heart of the City (DOC, 1992a). A map of the port is shown in Figure D-21.

As stated in the Coast Pilot, the Port of Galveston offers a short route to the sea and, together with the deep and easily navigated channel and excellent port facilities, enables Galveston to handle cargo most expeditiously and economically (DOC, 1992a). A Federal project provides for an entrance channel and an outer bar channel both dredged to 12.8 m (42 ft), thence 12.2 m (40 ft) to Galveston. The Port of Galveston is a multi-terminal port complex located on the northeastern end of Galveston Island, only 15 km (9.3 mi) from the entrance buoy to the open sea. Overall tonnage reported for 1991 was 4,159,233 metric tons (4,584,723 tons), of which approximately 17 percent (703,511 metric tons or 773,862 tons) was containerized freight (over 70,000 20-ft equivalent units). Roughly 77 percent of the tonnage was dry and liquid bulk, much of it grain (AAPA, 1993).

The Port of Galveston is a separate utility of the City of Galveston with its powers established by the City Charter. The Charter provides that all city-owned wharf and terminal properties be set aside and controlled, maintained, and operated by a "Board of Trustees of the Galveston Wharves."



LEGEND

- Population Center
- State Route
- Geo Feature
- Large City
- Street, Road
- Major Street/Road
- State Route
- Railroad
- River
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

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"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-21 Map of the Port of Galveston, TX

Principal container lines and the areas they serve include: Lykes Brothers — North Europe, Mediterranean, Mexico and West Coast of South America; Deppe Line — North Europe; Companhia Maritima Nacional — Brazil/Mexico; Compania Chilean Navegacion Interoceanica — South America/Mexico, and Del Monte/Network Shipping — Guatemala/Mexico (Jane's, 1992; AAPA, 1993).

Pier 10 Container Terminal: This is Galveston's principal container handling facility. It is leased to a private operator, Container Terminal of Galveston, Inc., who operates the facility as a public terminal. This facility has two berths, four container cranes, and 19.83 ha (49 acres) of paved storage area. The Port of Galveston owns an additional ten (10) open-dock ship berths and 20 berths with shipside warehouses used for breakbulk and other cargoes. The Container Terminal of Galveston, Inc. has two berths with a total length of 410 m (1,346 ft). Depth alongside the Container Terminal of Galveston, Inc. at mean low water is 12.2 m (40 ft). Crane capacities on Container Terminal of Galveston, Inc's. Pier 10 are three 50.8 metric tons (56 ton) container cranes and one 61.0 metric tons (67 ton) container crane. All cranes are equipped with 40.6 metric tons (45 ton) capacity spreaders.

The Container Terminal of Galveston, Inc. has a controlled all-weather truck entrance and interchange area. The terminal is connected to Interstate Highway 45 on the mainland by the 9.3 km (5.8 mi), four-lane State Highway 87 and two 2.8 km (1.75 mi) causeways that cross the southwest end of Galveston Bay. The island portion of the limited access route is through densely populated built-up areas. The Container Terminal of Galveston, Inc. is served by four major railroads: the Burlington Northern, Santa Fe, Southern Pacific, and Union Pacific Lines. Galveston Railway, Inc., provides terminal connections and performs switching of all rail traffic. An intermodal container transfer terminal is located within the container terminal and trackage extends to within 30.5 m (100 ft) of ship berths (Jane's, 1992; AAPA, 1993; Schultz, 1993).

Other Pertinent Information: The Port of Galveston has its own security force that provides 24-hour surveillance of its terminals. Container Terminal of Galveston, Inc. is fenced and has controlled access. An area is provided for segregation and temporary storage of hazardous cargoes.

The Port of Galveston's Director of Operations was unaware of any regulations prohibiting the importation of spent nuclear fuel (Schultz, 1993). The port occasionally handles hazardous materials, including Class A explosives (Schultz, 1993). NRC records indicate the port has not handled foreign research reactor spent nuclear fuel since at least 1979 (NRC, 1993).

The container terminal operator is responsible for handling hazardous materials emergencies at the Container Terminal of Galveston, Inc. facility. The Port of Galveston relies on the Galveston Fire Department's hazardous materials team and/or highly trained hazardous materials personnel at refineries located some 16-24 km (10-15 mi) away. The West Gulf Employers Association holds training courses for longshoremen which Port of Galveston terminal personnel also attend (Schultz, 1993).

Galveston is a major resort and tourist center for the Southwest United States. There is a 2.95 ha (7.3 acres) waterfront tourist attraction at "Pier 21" close to the historic district. A hospital is located across the street from the general cargo berths (Schultz, 1993). A public park on Pelican Island, reached by causeway, is located across the Intracoastal Waterway from the port. A cruise ship terminal is located at Pier 25 in the heart of the port complex and there is a tanker terminal on Pelican Island across from the port at its southern end. The greatest source of potential conflict is the heavy tanker traffic utilizing the Galveston entrance channel en route to Texas City and the Port of Houston petroleum/petrochemical terminals. Houston is the third most active port in the United States in terms of tonnage handled

(IPA, 1993). The U.S. Coast Guard accident data for the period 1991-1993 indicate 52 reported accidents in the Galveston Bay area (USCG, 1994b). This includes ship traffic bound for the Houston area and also includes barge accident data.

Other than general heightened environmental awareness, there are no known sensitive environmental areas in the Port of Galveston area (Schultz, 1993). The port is subject to hurricane and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Galveston, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 census population within 16 km (10 mi) of the port terminals was 73,322. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 403,000; Oak Ridge Reservation, 337,000; Idaho National Engineering Laboratory, 526,000; Hanford Site, 575,000; and Nevada Test Site, 595,000. Populations along rail routes to these sites are slightly larger for Savannah River Site and Oak Ridge Reservation, but are slightly less for Idaho National Engineering Laboratory, Hanford Site, and Nevada Test Site. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,600 km (1,000 mi); Oak Ridge Reservation, 1,550 km (963 mi); Idaho National Engineering Laboratory, 3,070 km (1,911 mi); Hanford Site, 3,740 km (2,327 mi); and Nevada Test Site, 3,000 km (1,862 mi). Distances along rail routes are slightly longer.

Environmental Conditions

A large number of aquatic and terrestrial species frequent the Galveston Bay area. A variety of birds migrate, winter, and breed along the Texas Coast including shorebirds, songbirds, waterfowl, and raptors (Breslin, 1993; FWS, 1992). These endangered/threatened bird species include the brown pelican, peregrine falcon, bald eagle, Attwater's greater prairie-chicken, piping plover, and the eskimo curlew (State-threatened only). Endangered/threatened marine mammals and sea turtles also are found in the coastal bay systems and the Gulf of Mexico. Galveston Bay is within the range of the green, hawksbill, Kemp's ridley, leatherback, and loggerhead sea turtles. While no protected species are known to be located within the Port of Galveston, significant populations of the endangered brown pelican and the piping plover exist nearby (Werner, 1994). The U.S. Fish and Wildlife Service reported that as many as 600 brown pelicans have been sighted loafing on the north end of Little Pelican Island, which is approximately 5.6 km (3.5 mi) northwest of the port. In addition, approximately 125 pairs nested and produced 90 young for the year at this site in 1994, the first time that brown pelicans had successfully nested in Galveston Bay in over 40 years. Wintering populations of the threatened piping plover use the northeastern end of Galveston Island and the southeastern end of Bolivar Peninsula. Of the 3,187 birds observed during the first Gulf Coast count of wintering piping plovers, 1,904 were observed on the Texas coastline (Werner, 1994).

A great amount of commercial and recreational fishing occurs in Galveston Bay and the Gulf of Mexico. Shellfish are the most important commercial species, particularly shrimp, followed by eastern oysters and blue crabs (TPWD, 1989a). The most valuable finfish landed from the Galveston Bay system are black drum, and mullet. In 1988, a total of 5,077,170 kg (11,169,773 lbs) of shellfish valued at $13,489,146 was landed from the Galveston Bay System; a total of 224,536 kg (493,980 lbs) of finfish valued at $226,140 was also landed. The major recreational species of fish that were caught in the Galveston Bay system in 1987-1988 were: Atlantic croaker, sand seatrout, spotted seatrout, southern flounder, black drum, and red drum (TPWD, 1989b).

While the port area is highly developed, a wide variety of marine, estuarine, and lacustrine wetlands exist along Galveston Bay, including a large portion of Pelican Island, directly west of the port. Wetlands also occupy the majority of the far northern end of Galveston Island (FWS, n.d.a.).

Climatic Conditions

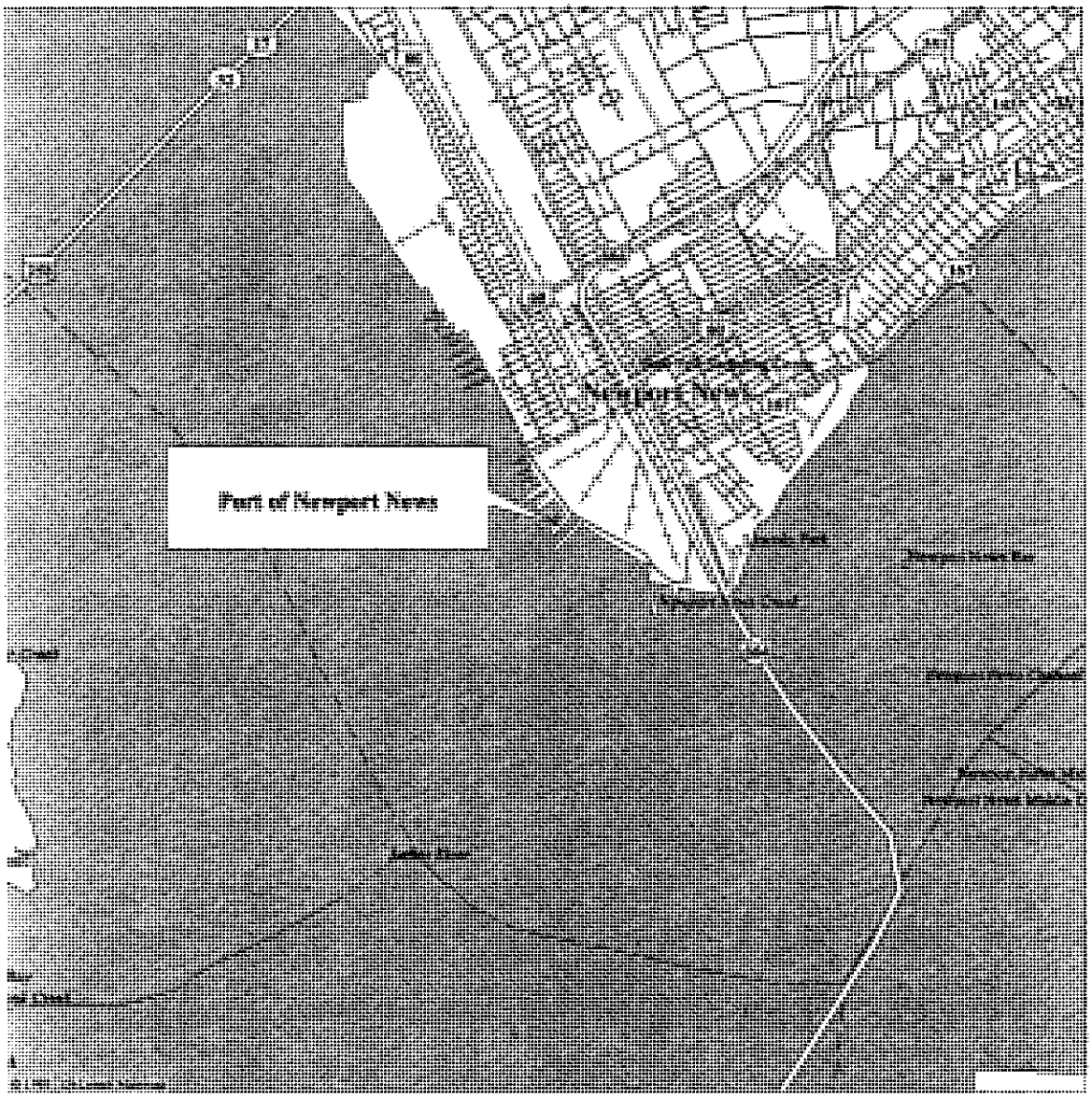
The City of Galveston is bounded on the southeast by the Gulf of Mexico and on the northwest by Galveston Bay. Thus, the climate of the Galveston area is predominantly marine, with periods of modified continental influence during the colder winter months when cold fronts from the northwest sometimes reach the Texas coast. Because of its coastal location, sub-freezing temperatures are rare, and higher than normal humidities prevail throughout the year. Summer rainfall is highly variable across the island due to thunderstorms and the local sea breeze circulation. Winter precipitation comes mainly from frontal activity and onshore flow, which produces slow, steady rains under a low stratus cloud deck. The island has been subject at infrequent intervals to major tropical storm systems with hurricane-force winds (NOAA, 1993c).

D.2.1.3 Hampton Roads, VA (Includes the Combined Terminals at Newport News, VA; Norfolk, VA; and Portsmouth, VA)

Hampton Roads is one of the world's foremost bulk cargo harbors. It is a multi-terminal port with privately and publicly owned marine cargo handling facilities located at the southwest corner of the Chesapeake Bay at the confluence of the James and the Elizabeth Rivers. The port is about 26 km (16 mi) from the Virginia Capes and the entrance from the Atlantic Ocean. The major terminals located on the Elizabeth and James Rivers are approximately another 10 to 13 km (6 to 8 mi) from the Bay (DOC, 1993c). The port includes the ports and cities of Norfolk, Portsmouth, and Newport News. Adjacent communities include the cities of Chesapeake and Virginia Beach. The maps of the port are shown in Figures D-22 (Newport News), D-23 (Norfolk), and D-24 (Portsmouth).

In 1992, Hampton Roads handled approximately 5.9 million metric tons (6.5 million tons) and 875,000 20-ft equivalent units of containerized cargo, including large amounts of radioactive materials (primarily uranium dioxide). The port ranks closely with the port of Charleston as the second or third most active container port for the East and Gulf Coasts (DOE, 1994d). The port is serviced by more than 75 ship lines that serve the port on a regular basis and provide approximately 4,000 sailings a year to many countries of the world, including Scandinavia, Europe, the Mediterranean, Near East, Mideast, Far East, Africa, Japan, and South America. A partial listing of lines include Alianca, American-Africa-Europe, American Transport, Argentine Line-ELMA, ACL, Atlantic Express, Bank, Ceylon Shipping, CGM, Chilean Line, Cho Yang Shipping, COSCO, DB Turkish Cargo Lines, Deppe, DSR Senator, Eimskip, Evergreen, Farrell, Hapag-Lloyd, Hoegh, Italian Line, Ivarian Lines, Jugolinja, K Line, Lloyd Basilero, Lykes Lines, Maersk, Mediterranean Shipping, Mitsui OSK, NSCSA, Nedlloyd, Neptune Orient, Netumar Lines, NYK, OOCL, Ocean Star Container Line, P & O, PT Djakarta Lloyd, Safbank Lines, Sea-Land Service, Shipping Corp. of India, Spanish Line, Tokai Shipping, Toko Kaiu Kaisha Ltd., Torm West Africa, United Arab Shipping, Venezuelan Line Walleniuis, Waterman, Wilhelmsen, Yang Ming Line, and Zim (Jane's, 1992; AAPA, 1993).

Because Hampton Roads and its approaches from the Virginia Capes handle a large amount of shipping, traffic separation schemes have been established for the control of maritime traffic. The controlling depth in the Deep Water Route from the Virginia Capes is 15.2 m (50 ft), except for one 14.3 m (47 ft) location. Projected depth for the Hampton Roads channel varies from 15.2 m to 16.7 m (50 to 55 ft). Depth alongside terminals at Newport News and Portsmouth is about 11.6 m (38 ft), while at the Norfolk terminal it is 12.5 m (41 ft) (DOC, 1993c; Jane's, 1992).



LEGEND

- Population Center
- State Route
- Geo Feature
- Large City
- Park
- Interstate, Turnpike
- US Highway
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Open Water

Scale 1:50,000 (at center)
5000 Feet
1000 Meters

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"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-22 Map of the Port of Newport News, VA

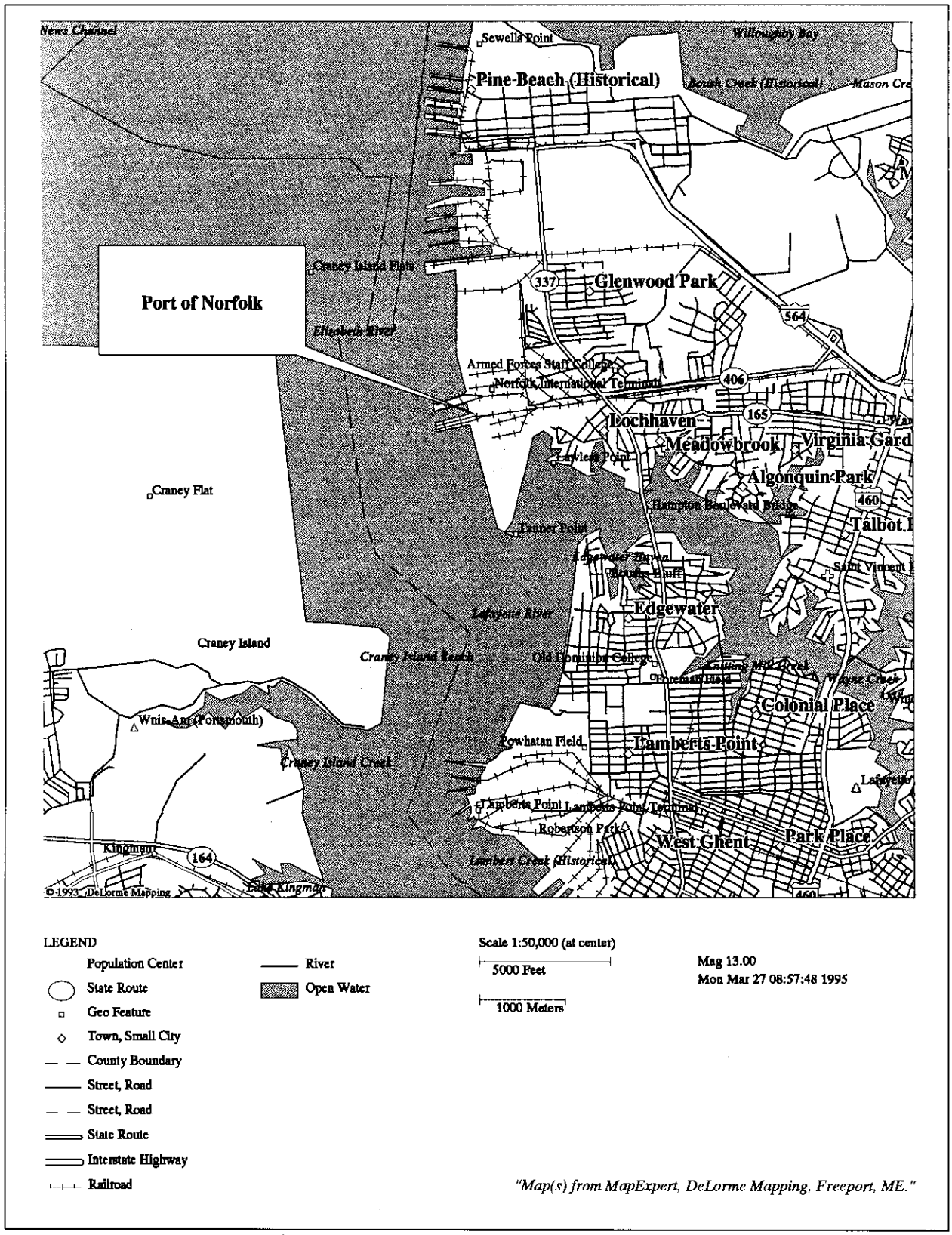


Figure D-23 Map of the Port of Norfolk, VA

Figure D-24 Map of the Port of Portsmouth, VA

Changing weather can also be a concern as noted in the U.S. Coast Pilot: "Weather deterioration in the lower bay is often sudden and violent and constitutes an extreme hazard to vessels operating or anchoring in this area. The proximity of the bridge-tunnel complex to main shipping channels adds to the danger. Currents in excess of 1.5 meters-per-sec (3 knots) can be expected in this area" (DOC, 1993c).

The presence of three major vehicle tunnels (Chesapeake Bay tunnels, and Hampton Roads Tunnel with associated bridges) under the shipping channels are also sources of risk from ship collisions, especially in fog or during bad weather. Overall, however, the transit is direct and well-managed (DOC, 1993c).

The terminals of primary interest are owned by the Virginia Port Authority, that is a state agency reporting to the Secretary of Economic Development. The Virginia Port Authority's three large, general cargo terminals within the Greater Hampton Roads harbor area include Norfolk International Terminals which is a large container port that includes Sewell's Point Docks (a breakbulk facility), Portsmouth Marine Terminal, and Newport News Marine Terminals. These Terminals are operated by the Virginia International Terminals (the operating arm of the Virginia Port Authority). Lambert's Point Docks, a large breakbulk terminal owned by Norfolk Southern Railroad is also located on the Norfolk waterfront, but lacks container cranes. All three terminals are located in commercial port districts of their respective cities, somewhat separated from other community activities, in areas dedicated primarily to port industrial usage. The three Virginia Port Authority terminals are discussed below in subsections by terminal (Jane's, 1992; AAPA, 1993; FHI, 1994b; DOE, 1994d; VPA, 1994).

Other Pertinent Information: There are no regulatory restrictions prohibiting the receipt and handling of spent nuclear fuel in the port. Compliance with hazardous materials regulations (49 CFR) is the primary requirement. The Portsmouth Marine Terminal has had extensive experience in the receipt and handling of spent nuclear fuel shipments in the recent past, and Norfolk International Terminal and Newport News Marine Terminal also have had some experience (SNL, 1994; NRC, 1993). There appears to be little or no conflict with other hazardous cargoes, including petroleum products, naval weapons depots, etc., in the immediate vicinity of the three Virginia Port Authority terminals. The Virginia Port Authority depends on the Hampton Roads Emergency Team for response to hazardous materials accidents within its terminals. Hampton Roads Emergency Team consists of the fire departments of Norfolk, Portsmouth, and Virginia Beach, in liaison with the U.S. Coast Guard. Chief White of the Portsmouth Fire Department is in charge of the team, which also has ties to the State Emergency Team. All of the Virginia Port Authority terminal operating personnel and longshoremen are currently trained in hazardous materials awareness. Security for the port is provided by perimeter fences and the Virginia Port Authority's Port Police, which maintain 24-hour patrol and surveillance at all three terminals. The state of Virginia's Safety Manual sets forth the rules and policies for operations, including, among other things, hazardous cargoes, container control, emergency procedures and general safety, and provides the policy for receipt and handling of radioactive materials, including emergency response, personnel protection, facility protection, environmental protection and cargo protection (Edwards and Drews, 1993).

All three terminals are located in a large urban area in which congestion is to be expected. Of the three terminals, Portsmouth Marine Terminal is located closest to residential and downtown areas; however, Portsmouth is a relatively small city in both area and population, and it is only a short distance from the terminal to more sparsely populated rural areas. Conversely, truck shipments from Norfolk International Terminals, the terminals closest to the sea, must travel about 38 km (24 mi) of heavily trafficked Interstate through built-up sections of Norfolk, Virginia Beach, and Chesapeake before reaching Bowers Hill (a rural area). The comparable distance from Portsmouth is about 6 km (4 mi).

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Hampton Roads, the Uniform Building Code requires buildings to withstand wind speeds up to 140 km/hr (90 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

Environmental Conditions

The lower Chesapeake Bay - Hampton Roads area is located on the coastal plain of southeastern Virginia. This area is rather flat and is dissected by numerous bays, rivers, creeks, and wetlands including saltwater marshes, bogs, and swamps (DOE, 1995). However, the areas in the vicinity of the Ports of Newport News, Norfolk, and Portsmouth are highly developed and the waterfronts consist largely of piers and bulkheads associated with the various shipyards, shipping terminals, warehouses and railroad yards that comprise this heavily utilized harbor area. The U.S. Fish and Wildlife Service reports that there are no Federally-listed or proposed listing for endangered or threatened species within a one-mile radius of the Portsmouth or Newport News Terminals (Mayne, 1994). However, rare, threatened, and endangered plant and animal species may be found around the cities of Newport News, Norfolk, and Portsmouth (O'Connell, 1994).

The waters of the Chesapeake Bay and Hampton Roads can be classified as a high-salinity estuarine habitat (generally 16.5-30 parts per thousand), while the Elizabeth and James Rivers, in the vicinity of the ports addressed in this section, are classified as mid-salinity estuarine habitat (generally 5-16.5 ppt) (FWS, 1980d). A number of aquatic species can generally be found in mid-salinity estuarine habitat in this area. Of particular note is the Atlantic sturgeon, a state-endangered species that migrates through these areas. In addition, the eastern oyster is common in the Elizabeth River, in the vicinity of the Norfolk International Terminal and the Portsmouth Marine Terminal. Blue crabs, eastern oysters, and hard clams are also found in the vicinity of the Newport News Marine Terminal (FWS, 1980d). However, Hampton Roads, the Elizabeth River, and portions of the James River (including that portion along the port of Newport News) have been designated as the "Condemned Shellfish Area No. 7" by the Virginia State Department of Health (West, 1994). Shellfishing is either restricted (special permits) or prohibited in this area. The State of Virginia also reports that a fishing health advisory is in effect for the James River and its tributaries due to Kepone contamination. In addition, the Southern Branch of the Elizabeth River, which is upstream of the Norfolk International and Portsmouth Marine Terminals, only partially supports water column standards for dissolved oxygen and sediment standards for lead. The waters in the Hampton Roads area are considered to be "swimmable" by the State of Virginia (West, 1994). The Tidewater area is part of the Mid-Atlantic flyway, and the numerous waterways and wetlands in this area are utilized by many migratory birds that pass through or winter in this region. There is generally a lack of suitable habitat or forage areas in the immediate vicinity of these ports. However, the Ragged Island Wildlife Management Area, located across the James River from the Port of Newport News, is used as a migratory area for waterfowl. Nesting areas for the great blue heron and the yellow-crowned night heron, both State-protected species, are reportedly located on the Lafayette River, approximately 3.2 km (2 mi) upstream of the Norfolk International Terminal (FWS, 1980d). More recently, the U.S. Fish and Wildlife Service indicated that a yellow-crowned night heron rookery consisting of eight nests was documented in the vicinity of the Norfolk Terminal (Mayne, 1994).

There are no known areas of special environmental concern other than the growing interest in preservation of the Chesapeake Bay and its tributary rivers. While the Dismal Swamp National Wildlife Refuge is located about 16 km (10 mi) from the two terminals on the Elizabeth River, the water drainage from the swamps is toward the port, and would not normally carry waterborne radioactivity into the swamp. Further, the swamp is far enough from the terminals so that any radiological impacts from airborne

releases (e.g., fires) would be expected to be negligible. In port, any potential negative impacts of low-probability, severe accidents on wildlife populations would be limited to the immediate area around the terminals.

Climatic Conditions

The Port of Hampton Roads, VA is located at the confluence of the James River and the Chesapeake Bay, approximately 29 km (18 mi) west of the Atlantic Ocean. The average elevation of this region is approximately 4 m (13 ft) above sea level.

The geographic location of this region is especially favorable, tending to be located south of the predominant winter extratropical cyclone tracks which originate at higher latitudes and north of the usual tropical cyclone (e.g., tropical storms and hurricanes) paths. In general, the winters are mild with slightly warmer temperatures during the spring and fall seasons. The summer season is warm and long, but is characterized by frequent cool periods, generated by cool northeasterly winds off of the North Atlantic. Extreme cold waves are infrequent, and temperatures below 18°C (0°F) are almost nonexistent. In general, winters pass without measurable snowfall, and most snowfall melts within 24 hrs. The average first sub-freezing day in the fall is November 17 and the last occurrence in the spring is March 23. The predominant wind directions since 1984 are from the south-southwest (about 30 percent) and north-northeast (about 25 percent) and vary seasonally (NOAA, 1992a).

D.2.1.3.1 Newport News Marine Terminal

This terminal is located on the north shore of the Port of Hampton Roads on the James River. It is a combination container, roll-on/roll-off, and breakbulk terminal. The facility has two piers (B and C), a total area of 56.9 ha (141 acres), five berths [two container berths, (each 284 m (930 ft) long), three breakbulk berths (totaling 667 m or 2,190 ft)], and four container cranes [(two 40.6 metric ton (45 ton) and two 30.5 metric ton (33.6 ton)]. Pier B is 189 m (620 ft) long and 168 m (550 ft) wide with three ship berths. Pier C is 285 m (985 ft) long and 165 m (540 ft) wide with equal dimensions of both the north and south sides. There is covered storage on both piers (36,620 m^2, or 394,200 ft^2) and the container handling terminal has storage for 790 stacked containers and 1,210 containers on chassis. The Virginia Port Authority is improving this terminal with a new 9,300 m^2 (100,000 ft^2) warehouse (Jane's, 1992; AAPA, 1993; FHI, 1994b).

Newport News Marine Terminal has immediate access to Interstate-664 outside the terminal. I-664 connects with I-64 Northbound, bypassing the Hampton Roads Tunnel, en route to the Richmond bypass, I-295 South, which connects with I-95 and I-85 Southbound. The terminal is served shipside via CSX Railroad, with direct rail service (Jane's, 1992, FHI, 1994b; AAPA, 1993).

The 1990 population within 16 km (10 mi) of the port terminals was 430,757. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 181,000; Oak Ridge Reservation, 209,000; Idaho National Engineering Laboratory, 628,000; Hanford Site, 677,000; and Nevada Test Site, 691,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 840 km (519 mi); Oak Ridge Reservation, 890 km (553 mi); Idaho National Engineering Laboratory, 4,010 km (2,492 mi); Hanford Site, 4,680 km (2,908 mi); and Nevada Test Site, 4,172 km (2,595 mi). Distances along rail routes are slightly longer.

D.2.1.3.2 Norfolk International Terminals

This is the Virginia Port Authority's largest container handling facility, located on the south side of the Port in Norfolk, adjacent to the Navy Base on the Elizabeth River Channel. The Norfolk International Terminals have a wharf area of 328 ha (811 acres), 4 container berths, 7 container cranes, room for stacking 23,930 20-ft equivalent units four high, chassis stackers for 702 chassis, a roll-on/roll-off berth and covered pier storage of 83,640 m^2 (900,000 ft^2) plus associated container terminal handling equipment. Sewell's Point Terminal, located at the north end (seaward) of the Norfolk International Terminals' container berths consists of 12.14 ha (30 acres) of land area, two piers, and covered storage for breakbulk cargoes. The Norfolk International Terminals have four container berths, 1,289 m (4,230 ft) in length, six 40.6 metric ton (45 ton) container cranes, and one 30.5 metric ton (33.6 ton) container crane.

The terminal is located approximately 2.9 km (1.8 mi) from Interstate 64 via International Terminal Boulevard (a multi-lane industrial roadway bordering the Norfolk Naval Base). It is assumed that travel on I-64 would be southbound only because of the Hampton Roads Tunnel on I-64 North. Southbound routing requires crossing several bridges over the Eastern and Southern Branches of the Elizabeth River and dealing with frequent traffic congestion on the heavily traveled Interstate. The Norfolk International Terminals is served directly (shipside) by the Norfolk Southern Railroad, and indirectly via the Norfolk and Portsmouth Belt Line Railroad, with CSX and Eastern Shore Railroads (AAPA, 1993; FHI, 1994b).

The 1990 population within 16 km (10 mi) of the port terminals was 681,864. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 131,000; Oak Ridge Reservation, 174,000; Idaho National Engineering Laboratory, 631,000; Hanford Site, 694,000; and Nevada Test Site, 694,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 800 km (498 mi); Oak Ridge Reservation, 880 km (550 mi); Idaho National Engineering Laboratory, 4,070 km (2,530 mi); Hanford Site, 4,740 km (2,949 mi); and Nevada Test Site, 4,240 km (2,633 mi). Distances along rail routes are slightly longer.

D.2.1.3.3 Portsmouth Marine Terminals

This is the Virginia Port Authority's second largest marine container handling facility, located further upstream at the confluence of the Elizabeth River and its Western Branch in the City of Portsmouth. The terminal has 3 berths that handle container, breakbulk and roll-on/roll-off cargoes, and a total land area of 88.7 ha (219 acres). It has four marginal berths with a total length of 1,080 m (3,540 ft), with 759 m (2,490 ft) of container berths. The terminal has a storage capacity of 1,770 stacked containers and 2,000 containers on chassis. The terminal also has 14,900 m^2 (160,400 ft^2) of warehouse space. The terminal has three 30.5 metric ton (33.6 ton) container cranes, one 40.6 metric ton (45 ton) container crane, and one 48.8 metric ton (54 ton) container crane.

The Portsmouth Marine Terminals are located approximately 4 km (2.5 mi) from the entrance ramp to I-264, a beltway that links up with U.S. Route 58 westbound in the rural Bowers Hill area of Chesapeake en route to I-95 or I-85 south. The assumed route from the Portsmouth Marine Terminals to the Beltway would be via Harbor Drive and Turnpike Road (State Highway 337), which runs through an area of mixed, small businesses and low-density housing for about 1.6 km (1.0 mi). The Portsmouth Marine Terminals are served directly (shipside) by the CSX Railroad with connections to the other rail lines via the Norfolk and Portsmouth Belt Line Railroad (AAPA, 1993; FHI, 1994b).

The 1990 population within 16 km (10 mi) of the port terminals was 665,700. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 135,000; Oak Ridge Reservation, 257,000; Idaho National Engineering Laboratory, 670,000; Hanford Site, 718,000; and Nevada Test Site, 732,000. Populations along rail routes to these sites are about the same for Eastern sites and slightly larger for Western sites. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 810 km (501 mi); Oak Ridge Reservation, 780 km (487 mi); Idaho National Engineering Laboratory, 4,040 km (2,514 mi); Hanford Site, 4,710 km (2,930 mi); and Nevada Test Site, 4,210 km (2,617 mi). Distances along rail routes are slightly longer.

D.2.1.4 Jacksonville, FL

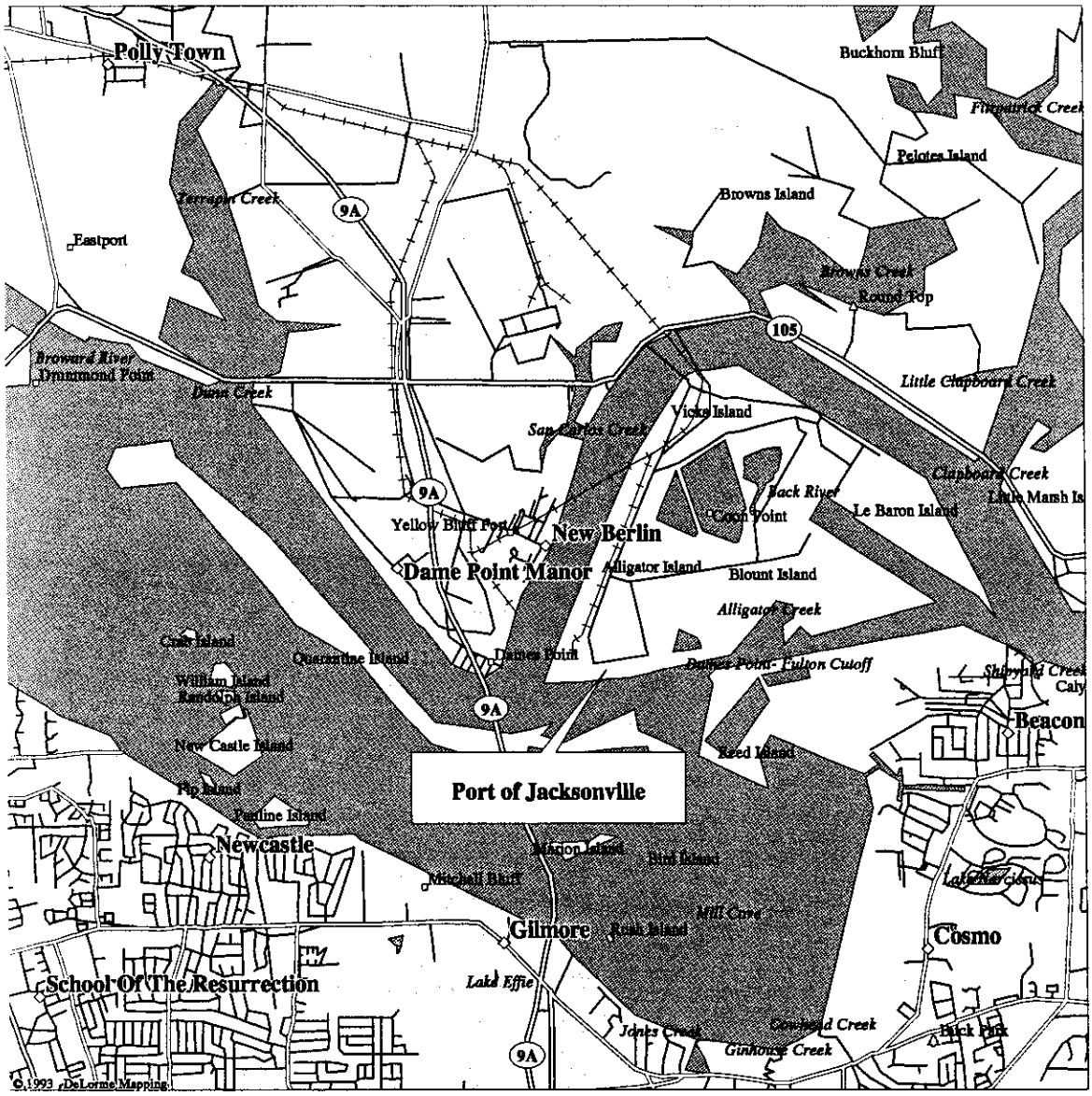
The Port of Jacksonville is located on the Atlantic Coast of Northern Florida, along the St. Johns River. It is a geographically large city (1,967 km^2 or 760 mi^2), ranging from the town of Orange on the east side of the river to Julington Creek on the west side. Most of the marine terminals are on the west side of the river, about 34 km (21 mi) from the ocean entrance. However, the Blount Island container terminal is well-separated from the city, and is only about 11 km (7 mi) from the harbor entrance. A Federal Project maintains a channel depth of 12.2 m (40 ft) to 12.8 m (42 ft) at the entrance to the river. The depth gradually decreases to about 9.1 m (30 ft) at the railroad bridge in Jacksonville. The Blount Island Terminal is located downstream from the railroad bridge in a deeper part of the channel (DOC, 1993d; Jane's, 1992; AAPA, 1993; Southern Shipper, 1993). A map of the port is shown in Figure D-25.

The St. Johns River has a deep, steep-sided channel cut through the rock in some areas. This channel configuration, combined with the increased size and draft of vessels entering the port makes the river difficult to navigate. Tidal currents in the river are strong as far as Jacksonville, approaching 1.5 meters-per-sec (3 knots) in several places (DOC, 1993d).

The Jacksonville Port Authority (Jaxport) operates two deep water container/general cargo terminals: Blount Island, located approximately 11 km (7 mi) from the harbor entrance and Talleyrand Docks and Terminals, located about 34 km (21 mi) from the entrance. Both terminals are equipped with modern entrance cranes, handle breakbulk and other types of cargo, and have transit sheds, warehouses, and open storage areas. Of the two, Blount Island is preferred because of its separation from the high density downtown area and closer proximity to the sea. A new terminal is under consideration adjacent Blount Island at Dames Point (Southern Shipper, 1993).

Both terminals serve a number of major general cargo and container ship lines from around the world including Sea-Land, NYK, Hyundai, and Mitsui OSK, that offer worldwide cargo services, and Columbus and Blue Star Line (Australia service). These lines provide service to many regions of the world, including Europe, the Mideast, South America, and Australia (Southern Shipper, 1993; Jane's, 1992).

Blount Island Terminal: Blount Island is a 356 ha (880 acre) facility with 1,920 m (6,299 ft) of berthing space, of which Berth 12 is the longest [351 m (1,150 ft)]. Blount Island Berths 7-13 have 11.6 m (38 ft) of water alongside at mean low water, and five 40.6 metric ton (45 ton) container cranes. It has 34,000 m^2 (360,000 ft^2) of transit sheds/warehousing and 149 ha (367 acres) of open storage. This terminal is connected to the mainland via a fixed highway bridge that joins State Highway 105 (Necksher Drive) and connects with I-95 and Route 17 about 8 km (5 mi) north of the City of Jacksonville. A new eight lane truck security plaza was dedicated in 1992. Blount Island has pierside service by the CSX Railroad that connects with the Norfolk Southern Railroad (Southern Shipper, 1993).



LEGEND

- Population Center
- State Route
- Geo Feature
- Town, Small City
- Street, Road
- Major Street/Road
- State Route
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:06:37 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-25 Map of the Port of Jacksonville, FL

Talleyrand Terminal: Talleyrand Docks is a 70 ha (172 acre) facility with 1,250 m (4,100 ft) of marginal wharf on deep water [11.6 m (38 ft)] at mean low water. It has two 40.6 metric ton (45 ton) container cranes, and two large gantry whirley cranes [50.8 metric ton (56 ton) and 102 metric ton (112 ton)], and 14,900 m^2 (160,000 ft^2) of transit sheds/warehousing with 49 ha (120 acres) of paved open storage (fenced and lighted). Talleyrand Terminal is located in downtown Jacksonville's shopping and commercial zone, about 2.9 km (1.8 mi) downstream of the John R. Matthews Bridge (alternate U.S. Route 90), and less than 1 km (0.6 mi) via city streets to the Expressway (alternate U.S. Route 1) (Southern Shipper, 1993).

Other Pertinent Information: The Port Authority is not aware of any local regulatory restrictions on receipt and handling of spent nuclear fuel (Castiel, 1993). The terminals have no prior experience handling spent nuclear fuel (SNL, 1994; NRC, 1993) or hazardous wastes, but do handle hazardous cargoes such as poisons, corrosives, and Class B explosives. Jaxport is a member of Jacksonville Spillage Control and the City of Jacksonville's Hazardous Materials Team. Terminal operating personnel and longshoremen receive basic instruction in the handling of hazardous cargoes. Around the clock security is provided to both terminals by the Jacksonville Port Authority, with secure, short-term storage available if needed.

There are several tanker terminals and petroleum storage depots downstream and immediately adjacent Talleyrand Docks and Terminals. Blount Island Terminal appears to have no petroleum terminals or other conflicting cargo activities (Castiel, 1993).

While the entire State is environmentally aware, there are no known sensitive wildlife sanctuaries in the immediate area of Jaxport. Blount Island is surrounded by extensive marsh and wetlands. The port is subject to severe hurricanes and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Jacksonville, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 334,212. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 46,900; Oak Ridge Reservation, 175,000; Idaho National Engineering Laboratory, 576,000; Hanford Site, 643,000; and Nevada Test Site, 639,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 607 km (377 mi); Oak Ridge Reservation, 912 km (567 mi); Idaho National Engineering Laboratory, 4,030 km (2,504 mi); Hanford Site, 4,700 km (2,924 mi); and Nevada Test Site, 4,190 km (2,607 mi). Distances along rail routes are about the same.

Environmental Conditions

The area between the mouth of the St. Johns River and Blount Island is characteristic of typical coastal lowlands found along the southeastern United States. Numerous creeks meander through large expanses of marshes and swamps. With the exception of the U.S. Naval Station Mayport and the village of Mayport, which occupy the first several kilometers along the southern bank of the river, the land bordering the lower portion of the river is largely undeveloped, with the exception of riverfront residences, mainly along the northern bank. In fact, most of the land to the north of the river between Blount Island and the coast is part of the Nassau River - St. Johns River Marshes Aquatic Preserve. The Fort Caroline National Memorial is located southeast of Blount Island on the southern bank of the river. The Little Talbot Island State Park is located approximately 1.6 km (1 mi) north of the channel entrance.

The lower 24.2 km (15 mi) of the St. Johns River has been designated as critical habitat for the manatee, a listed endangered species. The river is also used as a migratory area for the shortnose sturgeon, a listed endangered species (FWS, 1980e). According to the Florida Natural Areas Inventory, the following rare species have been reported within 3.2 km (2 mi) of the Blount Island Terminal: West Indian Manatee (State and Federal Listed Endangered Species), shortnose sturgeon (State and Federal Listed Endangered Species), Atlantic sturgeon (State Listed Species of Special Concern and Federal Listed Threatened Species), sea lamprey, and the opossum pipefish (Murray, 1994). In addition, the U.S. Fish and Wildlife Service reports that the following protected marine species may occur in Duval County: west indian manatee (endangered), shortnose sturgeon (endangered), Kemp's ridley sea turtle (endangered), leatherback sea turtle (endangered), loggerhead sea turtle (threatened), hawksbill sea turtle (endangered), and the green sea turtle (threatened). Protected bird species include the bald eagle (endangered), wood stork (endangered), piping plover (threatened), and red-cockaded woodpecker (endangered) (Bentzien, 1994).

A variety of wading birds are also found in the vicinity of the Fort Caroline National Memorial. Several species of birds, including shorebirds, waterfowl, and gannets frequent the area around the jetties at the channel entrance. In particular, the brown pelican (a State Species of Special Concern) is found in this area. A variety of birds inhabit the Little Talbot Island State Park, including the American oystercatcher (a State Species of Special Concern). Loggerhead sea turtles (a listed endangered species) use the beaches along this portion of Florida as a nesting area (FWS, 1980e).

Climatic Conditions

The Port of Jacksonville, FL, is located along the lower 39.4 km (24.5 mi) of the St. Johns River. The terrain in this area is relatively level, providing very little change in relief proceeding inland from the coastal region.

As with the other more northern ports, the climate of this area is also modified by the influence of the Atlantic Ocean. Easterly winds occur roughly 40 percent of the time, producing a true maritime climate for the Jacksonville area. The greatest rainfall occurs during summer, usually associated with afternoon and evening thunderstorms. During summer, measurable precipitation can be recorded nearly every two days. The prevailing winds are northeasterly in the fall and winter months, becoming more southwesterly during spring and summer. Although Jacksonville is along the eastern United States coast, it has been very fortunate in escaping hurricane-force winds. The majority of systems in recent years that have reached this latitude have moved parallel to the coastline, keeping well offshore. Others have weakened significantly moving over land prior to reaching the Jacksonville area. The combination of these two factors has spared the area from any major devastation due to tropical systems in recent years (NOAA, 1992e).

D.2.1.5 Military Ocean Terminal, Sunny Point, NC

The Military Ocean Terminal at Sunny Point (MOTSU) is a defense transportation facility used to move military cargo (principally munitions) into and out of the United States. The terminal is located approximately 16 km (10 mi) upstream from the mouth of the Cape Fear River on the Atlantic Coast near Southport, NC. A map of the port is shown in Figure D-26. The port is easily accessed from the ocean, and all commercial vessels bound for Wilmington, NC must pass by MOTSU. It is served by a 12.1 m (40 ft) deep by 152 m (500 ft) wide channel from the ocean (DOE, 1994d).

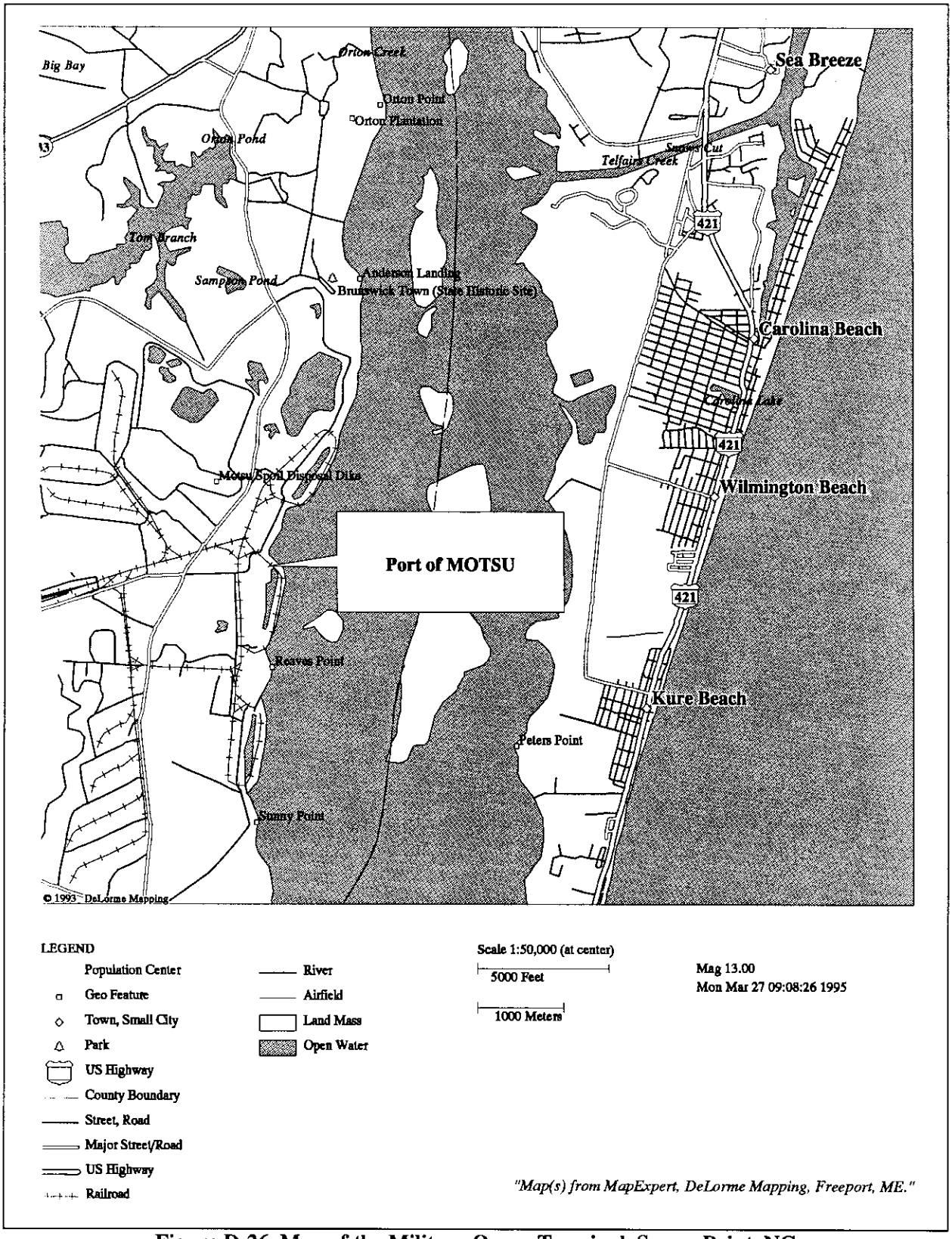


Figure D-26 Map of the Military Ocean Terminal, Sunny Point, NC

Since the majority of cargoes handled at MOTSU are explosive, the terminal is laid out such that an explosion at one wharf will not seriously impact activities at an adjacent wharf. This would permit containerized spent nuclear fuel carried in a commercial vessel (without explosive or hazardous cargoes on-board) to be safely received and transported from the terminal, even though there are conflicting activities within the terminal. Further, after many years of service, MOTSU has never had an explosion accident, so the risks are believed to be small. However, unloading of spent nuclear fuel would be scheduled during periods when explosives were not being unloaded. On average, MOTSU receives about 70 vessels per year, and moves approximately 433,000 metric tons (476,000 tons) of cargo through the port.

While regularly scheduled commercial container or breakbulk vessels do not call at MOTSU, commercial container vessels chartered by defense agencies routinely call at the port. The water depth (channel and alongside the wharves) of 10.3 m (34 ft) mean low water is adequate for most commercial breakbulk, roll-on/roll-off, and container ships. The terminal has three 606 m (2,000 ft) wharves, each with three berths. All wharves have three parallel sets of rail tracks. Berth 1, on the south wharf, has two 45.3 metric ton (50 ton) container cranes capable of off-loading container or container/breakbulk vessels. Berth 3 has been modified with a 30 m (100 ft) wide, reinforced concrete apron that permits breakbulk and roll-on/roll-off operations in addition to containerized cargoes (DOE, 1994d).

MOTSU is serviced by well-maintained roads which are primarily two-lane roads providing connections to multi-lane controlled access highways. In the event that MOTSU was utilized for receipt of foreign research reactor spent nuclear fuel, all transport of spent nuclear fuel over these roadways would be in conformance with State regulations for normal truck traffic between MOTSU and other locations to avoid overloading roadways and bridges. Truck access is provided by State Route 87 from the northwest and State Route 133 from the north. Route 87 provides access to U.S. 17, which runs southwest or northeast. The distance from the terminal gate to Route 133 is about 3.7 km (2.2 mi). Route 133 runs directly to U.S. 17 just outside Wilmington, NC. From Wilmington, U.S. 74 runs west 120 km (75 mi) to Interstate 95, the nearest major north-south highway (DOE, 1994d). A dedicated 157 km (97.4 mi) U.S. Army rail line connects the CSX network directly to the terminal.

Other Pertinent Information: At the present time, there are no regulatory restrictions on receipt, handling, and transhipment of foreign research reactor spent nuclear fuel at MOTSU. MOTSU is the only port in the contiguous United States which has current experience with foreign research reactor spent nuclear fuel receipt and handling, with two shipments received in October 1994 under the Urgent Relief Environmental Assessment.

Cargo handling at the terminal, including explosives, is performed by members of the International Longshoremen Association. Port security is maintained on land by security guards, and on water by dedicated patrol boats.

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For MOTSU, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 7,995. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 34,200; Oak Ridge Reservation, 128,000; Idaho National Engineering Laboratory, 463,000; Hanford Site, 548,000; and Nevada Test Site, 619,000. Populations along rail routes to these sites are

slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 402 km (250 mi); Oak Ridge Reservation, 798 km (496 mi); Idaho National Engineering Laboratory, 3,873 km (2,407 mi); Hanford Site, 4,615 km (2,868 mi); and Nevada Test Site, 3,953 km (2,457 mi). Distances along rail routes are slightly longer.

Climatic and environmental information for MOTSU is similar to that for the Port of Wilmington, NC, as listed in Section D.2.1.10.

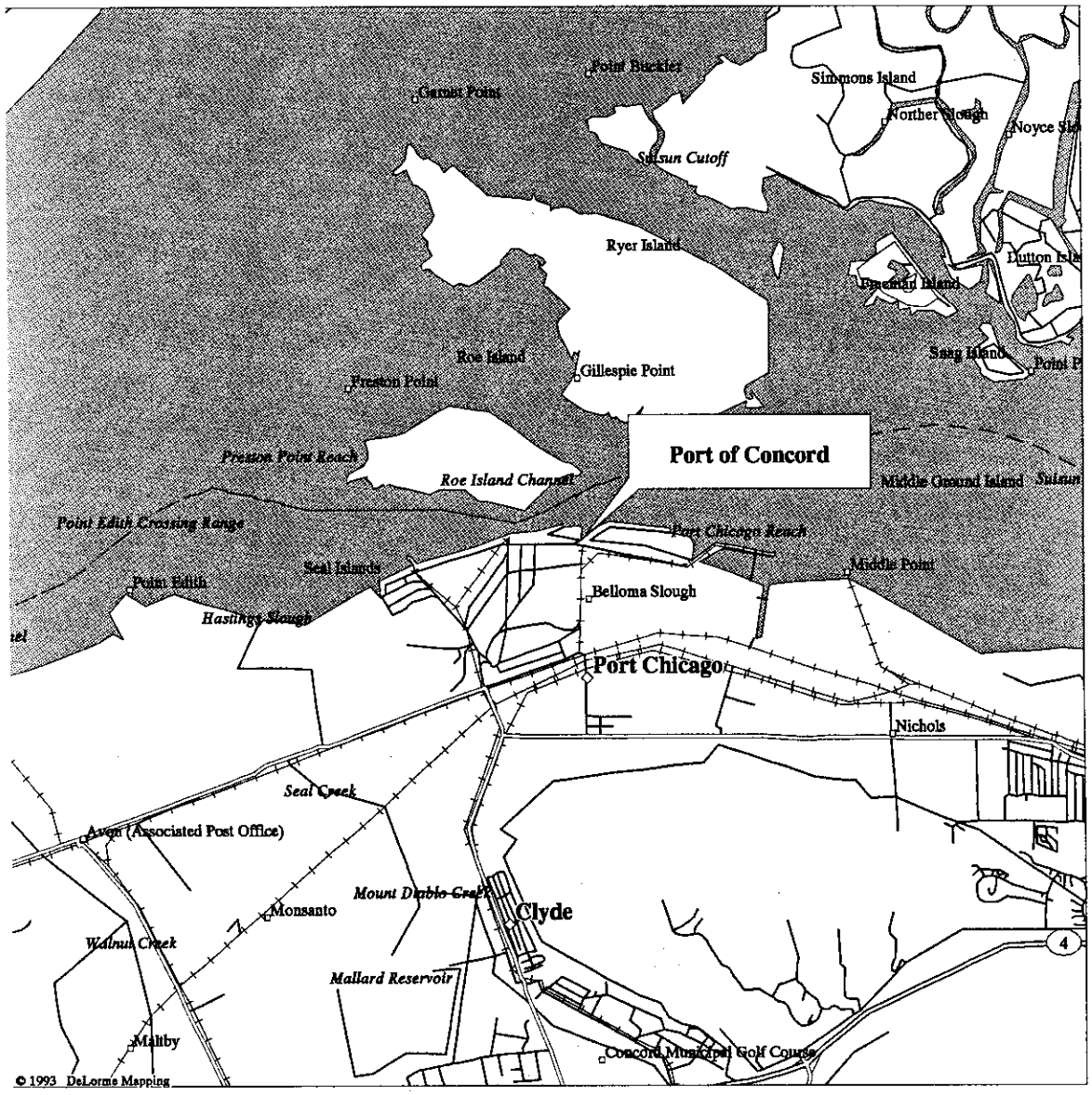
D.2.1.6 Naval Weapons Station (NWS) Concord, Concord, CA

Concord NWS is located on the western edge of Suison Bay, an esturine area carrying the flows of the Sacramento and San Joaquin Rivers to San Pablo Bay through the Straits of Carquinez. By sea, the transit is approximately 55 km (35 mi) northeast of the Golden Gate Bridge. Concord NWS is about 5 km (3 mi) north of the city of Concord, CA. The wharf at Concord NWS is about 8 km (5 mi) north of the city. The site tidal areas surrounding the pier areas (including small off shore islands) comprise about 3,1000 ha (7,648 acres) to provide a large separation from nearby communities, and security for the site. A map of the area is shown in Figure D-27. Concord NWS is aligned under the Pacific Division Division of the Naval Ordnance Center. Concord NWS is aligned under the Pacific Division of the Naval Ordnance Center. The Pacific Division is located at Seal Beach, CA and the Naval Ordnance Center is located at Indian Head, MD (Yocum, 1994b).

The Station currently is a breakbulk facility (primarily munitions, naval ordnance, and other high explosives), with limited container handling capabilities. Most vessels servicing the facility are self-supporting, with on-board cranes for handling cargo. Concord NWS currently has a 100-metric ton (112-ton) floating crane and a truck mounted mobile 82 metric ton (90 ton) crane used to service the Station as needed (several cranes are mounted on a crane ship that ties up alongside cargo vessels and loads or offloads cargoes). The facility also has a roll-on/roll-off berth for stern ramps, and a substantial barging pier. A $57 million modernization program has been approved for completion by 1999, which will add to Pier 3 two new 36 metric ton (40 ton) container cranes and gantry crane rails outside of the existing pier structures. The improvements will permit more efficient handling of containerized cargo. The Station is very similar to the MOTSU facility, with three well-separated wharves with two berths on each [about 360 m to 370 m long (1,180 ft to 1,220 ft) at Pier 3]. Separation is designed to protect adjacent vessels from severe damage (or additional explosions) in the unlikely event of an explosion on one ship. Like MOTSU, each pier has three parallel rail lines. Depth alongside the breakbulk/container wharves is 10.6 m (35 ft) at mean low water, which is adequate for most breakbulk/container vessels. Vessel size is limited by the height limit of 33 m (135 ft) under the bridges over the Strait, and the width of the channel [about 90 m (300 ft) minimum] (Yocum, 1994a and 1994b).

Truck access to Interstate 680 is about 10 km (6 mi) via State Route 4. The site has about 127 km (79 mi) of paved roads and 165 km (103 mi) of rail tracks. Concord NWS is served by the Union Pacific, Southern Pacific, and Santa Fe Railroads (all of which have mainline tracks through the tidal area) and has the equivalent of a small intermodal railyard in the immediate vicinity of the pier where railcars can be brought after loading on the piers. Rail routes include direct movement to the Hawthorne Army Ammunition Depot in Hawthorne, NV, where spent nuclear fuel could be off-loaded to trucks for direct shipment to the Nevada Test Site (Yocum, 1994b).

Other Pertinent Information: Since it is a military facility and an explosives operating area, the entire pier operations waterfront is surrounded by barbed wire fencing with access through military posted gates. The facility has areas for staging (and short-term secure storage) shipments by truck or rail near the pier areas.



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- County Boundary
- Street, Road
- Major Street/Road
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Sun Mar 26 13:10:59 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-27 Map of the Concord Naval Weapons Station, Concord, CA

When explosives are being handled, the explosive safety arc is approximately 3,400 m (11,200 ft) around the pier area. The existing State highway through the site is closed off about 3 km (2 mi) from the piers at the small town of Clyde (population about 485) adjacent to the Station's Administrative areas (Yocum, 1994b). Concord NWS area has its own full-time security force, with a U.S. Coast Guard facility onsite to provide some explosive oversight services during loading and unloading activity. There is a fire station in the immediate vicinity of the pier areas, with an estimated 3-minute response time for first responders.

The primary mission of the port is to support all branches of the military in shipping munitions. No concurrent non-explosives cargo handling, such as foreign research reactor spent nuclear fuel, would be allowed when explosives are being handled. Scheduling of foreign research reactor spent nuclear fuel shipments would have to be done for times when no explosives handling is anticipated. Unscheduled activities or activities with little advance notice involving the military mission would require re-scheduling or re-routing of the foreign research reactor spent nuclear fuel. The foreign research reactor spent nuclear fuel handling would not be the first priority of the port.

Conflicting activities are expected to be avoided by proper scheduling (normally only one ship at a time is in port).

Parts of the tidal area are leased to local cattle growers to keep the grass down for fire protection purposes. The station is a wildlife sanctuary for migratory birds (about 1,200 ha or 3,000 acres) of the tidal area) and hosts native Tule elk, which were formerly on the endangered species list (Yocum, 1994b).

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Concord NWS, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located on the edge of a very high seismic zone with an acceleration of 0.45 g.

The 1990 population within 16 km (10 mi) of the port terminals was 381,070. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,040,000; Oak Ridge Reservation, 742,000; Idaho National Engineering Laboratory, 271,000; Hanford Site, 263,000; and Nevada Test Site, 437,000. Populations along rail routes to these sites are slightly smaller for Oak Ridge Reservation, Idaho National Engineering Laboratory and Nevada Test Site, and slightly larger for Savannah River Site, and Hanford Site. These populations are shown in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,476 km (2,784 mi); Oak Ridge Reservation, 4,111 km (2,557 mi); Idaho National Engineering Laboratory, 1,516 km (943 mi); Hanford Site, 1,376 km (856 mi); and Nevada Test Site, 1,145 km (712 mi). Distances along rail routes are about the same for Idaho National Engineering Laboratory, and slightly longer for Savannah River Site, Oak Ridge Reservation, Hanford Site, and Nevada Test Site.

Environmental Conditions

Concord NWS occupies 5,233 ha (12,931 acres) of land adjoining south Suison Bay. Of this total acreage, 2,135 ha (5,276 acres) are inland, while 3,097 ha (7,653 acres) are more tidal in nature. Wetlands comprise approximately 1,215 ha (3,002 acres) of the tidal area (Yocum, 1994b). Wetlands occupy large areas of land bordering all sides of Suison Bay and Grizzly Bay, which is located directly north of Suison Bay. The waters of Suison Bay are characterized as a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). Chinook salmon (endangered), steelhead trout, striped bass, sturgeon, and American shad are typically found in this area (FWS, 1981e; FWS, 1981f).

Portions of the inland area at Concord NWS serve as a sanctuary for Tule elk, a formerly endangered species (Yocum, 1994b). Other terrestrial species found in the area include the river otter, the salt-marsh harvest mouse (a Federally protected species), and the white-tailed kite (FWS, 1981e; FWS, 1981f). Adult concentrations and nesting areas of the California clapper rail (a Federally protected bird species) and the California black rail (a State protected species) are also found in this area. The U.S. Fish and Wildlife Service reports that the following Federally-listed, protected species may occur in Contra Costa County: winter-run chinook salmon (endangered), delta smelt (threatened), bald eagle (endangered), American peregrine falcon (endangered), Aleutian Canada goose (threatened), California brown pelican (endangered), California clapper rail (endangered), California least tern (endangered), and the salt marsh harvest mouse (Medlin, 1994). The Federally and State protected figwort plant family is also found in the vicinity of Concord NWS. In general, the greater San Francisco Bay area annually supports large numbers of shorebirds, wintering waterfowl, raptors, seabirds, and passerlings. In addition, shorebirds, wading birds, waterfowl, seabirds, and songbirds migrate through this coastal area.

Climatic Conditions

Currently, there is no operational National Weather Service station located in Concord, CA. However, the National Weather Service does operate stations at the San Francisco International Airport (37° 37' N; 122° 23' W) and at Stockton, CA (37° 54' N; 121° 15' W). Because of the influence of the California Coast Ranges, which exist between San Francisco and Concord (trending northwest-southeast), the National Weather Service data at Stockton, CA, is considered a better surrogate for the climatological conditions at the Concord Naval Weapons Station.

Concord is located on the westernmost edge of the Great Valley of California, in the eastern foothills of the Coast Ranges. The region is comprised of rich agricultural land, located on the broad delta formed by the confluence of the Sacramento and San Joaquin Rivers. Well to the east of this region are the foothills of the Sierra Nevada Mountains and to the west are the California Coast Ranges. The coast ranges are important in providing an effective barrier from the maritime air masses that greatly influence the San Francisco-Oakland area. However, several gaps in the Coast Ranges do permit the inland migration of the sea breeze circulation, which tends to moderate daytime high temperatures in the summer months. In general, the area is characterized in summer by warm, dry days and relatively cool nights with clear skies and little rainfall. Winter brings relatively milder temperatures, with light precipitation and frequent heavy fog events, which often have long durations in December and January. Ninety percent of the precipitation falls between November and April, with thunderstorms extremely infrequent (4 days per year) and snowfall almost nonexistent (NOAA, 1992j).

D.2.1.7 Portland, OR

The Port of Portland is located about 160 km (97 mi) above the mouth of the Columbia River on the Willamette River tributary. Portland is the principal city of the Columbia River system and one of the major ports on the Pacific Coast. The preferred container terminal (T6) is located approximately 170 km (90 mi) from the entrance of the Columbia River. Federal project depths in the Columbia River are 14.6 m (48 ft) over the bar, and 12 m (40 ft) to Portland (DOC, 1992b). However, a port official indicated the actual channel depth is 13.11 m (43 ft), and the channel width is 183 m (600 ft) from the coast to the port (Magness, 1993).

There are a number of cautions concerning entering and navigating the Columbia and Willamette Rivers. The U.S. Coast Pilot warns that entry into the Columbia River can be dangerous because of sudden and unpredictable changes in the currents often accompanied by breakers. It is reported that "ebb [tide] currents on the [North] side of the bar attain velocities of [3.1 to 4.2 meters-per-sec] 6 to 8 knots . . . In the

entrance the currents are variable, and at times reach a velocity of [2.6 meters-per-sec] 5 knots on the ebb; on the flood [tide] they seldom exceed [2.1 meters-per-sec] 4 knots. Since logging is one of the main industries of the region, free floating logs and submerged deadheads or sinkers are also a source of danger. The danger is increased during spring freshets" (DOC, 1992b).

U.S. Coast Guard statistics for 1990 through 1993 indicate that the transit from the Pacific Ocean to the Port of Portland is hazardous, with a reported total of 112 ship collisions and 145 (hard) groundings (USCG, 1994b). It is noted that a large number of oceangoing vessels make the transit on a routine basis without incident. Since some of these accidents were most likely associated with barges, it is believed that the actual rate for oceangoing vessels is probably lower.

The Port of Portland owns and operates Terminal T6, a deep-water dedicated container facility located on Pearcy Island, at the confluence of the Columbia and Willamette Rivers, about 140 km (90 mi) from the ocean entrance to the Columbia River. The port also owns other terminals (including T2, a container/breakbulk facility), all of which lie further upstream of Terminal T6. Terminals are situated in an industrial port district northwest and seaward of downtown Portland (POP, 1994). A map of the port is shown in Figure D-28.

The port is served by several container lines including Australia New Zealand Direct Line, Evergreen Line, Hanjin Shipping Co., LTD., Hawaiian Marine Lines, Hyundai, International Marine Transport Lines, Italian Line, d'Amico Line, Jebson's International, "K" Line, Mitsui OSK, Neptune Orient, NYK Line, Pacific Commerce Line, Safbank Line, and United Yugoslav Line (Jane's, 1992; AAPA, 1993; POP, 1994).

Terminal T6: This terminal has three berths, five container cranes [two 36.3 metric ton (40 ton) and one 50 metric ton (55 ton)], a container freight station, distribution warehouse, and rail/barge service. It has about 869 m (2,851 ft) of marginal wharf, with 12.2 m (40 ft) of water alongside at mean low water. Truck access to Interstate 5 is via North Marine Drive and N. Lombard Street, both of which connect with I-5 about 5.5 km (3.4 mi) from the terminal entrance. North Marine Drive is an industrial use roadway that connects with I-84, the assumed route to Idaho National Engineering Laboratory, north and across the Willamette River from downtown Portland. T6 is served by the Burlington Northern and Union Pacific Railroads, whose tracks reach to within about 0.5 km (1,500 ft) of the container berths; an intermodal, container-on-flat-car rail yard is an integral part of the terminal. T6 has reciprocal switching arrangements with the Southern Pacific Railroad (AAPA, 1993; POP, 1994).

T6 is located north (downstream) of the port's other marine terminals and has no apparent conflict with other hazardous cargoes. It is currently operated by the port, but the port is considering an operations contractor for the future (Hachey et al., 1994).

Terminal T2: This terminal has about 833 m (2,730 ft) of marginal wharf, with 12.2 m (40 ft) of water alongside at mean low water, and four container cranes with capacities ranging from 33 metric tons (36 tons) to over 77 metric tons (85 tons). Truck access to Interstate 84 is via Interstate 5 South to U.S. Route 30 West, connecting with I-84 at Maywood Park, a total distance of about 19 km (12 mi). The terminal is served by the Portland Terminal Railroad and the Burlington Northern, and has direct ship-to-rail transfer capability. T2 also has reciprocal switching arrangements with the Southern Pacific (AAPA, 1993; POP, 1994). T2 is located near several large bulk petroleum terminals that are undoubtedly supplied by tankers. Such traffic was not considered to be a major risk factor for the transportation of spent nuclear fuel to Portland. However, because of the potential conflicting uses, Terminal T6 is the preferred facility.

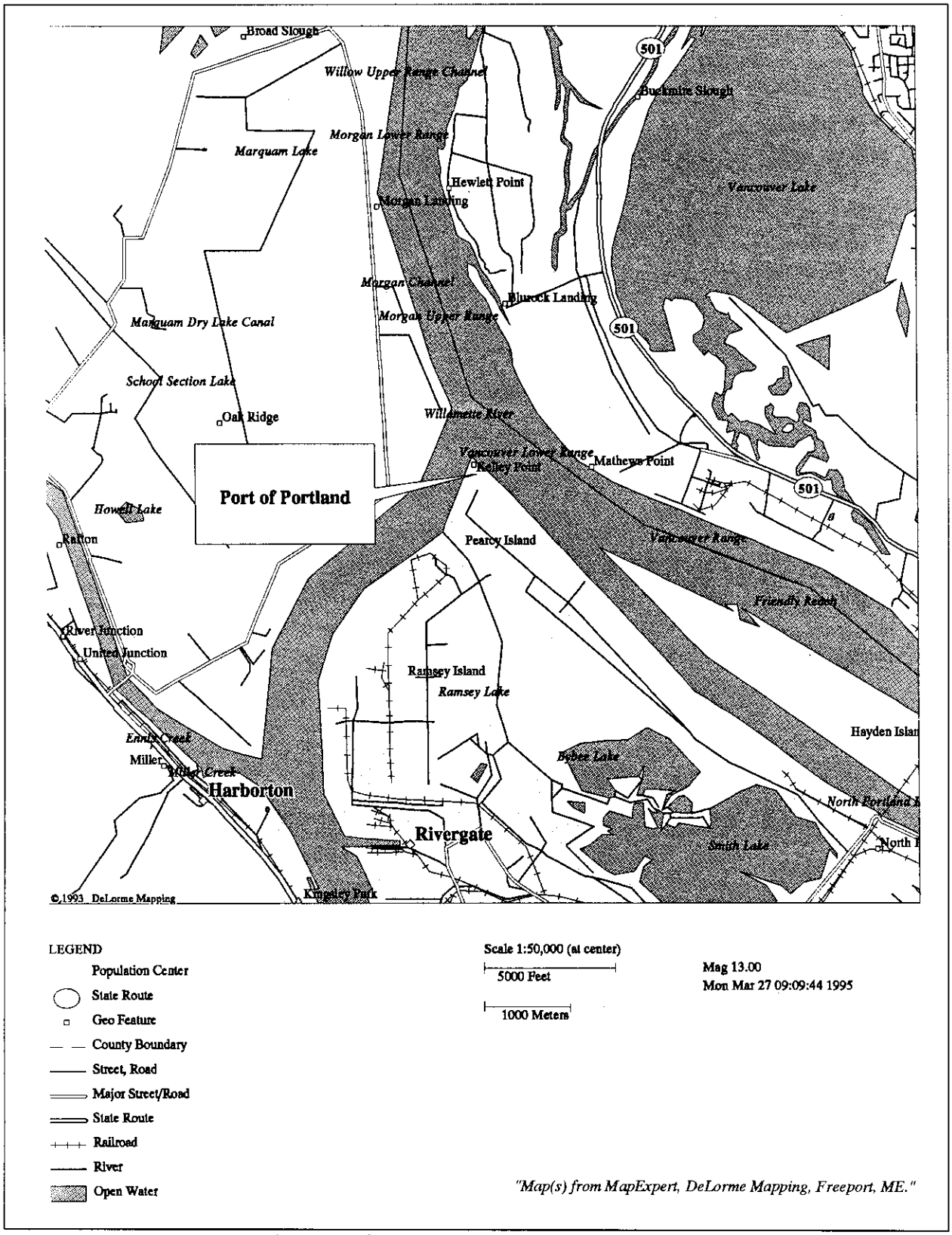


Figure D-28 Map of the Port of Portland, OR

Other Pertinent Information: Security is provided by perimeter fencing and the port's police force, which maintains a 24-hour patrol and surveillance function at both terminals.

There are no restrictive regulations currently affecting the potential receipt and transport of foreign research reactor spent nuclear fuel through the port. The Manager of Marine Market Development indicated that the port has not handled spent nuclear fuel since 1985, and there is opposition to handling nuclear materials by the port's labor unions (Magness, 1993). It is noted that while most of the spent nuclear fuel shipped through Portland had been shipped by the end of 1985, other data sources indicate the port also handled additional spent nuclear fuel in 1989 (NRC, 1993; SNL, 1994). There are no restrictions on Class A or B explosives, and the Coast Guard does not make radiation surveys of radioactive cargoes. Recently, the port could not get shippers to handle naturally radioactive columbium concentrate from British Columbia even if it is not unloaded (Hachey et al., 1994). While this does not preclude foreign research reactor spent nuclear fuel shipments, this indicates there is the potential for delays which could result in failing to "expeditiously transfer" foreign research reactor spent nuclear fuel from the port to a selected storage site.

Portland has a Port Evacuation Plan and a hazardous materials advisory staff (Hachey et al., 1994). The State Health Division with whom the port confers, has a resident nuclear physicist for technical assistance. The port is also a member of the Maritime Fire and Safety Association (an industrial association representing 27 terminal operators) and nine fire departments on the Columbia and Willamette Rivers. The nearest fire station can respond within about six minutes (Hachey et al., 1994). The Association has developed emergency response plans and is implementing a radio communications system covering the entire river system from Astoria to Portland. The City fire department and Coast Guard respond to accidents involving hazardous materials cargoes. Port operating personnel and longshoremen receive general instruction concerning handling of hazardous materials cargoes (Magness, 1993). In addition, the port has contractors ready to respond to hazardous materials accidents when necessary (Hachey et al., 1994). There has not been a severe container accident in at least 10 years, so no port accident statistics were available (Hachey et al., 1994). The port is located several miles downstream from Portland's business and residential districts in an area that appears dedicated to port industrial usage, but as already noted, has excellent connections with highways and rail service.

There are no known areas of special environmental concern in the immediate vicinity of the port, although concern for the environment runs high throughout the Pacific Northwest. A "critical habitat" adjacent to Terminal 6 will have to be mitigated with the planned expansion at T6, but there are no plans to fill wetlands between T6 and populated areas about 1 or 2 km (0.6 or 1.2 mi) away (Hachey et al., 1994).

The port is subject to earthquakes and volcanism. The likelihood of severe natural phenomena, such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Portland, the Uniform Building Code requires buildings to withstand wind speeds up to 140 km/hr (90 mph). The port is located in a moderate seismic zone with an acceleration of 0.20 g. There have been two major earthquakes in the Puget Sound area this century; a Modified Mercalli Intensity (MMI) VIII on April 13, 1949, and an MMI VII-VIII on April 29, 1965 (Bolt, 1978). On May 18, 1980, nearby Mount St. Helens suffered a major volcanic eruption (IPA, 1993). All the mountains along the Cascade Range are volcanic in origin and prone to eruption (Foster, 1971; Hamilton, 1976; IPA, 1993).

The 1990 census population within 16 km (10 mi) of the Terminal was 356,064. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 686,000; Oak Ridge Reservation, 519,000; Idaho National Engineering Laboratory, 143,000; Hanford Site, 85,700; and Nevada Test Site, 375,000. Populations along rail routes to these sites are

slightly smaller for Nevada Test Site and Idaho National Engineering Laboratory, but slightly larger for Savannah River Site, Oak Ridge Reservation, and Hanford Site. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,630 km (2,879 mi); Oak Ridge Reservation, 4,200 km (2,609 mi); Idaho National Engineering Laboratory, 1,190 km (738 mi); Hanford Site, 407 km (253 mi); and Nevada Test Site, 2,040 km (1,270 mi). Distances along rail routes are slightly longer, with the exception of Hanford Site, which is slightly less.

Environmental Conditions

The areas surrounding the Terminal are in river-oriented industrial land use. Wildlife habitat along the Oregon Slough is limited because of the industrial development, although some waterfowl use the area. While the primary uses in the Terminal area are commercial navigation and industry, some recreational fishing and boating occurs in Oregon Slough and the Columbia River (Kurkoski,1994).

The U.S. Fish and Wildlife Service's Ecological Inventory for the Vancouver, Washington-Oregon area indicates that the Columbia River generally includes the following fish species: salmonids, chinook salmon, coho salmon, chum salmon, pink salmon, sockeye salmon, steelhead trout, Dolly Varden, smelts, river lamprey, white sturgeon, American shad, eulachon, and cutthroat trout (FWS, 1981d). South of Portland, the various islands and wetlands along the Columbia River provide habitat for a wide variety of terrestrial organisms. Areas of special interest include the Sauvie Island Game Management Area, which is located approximately 8 km (5 mi) downriver of Terminal 6, and the Ridgefield National Wildlife Refuge, which is approximately 16 km (10 mi) downriver.

The U.S. Army Corps of Engineers reports that raptors such as the red-tail hawk, bald eagle, and peregrine falcon are occasional visitors to this area and that the U.S. Fish and Wildlife Service has indicated that the endangered American peregrine falcon and threatened bald eagle may winter in this area. In addition, the National Marine Fisheries Service has listed the Snake River sockeye salmon as endangered, and two Snake River chinooks stocks as threatened (Kurkoski, 1994). The State of Oregon's Natural Heritage Program reports that there are at least two rare species that occur in the vicinity of Terminal 6 (Gaines, 1994). These species are the painted turtle (a State-Sensitive-Critical species) and the Columbia water-meal.

Climatic Conditions

The city of Portland is situated midway between the northerly oriented low coast range on the west and the higher Cascade range on the east. The Cascade range provides a steep slope for orographic uplift of moisture laden air arriving on westerly winds from over the Pacific Ocean, resulting in moderate rainfall events in the area. The prevailing winds are generally northwesterly during spring and summer, becoming more southeasterly in fall and winter. The Portland area is characterized by a winter rainfall regime, where approximately 88 percent of the annual total falls during October through May. Thus, the winter season is dominated by relatively mild temperatures, cloudy skies and rain accompanied by southeasterly surface winds. Summer produces pleasantly mild temperatures, northwesterly winds and very little precipitation. Fall and spring are traditional seasons with variable characteristics. Fog generally occurs most frequently during fall and early winter. Destructive storms are infrequent in this region of the United States, and surface winds rarely exceed gale force. Thunderstorms occur monthly through the spring and summer, with gentle rains occurring almost daily during the winter months. Based on the 1951-1980 climatology, the first frost occurs on average around November 7, with the last spring frost occurring near April 3 (NOAA, 1992f).

D.2.1.8 Savannah, GA

The Port of Savannah is located on the South Bank of the Savannah River, about 35 km (22 mi) above the entrance from the Atlantic Ocean. Savannah is the third largest city in Georgia, and is the chief port of the State of Georgia. A Federal Project maintains 12.2 m (40 ft) of water through Tybee Roads, then 11.6 m (38 ft) for about 16 mi in the main channel to the turning basin at Kings Island (DOC, 1993d). A map of the port is shown in Figure D-29.

Under normal conditions, currents at the entrance to Savannah are 1.1 to 1.5 metric-sec (2.2 to 3.1 knots) during the ebb tide, and 0.8 to 1.2 metric-sec (1.6 to 2.4 knots) during the flood tide. It has been reported that currents in the river can reach 3.6 to 4.1 metric-sec (7 to 8 knots) in the vicinity of Garden City Terminal just below the Route 17A bridge and at the Colonial Oil Berths, about 4 km (2.5 mi) above the bridge. Access to the port can be complicated due to some relatively narrow sections of the channel combined with high currents (DOC, 1993d).

The Georgia Ports Authority (GPA) operates three large cargo terminals on the South Bank of the Savannah River. Ocean Terminal, located approximately 41 km (25 mi) from the river entrance in the City of Savannah, is a combination breakbulk and container handling facility; Garden City Terminal is about 4.6 km (2.9 mi) further upstream from Ocean Terminal. Containerport, part of the Garden City terminal complex, is a dedicated container handling facility. The depth alongside both container terminals is 11.6 m (38 ft), and dredging to 12.7 m (42 ft) is in progress.

The port is served by more than 50 container and breakbulk ship lines, including several major container carriers, with itineraries to some 100 countries in the world, including many in Europe and the Far East, as well as Japan, and Australia (Jane's, 1992; AAPA, 1993; Southern Shipper, 1993)

Ocean Terminal: This facility has 10 berths, a 61 m (200 ft) apron, extensive Transit sheds and warehouse space, with 34 ha (83 acres) of open storage. It has one 41 metric ton (45 ton) single hoist container crane and four gantry cranes of greater capacity, and 1,825 m (5,990 ft) of marginal wharf; Berth 13, the longest, is 297 m (975 ft) long. The terminal has almost immediate access to U.S. Route 17 (north/south), and connects with I-16 a few city blocks from the terminal. The terminal is served by the Norfolk Southern and CSX railroads (AAPA, 1993; Southern Shipper, 1993; Jane's, 1992). Due to its close proximity to the City, it is not a preferred container terminal.

Containerport: This is the preferred container terminal, due to its better separation from the City and modern facilities. It is located about 40 km (25 mi) from the Atlantic. It has 6 container ship berths, a 61 m (200 ft) apron, and a 457 m (1,500 ft) by 488 m (1,600 ft) turning basin. The terminal has a large Container Freight Station comprising 51,280 m^2 (552,000 ft^2) for stuffing and stripping containers, with areas for segregating hazardous cargoes. The facility has 1,676 m (5,500 ft) of marginal wharf, and nine 40.8 metric ton (45 ton) single hoist container cranes. Truck access to the terminal is via limited access roads and lightly populated areas to the following expressways: about 13 km (8 mi) to I-16 (east/west); about 9.6 km (6 mi) to I-95 north and about 18 km (11 mi) to I-95 south. Containerport has excellent shipside rail service, consisting of a series of rail spurs at right angles to the container berths, providing direct ship-to-rail transfers. Onsite switching is provided by GPA's Savannah State Docks Railroad, connecting with the CSX and Norfolk Southern rail systems (AAPA, 1993; Southern Shipper, 1993; Jane's, 1992; GPA, 1994).

Other Pertinent Information: The port's Director of Public Relations was unaware of any ordinances or regulations prohibiting the receipt and handling of spent nuclear fuel, and the port does handle radioactive materials (Moore, 1993). The official did not know if the port had handled spent nuclear fuel, but data

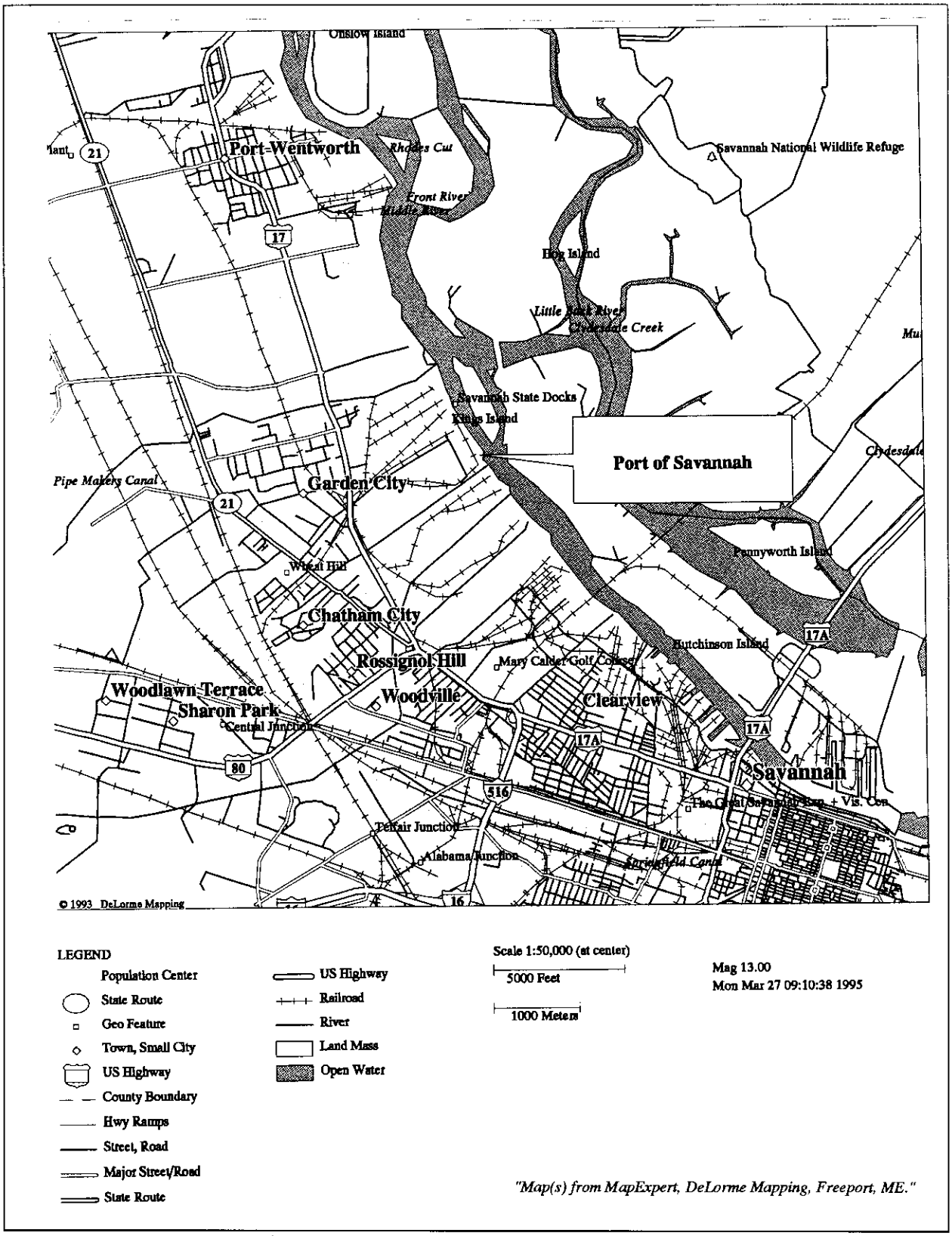


Figure D-29 Map of the Port of Savannah, GA

show it has (SNL, 1994; NRC, 1993). The port has a hazardous materials training staff but no Emergency Response Team. Reportedly, the GPA contracts with outside firms to respond to oil and other hazardous materials accidents.

There are tanker berths and petroleum storage facilities adjacent to Containerport's facilities, and there are several private bulk liquid storage facilities downstream of Containerport (towards the City), including a liquid natural gas terminal a few miles above the Pilot station. The presence of these terminals along a river channel only 152 m (245 ft) wide with swift currents, and the increasing number of container ships with lengths in excess of 250 m (820 ft) heighten the possibility of potentially serious conflicts within the port.

The port is subject to severe hurricanes and tropical storms, and given its proximity to Charleston, SC may have a slightly higher risk of earthquakes than the rest of the State of Georgia. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Savannah, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 155,166. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 37,300 ; Oak Ridge Reservation, 101,000; Idaho National Engineering Laboratory, 553,000; Hanford Site, 602,000; and Nevada Test Site, 616,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 400 km (250 mi); Oak Ridge Reservation, 720 km (449 mi); Idaho National Engineering Laboratory, 3,860 km (2,398 mi); Hanford Site, 4,530 km (2,813 mi); and Nevada Test Site, 4,020 km (2,501 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The lower Savannah River has multiple branches that meander through a variety of coastal lowlands including salt marshes, tidal creeks, freshwater marshes and freshwater impoundments. South Carolina has classified the portion of Savannah Harbor within its boundaries upstream from Fort Pulaski as Class B and the portion oceanward as Class SA. Class B waters are fresh waters suitable for secondary contact recreation, as a drinking water source following conventional treatment, fishing, industrial, and agricultural use. Class SA waters are defined as tidal saltwaters suitable for primary contact recreation, and for all the uses listed in Class B. The State of Georgia has classified the Savannah River from Mi 0 at Fort Pulaski north to Mi 5 at Field's Cut as recreation waters. North of Field's Cut, the waters are classified as Coastal Fishing (U.S. Army, 1991). According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for Savannah, the Containerport is located in a transitional estuarine habitat where the salinity ranges from low (generally 0.5 to 5 parts per thousand) to mid-salinity (generally 5 to 16.5 parts per thousand) (FWS, 1980c).

A large number of aquatic and terrestrial species are found in and around the Savannah River near Garden City. The U.S. Fish and Wildlife Service indicated that the protected species most likely to be found on or near this general area include: the bald eagle, wood stork, shortnose sturgeon, west indian manatee, and eastern indigo snake. The bald eagle and wood stork occur on the Savannah National Wildlife Refuge. West indian manatees are known to use the lower Savannah River and are seen fairly often in the river and harbor area. The shortnose sturgeon uses the Savannah River as a migratory area. The lower Savannah

River also is reported to be an important spawning area for striped bass (Laumeyer, 1994). In addition, the loggerhead turtle, bald eagle, and the American alligator are found along the lower reaches of the Savannah River (FWS, 1980c).

Both invertebrate and fish species of commercial and recreational value found in the Savannah River. Commercial fishing is primarily for American shad, sturgeon, shrimp, and blue crab. Public shellfishing is allowed in some areas near the mouth of the Savannah River in the vicinity of Fort Pulaski. The Savannah River is host for the migration of several important commercial and game fishes including the American shad, the hickory shad, and the blueback herring. Game species include the spotted seatrout, red drum, croaker, spot, striped bass, flounder, silver perch, white catfish, channel catfish, large mouth bass, sunfish, and crappies. The State of Georgia has closed the striped bass fishery for population recovery purposes. Results of a seasonal creel survey of the Savannah River estuarine fishery, conducted by the Georgia Department of Natural Resources from October 20, 1992 to February 16, 1993 found that the estimated angler harvest for that time period was 10,893 fish. White catfish (28.4 percent), spotted seatrout (27.9 percent), red drum (17.9 percent), and silver perch (10.4 percent) represented approximately 85 percent of the fish harvested from the Savannah River during this time period (Schmitt, 1993).

There are several wildlife refuges and/or game management areas located along the lower portion of the Savannah River. Tybee National Wildlife Refuge is located at the mouth of the Savannah River at the confluence with the Atlantic Ocean. Just north of Tybee National Wildlife refuge is the Turtle Island Game Management Area. The Containerport itself is located across the river from the southern end of the 10,371 ha (25,608 acre) Savannah National Wildlife Refuge. The Savannah National Wildlife Refuge and the Tybee National Wildlife Refuge are managed by the U.S. Fish and Wildlife Service.

Climatic Conditions

The Port of Savannah, GA, is located in Chatham County on the Savannah River. The city of Savannah is surrounded by low, flat terrain that is marshy to the north and east and rises to a few meters (several ft) above sea level to the west and south.

The area has a temperate climate which, again, is greatly influenced by winds coming into the area off of the surrounding ocean. Nominally, 50 percent of the rainfall occurs during thunderstorms with the remainder being equally distributed over the year and generally related to frontal passages. Severe tropical systems affect the Savannah, GA, area roughly once every 10 years and cause heavy, sustained precipitation, high winds, and extreme localized coastal flooding. Rainfall measurements in excess of 51 cm (20 in) have been observed as a result of tropical systems impacting the area. Based on the 1951-1980 climatology, the first freeze occurs on average around November 15th and the last freeze occurs near March 10th (NOAA 1992d).

D.2.1.9 Tacoma, WA

The Port of Tacoma is located in the southeastern corner of Puget Sound on the deep waters of Commencement Bay, about 5 km (3 mi) from the Sound. It is a rapidly expanding major port, second only to Seattle in maritime importance on Puget Sound. Like Seattle, access is gained via the Straits of Juan de Fuca and Puget Sound. The distance from the entrance into Puget Sound is approximately 130 km (80 mi). While the transit is open with deep wide channels, it is a relatively long distance on an inland waterway (DOC, 1992b). The port currently handles about 1,054,000 20-ft equivalent container units, amounting to 6.7 million metric tons (7.4 million tons) of cargo (AAPA, 1994). A map of the port is shown in Figure D-30.

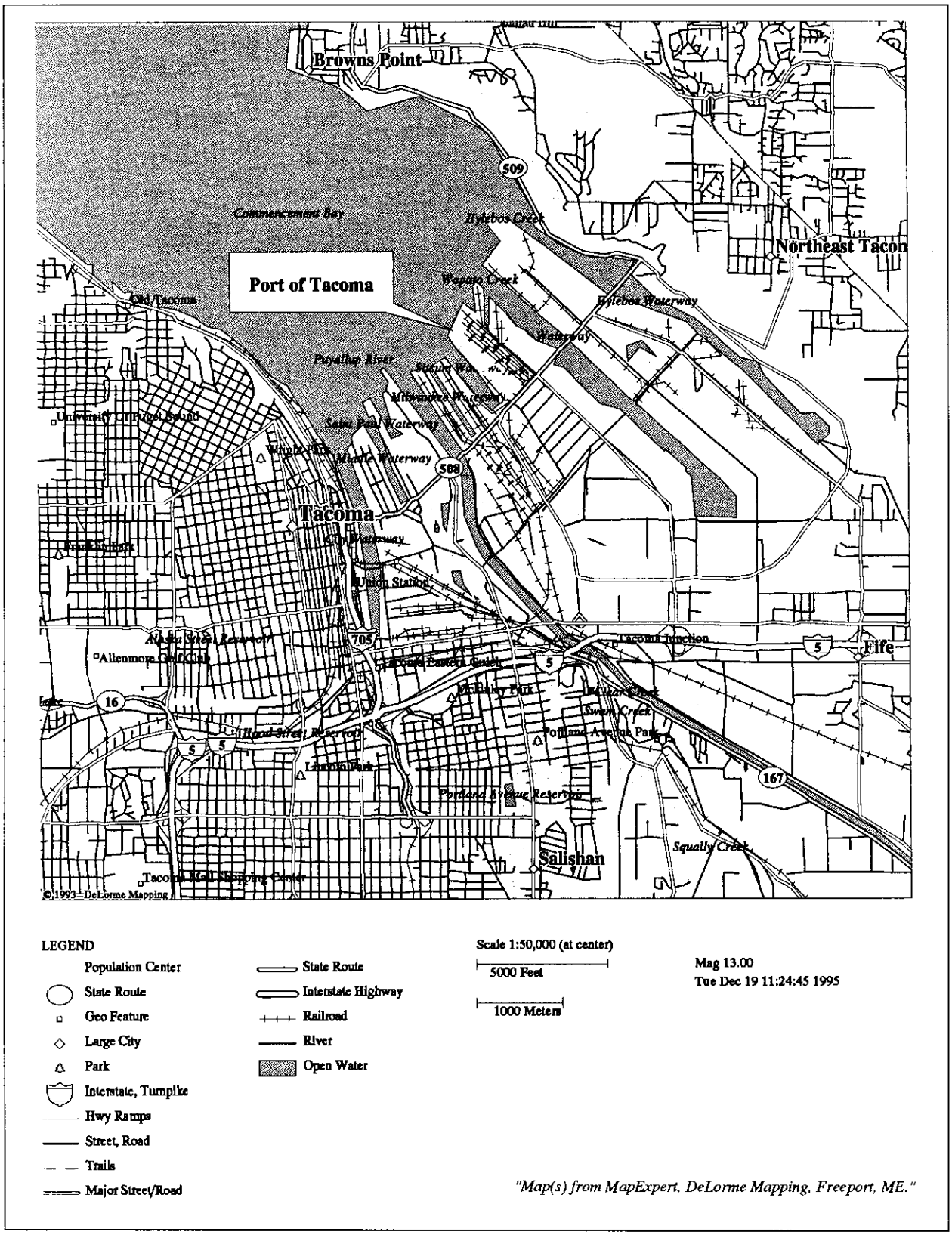


Figure D-30 Map of the Port of Tacoma, WA

The port functions as a special purpose district operation under State enabling legislation and is governed by a Board of Commissioners. The Commission owns and operates several terminals, including container and roll-on/roll-off facilities. Stevedoring is performed by private contractors and/or by ship lines leasing facilities from the port.

Commencement Bay has been designated a "Superfund Site" by the Environmental Protection Agency. However, since the acceptance of spent nuclear fuel through the Port of Tacoma would neither affect the activities being conducted in response to the "Superfund Site" designation, nor would it add any additional burden to this designation, the "Superfund Site" designation has no bearing on the proposed action.

Berths A, B, and C of Terminal 7 are primarily public general cargo facilities handling breakbulk and dry bulk cargoes. Depths alongside range from about 12.2 m to 15.2 m (40 to 50 ft), and it has two 36 metric ton (40 ton) gantry cranes and one 36 metric ton (40 ton) multi-purpose bulk unloading crane. Terminal 7, Berth D (Husky Terminal) is the primary container terminal, and has one 274 m (900 ft) long container berth, 3 container cranes [two 45 metric ton (50 ton) and one 50 metric ton (55 ton)), and 15.2 m (50 ft) of depth alongside at mean low water. It has 14 ha (33 acres) of terminal area with access to the 9,512 m^2 (102,400 ft^2) container freight station and a 8,920 m^2 (96,000 ft^2) transit warehouse located near Berths A and B.

The Husky Terminal is about 4.8 km (3 mi) from the Port of Tacoma road access (Exit 136) to Interstate 5 immediately outside the port complex. While a somewhat longer route, Interstate 5 South connects with I-84 East near Portland, OR, avoiding the added risks of trucking spent nuclear fuel over Snoqualmie Pass to Eastern Washington during the winter. Ship berths are served by the Port Belt Line Railroad, and the port is served by the Burlington Northern and Union Pacific Railroads, which interline with eastern and southern railroads. All Terminal 7 berths are adjacent to the North Intermodal Railroad Yard, which consists of 10.4 ha (26 acres) of yard area and 5,340 m (17,500 ft) of trackage. A second intermodal rail terminal, the South Intermodal Rail Yard, is also located within the port for use by all port shippers (Jane's, 1992; AAPA, 1993; POT, 1994).

Tacoma is served by over a dozen containership and breakbulk ship lines including ELMA, Evergreen, Hyundai, IMT, "K" Line, Maersk, MOL, Navianca, Naviera Pacifico, NOSAC, PCL, Sea-Land, South Pacific Interline, Totem Ocean Trailer Express, Wallenius, and Wallno (Jane's, 1992; AAPA, 1993). These lines provide service with most of the Pacific Rim, including Australia and Japan, and also have service with the Mediterranean (Jane's, 1992; AAPA, 1993).

Other Pertinent Information: According to the port's Director of Risk Management, shipments of spent nuclear fuel could be prohibited by the City of Tacoma Harbormaster's Office, but no formal regulatory restriction was identified. The port has had no identifiable experience with shipment of spent nuclear fuel (SNL, 1994; Paulsen, 1993; NRC, 1993). Security is maintained at Terminal 7 by the port Police around the clock, with locations for segregation and temporary storage of hazardous cargoes (special guards would have to be provided by the shipper for spent nuclear fuel) (Paulsen, 1994). The Tacoma Fire Department provides response for accidents, and the port security personnel are trained in emergency response in cooperation with the Fire Chief (McLendon, 1994). There is also the possibility that the Husky Terminal may begin handling ammonium nitrate in bulk, which (because of the explosion potential) would have to considered in the event the port were to receive spent nuclear fuel shipments (Paulsen, 1994). The U.S. Coast Guard accident statistics for the period 1991-1993 for the Puget Sound indicate a total of 54 reportable accidents (USCG, 1994b). Given the high volume of ship traffic, the accident frequency is considered to be low.

As is the case with Seattle, there is substantial environmental concern about environment damage, and the entire Puget Sound area is subject to severe earthquakes and volcanism. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Tacoma, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a high seismic zone with an acceleration of 0.30 g. There have been two major earthquakes in the Puget Sound area this century; a modified Mercalli intensity VII on April 13, 1949 and a modified Mercalli intensity of VIII on April 29, 1965 (Bolt, 1978). On May 18, 1982, Mount Saint Helens suffered a major volcanic eruption (IPA, 1993). All the mountains along the Cascades Range, from Canada to Northern California, are volcanic in origin, and potentially active (Foster, 1971; Hamilton, 1976; IPA, 1993)

The 1990 population within 16 km (10 mi) of the port terminals was 511,575. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 601,000; Oak Ridge Reservation, 431,000; Idaho National Engineering Laboratory, 157,000; Hanford Site, 98,600; and Nevada Test Site, 379,000. Populations along rail routes to four of these sites are slightly larger, but the population along the rail route to Nevada Test Site is slightly smaller (this is largely due to primary use of interstate highways through Salt Lake City and Las Vegas, use of U.S. 95 south from Oregon would reduce the population along truck routes. These populations are shown in Tables D-6 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,720 km (2,931 mi); Oak Ridge Reservation, 4,280 km (2,658 mi); Idaho National Engineering Laboratory, 1,310 km (815 mi); Hanford Site, 399 km (248 mi); and Nevada Test Site, 2,160 km (1,344 mi). Distances along rail routes are longer.

Environmental Conditions

A variety of aquatic species are found in Puget Sound. Several animal species, with special status, may also be found in this area. A variety of marine mammals can be found in the central Puget Sound including the Pacific harbor seal, California sea lion, killer whale, Dall porpoise, and harbor porpoise. In 1991, the U.S. National Marine Fisheries Services reported that the following endangered and/or threatened species may occur in the Puget Sound: the endangered gray whale, the endangered humpback whale, the threatened Stellar sea lion, and the endangered leatherback sea turtle (DOE, 1995). These species are not reported at the port. The U.S. Fish and Wildlife Service reported that the bald eagle and marbled murrelet, both listed protected species, may occur in the vicinity of the port (Frederick, 1994). Bald Eagles can be found throughout this coastal zone, and American peregrine falcons are uncommon winter visitors (FWS, 1981a). The U.S. Fish and Wildlife Service's Ecological Inventory for the Puget Sound area indicates that the habitat of Commencement Bay is used by a variety of birds including: shorebirds, gulls, sandpipers, turnstones, yellowlegs, herons, rails, great blue herons, waterfowls, loons, grebes, swans, geese, dabbling ducks, diving ducks, mergansers, American widgeons, pintails, mallards, seabirds, cormorants, alcids, common murre, and the pigeon guillemot. Adult concentrations of all of these species may be found in the Bay. Some of these species may also use this area as an overwintering area, a migratory area, and/or a nesting area (FWS, 1981a). It is also indicated that adult concentrations of Chinook salmon, coho salmon, chum salmon, and pink salmon are found in the Puyallup Waterway/River and use this water body and upstream segments as migratory and nursery areas.

According to the State of Washington's Department of Wildlife, a number of seabird colonies exist along the shoreline of Commencement Bay. Areas of the Puget Sound, north of Commencement Bay, are also used as haulouts by the California sea lion. Areas of estuarine wetlands are located along the northern shore of Commencement Bay (WDW, 1994b).

Commencement Bay has been designated a "Superfund Site" by the Environmental Protection Agency. However, since the acceptance of spent nuclear fuel through the Port of Tacoma would neither affect the activities being conducted in response to the "Superfund Site" designation, nor would it add any additional burden to this designation, the "Superfund Site" designation has no bearing on the proposed action.

Climatic Conditions

See Section D.2.2.21 (Seattle) for climatic information, since conditions in Tacoma, WA are essentially the same.

D.2.1.10 Wilmington, NC

The Port of Wilmington, NC is located on the east bank of the Cape Fear River, about 42 km (26 mi) above its mouth on the Atlantic Ocean. It is the leading port of North Carolina, and its major export is wood pulp. It handles about 110,000 20-ft equivalent units per year, representing about 30 percent of total tonnage. The major terminals are down river from the city. A Federal project maintains a 12.2 m (40 ft) channel over the ocean bar into the Cape Fear River, and then 11.6 m (38 ft) to the port. A new dredging program will deepen the approach channel to 12.2 m (40 ft). The approach to Wilmington, up the Cape Fear River, is more open than many river approaches but has restricted segments. The minimum channel width is about 120 m (400 ft). Currents in the river conform to the channel (DOC, 1993d; FHI, 1993c; NCSPA, 1994). A map of the port is shown in Figure D-31.

The port is owned and operated by the North Carolina State Ports Authority (NCSPA), a State agency. It is a modern container and general cargo facility with over 92,900 m^2 (more than a million ft^2) of covered, sprinklered storage and a total of 11 berths, two of which are open. The port has over 40 ha (100 acres) of paved, open area and 10 ha (25 acres) of semi-improved storage area. The Wilmington wharves are of concrete pile construction, rubber fendered, with a total frontage of about 2,000 m (6,568 ft). Berths 6 to 9 are dedicated containership berths with the remaining berths used for various kinds of general cargo. All of the main cargo berths have a depth alongside at mean low water of 11.6 m (38 ft). The terminal has three, 40.6 metric ton (45 ton) container cranes and two, 50.8 metric ton (56 ton) container cranes, plus three gantry cranes ranging from 40.8 metric ton (45 ton) to 204 metric ton (225 ton) (Jane's, 1992; AAPA, 1994; FHI, 1993c; NCSPA, 1994).

Terminal access via truck is through the controlled South Gate Container Entrance. Truck shipments of spent nuclear fuel from Wilmington to southern destinations are from U.S. Routes 17, 74, 76 and 421 to Interstates 95 and 40. The local routes are accessed about 3 km (2 mi) north of the terminal where they cross the Cape Fear River using the lift bridge. Northern and western long distance routes are via Interstate 40 which connects with State Highway 132 about 16 km (10 mi) north of the city. Wilmington container berths are served shipside by the port rail system and the CSX Railroad. There is also an intermodal trailer-on-flat-car and container-on-flat-car rail yard within the container port. While not currently operational, the port is negotiating with CSX for resumption of intermodal rail service at that facility. At the present time, most rail cargo which requires intermodal connections is trucked to the Charlotte Intermodal Terminal (Wilson, 1995).

The port is serviced by over 30 container lines, including Yang Ming, Polish Ocean, Allied Scandinavian, Central Gulf, Zim, Hanjin and Turkish Cargo Line, plus several regularly scheduled breakbulk shipping companies. These lines provide service from Northern Europe, the Mediterranean, Mideast, East and South Africa, South America, the Far East, Australia, and other shipping centers of the world (Southern Shipper, 1993; Jane's, 1992).

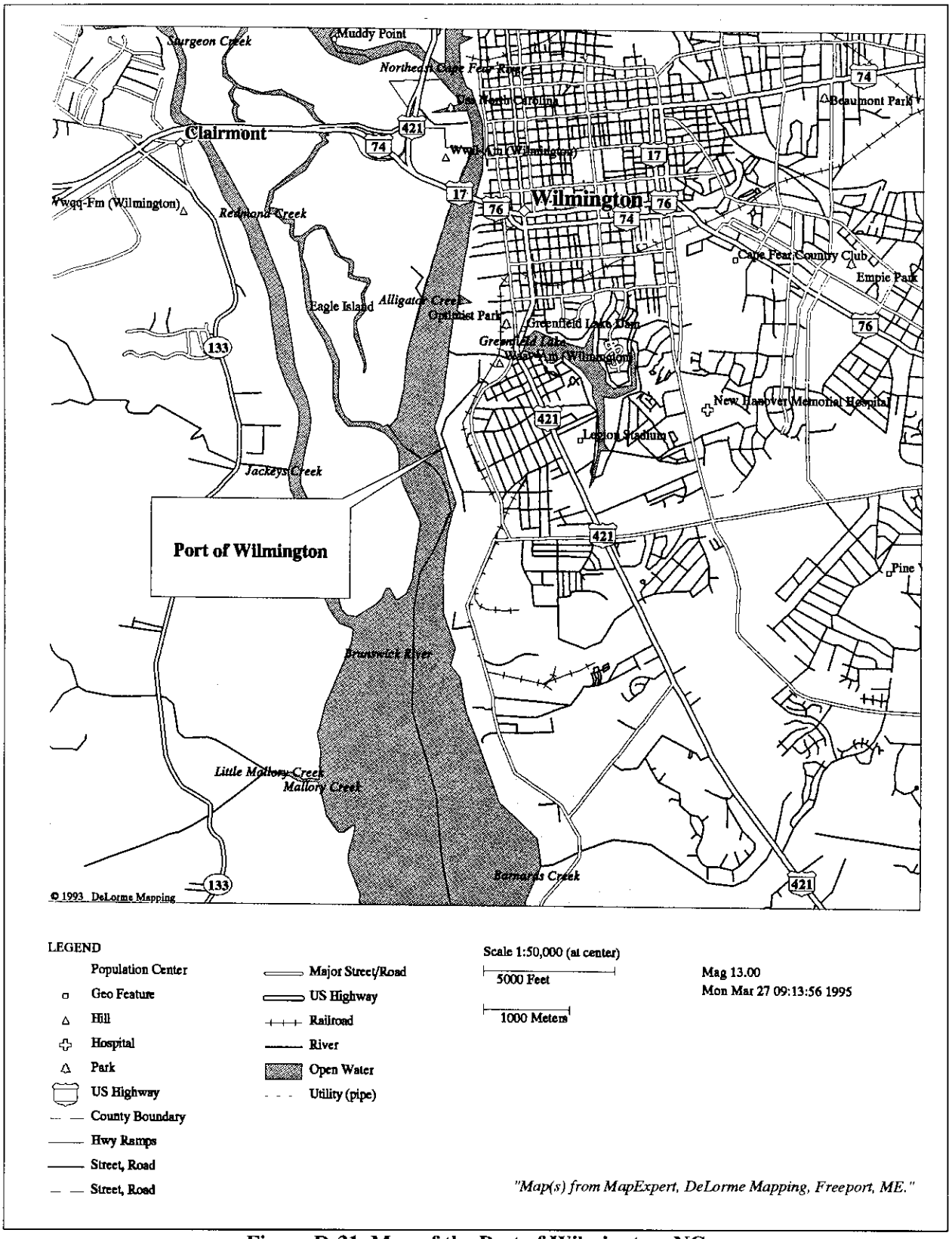


Figure D-31 Map of the Port of Wilmington, NC

Other Pertinent Information: There are no known restrictions on the receipt and handling of spent nuclear fuel through the port, although the Mayor has provided notice that the city is not convinced that the port is desirable for spent nuclear fuel shipments (Betz, 1994). This position was echoed by the Port's Executive Director, who noted that permission to visit the port must come from the State Port Commission, and that the Governor was opposed to handling spent nuclear fuel at State ports (Scott, 1994). Wilmington has handled the import shipments of enriched uranium for nuclear fuel fabrication consigned to a General Electric commercial nuclear fuel fabrication plant north of Wilmington, the exports of the finished nuclear fuel assemblies, and has also handled containerized Class A explosives (Wilson, 1993). The Sandia National Laboratories Radioactive Materials Postnotification Database was queried in April 1994, and the data showed that Wilmington received two shipments of spent nuclear fuel from Japan on February 3, 1986 and transhipped the casks to Savannah River Site the same day (SNL, 1994).

The port is located several miles downstream of the business district in an area of increasing industrial development, although there is some residential housing bordering the complex. The Military Ocean Terminal at Sunny Point is also located on the Cape Fear River, north of Southport, NC, and south of Wilmington, NC.

Port officials are part of an emergency response team headed by the Coast Guard and the Wilmington Fire Department. There are two fire stations within 3 km (2 mi) of the port, with a 5-minute response time (Scott, 1994). All operational personnel working within the terminal, including longshoremen, are given basic hazardous materials training, but training does not deal specifically with spent nuclear fuel.

Security at the port is provided by a 2 m- (6 ft-) high perimeter fencing topped with barbed wire, and a North Carolina State Ports Authority Police Force, which maintains a 24-hour patrol and surveillance. Armed officers are commissioned by the City Police Department, and unarmed guards at the gates are employed by the port (Scott, 1994). A North Carolina State Ports Authority Safety Manager reports to the Director of Operations and is responsible for all safety aspects of the terminal. A tanker terminal and petroleum storage depot are located immediately adjacent downstream of the port. Immediately north of the terminal, on the same side of the river, is an asphalt and chemical storage marine terminal. There is little ship traffic on the River, north or south of the State docks, and therefore there is little conflicting traffic or cargoes.

There are no known environmentally sensitive areas in the immediate vicinity of the terminal, but due to area resorts and recreational activity, there is heightened environmental awareness. The port is subject to hurricanes and tropical storms, as discussed below.

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Wilmington, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 115,057. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 64,700; Oak Ridge Reservation, 128,000; Idaho National Engineering Laboratory, 507,000; Hanford Site, 556,000; and Nevada Test Site, 570,000. Populations along rail routes to these sites are slightly longer. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 500 km (310 mi); Oak Ridge

Reservation, 820 km (509 mi); Idaho National Engineering Laboratory, 4,100 km (2,546 mi); Hanford Site, 4,770 km (2,963 mi); and Nevada Test Site, 4,260 km (2,650 mi). Distances along rail routes are slightly longer for Western sites, but about the same for Eastern sites.

Environmental Conditions

North Carolina has given the lower portion of the Cape Fear River three different stream classifications. From the Northeast Cape Fear River to the confluence with the Cape Fear River the waters are classified as SC-swamp. From the mouth of the Northeast Cape Fear to a point between Snow and Federal Points, the waters are classified as SC. From Snow and Federal Points oceanward the waters are classified as SA. SC waters are tidal waters suitable for fishing, fish and wildlife propagation, secondary recreation and other water uses requiring lower quality. The term "swamp" denotes waters with slow velocity. Class SA waters are suitable for shellfishing, primary recreation, as well as all of the activities approved for Class SC waters (NCDEHNR, 1992). According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for Beaufort, NC, the Port of Wilmington is located in a low salinity estuarine habitat (generally 0.5 to 5 parts per thousand) and tidal freshwater habitat. Below Wilmington at Campbell Island, the river changes to a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). The Cape Fear River near MOTSU changes once again to a high salinity estuarine habitat (generally 16.5 to 30 ppt) (FWS, 1980a).

The lower Cape Fear River supports a large number of aquatic and terrestrial species. There are both invertebrate and fish species of commercial and recreational value found in the Cape Fear River near the Port of Wilmington. Species sought by commercial and recreational fisherman include flounder, trout, spot, croaker, bluefish, Spanish mackerel, and king mackerel. Shellfish sought include penaeid shrimp and blue crabs.

The Natural Heritage Program of the North Carolina Department of Environment, Health, and Natural Resources reports that the area around the state port has not been systematically inventoried for rare species (Smith, 1994). However, DEHNR reports that the lower Cape Fear River, from Wilmington to the mouth of the river at Smith Island, is brackish and contains numerous rare animals. The shortnose sturgeon (State and Federal Endangered Species) rarely occurs in the river, whereas manatees (State and Federal Endangered Species) occasionally occur, especially in the summer. American alligators (State and Federal Threatened Species) can be found in tributary streams. The freckled blenny, spinycheek sleeper, opossum pipefish, and marked goby are other rare marine fishes that inhabit the river.

A large number of aquatic species may be found in the lower Cape Fear River and along the southern coast of North Carolina (Horning, 1994; U.S. Army, 1993; FWS, 1980a). There are many animals with special status in this area, including various types of whales, sea turtles, and birds. State or Federally protected endangered or threatened aquatic species in this area include the shortnose sturgeon (fish), finback whale, Florida manatee, humpback whale, right whale, sei whale, and sperm whale (mammals), Arctic peregrine falcon, bald eagle, piping plover, red-cockaded woodpecker, wood stork (birds), and the American alligator, green sea turtle, hawksbill sea turtle, Kemp's ridley sea turtle, leatherback sea turtle and the loggerhead sea turtle (reptiles and amphibians).

Climatic Conditions

The elevation of this region is approximately 12 m (40 ft) above sea level and is more variable than the coastal plain surrounding the Norfolk, VA, area.

The maritime location of the Wilmington, NC, area makes the climate unusually mild for its northern latitude. All wind directions from the east-northeast through the southwest have some moderating effect on the local climate, due to the relatively warm and cool ocean in the winter and summer seasons, respectively. The area rarely experiences cold episodes where the temperature falls below -18°C (0°F). However, cold air outbreaks do occur, causing sharp fluctuations in winter temperatures. Rainfall in the area is generally considered ample and evenly distributed throughout the year, with the bulk of the precipitation occurring during the summer months. The bulk of this rainfall is generally associated with afternoon and evening thunderstorms. In contrast, the winter rains tend to be of the slow, steady type lasting, generally, one to two days. As is common at Atlantic coastal localities at this latitude, the late summer and early fall months bring the possibility of hurricanes and tropical storms to the Wilmington, NC area. These storms are capable of generating high winds, above normal tides, and torrential rains. The latter two are also capable of creating widespread local flooding of low-lying coastal areas (NOAA, 1992b).

D.2.2 Other U.S. Ports Meeting the Appropriate Experience Criteria

The ports described in this section are those that were initially identified as acceptable based on experience with containerized cargo, but that were subsequently dropped from further consideration based on evaluation against other criteria. Those criteria and the evaluation process are described in Section D.1.

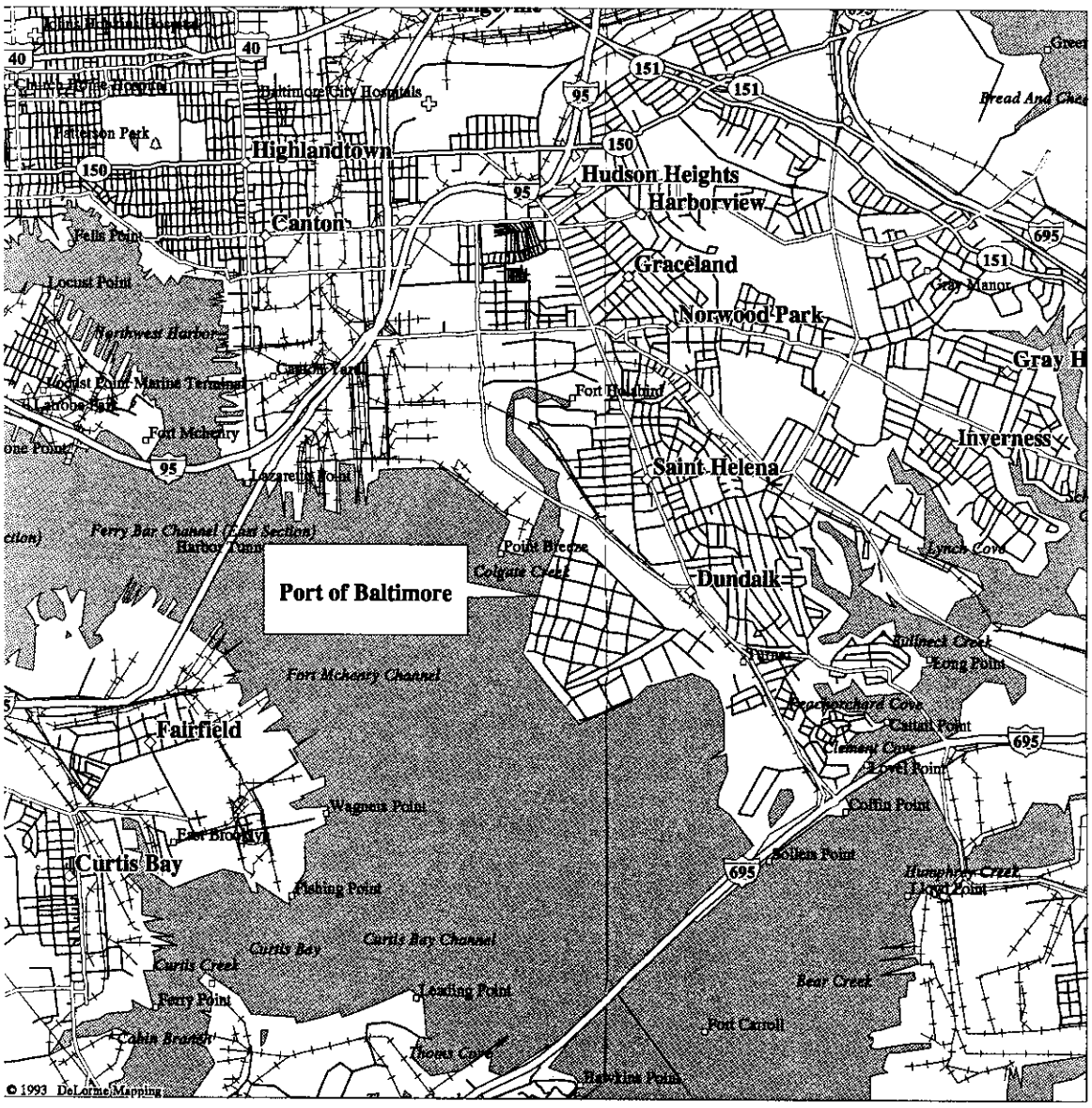
D.2.2.1 Baltimore, MD

The Port of Baltimore, one of the major ports of the United States, is established on the upper Chesapeake Bay at the head of tidewater navigation on the Patapsco River. It is situated 13 km (8 mi) from the entrance to the Patapsco, 240 km (150 mi) from the Virginia Capes, and 1670 km (104 mi) from the Delaware Capes.

Depths within the harbor range from 15.2 m (50 ft) to 12.2 m (40 ft). Federal project depths are 15.2 m (50 ft) in the main channel between the Virginia Capes and Fort McHenry in the Baltimore Harbor. Access to the port can be gained via the Delaware Bay and River, and the Chesapeake and Delaware Canal, although this route is not recommended due to the restrictive nature of the transit. The preferred route is via the Virginia Capes and Chesapeake Bay. The Chesapeake Bay is considered part of the transit to Baltimore; this includes 240 km (150 mi) up the Bay under the Chesapeake Bay Bridge to the Patapsco River (DOC, 1993c). A map of the Port of Baltimore is shown in Figure D-32.

In the Port of Baltimore, the Maryland Port Administration (MPA) owns and operates the two principal general cargo terminals — Seagirt and Dundalk Marine Terminals. Together these terminals provide a total of 338 ha (835 acres) of cargo handling space on deepwater channels. Both terminals are ports of call for many of the world's largest container and roll-on/roll-off shipping lines. Seagirt is capable of handling (length and draft) the largest post-Panamax vessels currently in service (Jane's, 1992; AAPA, 1993; D&B, 1993).

Seagirt Terminal: Seagirt is MPA's newest and most modern container facility with three container ship berths, eight cranes, and 107 ha (265 acres) of developed terminal area. Seagirt also has a deeper water depth alongside berths; the three container berths have a depth alongside at mean low water of 12.8 m (42 ft). The terminal has two 313 m each (1,020 ft each) marginal wharves and one 326 m (1,070 ft) marginal wharf. Crane capacity at the terminal consists of four 50.8 metric ton (56 ton) single hoist container cranes, three 50.8 metric ton (56 ton) double hoist container cranes, and one 27.9 metric ton (30.7 ton) single hoist container crane (AAPA, 1993 and 1994; Jane's, 1992; D&B, 1993).



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- Interstate, Turnpike
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- Interstate Highway
- Railroad
- River
- Open Water
- Intermittent River
- Utility (powerline)

Scale 1:50,000 (at center)
5000 Feet
1000 Meters

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"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-32 Map of the Port of Baltimore, MD

Dundalk Marine Terminal: Dundalk Marine Terminal, located adjacent and eastward of Seagirt, has approximately 231 ha (570 acres) of terminal area and is a combination container, roll-on/roll-off, and breakbulk handling facility. The terminal has 13 barge and ship berths, 11 cranes, and covered storage shed space of more than 37,000 m^2 (400,000 ft^2). Marginal wharves consist of three 808 m total (2,650 ft), two 553 m total (1,820 ft) container berths, and one 305 m (1,000 ft) container berth. Containership berths have a depth alongside of 11.5 m (38 ft). Crane capacity at the terminal includes nine 40.6 metric ton (45 ton) single hoist container cranes and two 61 metric ton (67 ton) gantry whirley cranes (Jane's, 1992; D&B, 1993; AAPA, 1993 and 1994).

The Seagirt and Dundalk Terminals are located in the Dundalk section of the City of Baltimore, east and south of the central business district. The access road to both terminals is bordered primarily by heavy industrial types of businesses with relatively good interstate highway connections. Southbound, the distance from Seagirt to I-695 is roughly 4 km (2.5 mi). The entrance to the Seagirt Marine Terminal is approximately 1.6 km (1 mi) from I-95 connected by Bruening Highway, an industrial roadway that also serves as the main truck access to both terminals. Access to other major interstate highways is via the I-695 Beltway, which would be used to bypass harbor tunnels for Savannah River Site or other southern destinations. Routing and connect time for Dundalk traffic would be virtually the same due to proximity of location to the Seagirt terminal. Seagirt is served by the CSX Railroad, which operates a 16.2 ha (70 acre) intermodal container transfer facility inside the terminal and within 0.3 km (1000 ft) of the ship berths. Conrail serves the Dundalk Terminal for breakbulk cargoes (D&B, 1993; AAPA, 1993 and 1994).

Other Pertinent Information: Security of both terminals is maintained by the MPA Port Police and is deemed to be excellent. There are secure areas for temporary segregation and storage of containers if necessary.

There are no port restrictions against handling spent nuclear fuel. A port safety officer stated that spent nuclear fuel shipments go out of the port with an armed escort (normally at night), and that the port also handles casks (cylinders) of uranium hexafluoride (UF_6) shipments quite frequently. Although there are no known conflicts with other hazardous materials in the immediate terminal area, there is a diversity of marine terminals and ship traffic activity on the Patapsco River which are not deemed to represent a major hazard factor.

There are no known special environmental issues with regards to handling spent nuclear fuel at Baltimore. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Baltimore, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The MPA relies on the hazardous materials teams of the Baltimore City and County fire departments as well as the Coast Guard for response to hazardous materials accidents. The Maryland Department of the Environment also has input on hazardous materials problems. The MPA has an ongoing hazardous materials training program for all port operating personnel, including the longshoremen. Instruction includes dealing with hazardous wastes (but not spent nuclear fuel in particular) in the soil and groundwater due to the former use of the port site.

The 1990 population within 16 km (10 mi) of the port terminals was 1,182,024. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 308,000; Oak Ridge Reservation, 246,000; Idaho National Engineering Laboratory, 482,000; Hanford Site, 531,000; and Nevada Test Site, 665,000. Populations along rail routes to these sites are

much larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,020 km (636 mi); Oak Ridge Reservation, 925 km (575 mi); Idaho National Engineering Laboratory, 3,790 km (2,354 mi); Hanford Site, 4,460 km (2,770 mi); and Nevada Test Site, 4,060 km (2,526 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The Gunpowder Falls State Park is located approximately 22 km (13 mi) northeast of the port. The Remington Farms Wildlife Reserve, on the Eastern Shore of Maryland, is approximately 35 km (23 mi) east of the port. The Eastern Neck National Wildlife Refuge is located in the Chesapeake Bay, about 38 km (25 mi) southeast of the port area. The Fort McHenry National Monument and Historic Shrine is located on a point of land approximately 4 km (2 mi) west of the Baltimore port. Numerous State Parks and other wildlife refuges are located along the passageway in the Chesapeake Bay south of the port.

The endangered peregrine falcon occurs in the vicinity of the Port of Baltimore (Wolflin, 1994). These birds feed, in part, on shorebirds and other waterbirds using the waters of the Port of Baltimore. The endangered Delmarva fox squirrel and the great blue heron (State-protected) nest on the Eastern Neck Island (FWS, 1980g). The bald eagle (endangered) also nests in the Eastern Neck Island area.

The Bay contains many beds of commercially valuable oysters and soft-shelled clams. Blue crabs are harvested extensively throughout the Bay area. Commercial harvesting of channel catfish and menhaden also is important in the Bay area. Numerous types of fish use the Bay area, including the waters around the port, for nursery areas. Common fish species include the American eel, blueback herring, hickory shad, alewife, gizzard shad, perch, striped bass, drum, flounder, and others. Sport fishing for these fish is also common. State-protected species include the Atlantic sturgeon and American shad (FWS, 1980g). The western bank of the Eastern Shore is a migratory area for the dabbling duck (nonendangered) and a heavily used migration pathway for geese.

Climatic Conditions

Baltimore is in a region about midway between the rigorous climates of the North and the mild climates of the South and adjacent to the modifying influences of the Chesapeake Bay and Atlantic Ocean to the east and the Appalachian Mountains to the west. The net effect is to produce a more equable climate compared to inland locations of the same latitude.

Rainfall distribution throughout the year is rather uniform; however, the greatest intensities are confined to the summer and early fall, the season for hurricanes and severe thunderstorms. Rainfall during this period occurs principally in the form of thundershowers, and rainfall totals during these months vary appreciably, depending on the number of thundershowers that occur largely by chance in a given locality. Hurricane-force winds, however, may occur on rare occasions due to a severe cold front or a severe thunderstorm. The greatest damage by hurricanes is that produced along waterfronts and shores by the high tides and waves.

In summer, the area is under the influence of the large semipermanent high-pressure system commonly known as the Bermuda High and centered over the Atlantic Ocean near latitude 30°N. This high-pressure system brings a circulation of warm, humid air masses over the area from the deep South. The proximity of large water areas and the inflow of southerly winds contribute to high relative humidities during much of the year.

January is the coldest month, and July the warmest. Winter and spring have the highest average windspeeds. Snowfall occurs on about 25 days per year on the average; however, an average of only 9 days annually produce snowfalls greater than 1.0 in. Although heaviest amounts of snow generally fall in February, occasional heavy falls occur as late as March. Records for the period August 1950 through December 1967 indicate that the average date of the last temperature as low as 32° in the spring is April 15, while the average date of the first temperature as low as 32° in the autumn is October 26 (NOAA, 1993a).

Glaze or freezing rain occurs on an average of two to three times per year, generally in January or February, although some occurrences have been noted in November and December. Some years pass without the occurrence of freezing rain, while in others it occurs on as many as eight to ten days. Sleet is observed on about five days annually. The sleet season begins as early as November in some years, and ends as late as March in some cases, with the greatest frequency of occurrence in January (DOC, 1993c).

D.2.2.2 Boston, MA

The Port of Boston is located on Massachusetts Bay about 93 km (50 mi) west of Cape Cod and is the largest seaport in New England. Boston North Channel is the main entrance to Boston Harbor and Boston South Channel and The Narrows are alternative entrances. A Federal project on the North Channel (to the Mystic River) provides for a channel width of 460 m (1,500 ft) and a depth of 12.2 m (40 ft) in the eastern section, and a width of 270 m (900 ft) and depth of 10.7 m (35 ft) in the western section (DOC, 1993a).

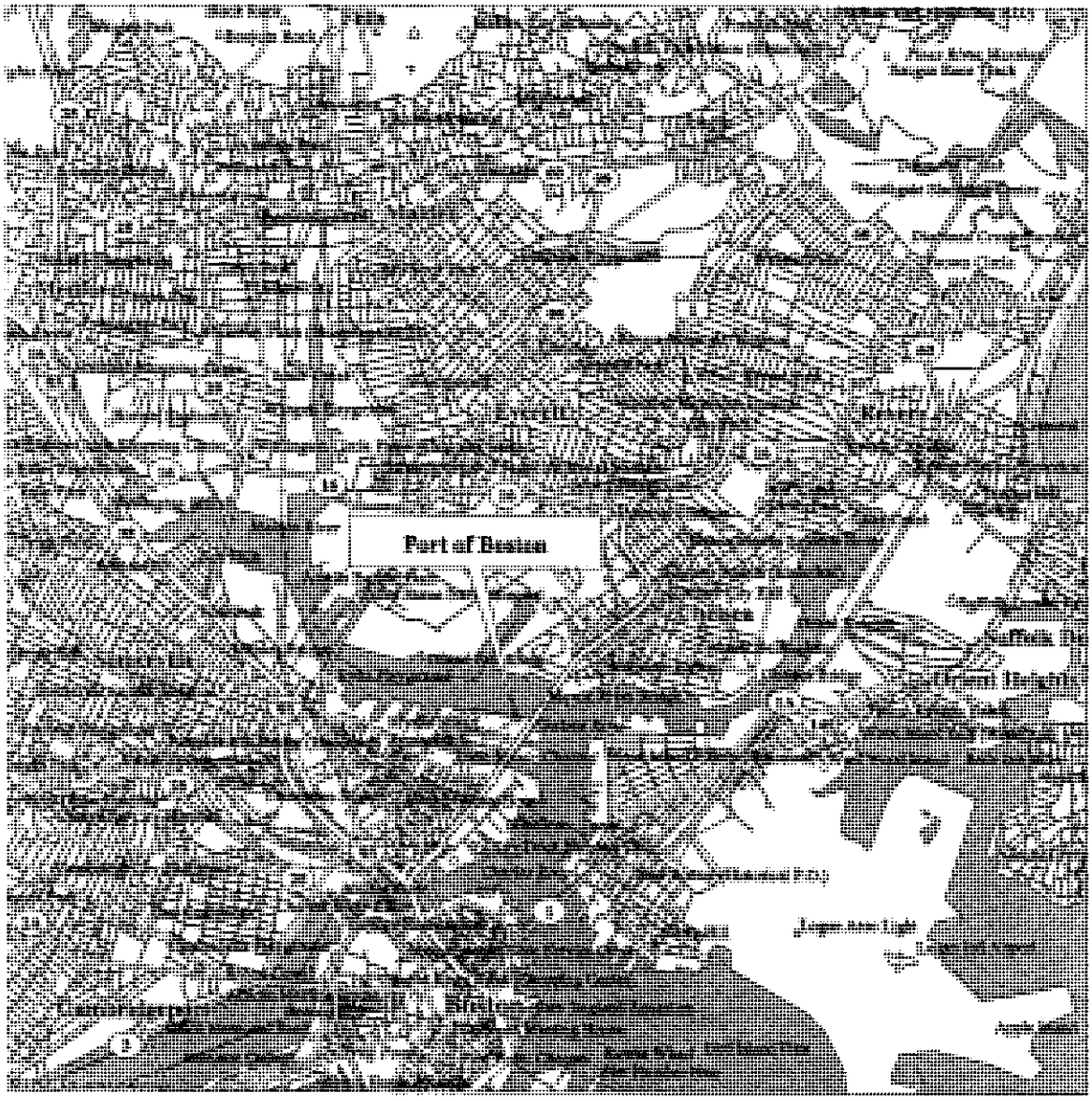
Although there are many obstructions in the Harbor approaches, they are marked by a number of powerful lights, and the principal dangers are buoyed. Because of the heavy traffic to the Harbor, there is a traffic separation scheme extending over 160 km (100 mi) out to sea (DOC, 1993a).

The Massachusetts Port Authority (Massport) is a quasi-governmental authority created by the State Legislature in 1956. The Maritime Division is responsible for the operation, development, and maintenance of the port's three public terminals, including two container terminals (Moran and Conley) and one general cargo facility (Harbor Gateway Terminal) (POB, 1993). A map of this port is provided in Figure D-33.

Moran Terminal: Moran Terminal is located about 7.2 km (4.5 mi) upstream of the Inner Harbor Entrance, on the left side of the ascending bank of the Mystic River in Boston's Charlestown section. It is the largest container terminal in New England and is operated by Massport. The facility consists of 20.2 ha (50 acres) of open storage space, storage capacity for 4,000 20-ft equivalent units, and two container cranes [46 and 71 metric tons (51 and 78 tons)]. It has 335 m (1,100 ft) of marginal wharf and depth alongside of 12.2 m (40 ft). Vessels are limited by the 41.2 m (135 ft) clearance under the Tobin Memorial Bridge (Jane's, 1992; AAPA, 1993 and 1994; D&B, 1993; POB, 1993).

Moran Terminal is situated about 1.6 km (1 mi) from the intersection with I-93 with access via city streets through the densely populated Charlestown area. The terminal is served by the Boston & Maine Railroad, whose tracks enter the terminal and extend to the pier apron.

Conley Terminal: The Conley Terminal on Castle Island is less than 6.4 km (4 mi) from the designated entrance to Boston Harbor. The northern approach to the terminal is obstructed by islands and shoals that extend 6.4 km (4 mi) from the entrance for a combined distance of about 13 km (8 mi). It is located at the entrance of the Inner Harbor on the South Boston waterfront. The Terminal is operated by a subcontractor to Massport. It has 305 m (1,000 ft) of marginal wharf, and consists of Berths 11-15, and Berth 17. The depth alongside is 12.2 m (40 ft).



LEGEND

- Population Center
- State Route
- Geo Feature
- Town, Small City
- Hill
- Hospital
- Park
- Interstate, Turnpike
- US Highway
- County Boundary
- Hwy Ramps
- Street, Road
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Land Mass

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

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"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-33 Map of the Port of Boston, MA

The container terminal, Berth 11, has two 41 metric ton (45 ton) container cranes and an open storage area of 4 ha (9.9 acres). Berth 12 is presently undergoing a $50 million improvement program (to be completed in 1995), and Berths 13-15 are leased to automobile importers. Berths 16 and 17 are served by one container crane (31 metric ton) and are also leased by automobile importers (Jane's, 1992; AAPA, 1993 and 1994; D&B, 1993; POB, 1993). The terminal is approximately 3.2 km (2 mi) from Route I-93, which is part of the Greater Boston Beltway, which then connects with I-95 and I-90 (the Massachusetts' Turnpike). Access to the terminal is via East and West Broadway, a busy South Boston thoroughfare running through an area of primarily small businesses with some old residential housing. Construction of the Third Harbor Tunnel/Seaport Access Road began in 1992 for better interstate access. The terminal is served by Conrail whose tracks are located outside and at the rear of the terminal.

Massport Marine Terminal: This is a 16 ha (40 acre) facility used for cruise ships and the discharge of automobiles (roll-on/roll-off) and bulk cargo. This terminal is about 1.6 km (1 mi) from I-93 via Northern Avenue (a truck route to the Boston Fish Pier) and other industrial users along the waterfront (Jane's, 1992; AAPA, 1993 and 1994; D&B, 1993; POB, 1993).

Other Pertinent Information: Massport has its own security force, which has police powers at State-owned terminals. Although there is no officially designated space for segregating hazardous materials, the port would provide one if necessary. There are no known regulatory restrictions against handling of spent nuclear fuel at Massport terminals; the Deputy Port Director for Operations did not know if the port has ever handled spent nuclear fuel (Moriconi, 1993).

Massport relies on its fire department, which also has a fireboat, for emergency response for hazardous materials accidents, and on Coast Guard supervision. The port also coordinates its activities with State hazardous materials safety personnel. Massport has a training program for terminal workers at Moran, and recently began an introductory course for longshoremen. Training at leased facilities, like Moran Terminal, is the responsibility of the terminal operator (Moriconi, 1993).

Moran Container is located in the densely populated Charlestown area on the Mystic River across from petroleum and natural gas terminals, and a residential condominium/marina complex. Conley Terminal is in an industrial area with less conflicting use, but access is through South Boston, also a densely populated commercial/residential area.

There are no known special environmentally sensitive areas within the port. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Boston, the Uniform Building Code requires buildings to withstand wind speeds up to 140 km/hr (85 mph). The port is located in a moderate seismic zone with an acceleration of 0.30 g.

The 1990 population within 16 km (10 mi) of the port terminals was 1,466,233. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,080,000; Oak Ridge Reservation, 912,000; Idaho National Engineering Laboratory, 716,000; Hanford Site, 785,000; and Nevada Test Site, 796,000. Populations along rail routes to these sites are much larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,734 km (1,079 mi); Oak Ridge Reservation, 1,600 km (995 mi); Idaho National Engineering Laboratory, 4,180 km (2,600 mi); Hanford Site, 4,850 km (3,016 mi); and Nevada Test Site, 4,560 km (2,832 mi). Distances along rail routes are about the same for Hanford Site and Idaho National Engineering Laboratory, and are slightly longer for Savannah River Site, Oak Ridge Reservation, and Nevada Test Site.

Climatic Conditions

Three important influences are responsible for the main features of Boston's climate (DOC, 1993a). First, the latitude 42°N places the city in the zone of prevailing west to east atmospheric flow, which encompasses the northward and southward movements of large bodies of air from tropical and polar regions. This results in variety and changeability of the weather elements. Secondly, Boston is situated on or near several tracks frequently followed by systems of low air pressure. The consequent fluctuations from fair to cloudy or stormy conditions reinforce the influence of the first factor, while also ensuring a rather dependable precipitation supply. The third factor, Boston's east coast location, is a moderating factor affecting temperature extremes of winter and summer.

Hot summer afternoons are frequently relieved by the locally celebrated "sea-breeze," as airflows inland from the cool water surface to displace the warm westerly current. This refreshing east wind is more commonly experienced along the shore than in the interior of the city or the western suburbs. In winter, under appropriate conditions, the severity of cold waves is reduced by the nearness of the then relatively warm water. The average date of the last occurrence of freezing temperature in spring is April 8; the latest is May 3, 1874 and 1882. The average date of the first occurrence of freezing temperature in autumn is November 7; the earliest on record is October 5, 1881. In suburban areas, especially away from the coast, these dates are later in spring and earlier in autumn by up to one month in the more susceptible localities.

Boston has no dry season. For most years the longest run of days with no measurable precipitation does not extend much more than two weeks. This "dry spell" may occur at any time of year.

Much of the rainfall from June to September comes from showers and thunderstorms. During the rest of the year, low-pressure systems pass more or less regularly and produce precipitation on an average of roughly one day in three. Coastal storms, or "northeasters," are prolific producers of rain and snow. The main snow season extends from December through March. The average number of days with four inches or more of snowfall is four per season, and days with seven inches or more come about twice per season. Periods when the ground is bare or nearly bare of snow may occur at any time in the winter.

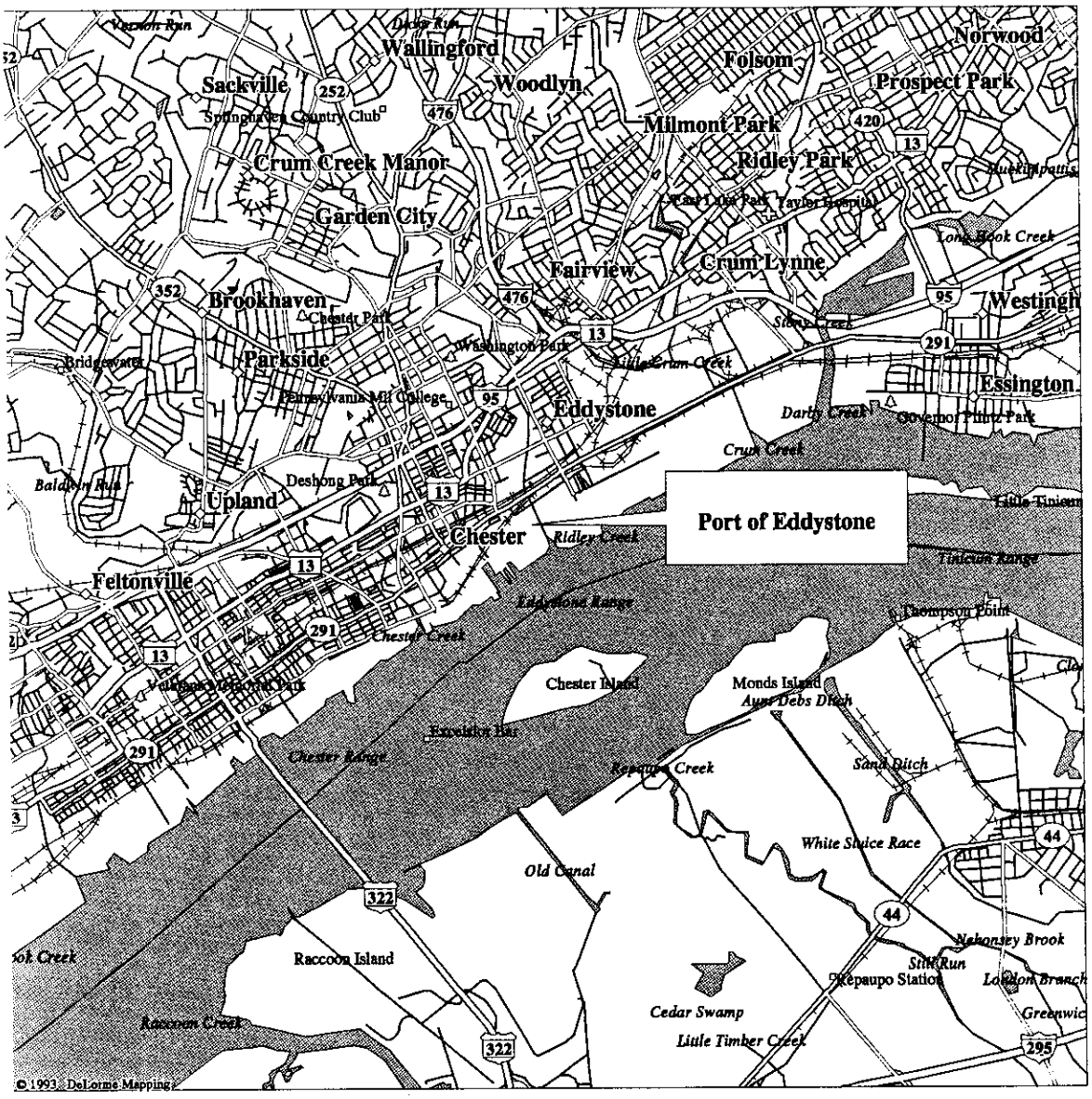
Relative humidity has been known to fall as low as five percent (May 10, 1962), but such desert dryness is very rare. Heavy fog occurs on an average of about two days per month, with its prevalence increasing eastward from the interior of Boston Bay to the open waters beyond. Winds from the east to southwest bring fog and westerly and northerly winds clear the fog away.

At all seasons the heaviest gales are usually from the northeastward or eastward. Although winds of 32 mph or higher may be expected on at least 1 day in every month of the year, gales are both more common and more severe in winter (DOC, 1993a).

D.2.2.3 Eddystone, PA

The location of Penn Terminals at Eddystone is on the former site of Pennsylvania Shipbuilding Company's North Yard Wharf, just upstream of the entrance to Ridley Creek in Eddystone, PA. It is located approximately 18 km (11 mi) south of Philadelphia. It is approximately 1.5 km (1 mi) above the small port of Chester, PA. Penn Terminals, Inc. is one of several independent Delaware River port terminal operators who come under the marketing umbrella of the Delaware River Port Authority (AAPA, 1994). A map of the port is shown in Figure D-34.

Geophysical and navigational data for Penn Terminals is, with the exception of the approach to the Terminal, the same as for Philadelphia and other Delaware River Ports in the immediate vicinity (AAPA, 1994).



LEGEND

- Population Center
- State Route
- Geo Feature
- Town, Small City
- Park
- Interstate, Turnpike
- US Highway
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)
5000 Feet
1000 Meters

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"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-34 Map of the Port of Eddystone, PA

Penn Terminals was founded in 1986 to manage containerized cargoes but subsequently expanded its scope of services to include breakbulk and project cargoes. Penn Terminals' brochure states that they handle about 250 ship calls a year (PT, 1994). A port official reported that the Terminal handles 30,000 to 50,000 20-ft equivalent units a year, including some hazardous and radioactive materials (Davis, 1994). According to the Sandia National Laboratory's Radioactive Materials Postnotification (RAMPOST) Database, on April 17, 1991, this port was used for receipt of about 1.4×10^{16} Bq [366,000 curies (Ci)] of cobalt-60 for shipment to Dickerson, MD, in a Type B cask comparable to those used for spent nuclear fuel shipments (SNL, 1994). There was no indication of foreign research reactor spent nuclear fuel receipts since October 1984, when the database was established.

The Terminal features 40.7 ha (71 acres) of storage area, including 23,200 m^2 (250,000 ft^2) of covered storage. The terminal has 335 m (1,100 ft) of marginal wharf, container gantry cranes, a 27 metric ton (30 ton) and a 41 metric ton (45 ton) and a heavy lift truck crane with a capacity of 220 metric tons (240 tons). Rail service is provided by Conrail. Access to Interstate 95 is about 1.6 km (1.0 mi) from the terminal via industrial and old residential streets (PT, 1994; AAPA, 1994).

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Eddystone, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 609,241. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 400,000; Oak Ridge Reservation, 300,000; Idaho National Engineering Laboratory, 600,000; Hanford Site, 600,000; and Nevada Test Site, 700,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,170 km (725 mi); Oak Ridge Reservation, 1,060 km (660 mi); Idaho National Engineering Laboratory, 3,930 km (2,440 mi); Hanford Site, 4,590 km (2,850 mi); and Nevada Test Site, 4,180 km (2,600 mi). Distances along rail routes are about the same.

Environmental Conditions

Monds Island and Chester Island are directly across the shipping channel from the port, and Little Tinicum Island is 1.8 km (1.1 mi) upriver. Tidal flats surround these islands, which are comprised of marshes and wetlands. The Tinicum National Environmental Center, located approximately 3.5 km (2.2 mi) to the northeast on Darby Creek, is a nationally recognized wetlands and environmental education center.

The Port of Eddystone is located within Zone 4 (tidal river) of the Delaware River. Protected water uses for Zone 4, which encompasses River Miles (RM) 79-95, are water supply (industry), wildlife, resident fish maintenance, anadromous fish passage, secondary contact, and navigation (DRBC, 1994). However, several uses within Zone 4 are currently impacted, including: fish and other aquatic life due to low dissolved oxygen levels from point source discharges, and fish and shellfish consumption due to chlordane and PCB contamination from point and nonpoint source discharges.

The Delaware River at Eddystone is classified as a low salinity estuarine (generally 0.5 to 5 ppt) and tidal freshwater habitat. Aquatic organisms that are typically found in the waters of this area include: American shad, Atlantic sturgeon, American eel, blueback herring, shad, alewife, white catfish, brown bullhead, perch, striped bass, bluegill, crappie, pumpkinseed, largemouth bass, carp, and chain pickerel

(FWS, 1980f). In addition, the Delaware River is used as a migratory area by the shortnose sturgeon, a Federally listed endangered species. The Water Quality Section of the Pennsylvania Department of Environmental Resources reported that 67 species of fish are full or part-time residents of this part of the Delaware estuary (Boyer, 1994). Most importantly, the area of the river between Monds Island, Chester Island, and Little Tinicum Island and the islands' backwaters, is an important spawning site for the striped bass.

This area of the Delaware River serves as a sport fishery for striped bass, American shad, blue-claw crabs, white perch, largemouth bass, and catfish. There is also limited commercial fishing for American eels and American shad. There is only low to medium recreational use of this part of the Delaware River due to the high volume of tanker and freighter traffic (Boyer, 1994).

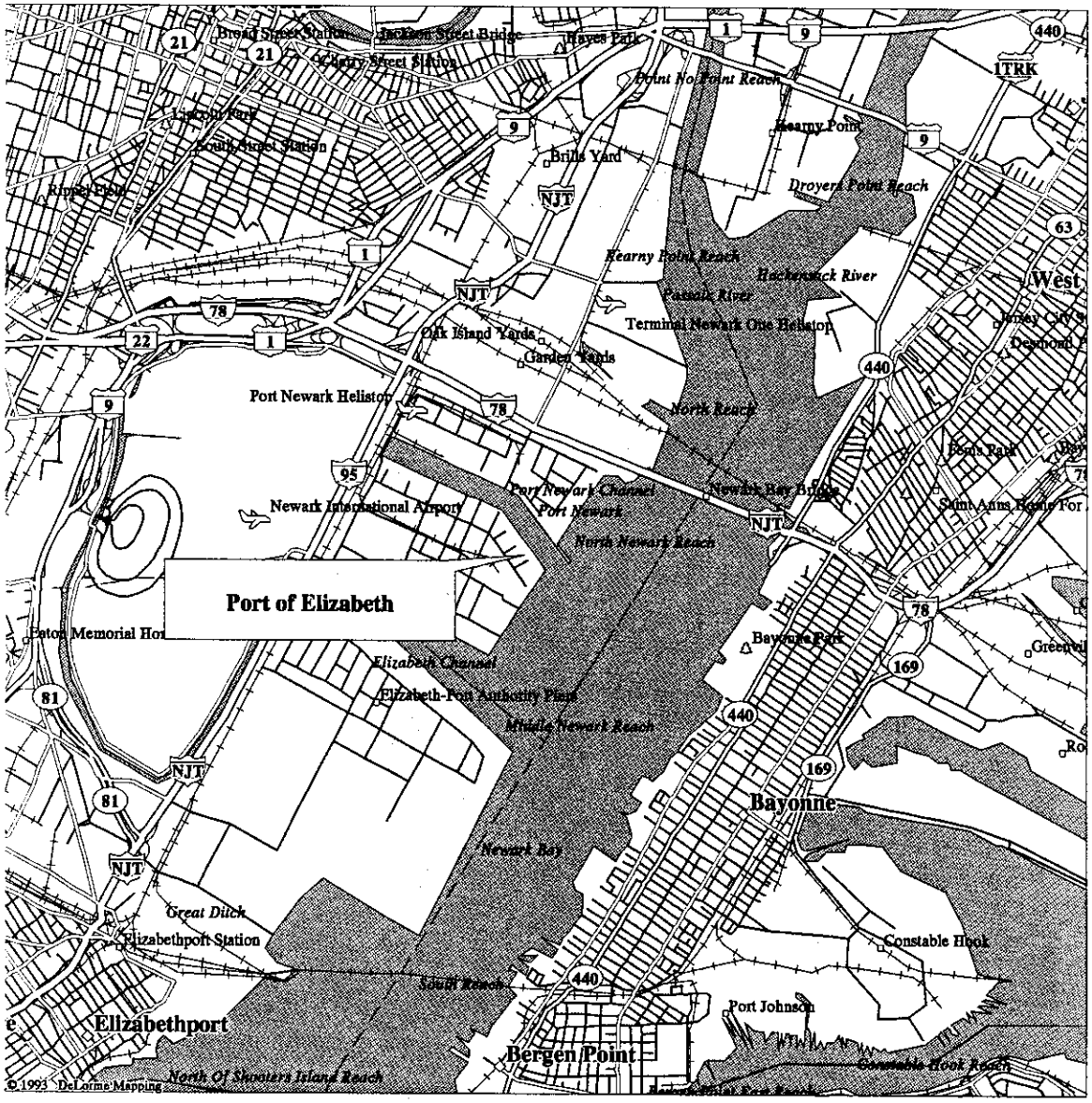
The U.S. Fish and Wildlife Service reported that except for occasional transient species, no Federally listed or proposed threatened or endangered species under their jurisdiction are known to exist in the port's impact area (Perry, 1994). Similarly, the Pennsylvania Natural Diversity Inventory reported that it did not expect any impact on rare, threatened, or endangered plant species in this location (PNDI, 1994).

Climatic Conditions

The climate of the Eddystone region is similar to that of Philadelphia, PA. The area is moderated by the Appalachian Mountains to the west and the Atlantic Ocean to the east. These geographic features cause periods of extreme temperatures to be short-lived in this region (generally, four days). On occasion during the summer months, the area is dominated by maritime tropical air masses, which contribute to elevated local temperature and humidity levels. The average annual precipitation of 105.2 cm (41.42 in) is relatively evenly distributed throughout the year, with maximum amounts occurring during the late summer months. The summer precipitation regime is dominated by localized thunderstorms and are subject to the influence of the urban heat island effect and local topography, which create varying rainfall amounts across the city for an individual event. Singular snowfall events that generate accumulated totals of greater than 25.4 cm (10 in) have a 5-year recurrence interval on average. The prevailing wind direction has a bimodal distribution, being southwesterly during summer and northwesterly in the winter months. The annualized average prevailing wind direction is from the west-southwest. Due to this region's inland location, destructive winds are comparatively rare from tropical cyclones and tornadoes. High winds are generally associated with frontal passages/low pressure systems and thunderstorms in the winter and summer months, respectively. However, flooding on the Schuylkill River normally occurs twice annually, usually associated with strong thunderstorms, with the duration of these events generally lasting less than 12 hrs. The Delaware River is rarely observed at or above flood stage (NOAA, 1992h).

D.2.2.4 Elizabeth, NJ

New York Harbor is the principal entrance by water to New York City and the surrounding ports. The harbor is divided by the Verrazano Narrows into the Lower Bay and Upper Bay. Using the Verrazano Narrows Bridge as a reference point, Port Elizabeth is approximately 18 km (11 mi) from the Lower Bay and the Atlantic Ocean via Kill Van Kull. The Battery, the southern tip of Manhattan, is at the junction of the East River and Hudson River. New York Harbor includes New York City, Staten Island, and the New Jersey principal ports of Perth Amboy, Port Elizabeth, Port Newark, and Bayonne. The project depth of the channels leading from the sea through the Lower Bay, Narrows and Upper Bay is 13.7 m (45 ft). Depths in the Kill Van Kull leading to the New Jersey container terminals is 10.7 m (35 ft). The approaches to New York Harbor are open, but highly trafficked. The 13 km (8 mi) down the Kill Van Kull to Port Elizabeth is restricted (DOC, 1993b). A map of the port is provided in Figure D-35.



LEGEND

- Population Center
- State Route
- Geo Feature
- Town, Small City
- Park
- Interstate, Turnpike
- US Highway
- Airfield
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Open Water

Scale 1:50,000 (at center)
5000 Feet
1000 Meters

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"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-35 Map of the Port of Elizabeth, NJ

Unlike many ports of the world, no single governmental or public agency in New York is responsible for controlling the overall operation of the port. Port administration is divided among many organizations, both private and public, which have an institutional interest in port activities.

The Port Authority of New York and New Jersey (Port Authority) is a quasi-public agency established in 1921 by treaty between the states of New York and New Jersey to deal with the planning and development of terminal and transport facilities, and to improve and protect the commerce of the port district.

The Port Authority's main maritime facilities are located in Elizabeth, Port Newark, and Hoboken, New Jersey, and in New York, at Erie Basin and Columbia Street terminals in Brooklyn. The City of New York owns the South Brooklyn Marine Terminal, Red Hook Marine Terminal (also in Brooklyn), and Howland Hook Marine Terminal on Staten Island. The latter is a major container terminal now leased to the Port Authority. Global Terminal, a privately owned and operated container facility, is located in Jersey City. All told, there are five separate container areas within the harbor equipped with a total of 35 container cranes along a total quay length of 8,000 m (approximately 5 mi), and a total berth area of about 500 ha (1,236 acres) (Jane's, 1992; AAPA, 1993).

With the exception of Global Terminal, all of the foregoing terminals are leased from the Port Authority or the City of New York and operated by terminal operating companies or steamship lines. Since virtually any one of these terminals would be physically capable of handling containers of spent nuclear fuel, description of the port's capabilities is limited to a single terminal, the Port Authority Marine Terminal, within the Port Authority's Port Elizabeth/Port Newark container complex. The Port Elizabeth/Newark area has direct access to the New Jersey Turnpike and is farthest removed (relatively) from centers of population.

Sea-Land Terminal (Elizabeth NJ): Berths 88-98 on the southeast corner of Elizabeth Channel have 1,403 m (4,603 ft) of marginal wharf. The terminal has 12.2 m (40 ft) depth alongside at mean low water. Sea-Land has crane capacities of six 40.6 metric ton (44.8 ton) container gantry cranes. Truck access to the New Jersey Turnpike (I-95) is via Port Newark (Exit I4) or Exit 13A in Elizabeth. The latter is reached via McLester Street to State Route 81 to the Turnpike. The route is almost entirely within the Port Authority Marine Terminal complex and distance traveled is estimated to be about 4.8 to 6.4 km (3 to 4 mi), respectively. The Sea-Land Terminal is adjacent to the Conrail Portside and the Port Authority intermodal rail yards.

In addition to Sea-Land, the Terminal is used by Hanjin Shipping Lines, Ltd., Italia Line, Nedlloyd, P & O Containers, Samskip, S.C.I. Line, Spanish Line, and Transroll Navegacao, SA. The list of container lines calling at other terminals is extensive and represents the major container carriers of the world (Jane's, 1992; AAPA, 1993).

Other Pertinent Information: Individual Terminals are responsible for their own security arrangements. However, it is believed that the New York Port Association controls and may serve as watchmen. All terminals are fenced with controlled access and 24-hour surveillance. A port official did not know what type of short-term storage arrangements exist at the Sea-land Terminal; however, he believed there is provision for segregating hazardous cargoes. He also did not know if there are any restrictive ordinances pertaining to spent nuclear fuel or if the port has handled it (Hennessy, 1993). Available data indicates spent nuclear fuel shipments have not been handled at least since 1979 (NRC, 1993; SNL, 1994).

The Kill Van Kull waterway, serving Port Elizabeth/Newark terminals, is also the approach route to the refineries and petroleum storage depots located along the Arthur Kill to the south. There is a great diversity of traffic and cargoes in the harbor but, due to the layout of the terminals, this diversity and traffic are not considered a major concern.

The Coast Guard and fire departments from the cities of Elizabeth and Newark, respond to hazardous materials incidents within terminals located within their municipalities. The Union County hazardous materials team responds to accidents in Port Elizabeth, and the Newark hazardous materials team in Port Newark. Sea-Land and other terminal operators have contracts with private companies for oil and chemical spill cleanup and/or decontamination work. It is not known what type of hazardous materials training is provided by terminal operators and/or the Port Authority (Hennessy, 1993). Training normally is provided in such large port operations. This was not investigated further because the port was not included in the final list selected for detailed assessment due to the extremely large populations around the port.

The Port Elizabeth/Newark terminals are separated from the urban city centers bearing their names. However, both are adjacent to Newark Airport and areas of heavy industrialization and heavy traffic on the Turnpike. There are also areas of dense population on the east side of Newark Bay in the cities of Bayonne and Jersey City. The 1990 population within 16 km (10 mi) of the port terminals was 3,223,038. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 316,000; Oak Ridge Reservation, 290,000; Idaho National Engineering Laboratory, 536,000; Hanford Site, 585,000; and Nevada Test Site, 782,000. Populations along rail routes to these sites are much larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,320 km (828 mi); Oak Ridge Reservation, 1,190 km (738 mi); Idaho National Engineering Laboratory, 3,860 km (2,396 mi); Hanford Site, 4,520 km (2,812 mi); and Nevada Test Site, 4,300 km (2,672 mi). Distances along rail routes are slightly longer.

There are no known special environmental concerns in the greater New York/New Jersey port area. The likelihood of severe natural phenomena, such as high winds and earthquakes, are reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Elizabeth, the Uniform Building Code requires buildings to withstand wind speeds up to 140 km/hr (85 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

Climatic Conditions

New York Harbor is located on the Atlantic coastal plain at the mouth of the Hudson River. The terrain is flat and diversified by numerous waterways; all but one of the city's five boroughs are situated on islands. Elevations range from less than 15 m (50 ft) over most of Manhattan, Brooklyn, and Queens to almost 90 m (300 ft) in the northern part of Manhattan and the Bronx, and over 120 m (400 ft) in Richmond (Staten Island).

Despite its nearness to the ocean and the numerous bays and rivers nearby, the port has a climate which more closely resembles the continental type of climate than it does the maritime type. Its modified continental climate follows from the fact that weather conditions affecting the city usually approach from a westerly direction and not from the ocean on the east. Some important exceptions to this must be noted, since the oceanic influence is by no means entirely absent. During the summer, local "sea breezes," winds

blowing onshore from the cool water surface often moderate the afternoon heat; and most often in winter, coastal storms, accompanied by easterly winds, produce, on occasion, considerable amounts of precipitation.

From November through April the prevailing winds are from the northwest; for the remainder of the year the prevailing winds are southwesterly. Gales with velocities of 64 km/hr (40 mph) or more are predominantly from the northwest.

The mean annual temperature is slightly higher than that of most places in the United States of the same latitude, with the exception of localities near the Pacific coast. Precipitation is both moderate and distributed evenly throughout the year. Most of the rainfall from June through September comes from thunderstorms, therefore, is usually of brief duration, but relatively intense. From October to April, however, precipitation is generally associated with widespread storm areas, so that day-long rain or snow is common. Over the entire year, the city receives 59 percent of the sunshine hours possible at its latitude. This value compares favorably with that for any region east of the Mississippi, except the Southeast. Relative humidity averages about 66 percent for the year, showing that the city has a relatively damp climate.

Winds play an important role by affecting currents in the harbor. During the winter, west and northwest winds prevail, with northerlies and southwesterlies in secondary roles. The strongest winds are out of the west through northwest at 13 to 15 knots, from January through April. The sheltering effect of the land is apparent when looking at frequencies of winds of 28 knots or more. These winds blow at Ambrose Light about eight to nine percent of the time compared to one percent at Kennedy Airport and Floyd Bennet Field. Summer winds are often out of the south and southwest with a 10 to 12 knot afternoon peak. Fog in the harbor area is more closely related to land-type fogs. In winter, fog is common on clear, calm mornings and more frequent than at Ambrose Light. Southerlies can also bring winter fogs of the advection type. During the spring and early summer, the harbor and its approaches are susceptible to advection fog, riding in on east through south winds. A morning peak still exists in the harbor, while Ambrose Light exhibits an afternoon maximum (DOC, 1993b).

D.2.2.5 Fernandina Beach, FL

The Port of Fernandina is located about 9.3 km (5 mi) above the Entrance Seabuoy to the St. Marys River and Cumberland Sound. The entrance is bordered by two jetties on the approach to the cities of Fernandina Beach (located on Amelia Island) and St. Marys, GA, the Naval Submarine Base in Kings Bay, and an inland passage to St. Andrew sound via the Cumberland River (DOC, 1993d). The entrance is approximately 37 km (20 mi) north of the entrance to the Port of Jacksonville, which is located on the St. Johns River. A map of the port is shown in Figure D-36. Amelia Island is a small, historic, coastal resort town. Fort Clinch, a State Park, museum, and recreation area is located on the north end of Amelia Island at the inshore end of the south entrance jetty (DOC, 1993d).

The Port of Fernandina is a forest products and general cargo container port. It handles around 25,000 20-ft equivalent units of containerized freight and about 272,000 metric tons (299,000 tons) of forest products annually, but the container volume has varied considerably from year to year. Much of the port's trade is with South and Central America. There is also eastbound monthly service to the Mediterranean (Southern Shipper, 1993; American Shipper, 1994; Stubbs, 1994). Reportedly, the current controlling depth of the entrance, and that of approach channel to the submarine base, is 14.3 m (47 ft) and the controlling width is 122 m (400 ft). The same width channel with 10.3 m (34 ft) controlling depth is available to the Ocean Highway and Port Authority Terminal in Fernandina Beach. There is a 1.8 m (6 ft)

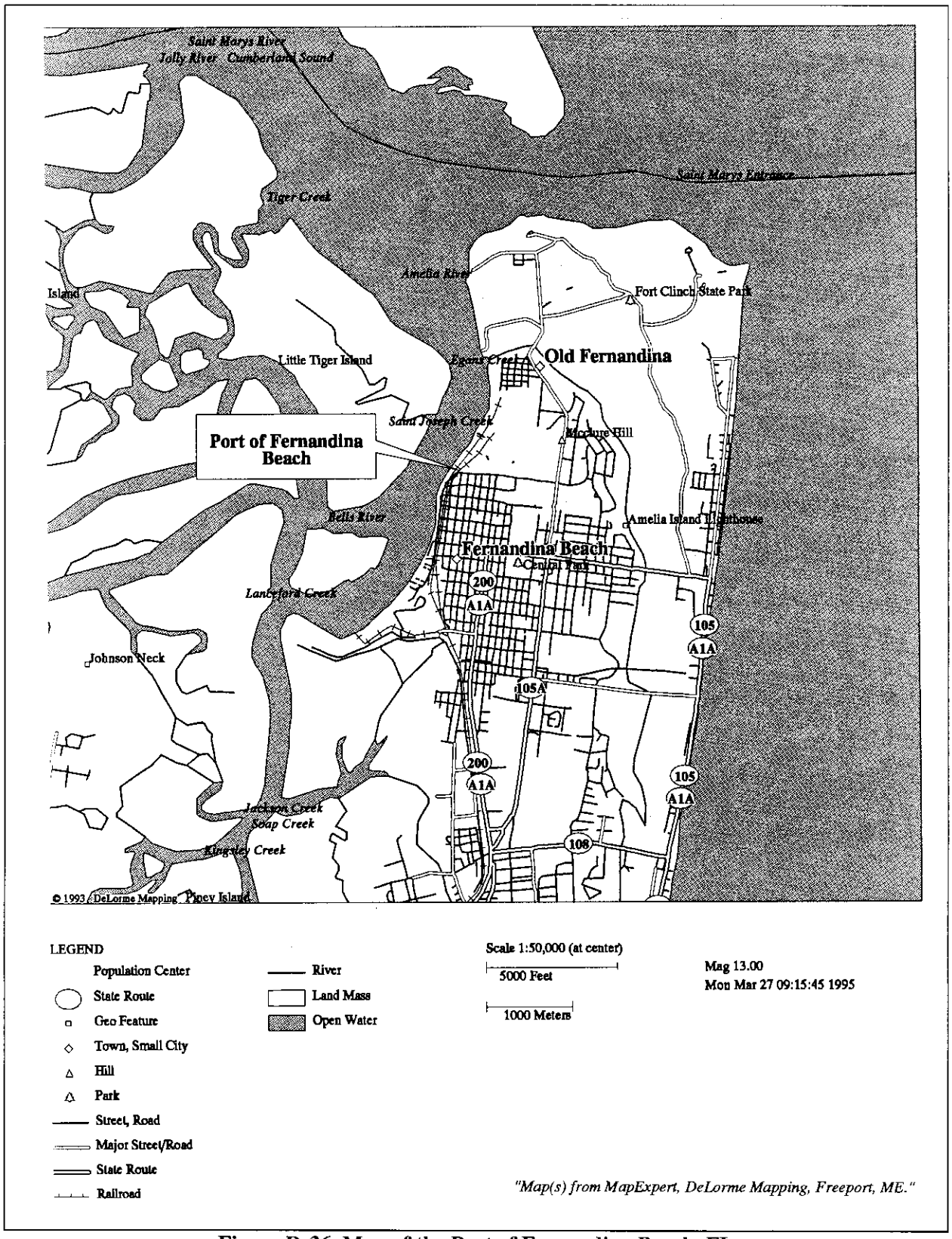


Figure D-36 Map of the Port of Fernandina Beach, FL

range of tide at Fernandina Beach. Tidal currents at the entrance have considerable velocity and are dangerous at times. A strong current "set" occurs at the St. Mary's entrance under certain weather conditions, it has been reported as high as 2.6 mi-per-sec (5 knots) (DOC, 1993d).

The Forest Products Terminal, located about 12 km (8 mi) above the channel entrance, is a publicly owned facility operated by Nassau Terminals, a private terminal operating and stevedoring company. Following a port expansion in 1992, the Terminal consists of 366 m (1,200 ft) of useable berthing situated on the left ascending bank of the Amelia River. The new capacity of the port is about 50,000 20-ft equivalent units per year. The Terminal is equipped with two 36 metric ton (40 ton) container cranes and other container handling equipment, a 4,645 m^2 (50,000 ft^2) container freight station, 2.0 ha (5 acres) of open storage area, and is served by the CSX Railroad with pierside rail trackage (DOC, 1993d; Southern Shipper, 1993). The port handles an average of two vessels a day, typically a cruise vessel and a cargo vessel. The only products normally handled by the port are forest products for a paper mill located in the area, and containers loaded with food and paper products. The passenger or cruise ship business is small, using smaller vessels for cruises in the near islands and offshore (Robas, 1994).

The port terminal is located in the downtown section of the town of Fernandina Beach. Truck access to the port is through the downtown area and mixed residential/business structures for a distance of about 8 km (5 mi). Total distance to Interstate 95 is about 24 km (15 mi), much of which is divided multi-lane highway of mostly rural character.

Other Pertinent Port Information: Terminal property is fenced and lighted and has 24-hour watchman service. Rail openings into the port are not secured. The port has little experience in handling hazardous materials, in that hazardous materials are not normally shipped in or out of the port (Robas, 1994).

The U.S. Army Corps of Engineers was to award a contract in October 1994 for deepening the harbor channel to 11 m (36 ft) and constructing a 366 m (1,200 ft) turning basin. The approach channel to the Terminal passes through a State aquatic preserve for the manatee and other marine animals. Nassau Terminals occasionally handles some containerized hazardous materials; however, a port official thought there would be considerable local opposition to handling spent nuclear fuel shipments for fear of the effect of adverse publicity on tourism in this popular resort area (Stubbs, 1994).

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Fernandina Beach, the Uniform Building Code requires buildings to withstand wind speeds up to 150 km/hr (95 mph). The port is located in a low seismic zone with an acceleration of 0.075 g or less.

The 1990 population within 16 km (10 mi) of the port terminals was 32,952. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 45,000; Oak Ridge Reservation, 185,000; Idaho National Engineering Laboratory, 590,000; Hanford Site, 650,000; and Nevada Test Site, 650,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 610 km (380 mi); Oak Ridge Reservation, 920 km (570 mi); Idaho National Engineering Laboratory, 4,000 km (2,500 mi); Hanford Site, 4,700 km (2,900 mi); and Nevada Test Site, 4,200 km (2,600 mi). Distances along rail routes are slightly longer for Western sites, and about the same for Eastern sites.

Environmental Conditions

The State of Florida has classified the Amelia River near the St. Mary's Entrance as a Class III water body. This classification indicates that the waters are suitable for recreation, and propagation and maintenance of a healthy, well balanced population of fish and wildlife (FL DEP, 1994). In addition, the State of Florida has designated certain waters in the vicinity of Fernandina Beach as "Outstanding Florida Waterways", which are afforded special protection. Outstanding Florida Waters are generally waters located within national parks, state parks, national seashores, marine sanctuaries, or aquatic preserves. Waters located near the Port of Fernandina Beach designated as Outstanding Florida Waters include Fort Clinch State Park, Fort Clinch State Park Aquatic Preserve, Nassau Valley State Reserve, and the Nassau River-St. Johns River Marshes Aquatic Preserve (FL DEP, 1994).

The Amelia River, in the vicinity of the Port of Fernandina Beach, is characterized as a mid- salinity estuarine habitat (generally 5 to 16.5 parts per thousand). There are both commercial and recreational fish and invertebrates found in the vicinity of the port. These aquatic species include: blue crabs, shrimp, American eel, Atlantic menhaden, tarpon, sea catfish, sheepshead, spotted seatrout, weakfish, spot, Atlantic croaker, kingfish, drum, flounder, silver perch, bluefish, mullet, pinfish, pigfish, ladyfish, and snapper (FWS, 1980e).

The Fort Clinch State Park and Fort Clinch State Park Aquatic Preserve are located on Amelia Island adjacent to Fernandina Beach. Birds that can be found in Fort Clinch State Park include various types of shorebirds, wading birds, waterfowl, raptors, songbirds, and seabirds. Endangered or threatened bird species in Fort Clinch State Park include: brown and white pelican, great egret, snowy egret, tricolored heron, little blue heron, black-crowned night heron, yellow-crowned night heron, least bittern, wood stork, white ibis, bald eagle, northern harrier, osprey, American kestrel, merlin, peregrine falcon, clapper rail, piping plover, American oystercatcher, least tern, black skimmer, royal tern, caspian tern, sandwich tern, worm-eating warbler, yellow-throated warbler, prairie warbler, Louisiana waterthrush, and American redstart (Fort Clinch State Park, 1994). Species with special status found in the area include the loggerhead sea turtle, the manatee, the American alligator, the least tern, and the burrowing four-o'clock (Murray, 1994). The loggerhead sea turtle, a Federally protected species, uses much of Amelia Island and Cumberland Island as nesting areas. In addition, the U.S. Fish and Wildlife Service reports that the following protected marine species may occur in Nassau County: west indian manatee (endangered), Kemp's ridley sea turtle (endangered), leatherback sea turtle (endangered), loggerhead sea turtle (threatened), hawksbill sea turtle (endangered), and the green sea turtle (threatened). Protected bird species include the wood stork (endangered) and red-cockaded woodpecker (endangered) (Bentzien, 1994).

Climatic Conditions

As with the other more northern ports, the climate of this area is also modified by the influence of the Atlantic Ocean. Easterly winds occur roughly 40 percent of the time, producing a true maritime climate for the area. The greatest rainfall occurs during summer, usually associated with afternoon and evening thunderstorms. During summer, measurable precipitation can be recorded nearly every two days. The prevailing winds are northeasterly in the fall and winter months and become more southwesterly during spring and summer. Although this region is located along the eastern Florida coast, it has been very fortunate in escaping hurricane-force winds. The majority of systems in recent years that have reached this latitude have moved parallel to the coastline, keeping well offshore. Others have weakened significantly moving over land prior to reaching the area. The combination of these two factors has spared the area from any major devastation due to tropical systems in recent years (NOAA, 1992e).

D.2.2.6 Freeport, TX

Freeport harbor is located about 64 km (40 mi) southwest of the Galveston, Texas harbor entrance, and about 5 km (3 mi) from the Gulf of Mexico on the Brazos River (DOC, 1992a), with the Gulf Intracoastal Canal crossing the river, making deepwater activity available. The main channel is maintained at 13.6 m (45 ft) and leads to a 364 m (1,200 ft) turning basin (D&B, 1993). Freeport is principally involved in petroleum and petrochemical transport (AAPA, 1994). However, in 1992, 188,400 metric tons (207,711 tons) of containerized cargo (approximately 20,000 20-ft equivalent units) were handled in the port. Primary inbound cargoes were bananas and fruit, and primary outbound cargoes were rice and chemicals (AAPA, 1994).

The harbor is regulated by the Navigation and Canal Commissioners of the Brazos River Harbor Navigation District, and is known locally as Brazosport (DOC, 1992a). The ship channel has been improved by construction of jetties on either side of the entrance. A map of the port is shown in Figure D-37.

Berth assignments at the Port of Freeport are made by the Terminal Superintendent. The port has five general breakbulk berths, 664 m (2,190 ft) in length with 10.9 m (36 ft) depth alongside. There is 19 ha (47 acres) of open storage adjacent to the wharves (D&B, 1993). The port has rail facilities with dual tracks on Berths 1, 1A (Brazos Harbor Public Facility Wharf), and 2 (Brazos River Harbor Wharf No. 2). Both facilities have substantial covered storage available for short-term storage. General cargo is usually handled by the ship's tackle, and no container cranes are available at the port [a floating 450 metric ton (500-ton) derrick is available for heavy lifts by special arrangement] (DOC, 1992a; AAPA, 1994).

Highway connection from the port is via State Highways 227 and 288, for approximately 56 km (35 mi) to Houston, where Interstate-10 is accessed.

Other Pertinent Information: There are no known restrictions on receipt of foreign research reactor spent nuclear fuel at the port, but there are substantial conflicting activities at the port, including petrochemicals and hazardous chemicals (AAPA, 1994). The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Freeport, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 census population of Freeport was 12,600. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 360,000; Oak Ridge Reservation, 300,000; Idaho National Engineering Laboratory, 480,000; Hanford Site, 530,000; and Nevada Test Site, 530,000. Populations along rail routes to these sites are slightly higher for Savannah River Site and Oak Ridge Reservation, but slightly lower for Idaho National Engineering Laboratory, Hanford Site, and Nevada Test Site. The approximate distances to the five potential sites on interstate routes are: Savannah River Site, 1,600 km (1,000 mi); Oak Ridge Reservation, 1,600 km (1,000 mi); Idaho National Engineering Laboratory, 3,100 km (1,900 mi); Hanford Site, 3,700 km (2,300 mi); and Nevada Test Site, 3,100 km (1,900 mi). Distances along rail routes are about the same.

Climatic Conditions

Weather in this area is only an occasional navigational problem. Winds blow at 28 knots (32 mph) or more approximately 3 to 4 percent of the time in November and from January through April. Average speeds are 12 to 14 knots (14 to 16 mph) during this period. Fog is also a winter problem, and visibilities drop below 160 m (0.25 mi) on approximately three to six days each month from November through April.

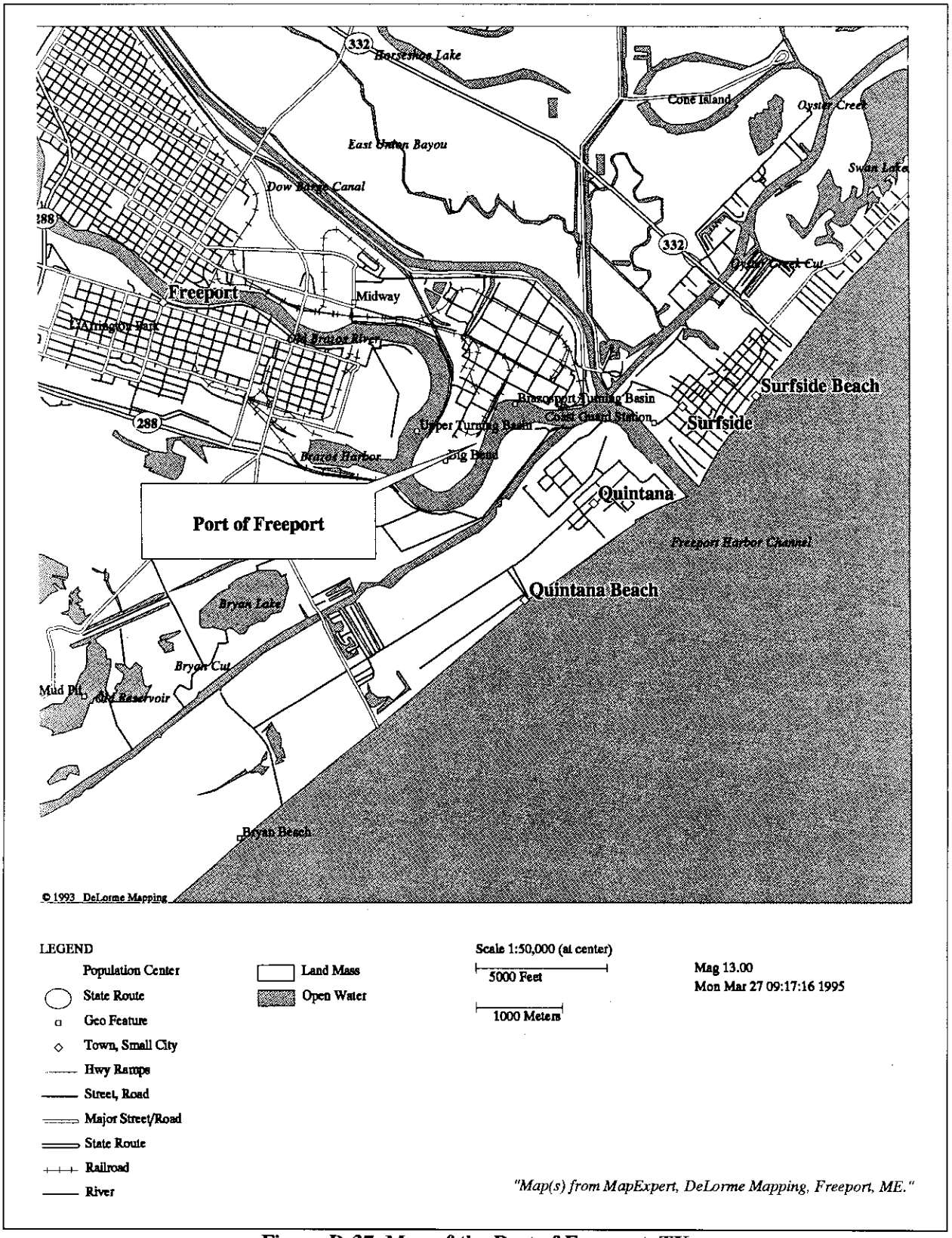


Figure D-37 Map of the Port of Freeport, TX

Thunderstorms are most frequent from April through September, during the afternoon and evening. These thunderstorms are usually air mass types as opposed to the less frequent but more severe thunderstorms that occur with fronts and squall lines from fall through spring. Tropical cyclones, particularly severe hurricanes, are most likely in August and September (DOC, 1992a).

D.2.2.7 Gulfport, MS

Gulfport, the seat of Harrison County, MS, is a seaport and tourist center located on the north side of Mississippi Sound, approximately 26 km (16 mi) from the entrance to the Ship Island Bar Channel on the Gulf of Mexico. Gulfport is located approximately 97 km (60 mi) east of New Orleans, LA. The approach to Gulfport is through a dredged channel marked by lighted buoys. Federal project depths are 9.7 m (32 ft) for the bar channel and 9.1 m (30 ft) for the Gulfport Channel and Harbor Basin. The harbor was deepened to 10.97 m (36 ft) mean low water in 1993 (DOC, 1992a; AAPA, 1993; Southern Shipper, 1993). A map of the port is shown in Figure D-38.

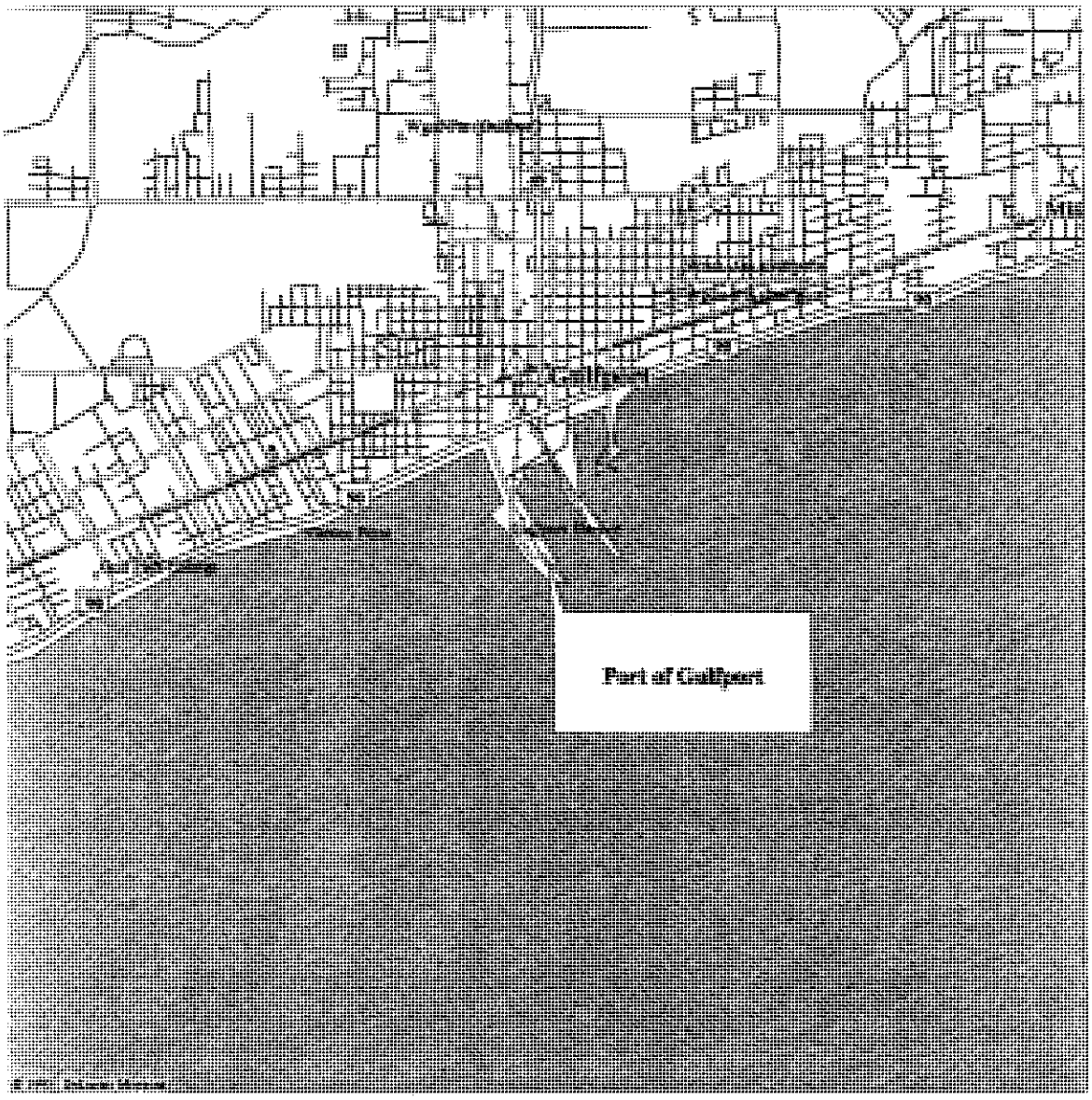
The State-owned Port of Gulfport is a small, but growing, niche port on the Gulf Coast primarily handling containerized banana imports and dry bulk commodities. The port has a growing general cargo outbound container tonnage as the fruit carriers fill otherwise empty containers on the return leg of voyages. Container traffic for fiscal year 1991 included 68,000 20-ft equivalent units amounting to approximately 664,973 mt (733,000 ton) of cargo (AAPA, 1993). By 1993, container volume increased to 736,100 mt (811,559 tons), or approximately 75,000 20-ft equivalent units (AAPA, 1994).

Gulfport has 10 berths with a total of 1,768 m (5,800 ft) of lineal berthing space. There is an open storage area of four ha (10 acres) and a shed area of 19,000 m^2 (204,500 ft^2). A second container berth (East Pier) is used for self-contained container ships. The port's West Pier container berth is approximately 750 m (2,460 ft) long, whereas the East Pier is approximately 200 m (656 ft) long. Gulfport has two 30.5 metric ton (34 ton) container cranes at its West Pier (AAPA, 1994).

The port is located immediately adjacent to the City of Gulfport, which forms the northern boundary of the terminal area. The Terminal has almost immediate access to U.S. Highway 90, and is about 5 mi from I-10, a major east-west roadway. U.S. Highway 49, which begins at the terminal gate and connects with I-10, runs through the center of the City. The port is served by the CSX and Mid-South Railroads with connections to the Norfolk Southern at Hattiesburg. Double trackage extends to the container berth.

Gulfport is presently served by one common carrier combination container/breakbulk ship operator, ABC Line, which operates five large ships on North European around-the-world trade routes. Three other containerized fruit carriers also regularly call at Gulfport (AAPA, 1994; Southern Shippers, 1993).

Other Pertinent Information: Gulfport employs a port security firm that maintains 24-hour guard service. The port is fenced with controlled access to vehicles and personnel. It does not appear that there are any regulations preventing the importation of spent nuclear fuel, although the port indicates that the Coast Guard may impose bans on especially hazardous shipments (Edwards and Burns, 1993). Gulfport has no prior experience handling spent nuclear fuel (NRC, 1993; SNL, 1994) and, as far as is known, there are no hazardous cargoes routinely handled at Gulfport. Port personnel provide First Response augmented by the Gulfport Fire and Police Departments. The Port of Gulfport conducts hazardous materials training of port personnel (Edwards and Burns, 1993). There is a former small cruise ship terminal at the East Pier as well as a floating casino located near the street entrance to the port. The United States Coast Guard indicated that the East Dock is presently slated for casino development, and there are two casinos on the north end of



LEGEND

- Population Center
- Geo Feature
- Large City
- Hill
- US Highway
- Street, Road
- Major Street/Road
- US Highway
- Railroad
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:18:12 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-38 Map of the Port of Gulfport, MS

the West Dock. As a result, the hazardous materials area at the north end of the West Dock has been eliminated for explosives. Also, the facility of particular hazard cannot be used for foreign research reactor spent nuclear fuel, and is not secure or well lit. (Brown, 1995)

There are no known sanctuaries or wildlife habitats in the immediate port area. However, to enter Gulfport, ships must pass close to the protected Gulf Islands National Seashore. The port is subject to severe hurricane and tropical storms. The likelihood of severe natural phenomena, such as high winds and earthquakes, are reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Gulfport, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of 0.075 g or less.

The 1990 population within 16 km (10 mi) of the port terminals was 113,153. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 194,000; Oak Ridge Reservation, 146,000; Idaho National Engineering Laboratory, 435,000; Hanford Site, 484,000; and Nevada Test Site, 683,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 910 km (565 mi); Oak Ridge Reservation, 920 km (573 mi); Idaho National Engineering Laboratory, 3,570 km (2,219 mi); Hanford Site, 4,240 km (2,635 mi); and Nevada Test Site, 3,530 km (2,195 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The Mississippi Sound is separated from the Gulf of Mexico by a series of uninhabited barrier islands. Vessels approaching the port from the Gulf enter the Ship Island Channel, which runs between the west end of Ship Island and Ceat Island and the east end of Cat Island. Ship Island, along with Horn Island and Petit Bois Island, which are further to the east, comprise the Gulf Islands National Seashore. These islands serve as a wintering and migratory area for the protected Peregrine Falcon as well as various shorebirds. Ships entering the Ship Island Channel from the Gulf pass directly north of the northern end of the Chandeleur Islands that comprise the Breton National Wildlife Refuge and Breton Wilderness. The Breton National Wildlife Refuge and Breton Wilderness, which includes areas of black mangroves, serves as a breeding area for the protected loggerhead sea turtle and brown pelican, and a migratory area for the protected peregrine falcon. This area is also home to a variety of shorebirds, wading birds, waterfowl, raptors, seabirds, and songbirds (FWS, 1982a).

The U.S. Fish and Wildlife Service reported that several Federally-listed protected species may occur in the Port of Gulfport area. These species include the endangered brown pelican and Kemp's ridley sea turtle and the threatened gulf sturgeon and loggerhead sea turtle (Goldman, 1994). According to the Mississippi Natural Heritage Program, four protected species of bird have been spotted feeding or loafing in the area of the Port of Gulfport. These species include the royal tern, black rail, reddish egret, and the piping plover (Gordon, 1994). Commercial harvesting areas for the eastern oyster are located throughout the Mississippi Sound, including several areas within a few miles of the port. Breeding areas for the State-protected Least tern also are located along the coast of Gulfport (FWS, 1982a).

The waters in the vicinity of the port have been classified by the State of Mississippi for "recreation" but not as a water supply (Reaves, 1994). The nearshore waters of the Mississippi Sound are characteristic of a middle salinity estuarine habitat (generally 5 to 20 parts per thousand). Aquatic organisms that are

typically found in the waters of this area include: shrimp, blue crab, seatrout, croaker, drum, spot, kingfish, flounder, catfish, mullet, Florida pompano, bluefish, Gulf menhaden, bay anchovy, Crevalle jack, blue runner, Alabama shad, and Atlantic bottlenose dolphin (FWS, 1982a).

Climatic Conditions

Because of Gulfport's geographic location, the local weather is greatly influenced by the Gulf of Mexico. Generally, summers are warm but temperatures are more moderate than those observed at inland locations because of the diurnal sea breeze circulation. Winter weather is generally mild, with the exception of the occasional cold air outbreak. These events occur at 3-10 day intervals between October and April in the Gulf of Mexico region, generally lasting less than three days. The annual rainfall in this region is among the highest in the continental United States. The precipitation is fairly evenly distributed throughout the year with a maximum coinciding with the summer thunderstorm season and minimum occurring during the late Fall months. However, extended rainy periods are rare in this region. Thunderstorm frequencies are highest in July and August, where they may occur every other day, but rarely do they reach intense or violent levels. However, the area is quite vulnerable to tropical systems (e.g., Hurricane Camille, 1969), which originate in the West Indies, West Caribbean, and the Gulf of Mexico (NOAA, 1992i; Wayland and Raman, 1989).

D.2.2.8 Houston, TX

Houston is the largest city in the State of Texas, and the Port of Houston is one of the largest ports in the United States (in terms of total tonnage handled). Morgans Point, approximately 37 km (23 mi) from the entrance to Galveston Bay, marks the beginning of an extensive industrial area lining the Houston Ship Channel. Houston is at the head of the channel, 71 km (44 mi) from the Gulf of Mexico. The transit of large ships is restricted to the ship channel across Galveston Bay and through parts of the San Jacinto and Buffalo Bayou. A Federal project provides for a 12.2 m (40 ft) channel from the entrance to the Gulf of Mexico to Houston (Brady Island) DOC, 1992a). The Houston Ship Channel is 12.2 m (40 ft) deep, and its width ranges from about 76 m to 120 m (250 to 400 ft), making the transit difficult for the large number of ship transits each year (D&B, 1993). A map of the port is shown in Figure D-39.

The Port of Houston is a 40.2 km-(25 mi)-long complex of diversified public and private facilities located on both banks of the Houston Ship Canal, which empties into and transits Galveston Bay. Bulk cargoes, dry and liquid, (including petroleum and petrochemicals) comprise the major share of tonnage handled by the port. Estimated tonnage for 1992 amounted to a total of 114.3 million metric tons (126 million short tons), of which bulk accounted for 72 percent, breakbulk 6.6 percent, and container 3.4 percent.

The Houston Port Authority owns and operates six public cargo facilities including: the Turning Basin Terminal (general cargo) located at the head of the Houston Ship Channel; Jacintoport Terminal (general cargo) located on the north side of the channel near Channelview, Texas; and Barbours Cut Container Terminal located at the head of Galveston Bay on the left ascending bank of the Houston Ship Channel (Jane's, 1992; AAPA, 1993; POHA, 1993).

Barbours Cut Container Terminal: About 40 km (25 mi) from the entrance to the ship channel, this terminal is designed to handle containers, roll-on/roll-off ships. The Barbours Cut Container Terminal is equipped with eight container cranes and five container berths [each is 305 m (1,000 ft) long], plus a separate roll-on/roll-off terminal. The terminal occupies 87 ha (215 acres) of developed land, including 17.8 ha (44 acres) of paved marshalling area for roll-on/roll-off cargoes. Barbours Cut Container Terminal

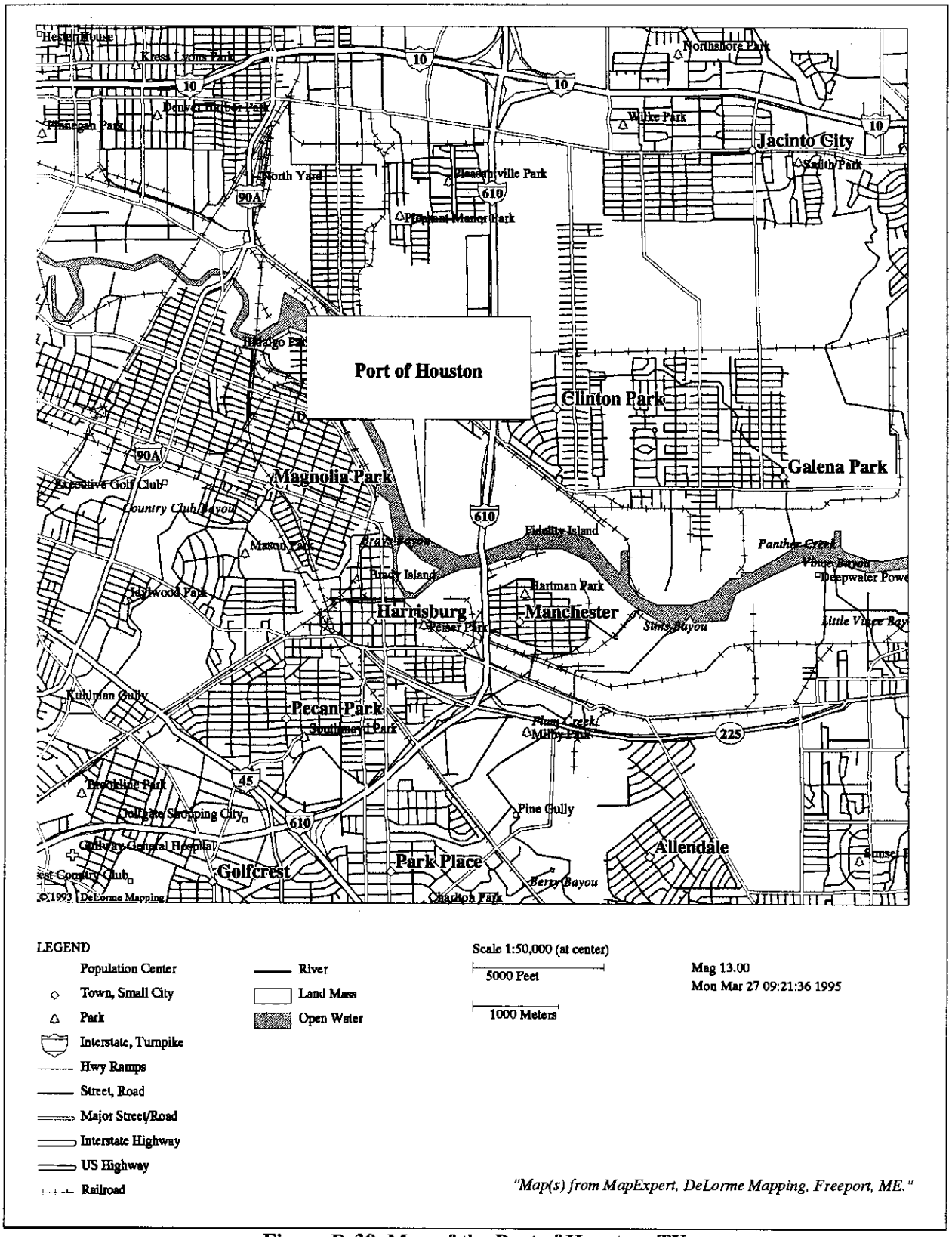


Figure D-39 Map of the Port of Houston, TX

has depths of 12.2 m (40 ft) at mean low water. Crane capacity for Barbours Cut Container Terminal is six 40.6 metric ton (45 ton) container cranes and two 30.5 metric ton (34 ton) container cranes (Jane's, 1992; AAPA, 1993).

Turning Basin Terminal: This terminal has several berths, the largest of which is 243.8 m (800 ft) long and can handle a 228.6 m (750 ft) ship. Turning Basin Terminal's depths are 10.97 m (36 ft) at mean low water. Crane capacity for this terminal is one 40.6 metric ton (45 ton) container crane and one 76.2 metric ton stiff-leg crane.

Barbours Cut Container Terminal has three entry points (gates) with a total of 21 truck lanes that are reached via Barbours Cut Boulevard, a multi-lane limited access roadway. Access to I-610, the Houston Beltway, and other interstate highways is via State Highway 146, which connects with State Highway 225 about 4.8 km (3 mi) from the Terminal. The Route 225 connector is an east-west highway about 22.5 km (14 mi) long. It appears that these routes run through commercial/residential areas with the opportunity for congestion. Barbours Cut Container Terminal is served by the Port Terminal Railroad Association and the Santa Fe Railroad. The Railroad Association connects with all other railroads including the Southern Pacific, Union Pacific, Burlington Northern, and the Houston Belt and Terminal Railroad. Trailer-on-Flat-Car shipments are possible within the terminal, but trackage does not extend to the container berths (Jane's, 1992; AAPA, 1993).

Barbours Cut Container Terminal is host to a large number of major international container and roll-on/roll-off ship lines. A partial listing includes: ABC Container Line, A. Bottacchi, ACL/Gulf Container Line, Afram Lines Ltd, America/Africa/ Europe Line, Atlantic Cargo Services, Baltic Shipping Co., Bank Line, Barber Blue Sea, CGM, CNAN, Columbus Line, COSCO, Costa Container Service, DB Turkish Cargo Line, Djakarta Lloyd, East Asiatic Ltd, Ellerman Line, Farrell Lines, Gulf Mideast Lines, Hapag-Lloyd, Hoegh Lines, Hyundai Merchant Marine, Italian Line, Ivaran Lines, Jugolinija, Kingwood Container Line, Maersk Line, Mediterranean Shipping Company, Nedlloyd Lines, SafBank Line, Sea-Land, Shipping Corp of India, Delmas-Vieljeux, Spanish Line, Torm Lines, Trans Freight Lines, United Arab Shipping Co., Waterman-Isthmian Line, and Zim Line (Jane's, 1992, AAPA, 1993).

Other Pertinent Information: The Houston Port Authority has its own 24-hour security force and all of its terminals are fenced with controlled access. A fireboat is stationed at Barbours Cut Container Terminal, which also has a full-service fire department. There is space within Barbours Cut Container Terminal for temporary segregation of hazardous cargoes (Horan, 1993).

A Port Authority Official was unaware of any regulations prohibiting the importation of spent nuclear fuel. The Houston Port Authority handles a lot of hazardous cargoes including radioactive substances, but the official did not know if the port has ever handled spent nuclear fuel (Horan, 1993). Available data indicates the port has not handled spent nuclear fuel at least since 1979 (NRC, 1993; SNL, 1994). The Houston Ship Channel and Galveston Bay are host to many petroleum and petrochemical berths and terminals served by a large amount of tanker and tank barge traffic. Many of these facilities are located upstream of Barbours Cut Container Terminal, which does not appear to have any conflicting use within its boundaries. The Houston Port Authority has its own emergency response team and fire department. The Houston Fire Department's hazardous materials team is used as a backup in emergencies. The Houston Port Authority has a hazardous materials training program for its terminal operating personnel. It is not known if longshoremen also receive this training (Horan, 1993).

There have been a number of ship accidents, tanker fires, and pipeline accidents at facilities near the Port of Houston in recent years. The United States Coast Guard data indicates that for the period 1991 to 1993, there were about 7,100 ship transits of the channel that resulted in 32 collisions, 33 allisions, 5 ship fires,

and 59 hard groundings (USCG, 1994b). Because the accident statistics also reflect barge traffic risks, the accident rate for oceangoing vessels is probably lower, but there is not data to refine that estimate available yet.

The Turning Basin Terminal is located at the terminus of the Ship Channel in a densely populated area above all other public and private terminal facilities within the port. Barbours Cut Container Terminal is remotely located from the City of Houston with relatively good separation from other terminals and traffic in the area (see II.E above). However, the two small communities of Morgan Point and La Porte (with a population of about 20,000) are located adjacent to the Terminal on the south.

There are no special plant, fish, or wildlife sanctuaries in the vicinity of Barbours Cut Container Terminal. The port is subject to hurricanes and tropical storms. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Houston, the Uniform Building Code requires buildings to withstand wind speeds up to 150 km/hr (95 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 census city population was 1,630,553, with the density estimated at 1,083 persons/km^2. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 349,000; Oak Ridge Reservation, 283,000; Idaho National Engineering Laboratory, 471,000; Hanford Site, 579,000; and Nevada Test Site, 524,000. Populations along rail routes to these sites are slightly smaller for Idaho National Engineering Laboratory, Hanford Site and Nevada Test Site, but slightly larger for Savannah River Site and Oak Ridge Reservation. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,550 km (964 mi); Oak Ridge Reservation, 1,480 km (918 mi); Idaho National Engineering Laboratory, 3,000 km (1,866 mi); Hanford Site, 3,610 km (2,282 mi); and Nevada Test Site, 2,930 km (1,818 mi). Distances along rail routes are about the same except for Savannah River Site, which is slightly longer.

Climatic Conditions

The climate of Houston is predominantly marine. The terrain includes numerous small streams and bayous, which together with the nearness to Galveston Bay, favor the development of both ground and advective fogs. Prevailing winds are from the southeast and south, except in January, when frequent passages of high-pressure areas brings invasions of polar air on prevailing north winds.

Temperatures are moderated by the influence of winds from the Gulf, which results in mild winters and, on the whole, relatively cool summer nights. Another effect of the nearness of the Gulf is abundant rainfall, except for rare extended dry periods. Polar air penetrates the area frequently enough to provide stimulating variability in the weather.

The average number of days with minimum temperatures of 32°F or lower is only about 7 per year at the city's National Weather Service office and about 15 per year at William P. Hobby Airport, which is about 16 km (10 mi) southeast of the city. Most freezing temperatures last only a few hours because they are usually accompanied by clear skies.

Monthly rainfall is evenly distributed throughout the year. In past years, about 75 percent of the total precipitation has been between 76.2 and 152.4 cm (30-60 in). Since thundershowers are the main source of rainfall, precipitation may vary substantially in different sections of the city on a day-to-day basis.

Records of sky cover for daylight hours indicate about one-fourth of the days per year as clear with maximum of clear days in October. Cloudy days are relatively frequent from November to May, and partly cloudy days are more frequent from June through September.

Snow rarely occurs; however, on February 14-15, 1895, 51 cm (20 in) of unmelted snow was measured. Heavy fog occurs on an average of 16 days a year, and light fog occurs about 62 days a year in the city, but the frequency of heavy fog is considerably higher at William P. Hobby Airport. Destructive windstorms are fairly infrequent, but both thundersqualls and tropical storms occasionally pass through the area (DOC, 1992d).

D.2.2.9 Lake Charles, LA

The city of Lake Charles, the seat of Calcasieu Parish, is located on the east side of the Lake. It is the center of large industries such as chemical, petroleum, natural gas, fish oil, synthetic rubber, salt, seafood, and rice. The Port of Lake Charles is situated 3 km (2 mi) south of the city on the east bank of the Calcasieu Lake, and is 52 km (32 mi) from the Gulf of Mexico (DOC, 1992a). A map of the port is shown in Figure D-40.

A Federal project provides for a channel 12.8 m (42 ft) deep across the outer bar, from 12.2 to 12.8 m (40 to 42 ft) through the jetties, and 12.2 m (40 ft) to the Port of Lake Charles.

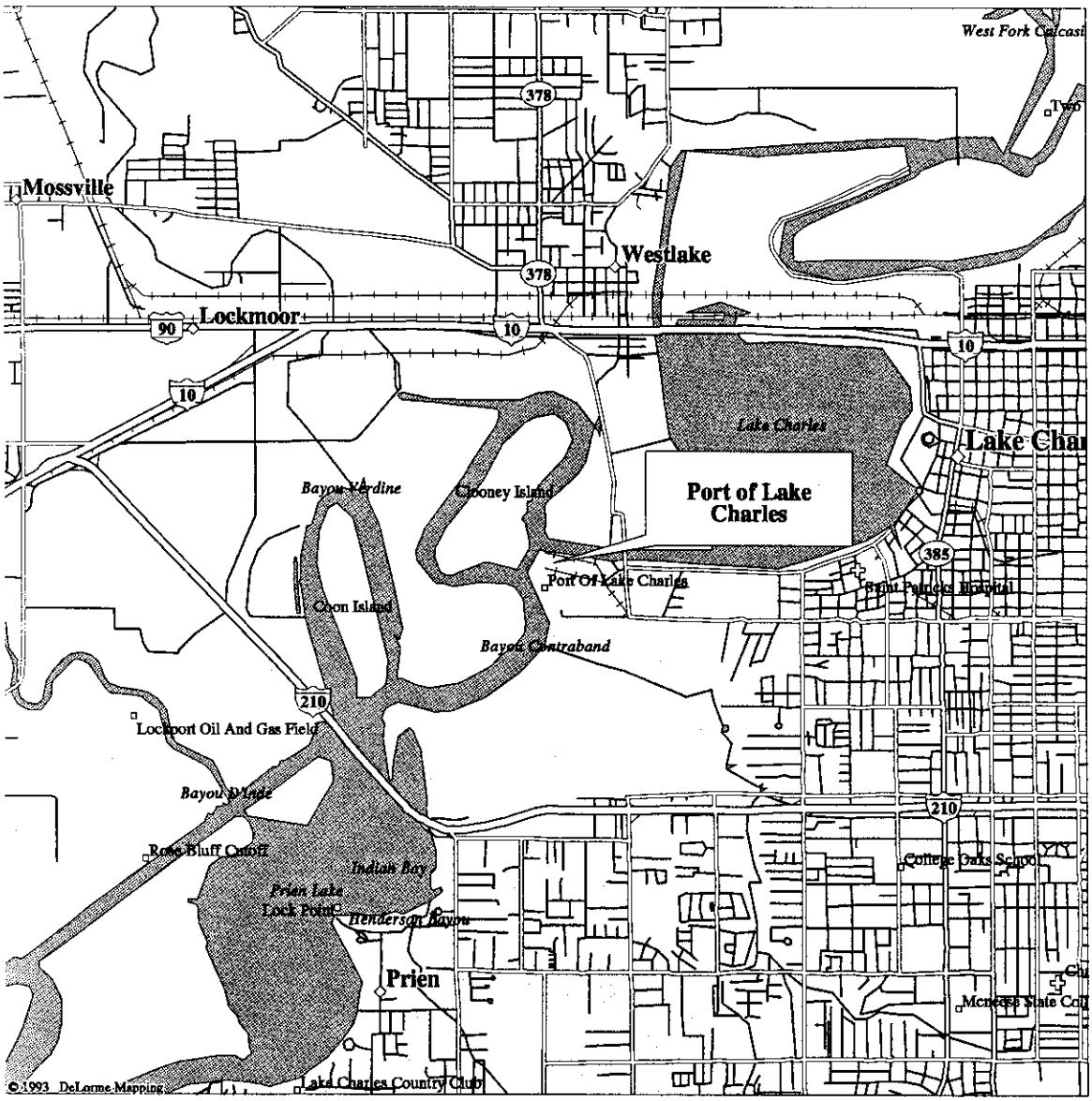
The United States Coast Pilot (DOC, 1992a) reports: "In recent years a substantial number of oceangoing vessels of increased size and draft have been entering the Calcasieu River Channel and proceeding to and from berths as far up the channel as the Port of Lake Charles. The channel, however, has not been appreciably widened in recent years. Based upon reported marine casualties to vessels and upon reported navigational problems arising from the increased oceangoing traffic, and after consultation with local marine interests, the Coast Guard Captain of the Port (COTP) has developed certain guidelines to enhance safe navigation."

The longest berth in the terminal is 274 m (900 ft). Lake Charles has no international container carriers serving the port and serves primarily as a breakbulk, dry bulk, and project cargo niche port (AAPA, 1993 and 1994; Southern Shipper, 1993). It can handle limited container traffic on breakbulk vessels (about 30,000 20-ft equivalent units in 1992) (Southern Shipper, 1993). Most of the area around Calcasieu Lake is wetlands, and ships entering the port pass by the Sabine National Wildlife Refuge.

Like all Gulf Coast ports, it is subject to severe hurricanes and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Lake Charles, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

Lake Charles, LA's climatic and environmental conditions are similar to those of the Port of New Orleans, LA. Port of New Orleans information is presented in Section D.2.2.14.

The 1990 census population estimate for this port vicinity was approximately 73,800 with a population density on the order of 940 persons/km^2 (2,434 persons/mi^2). The approximate distances to the five potential sites on interstate routes are: Savannah River Site, 1,100 km (700 mi); Oak Ridge Reservation, 960 km (600 mi); Idaho National Engineering Laboratory, 3,400 km (2,100 mi); Hanford Site, 4,000 km (2,500 mi); and Nevada Test Site, 3,200 km (2,000 mi). Distances along rail routes are about the same.



LEGEND

- Population Center
- State Route
- Geo Feature
- Town, Small City
- Large City
- Hospital
- Interstate, Turnpike
- US Highway
- Hwy Ramps
- Street, Road
- Trails
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Land Mass
- Open Water
- Intermittent River

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 18:37:13 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-40 Map of the Port of Lake Charles, LA

D.2.2.10 Long Beach, CA

Long Beach and Los Angeles Harbors, although divided by a political boundary, form a single geographic and economic port entity. The harbors occupy a major part of San Pedro Bay. The Port of Long Beach, one of the largest ports on the Pacific Coast, has extensive foreign and domestic traffic with modern facilities for the largest vessels. Most of the channels in Long Beach Harbor are maintained at more than the project depth of 10.7 m (35 ft). The entrance to Middle Harbor is 3.5 km (2.2 mi) from the Queens Gate entrance at the Pacific Ocean. The channel from the Pacific Ocean is straight, short, and direct (DOC, 1992b). A map of the port is shown in Figure D-41.

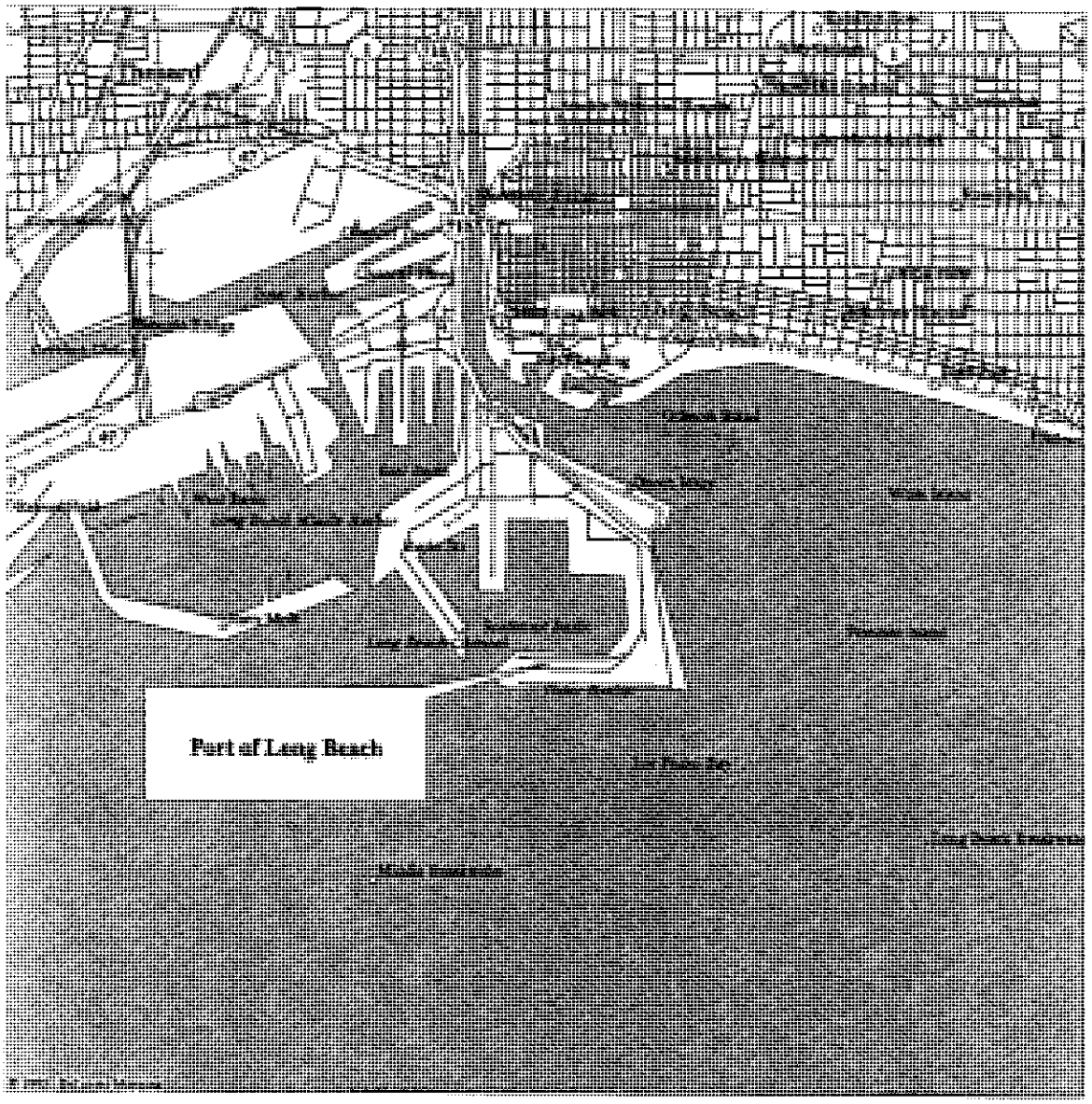
The Long Beach Harbor Department is a semi-autonomous agency of the City of Long Beach, CA. The Department is responsible for the operation, control, and development of the municipally owned port facilities. Long Beach is a large port (a load center) with 1,040 ha (2,816 acres) of land area, 12 piers, and 77 operational berths serving about 5,700 vessels annually. The port handles about 75 million metric tons (83 million tons) of revenue cargo annually, of which approximately 35 million metric tons (39 million tons) is containerized general cargo equivalent to 1.8 million 20-ft equivalent units (POLB, 1993a-d; AAPA, 1993).

Long Beach is a multi-terminal port and is host to seven container terminals with 38 container cranes and 243 ha (600 acres) devoted to container handling facilities. Additionally, there are facilities for petroleum and petroleum-related products, dry bulk materials, automobiles, steel, citrus, palletized general cargoes, and other commodities. The port functions as a "landlord" port leasing out its facilities to terminal and ship operating companies. Two of the container terminals (California United Terminals - Pier E, and Pacific Container Terminal - Pier J) are operated as "public" facilities. California United Terminals also has two roll-on/roll-off ramps and rail spurs (POLB, 1993a-d; AAPA, 1993; Janes's, 1992).

The dock/quay length available for cargo ships is as follows: California United, Pier E, Berths E24-E26 — 594 m (1,950 ft) long, and Pacific, Pier J, Berths J245-J247 — 1,006 m (3,300 ft) long. The corresponding depths alongside at mean low water are: California United with 14-15.2 m (46 to 50 ft), and Pacific Container with 14.9 m (49 ft). The five cranes at California United are all 40 metric ton (44 ton) container cranes. Pacific Container has six, 40 metric ton container cranes (Jane's 1992; AAPA, 1993; POLB, 1993a-d).

California United Terminals is served by an 11 lane main gate, which appears to be about 0.8 km (0.5 mi) from the "on" ramp to I-710 (the Long Beach Freeway), all within the confines of the port area. Pacific Container Terminal has similar ease of access to I-710, estimated to be a distance of about 1.9 km (1.2 mi), also within the port terminal. California United Terminals has shipside rail service provided by the Harbor Belt Line Railroad. Plans are to extend Belt Line rail service to Pacific Container Terminal by April 1994. The line connects with the major rail systems serving the Greater Los Angeles/Long Beach areas such as the Union Pacific, Santa Fe, and Southern Pacific Railroads. The latter operates a 97 ha (240 acre) intermodal container transfer facility which was built by the POLB to serve the marine terminals of both Long Beach and Los Angeles. The terminals are about 6.4 km (4.0 mi) from the double stack intermodal container transfer facility yard. The Santa Fe and Union Pacific railroads offer similar intermodal transfer facilities at their respective yards in east Los Angeles (D&B, 1993; Jane's, 1992; AAPA, 1993).

The port is served by a number of the world's largest container ship lines including: ACL, BHP/MTL, CCNI, COSCO, CGM, Cho Yang, Cool Carriers, DSR-Senator Line, EAC, Hanjin, Hapag-Lloyd, Hyundai, K-Line, Maersk Line, Nedlloyd Lines, OOCL, Philippine National Line, P & O, Sea-Land and TMM (Jane's, 1992; D&B, 1993; AAPA, 1993).



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- Hospital
- Park
- Interstate, Turnpike
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- Railroad
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:25:35 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-41 Map of the Port of Long Beach, CA

Other Pertinent Information: Terminal operators are responsible for the security of their respective facilities. Container terminals are fenced with controlled access and the security forces are port employees (Powell et al., 1994). The port contracts with the City of Long Beach for police and fire protection services. The City of Long Beach stations two fireboats within the port area. There are locations within the terminals for temporary storage of hazardous materials (Hilliard, 1993) but no special areas set aside (Powell et al., 1994).

There are no known environmentally sensitive areas within the harbor area. However, the port claims a long-term interest in maintaining a high quality environment and supports a number of programs to prevent contamination of air and harbor water quality. It was the first recipient of the American Association of Port Authorities Environmental Improvement and Protection Award, and enforces strict safety policies as well. "In the past 50 years, there have been no collisions between commercial vessels resulting in injuries . . . and no significant oil spills from oil transfers." (POLB, 1993b).

The port Marketing Manager did not know of any regulation prohibiting the handling of spent nuclear fuel (Hilliard, 1993). According to available data, the port has not handled spent nuclear fuel since at least 1979 (NRC, 1993; SNL, 1994). The Port of Long Beach does handle other hazardous cargoes and has a number of deep-draft petroleum and petrochemical terminals with attendant tanker traffic, including very large crude carriers. There appears to be good separation between these terminals and the two public container terminals at Pier E and Pier J (POLB, 1993a-d; Jane's, 1992).

Terminal operators contract with private hazardous materials response organizations to contain and control hazardous materials incidents on their premises. The Coast Guard and the Long Beach Fire Department's hazardous materials team are also used for emergency response. Hazardous materials training within the port is the responsibility of the port's Security Division. Port employees receive first responder training for hazardous cargo accidents (no Department of Transportation training), but the Fire Department is the responder for all port accidents (Powell et al., 1994). The Fire Department also calls on the county hazardous materials team as needed (Powell et al., 1994).

The port is physically separated from downtown Long Beach and has excellent highway connections. However, truck and/or rail passage from the terminals must pass through the heart of the adjoining communities that are large and densely populated, which makes the port less than ideal for spent nuclear fuel shipments.

The port is subject to severe earthquakes. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These requirements are shown in the Uniform Building Code (UBC, 1991). For the Port of Long Beach, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located in a very high seismic zone with an acceleration of 0.40 g (the highest Uniform Building Code ranking). Nearby San Fernando, CA was the site for one of the worst recorded earthquakes in the contiguous United States with a Modified Mercalli Intensity XI, in February 1971 (Bolt, 1978). Numerous other serious earthquakes with Intensities ranging from IX to X have also occurred in the last century. Long Beach was the site for a Intensity IX earthquake on March 10, 1933, which also resulted in numerous deaths, injuries, and building damage (Bolt, 1978).

The 1990 population within 16 km (10 mi) of the port terminals was 1,014,418. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 914,00; Oak Ridge Reservation, 823,000; Idaho National Engineering Laboratory, 692,000; Hanford Site, 617,000; and Nevada Test Site, 518,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-7 in Section D.1. The distances to

the five potential sites on interstate routes are: Savannah River Site, 3,940 km (2,443 mi); Oak Ridge Reservation, 3,610 km (2,246 mi); Idaho National Engineering Laboratory, 1,580 km (979 mi); Hanford Site, 2,000 km (1,241 mi); and Nevada Test Site, 645 km (401 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The environmental conditions for Long Beach are the same as those for Los Angeles. These are reported in Section D.2.2.11 below.

Climatic Conditions

Similar to the Los Angeles area, the climate of Long Beach, CA, is influenced significantly by the local topography. The Pacific Ocean has a moderating effect on the diurnal temperature range, which is greater than that observed further inland at the Los Angeles International Airport. In general, winter months are cool and wet followed by warm, dry summer months. Early morning clouds and fog, which are quite common during the late evening and early morning hours, generally burn off by late morning, resulting in sunny, pleasant daytime conditions during summer (NOAA, 1993f).

D.2.2.11 Los Angeles, CA

Los Angeles and Long Beach Harbors, although divided by a political boundary, form a single geographic and economic port entity. The harbors occupy a major part of San Pedro Bay. The Port of Los Angeles, one of the largest ports on the Pacific Coast, has a history of leading the Pacific Coast ports in terms of tonnage handled. It has extensive facilities to accommodate all types of traffic, and is the only southern California port at which passenger vessels call regularly (POLA, 1994).

The channel from the Pacific Ocean is straight, short, and direct. The Los Angeles Main Channel is maintained at 13.7 m (45 ft). The Super Tanker Channel to the deep draft facilities is maintained at 12.2 m (40 ft). The majority of the port facilities are located within 4.8 km (3 mi) of the harbor entrance (DOC, 1992b). A map of the port is shown in Figure D-42.

Worldport LA, the name adopted by the Los Angeles Harbor Department for the Port of Los Angeles, is a proprietary and self-supporting department of the City of Los Angeles reporting to a Board of Harbor Commissioners. The Worldport LA functions as a landlord operator administering its own budget, operations, and development programs (POLA, 1994).

Worldport LA is one of the country's largest, multi-terminal ports, and claims the title of the busiest container port in the United States. In fiscal year 1992, Worldport LA handled 2,154,890 20-ft equivalent units — the highest volume in the port's history. The port is also a cruise ship terminus handling over three-quarter million passengers in 1992 (AAPA, 1993; POLA, 1994).

Worldport LA encompasses approximately 1,684 ha (4,160 acres) of land area and 1,425 ha (3,520 acres) of sheltered waters. It has 36 cargo handling terminals, including six dedicated container terminals and three "Omni" terminals (which handle containers and breakbulk) with a total of 34 container cranes, on 45 km (28 mi) of waterfront. Of the three Omni terminals, Berths 142-146 (operated by Worldport LA) is a public facility with no "tenant" ship lines. The remaining two Omni terminals, RDP and Indies Terminals, are managed by private terminal operating companies but are open to public use (Jane's, 1992; AAPA, 1993; D&B, 1993). A multi-billion dollar, outer harbor Pier 300 development is underway with completion scheduled by the year 2010 (some terminals may be open during the period analyzed in this EIS) (POLA, 1994).

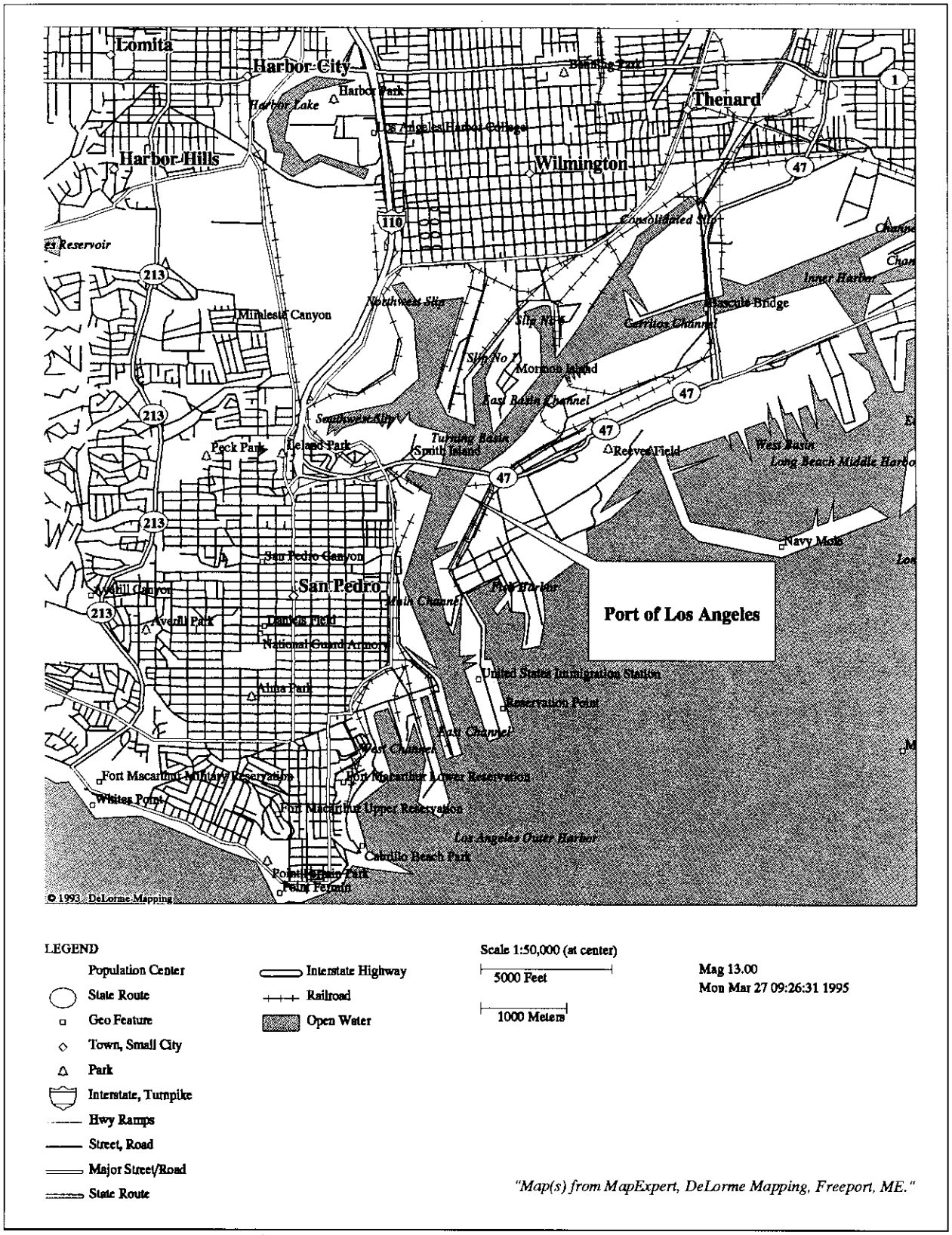


Figure D-42 Map of the Port of Los Angeles, CA

Worldport LA Berths 142-146: Dock/Quay lengths are 853 m (2,800 ft) and depths alongside at mean low water are 10.67-11.3 m (35-37 ft) at mean low water. Worldport LA berths have crane capacities of three 40.6 metric ton (45 ton) container cranes (Jane's, 1992; AAPA, 1993).

RDP Terminal: Berths 174-181 have lengths of 1,006 m (3,300 ft) and depths of 10.67 m (35 ft) at mean low water. Capacity of RDP cranes is two 40.6 metric ton container cranes (Jane's, 1992; AAPA, 1993).

Indies Terminal: Berths 216-227 have total lengths of 1,128 m (3,700 ft) and depths of 13.72 m (45 ft) at mean low water. The Indies berths have three 40.6 metric ton (45 ton) container cranes (Jane's, 1992; AAPA, 1993).

Los Angeles terminals are served by the Harbor Freeway (I-110) and Terminal Island Freeway (Route 47) which connect with Interstate Highways 5, 10, 15, and 40. The Harbor Freeway begins within the Worldport LA port complex. Worldport LA is connected to the Southern Pacific Transportation Co., Union Pacific, and Santa Fe railroads by the Harbor Belt Line Railroad, jointly owned by the Los Angeles Harbor Department and the three railroads. Belt Line tracks extend to cargo ship berths at each of the Omni Terminals. Intermodal connections are presently made at the intermodal container transfer facility described for the Port of Long Beach, which is approximately 8 km (5 mi) away. A new intermodal container transfer facility is under construction on Terminal Island and there are major infrastructure improvement projects underway to facilitate and expedite rail and truck traffic to the port through the Greater Los Angeles Metropolitan area (POLA,1994).

Worldport LA is host to more than a dozen cruise ship lines and about 40 cargo ship lines. A partial list of container lines calling at the port include: American President Lines, Australia-New Zealand Direct Line, Orient Overseas Container Line, Philippines, Micronesia & Orient Lines, Yang Ming Line, Mitsui O.S.K. Lines, Kawasaki Kisen Kaisha ("K" Line), Dole Fresh Fruit, Columbus Line, Blue Star PACE Ltd., Matson Navigation Co, NYK Line, Neptune Orient Lines, Evergreen Line, Barber Wilhelmsen, Blue Star Line, d'Amico Line, Italia Line, Nedlloyd Lines, and Slosna Plovba (Jane's, 1992; D&B, 1993).

Other Pertinent Information: The Port of Los Angeles has its own police force that patrols the waterfront around the clock by boat, helicopter, automobile, and bicycle (Leong, 1993). Port security is extensive and extremely well-organized. The port police are responsible for the safety and security of all passenger, cargo, and vessel operations at Worldport LA. They also monitor vessel berthings for possible wharf damage and issue hazardous cargo and dangerous goods permits. In addition, terminal operators have their own unarmed security personnel. All terminals also have areas for segregation and temporary storage of dangerous cargoes (Verhoef et al., 1994).

Fire protection is provided by the Los Angeles Fire Department which maintains five fire stations within the port and operates five fireboats. Two additional fireboat berths and stations are under construction. Response time is within five minutes. First responders for accidents receive Occupational Health and Safety Administration training but do not yet receive Department of Transportation training (Leong, 1993; Verhoef et al., 1994).

A port spokesperson did not know of any ordinances prohibiting the importation of spent nuclear fuel (Leong, 1993). A port spokesperson thought the port had handled spent nuclear fuel shipments in the past (Note: Database searches of shipments over the last decade do not show Los Angeles as a port for receipt of spent nuclear fuel; presumably these past shipments were other types of radioactive materials) (Leong, 1993). Item 1715 of Los Angeles Port Tariff No. 4, effective July 1, 1990, provides for the handling of radioactive and/or fissile materials, provided special written permission is received from the Executive

Director and U.S. Department of Transportation/Coast Guard Regulations are fully complied with (POLA, 1994). However, a spokesperson indicated that it was unlikely the port would accept spent nuclear fuel shipments (Verhoef et al., 1994).

The port police are the primary responders to hazardous materials incidents, backed up by the Los Angeles Fire Department and the United States Coast Guard. Based on Tariff Item 1715 referred to above, and the fact that radioactive shipments have occurred in the past, it is assumed port police have an adequate handling plan in place for radioactive materials. Worldport LA is an active participant in the Shoreline Emergency Network, a regional oil spill network organized to respond to coastal oil spill emergencies. The port police are trained in hazardous materials handling and are in charge of such operations. It is not known to what extent individual terminal operators are trained in hazardous materials response, but given the size and complexity of the port activities, it is assumed adequate hazardous materials training is provided. The combined ship accident history for the Ports of Long Beach and Los Angeles for the period of 1991-1993 is the lowest of all the major west coast ports (USCG, 1994b).

Worldport LA has a number of environmental programs underway that are designed to mitigate damage done to the marine environment in the past, and to prevent or lessen additional negative environmental impacts in the future. The port has a very active recreational/tourist component and, due in part to the presence of oil production facilities within the port, there is heightened environmental sensitivity on the part of the port community. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Los Angeles, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located in a very high seismic zone with an acceleration of 0.45 g. Like most Southern California cities, the port is subject to severe earthquakes. Two relatively recent severe earthquakes in Southern California (along the San Andreas fault system along the Pacific and North American tectonic plates) occurred March 10, 1993, in Long Beach (Modified Mercalli Intensity IX) and February 9, 1971, in nearby San Fernando (Modified Mercalli Intensity VIII-XI). Both resulted in numerous deaths and injuries and caused massive structural damage to buildings.

The 1990 population within 16 km (10 mi) of the port terminals was 1,124,493. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 936,000; Oak Ridge Reservation, 639,000; Idaho National Engineering Laboratory, 519,000; Hanford Site, 725,000; and Nevada Test Site, 334,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 3,930 km (2,440 mi); Oak Ridge Reservation, 3,550 km (2,204 mi); Idaho National Engineering Laboratory, 1,510 km (940 mi); Hanford Site, 2,070 km (1,286 mi); and Nevada Test Site, 580 km (359 mi). Distances along rail routes are slightly longer.

Environmental Conditions

Several wildlife refuges are located around the San Pedro Bay area. The Seal Beach National Wildlife Refuge is approximately 16 km (10 mi) east from the port area. The Bolsa Chica Ecological Reserve is located about 20 km (11 mi) east from the port area. The Abalone Cove Ecological Reserve is about 16 km (10 mi) around Point Fermin to the west of the port area. Several areas of biological significance lie along the passageway to the ports. These include the Channel Islands National Marine Sanctuary, about 60 km (43 mi) to the southwest of the port entrance, and the Santa Catalina Island Area of Special Biological Significance, about 28 km (17 mi) southwest of the port entrance (FWS, 1981c).

The water quality of the harbor waters at the ports is generally considered good. Areas in the inner harbor with limited waterflow experience infrequent periods of poor water quality. There is a wider range of salinity in the inner harbor than in the outer harbor, with higher values at the bottom than at the surface (U.S. Army, 1990).

The waters of the Los Angeles-Long Beach Harbor contain a variety of marine habitats, some natural and some manmade. Numerous fish species use the habitats in the harbor, including several recreational (e.g., barred sand bass and white croaker) and commercial (e.g., anchovy and halibut) value, for all or part of their life cycle (U.S. Army, 1990). Commercial fishing operations for crabs and spiny lobsters also are in San Pedro Bay. Other sport-fishing in the bay includes flatfish, grunions, California halibut, white seabass, kelp bass, Pacific bonito, and Pacific barracuda (FWS, 1981c). Shallow waters are important nursery areas for several fish species. At least 60 species of water-associated birds use the harbor, primarily for resting and foraging (U.S. Army, 1990).

Several threatened or endangered species are present at least seasonally in San Pedro Bay (Kobetich, 1994; U.S. Army, 1990). The endangered California least tern breeds in the area from April through August. California brown pelicans are present all year feeding on the fish in the harbor and resting on the breakwaters and other structures. Peregrine falcons are present in the region but are seldom sighted in the harbor. Other endangered birds around the ports include the light-footed clapper rail and the marbled morrelat. The western snowy plover (threatened) and the long-billed curlew, which is a candidate species, have been spotted infrequently in the harbor. Other candidate species, including the elegant tern, harlequin duck, loggerhead shrike, reddish egret, and white-faced ibis, can be found in the harbor area.

Within the Seal Beach National Wildlife Refuge, the wetlands portion supports a wide variety of fish and invertebrate as well as residential and migratory bird populations. The bay provides habitat for the light-footed clapper rail, Belding's savannah sparrow, the California least tern and the California brown pelican, all of which are endangered (U.S. Army, 1990). The Bolsa Chica Ecological Reserve provides habitat for the California least tern, the light-footed clapper rail, the California brown pelican, Belding's savannah sparrow (State protected), and the salt marsh bird's-beak, a member of the figwort family. The Belding's savannah sparrow is strictly associated with pickleweed, which is not found within the ports; therefore this species is not expected to be found directly in the ports. The reserve is also used by the coast horned lizard, monarch butterfly, snowy plover, and numerous bird species such as gulls, terns, sandpipers, herons, and egrets.

With regard to marine mammals, no species of cetaceans (whales, dolphins) actually inhabit regions in-shore of the breakwater, and their occurrence within the harbor is sporadic and infrequent. Visitors include the common dolphin, the Pacific white-sided dolphin, and gray whale (endangered). Groups of bottlenose dolphins have been observed swimming just outside the breakwater. The eastern Pacific gray whale migrates through California waters twice yearly in a route between the Bering Sea and Baja California. The southward migration occurs between November and February, while the northward return generally takes place off of California between March and May. While the gray whales usually stay outside the harbor mouth, approximately three to four accidently enter the harbor every year. The California sea lion and the harbor seal, both nonendangered, have been sighted in the area of the harbor. The California sea lion is known to occasionally haul-out on the harbor breakwater and sometimes can be seen swimming in the harbor. The harbor is not considered a birthing or important feeding habitat for the California sea lion, although sea lions could presumably forage within the harbor (U.S. Army, 1990).

Climatic Conditions

The dominant geographic influences on the climate of the Los Angeles basin are the Pacific Ocean and the southern California coastal mountain ranges. Marine air covers the coastal plain for the majority of the year, but inland air does occasionally migrate into the region. Pronounced differences in temperature, humidity, cloudiness, fog, sunshine, and rain occur over fairly short distances along the coastal plain due to the local topography and the decreased effect of the marine environment further inland. However, in general, temperature ranges are least and humidity highest close to the coast, while precipitation increases with elevation in the foothills. Prevailing daytime winds are from the west, with nighttime and early morning winds generally light and from the east and northeast. During the fall, winter and spring months, dry, gusty northeasterly winds (e.g., Santa Ana winds) blow over the southern California mountains. Precipitation occurs mainly during the winter months. Thunderstorms are rare along the coast, but increase in frequency as one approaches the coastal ranges. Fog and low visibility are frequent problems for aircraft navigation at the Los Angeles International Airport (NOAA, 1993e).

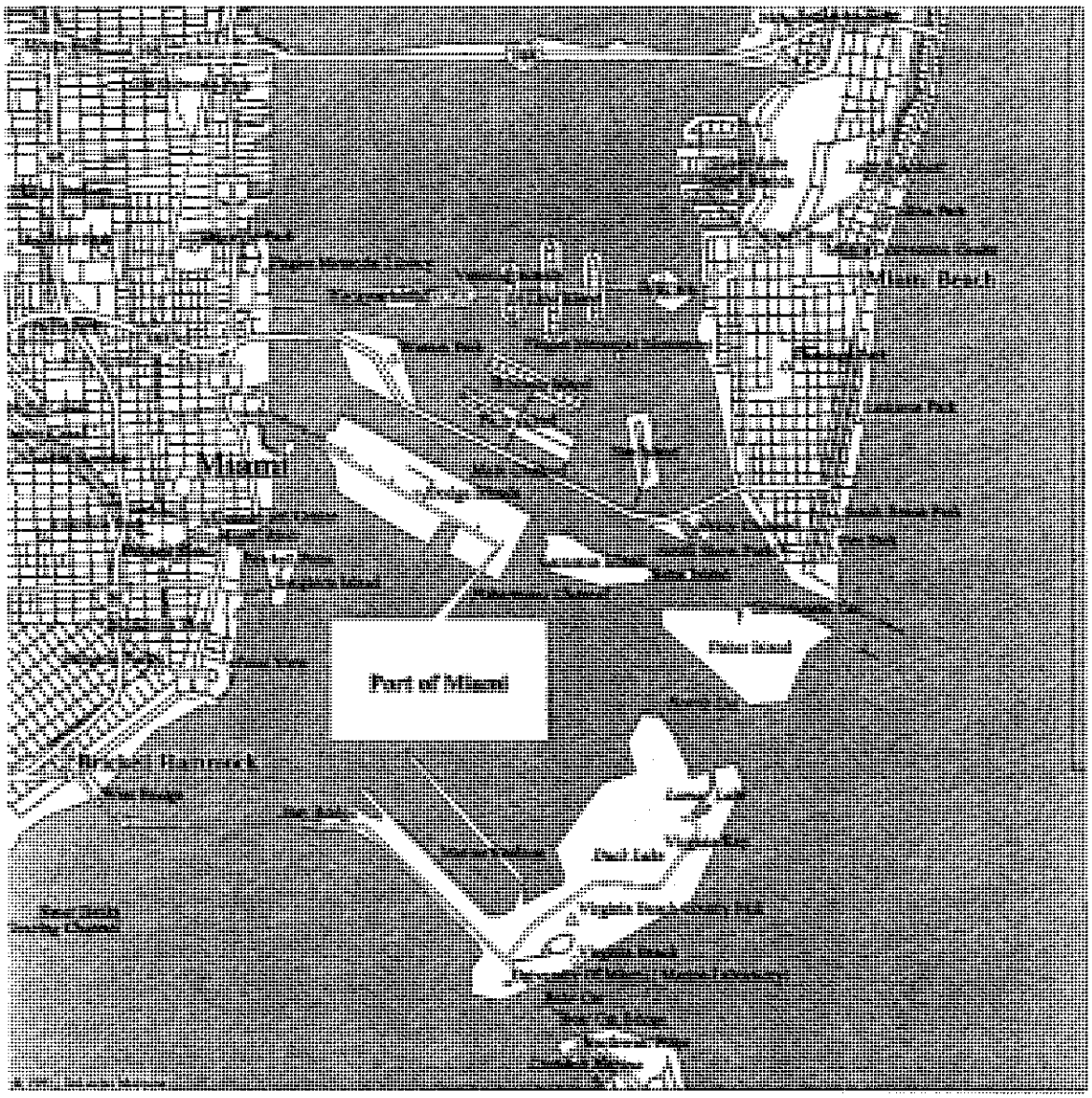
D.2.2.12 Miami, FL

Miami is Florida's most populous city and is located 8 km (5 mi) from the Gulf Stream on the east coast of Florida. It is an internationally famous winter resort and a popular yachting center. Miami is also a deepwater port; considerable foreign commerce passes through Miami and it is a major cruise port. Miami's cruise ship traffic has earned it the title of "Cruise Ship Capital of the World" (Southern Shipper, 1993). In addition to being a major shipping and cruise ship center, the Port of Miami is located in a popular resort area known for its beaches, fishing, recreational boating, and tropical landscape. The approach to Miami is open, but with strong tidal currents of 1.0 to 2.1 meters-per-sec (2 to 4 knots) in the entrance between the jetties. A Federal project provides for depths of 11 m (36 ft) to the main port facilities (DOC, 1993d; Southern Shipper, 1993; AAPA, 1993; Jane's, 1992). The port occupies 273 ha (675 acres) of land. It is situated on two interconnected islands, Dodge and Lummus, which lie in an east-west orientation due east of the City of Miami and west of the barrier island resort area of Miami Beach. The Miami Beach resort area forms the northern boundary of the harbor entrance. The major port facilities are within 5 km (3 mi) of the entrance from the Atlantic Ocean. A map of the port is shown in Figure D-43.

Miami's freight terminals serve as a hub for distribution and transshipment of cargo (largely tropical fruits and vegetables) to and from Latin America. The Port of Miami is an arm of the Dade County Seaport Department which functions as a "landlord" port. Almost 60 shiplines connect the port to most major countries in the world (Jane's, 1992; D&B, 1993; Southern Shipper, 1993). In 1994, nearly 520,000 20-ft equivalent-units were handled in the port (AAPA, 1994).

Lummus Island Terminal: The 91 ha (225 acre) terminal on the south side of the island is seaward of Dodge Island and just inside the entrance to the port. It is Miami's principal container handling facility with six container gantry cranes, including three new post-Panamax cranes and a new roll-on/roll-off berth. Activities at Dodge Island are primarily cruise ship, roll-on/roll-off, and breakbulk cargo oriented. Combined facilities consist of four container berths, 14 roll-on/roll-off berths, and 12 cruise ship berths. A private container terminal for shallow draft vessels is located on Causeway Island at the eastern end of the MacArthur Causeway, which parallels the ship channel north of Lummus and Dodge Islands (Southern Shipper, 1993).

This terminal has marginal wharf area of 1,067 m (3,500 ft). The roll-on/roll-off berths have 413 m (1,356 ft) of marginal wharf while the Dodge Island breakbulk has 853 m (2,800 ft) of marginal wharf. The passenger terminals have 2,373 m (7,785 ft) of marginal wharf. Depths alongside Dock/Quay are



LEGEND

- Population Center
- State Route
- Geo Feature
- Park
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- US Highway
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

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"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-43 Map of the Port of Miami, FL

noted as: Lummus Container Berths 1 and 2 with 12.8 m (42 ft) at mean low water. Berths 3 and 4 have 11.73 m (38.5 ft) at mean low water. On the north side of Dodge Island, the depth is 7.62-10.97 m (25-36 ft) at mean low water.

Crane capacities at Lummus Container Berths consist of three 50.8 metric ton (56 ton) container gantry cranes and three 40.6 metric ton (45 ton) container gantry cranes.

The Port of Miami is accessible via a five-lane, fixed bridge spanning the Intracoastal Waterway. It is approximately 1.2 km (0.75 mi) from the Biscayne Boulevard exit of I-395 to Dodge Island via NE 2nd Avenue in downtown Miami. I-395 is a connector to I-95 as well as all other south Florida highways. There are 5.2 km (3.2 mi) of trackage within the Port of Miami including a four-track marshalling yard. Rail connections are with the Florida East Coast and CSX Railroads.

Port users include Agromar, Argentine Line, Barber Blue Sea, Bottachi Line, Central American Shippers, CCNI, CGM, Chilean Line, Ecuadorian Line, Empremar, Flota Mercante Grancolombiana, Hapag Lloyd, Hoegh Line, Italian Lines, Ivaran Lines, Kirk Line, Lykes Line, Maersk Line, Navieras De Puerto Rico, Shipping Corp of India Transnave, Mexican Line, Spanish Line, Wallenius Transroll, and Zim Container Service (Jane's, 1992; Southern Shipper, 1993; D&B, 1993).

Other Pertinent Information: Containers discharged at Lummus Terminal must travel down the center of both islands and past the extensive cruise line terminals located on the north side of Dodge Island. Although travel on city streets on the mainland is for a very short distance, it is through an urban area which is believed to be heavily developed. The port recently completed a new $1.8 million, eight-lane security gate and cargo control facility on Dodge Island to the east of the bridge. Each lane is equipped with Regiscope photographic clearance systems (Southern Shipper, 1993).

Port officials did not respond to a faxed questionnaire or telephone calls for information, and it is not known if there is a designated area for temporary storage of hazardous cargoes. The port has no prior experience handling spent nuclear fuel (NRC, 1993; SNL, 1994). However, the Port of Miami is primarily a general cargo, container, and cruise ship port with no petroleum berths or other terminals for handling hazardous or dangerous goods. Passenger operations are considered a conflicting use. Since port officials did not respond to requests for information regarding emergency response capabilities, it is not known whether hazardous materials or spent nuclear fuel training exists for port workers.

There are no known wildlife habitats or sanctuaries in the immediate area; however, there is a high-level of environmental sensitivity in this area. The port's physically separate island locations, strictly controlled access, and limited use of city thoroughfares are very desirable features. However, the port is in relatively close proximity to the heavily populated Miami Beach area adjoining the harbor entrance (Government Cut) and roughly 0.8 km (0.5 mi) from downtown Miami.

The likelihood of severe natural phenomena, such as high winds and earthquakes, are reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Miami, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a low seismic zone with an acceleration of less than 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 833,057. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 336,000; Oak Ridge Reservation, 443,000; Idaho National Engineering Laboratory, 845,000; Hanford Site, 894,000; and Nevada Test Site, 908,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances

to the five potential sites on interstate routes are: Savannah River Site, 1,200 km (748 mi); Oak Ridge Reservation, 1,460 km (906 mi); Idaho National Engineering Laboratory, 4,570 km (2,842 mi); Hanford Site, 5,240 km (3,258 mi); and Nevada Test Site, 4,740 km (2,945 mi). Distances along rail routes are slightly longer, except for Savannah River Site, which is slightly less.

Environmental Conditions

The State of Florida has classified Biscayne Bay near Port of Miami as a Class III water body. This classification indicates that the waters are suitable for recreation, and propagation and maintenance of a healthy, well balanced population of fish and wildlife (FL DEP, 1994). In addition, the State of Florida has classified the Biscayne Bay, where the Port of Miami is located, as an "Outstanding Florida Waterway." As previously noted, Outstanding Florida Waters are generally waters located within national parks, state parks, national seashores, marine sanctuaries, or aquatic preserves. Other waters located near Port Everglades that are designated as Outstanding Florida Waters include Biscayne Bay National Park and the Bill Bags State Recreation Area. These waterways are afforded special protection by State environmental regulations (FL DEP, 1994).

The Biscayne Bay, in the vicinity of the Port of Miami, is characterized as a high salinity estuarine habitat (generally greater than 20 parts per thousand). There are both commercial and recreational fish and invertebrates found near the port. These aquatic species include: stone crabs, shrimp, spiny lobster, sharks, sand seatrout, drum, kingfish, mullet, Florida pompano, bluefish, mackerel, tarpon, ladyfish, snapper, grouper, grunts, jewfish, snook, greater amberjack, crevalle jack, silver perch, blue runner, Atlantic dolphin, short-finned pilot whale, false killer whale, and pygmy sperm whale (FWS, 1982c).

Protected species found near the Port of Miami include the loggerhead sea turtle and the West Indian manatee (Richards, 1994). The U.S. Fish and Wildlife Service reported that the Port of Miami is located in designated critical habitat for the endangered west indian manatee (Johnson, 1995). In addition, the U.S. Fish and Wildlife Service reports that the following protected, listed marine species are known to occur in Dade County: atlantic hawksbill turtle (endangered), atlantic ridley turtle (endangered), atlantic loggerhead turtle (threatened), atlantic green turtle (endangered), leatherback turtle (endangered), american crocodile (endangered), and the american alligator (threatened/similar appearance). Protected bird species in Dade County include the bald eagle (endangered), cape sable seaside sparrow (endangered), ivory-billed woodpecker (endangered), kirtland's warbler (endangered), arctic peregrine falcon (threatened), wood stork (endangered), everglades snail kite (endangered), bachman's warbler (endangered), roseate tern (threatened), and the piping plover (threatened) (Johnson, 1995).

Wildlife refuges located near the port area are the Bill Baggs Cape Florida State Recreation Area and the Biscayne Bay Aquatic Preserve. They are both located within 20 km (12 mi) of the Port of Miami. Protected species found in these areas include the loggerhead sea turtle, the West Indian manatee, and the peregrine falcon. Birds of interest found in these areas are: the spotted breasted oriole, songbirds, fulvuous whistling duck, and various shorebirds (FWS, 1982c).

Climatic Conditions

The climate of the southeast Florida region is essentially subtropical marine, which features long, warm summers with abundant rainfall, generally followed by a mild, dry winter. The influence of the ocean and numerous bays is seen in the small diurnal temperature range (generally <10°) and the rapid warming of any cold air masses that invade this portion of the State. The predominant windflow is from the east-southeast, which generates conditions right at the coast that are often different than those encountered further inland, due to land-induced frictional effects. Hurricanes occasionally effect the area, with the

months of September and October exhibiting the highest frequencies. However, destructive tornadoes (not associated with tropical systems) are rare. Waterspouts are frequently spotted offshore during the summer months, but rarely cause any loss of life or property damage (NOAA, 1993b).

D.2.2.13 Military Ocean Terminal, Oakland, CA

The Military Ocean Terminal, Bay Area, is located in the Outer Harbor of the Port of Oakland, adjacent to the east entrance to the Oakland Bay Bridge (descriptions of Oakland ship channels also apply to Military Ocean Terminal, Bay Area and are not repeated here). The facility is located approximately 16 km (10 mi) east of the Golden Gate Bridge, which spans the Pacific Ocean entrance to San Francisco Bay to the south and San Pablo Bay to the north. The single pier (Wharf 7) currently available for military cargo is directly opposite the commercial Sea-Land and Public Container Terminals, and located within the Oakland Army Base (MTMCTEA, 1990). The facility has the largest sealift workload of any military traffic ports on the West Coast, averaging on the order of 3,000 metric tons (3,300 tons) of cargo per year (the 1994 shipments of Patriot missiles to South Korea were shipped from Wharf 7). See the descriptions of the Port of Oakland for more information regarding truck and rail access, maps, populations, etc. A map of the terminal is shown in Figure D-44.

The Bay Bridge Terminal, adjacent to Military Ocean Terminal, Bay Area, operates Military Ocean Terminal, Bay Area wharves 6 and 6.5 as Berths 8 and 9 for its commercial operation (Jane's, 1992; MTMCTEA, 1990). Wharf 7 is 445 m (1,459 ft) long, with 10.6 m (35 ft) depth alongside. Wharf 7 has a single 91 metric ton gantry crane for all breakbulk operations and a container spreader that can be attached for limited container handling (MTMCTEA, 1990). A floating crane of comparable capacity is also available. Stern loading roll-on/roll-off operations are not feasible at the wharf.

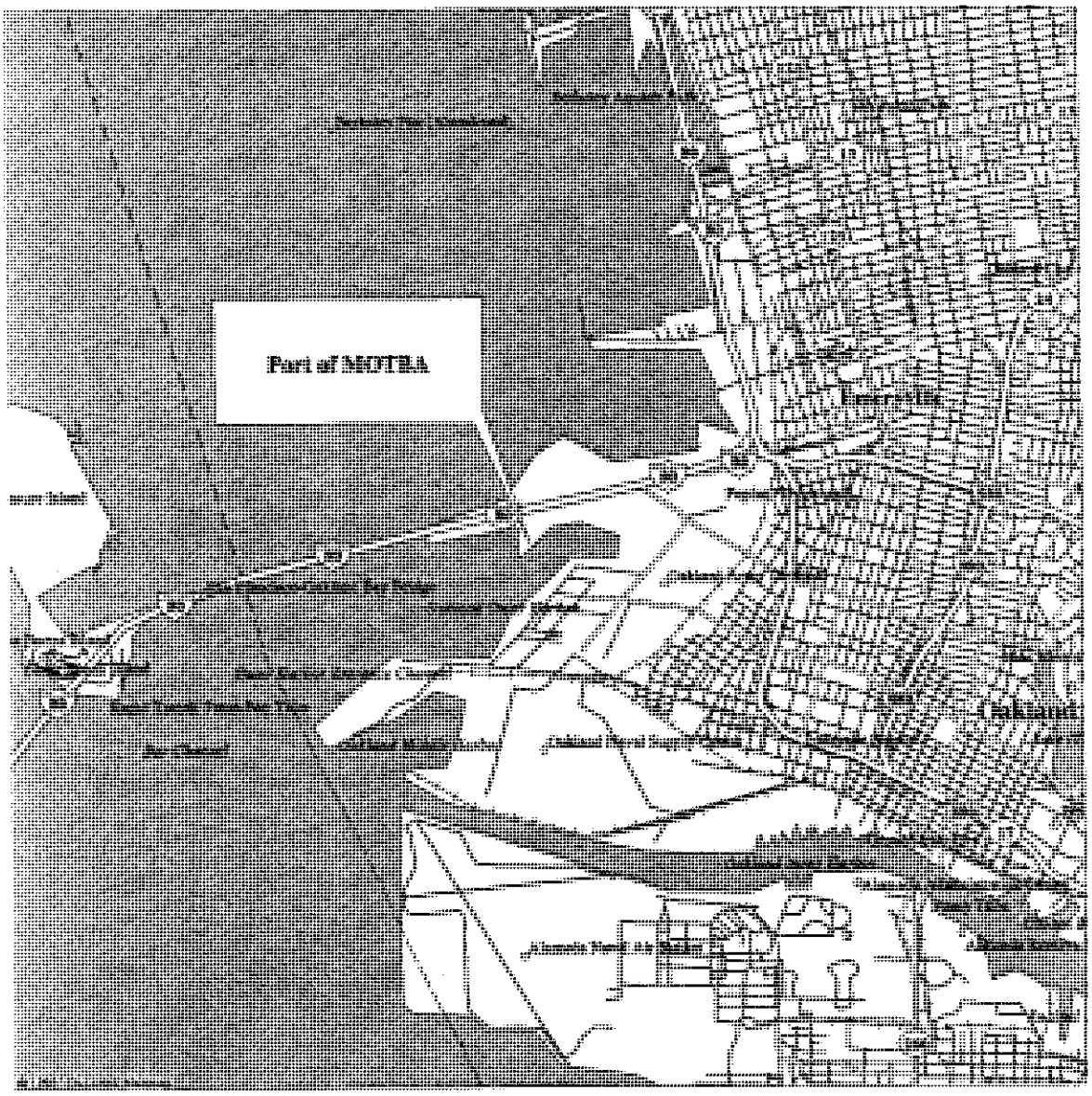
There are more than 8.1 ha (20 acres) of open storage space near the wharf, and a transit shed at the wharf provides more than 13,000 m^2 (141,000 ft^2) of covered storage. More than 65,000 m^2 (700,000 ft^2) of additional covered space is available on the Army Base (MTMCTEA, 1990).

Trucks can access the wharf for direct loading from ships at the facility. Access to Interstates 580, 680, or 880 is directly adjacent the Army Base through a largely industrial area at the Port of Oakland. Residential areas are within a few kilometers of the Base and the Port of Oakland.

The entire length of Wharf 7 is served by rail, making direct ship-to-rail loading possible for receipt of incoming cargo. Rail movements are carried out by two Base locomotives, which can move rail shipment to the adjacent and expanding Oakland Intermodal Terminal. The Intermodal Terminal is serviced by the Southern Pacific and Union Pacific rail systems and connections with the Atchison, Topeka, and Santa Fe Railroad intermodal yard about 19 km (12 mi) north of the port (MTMCTEA, 1990).

Other Pertinent Information: Since the facility is part of the Oakland Army Base, it is well lighted, fenced, and patrolled by gate guards and roving patrols. There are no full time longshoremen at the facility, and trained, experienced longshoremen are hired from the large pool of stevedores (1,000) normally working at the port.

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Military Ocean Terminal, Bay Area, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located in a very high seismic zone with an acceleration of 0.40 g (see seismic information for the Port of Oakland for more details).



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- Interstate, Turnpike
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- Railroad
- River
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:29:06 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-44 Map of the Military Ocean Terminal, Oakland, CA

Area 1990 census population and density figures are 1,110,549 and 1,323 persons/km^2 (511 persons/mi^2), respectively. The 1990 population within 16 km (10 mi) of the port terminals was 1,288,899. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,080,000; Oak Ridge Reservation, 786,000; Idaho National Engineering Laboratory, 367,000; Hanford Site, 359,000; and Nevada Test Site, 482,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,216 km (2,791 mi); Oak Ridge Reservation, 4,121 km (2,563 mi); Idaho National Engineering Laboratory, 1,548 km (963 mi); Hanford Site, 1,407 km (875 mi); and Nevada Test Site, 1,156 km (719 mi). Distances along rail routes are slightly longer.

The Military Ocean Terminal, Bay Area, is located in the Outer Harbor of the Port of Oakland. Climatic and environmental conditions for Military Ocean Terminal, Bay Area are the same as those for the Port of Oakland. These are presented in Section D.2.2.15.

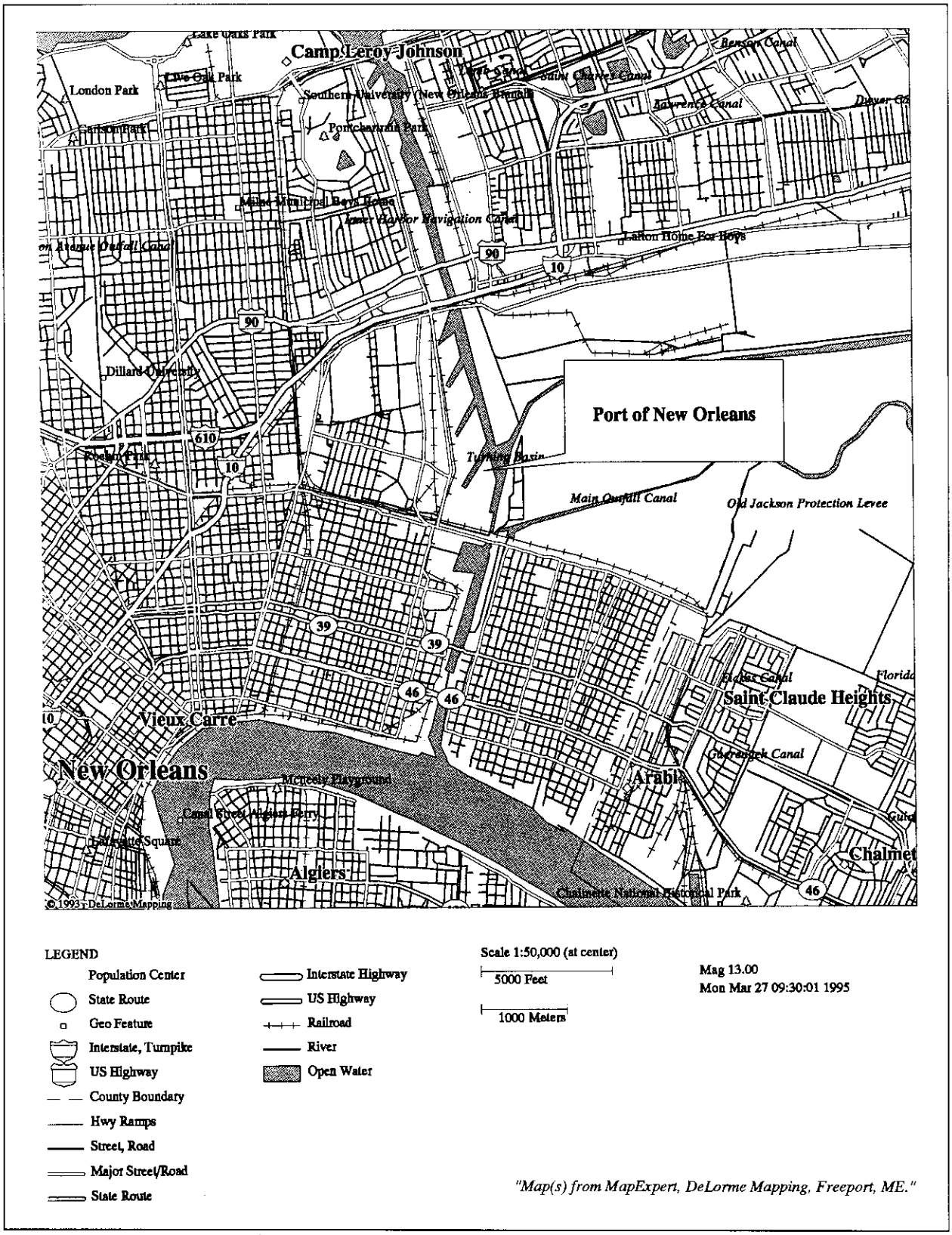
D.2.2.14 New Orleans, LA

The Port of New Orleans is one of the largest ports in the United States. It is located on both sides of the Mississippi River with its lower limit about 129 km (80 mi) above the Head of the Passes from the Gulf of Mexico, and its upper limit about 185 km (115 mi) above Head of the Passes. A Federal project provides for a channel 13.7 m (45 ft) deep over the bar through Southwest Pass to Head of the Passes, and on to New Orleans. The Port of New Orleans' lower limit is about 160 km (98 mi) from the Gulf of Mexico via Southwest Pass. Southwest Pass is straight and well-marked. From the Head of the Passes to New Orleans, the river has a least width of 550 m (1,800 ft) and a clear, unobstructed channel (DOC, 1992a). A map of the port is shown in Figure D-45.

The seven-person Board of Commissioners of the Port of New Orleans, is appointed by the Governor from a list of nominees drawn from industry, civic, and educational groups from the three parishes (counties) in which the Port of New Orleans' terminals are located. The Board, a state agency, sets policies and regulations for port operations. It also appoints the president and chief executive officer of the Port of New Orleans who, together with a staff of professional managers, are responsible for day-to-day operation of the port.

New Orleans is a multi-terminal port with predominantly publicly owned terminals and a few private terminals. The port is strictly a "landlord" operator, leasing all of its terminals to private operators and/or shipping companies. Most of the large publicly owned terminals are located along the banks of the Mississippi (on the New Orleans side of the River), which generally runs in an east-west direction in the vicinity of the City. In 1994, the port handled over 250,00 20-ft equivalent units of containerized cargo (AAPA, 1994).

France Road Container Terminal is the Port of New Orleans' principal container handling facility. It occupies 71.55 ha (177 acres) of land and is situated on the west bank of the industrial canal in the southwestern section of New Orleans at the intersection of the industrial canal with the Mississippi River/Gulf Outlet. Berths 1 and 4 are leased to Sea-Land and Navieras De Puerto Rico respectively, and Berths 5 and 6 are public terminals. Berth 1 has two 30.5 metric ton (34 ton) container cranes. Berths 5 and 6 are supported by three container cranes [one 30.5 metric ton (34 ton) container crane and two 40.64 metric ton (45 ton) container cranes], a marshalling yard of 195,077 m^2 (2.1 million ft^2), two container freight stations with 12,193 m^2 (131,120 ft^2) of consolidating space, and a roll-on/roll-off ramp



LEGEND

- Population Center
- State Route
- Geo Feature
- Interstate, Turnpike
- US Highway
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:30:01 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-45 Map of the Port of New Orleans, LA

at Berth 6. Berths 5 and 6 have a combined length of 518 m (1,700 ft) of marginal wharf with water depth alongside ranging from 9.75 - 10.97 m (32 - 36 ft) at mean low water (AAPA, 1993; Janes's, 1992; D&B, 1993; Southern Shipper, 1993; PON, 1994).

The France Road Public Container Terminal is located in a port industrial district that appears to be separate from residential areas and a considerable distance from the downtown New Orleans business district. This terminal is about 2.4 km (1.5 mi) from Interstate Highway 10 and U.S. Route 90 — major east-west highways — via Alvar Street or Florida Avenue, which are heavy truck routes. I-10 connects with I-49 to Shreveport, where it meets I-20. The Terminal has good truck and rail access, but waterway access is via the relatively narrow industrial canal with a lock near the entrance and several bridges en route. The city-owned Public Belt Railroad connects the France Road and other terminals on the Mississippi, Industrial Canal, and Mississippi/Gulf Outlet with the CSX, Illinois Central, Kansas City Southern, Norfolk Southern, Southern and Union Pacific Railroads. In the case of France Road Terminal, the Belt Railroad tracks serve the site, but not the pier apron.

The Port of New Orleans is port-of-call for over 50 steamship lines providing breakbulk and container freight service to virtually all of the world's major port cities. A partial list of these lines includes ABC Container Line, Argentine Line, ART Ocean Line, Atlantic Container Line, Baltic Shipping, Bank Line East Africa, Boss Line, China Ocean Shipping Co., Contship Container Line, Chilean Line, Daiichi-Chuo Shipping Line, Delmas/AAEL, Egyptian National, Forest Lines Inc., Hapag-Lloyd, Hoegh Line, Hyundai Merchant Marine, Industrial Maritime Carriers, Italia Line, Lykes Brothers Steamship Co., Maersk Inc., NCSCA, Pakistan National Shipping, Pan Ocean, Safbank, Sea-Land, Tecomar, Toko, Torm West Africa, Turkish Cargo Lines, United Arab Shipping, Waterman/LASH, Wilhelmsen Line, and Zim Container Line (Jane's, 1992; D&B, 1993).

Other Pertinent Information: The port has its own security force with police powers. The France Road Terminal is secured by fencing and controlled access. There are locations within the terminal for isolation of hazardous materials. The Port Harbor Police are the first line of defense with respect to hazardous materials accidents, followed by the Coast Guard and Louisiana State Police, who have primary responsibility for enforcing Department of Transportation Regulations. The port has an "elaborate" notification system in case of accidents on Port Authority terminals, beginning with the shipper or consignee of the goods. The New Orleans Fire Department also has a hazardous materials team. Hazardous materials training is the responsibility of the individual terminal operating companies. While the level of training at each terminal is uncertain, given the large quantities of hazardous materials passing through the port, some training is certain (Parker, Spalluto, and Cefalu, 1993).

Port officials know of no ordinances or regulations prohibiting the importation of spent nuclear fuel through the Port of New Orleans, and thought the port may have handled spent nuclear fuel in the past. However, other data indicate the port has not handled spent nuclear fuel since at least 1979 (NRC, 1993; SNL, 1994). The port spokesperson indicated that shipments of radioactive nuclear fuel (not spent nuclear fuel) have been shipped through the port and may still be coming in. Apparently these shipments were from South Africa (Parker, Spalluto, and Cefalu, 1993).

The Port of New Orleans is primarily a breakbulk and general cargo/container port. It is also a major river barge terminus for barge lines on the Mississippi River system. Although there is considerable tank ship and barge traffic on the River, tank terminals tend to be located on the opposite side of the river and/or outside the City limits. Conflicting use of the waterway is not considered a major factor with regard to handling spent nuclear fuel. However, a U.S. Coast Guard accident database established in 1990 shows that an extremely high number of accidents occur on the transit from the Gulf to the port (USCG, 1994a). During the period 1991 through the third quarter of 1993, there were 790 collisions, 825 allisions, and

1,065 hard groundings reported (see 46 CFR 4.05-1 for reporting requirements and definitions). The 2,680 accidents involved one of the following: vessel damage in excess of $25,000 and/or left the vessel unseaworthy, or without power or steering, or severe injury or death. The port 1993-1994 Annual Directory indicates that during this period, there were about 7,100 vessel transits (PON, 1994). Since the accident statistics include barge accidents (and New Orleans has large barge traffic), this number is rather high for oceangoing vessels, but no data are yet available yet to refine the information.

Other than flooding from severe hurricanes and tropical storms, and general environmental concerns, there are no known special environmental or wildlife issues in or near the port area. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of New Orleans, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 782,868. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 266,000; Oak Ridge Reservation, 256,000; Idaho National Engineering Laboratory, 455,000; Hanford Site, 504,000; and Nevada Test Site, 687,000. Populations along rail routes to these sites are slightly smaller for Idaho National Engineering Laboratory, Hanford Site, and Nevada Test Site and much larger for Savannah River Site and Oak Ridge Reservation. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,020 km (634 mi); Oak Ridge Reservation, 960 km (594 mi); Idaho National Engineering Laboratory, 3,510 km (2,184 mi); Hanford Site, 4,180 km (2,600 mi); and Nevada Test Site, 3,450 km (2,145 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The State of Louisiana has classified the waters of the Industrial Canal and the Mississippi River Gulf Outlet as suitable for primary and secondary water recreational activities and the propagation of fish and wildlife. The Mississippi River supports all of these uses in addition to being used as a drinking water supply source (Fabens, 1994).

The Mississippi River Gulf Outlet, in the vicinity of the France Road Terminal, is characterized as a high salinity estuarine habitat (generally greater than 20 parts per thousand). The entire canal travels through marshlands. Aquatic species found in these types of marshlands and surrounding areas in Louisiana include: shrimp, blue crab, eastern oyster seatrout, Atlantic croaker, drum, spot, kingfish, sheepshead, flounder, mullet, sea catfish, gulf menhaden, bay anchovy, crevalle jack, and Atlantic bottlenose dolphin (FWS, 1982d).

As ships approach the Mississippi River Gulf Outlet from the north they must travel past the Breton National Wildlife Refuge and Breton Wilderness. Birds of interest in these areas include: peregrine falcon, brown pelican, shorebirds, wading birds, herons, egrets, white ibis, least bittern, gallinules, waterfowl, bird hawks, osprey, magnificent frigate-bird, white pelican, songbirds, warblers and diving ducks. The peregrine falcon and brown pelican are protected species. Aquatic species found in these areas include: loggerhead sea turtle, spotted sea trout, drum, bluefish, cobia, and mackerel. The loggerhead sea turtle is a Federally protected species (FWS, 1982d). Travelling north into the Mississippi Gulf River Outlet towards the France Road Terminal ships must pass near the Biloxi Wildlife Management area.

Climatic Conditions

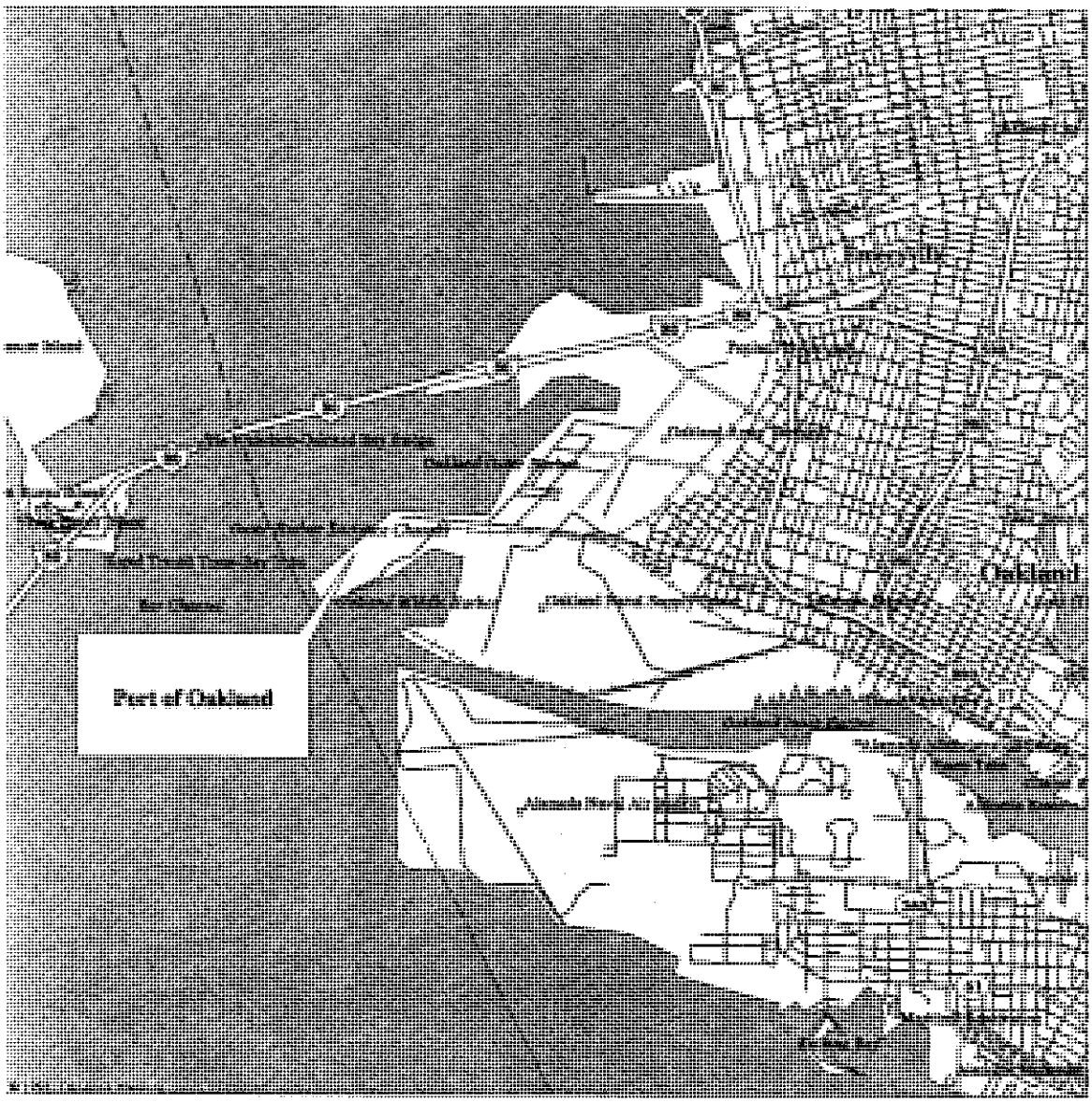
The city of New Orleans is essentially surrounded by water. Thus, the influence of the Gulf of Mexico and the surrounding bayous, lakes, and marshes are significant. The climate can best be described as humid, with the surrounding water significantly reducing the diurnal temperature range. Between mid-June and mid-September, almost daily, sporadic thunderstorms occur and prevent the temperature from rising much above 90°F. From mid-November through mid-March, the region is influenced alternatively by moist, tropical air masses from the south and from cold, dry continental air masses from the north. The general extratropical storm track is to the north of New Orleans but occasional systems do develop offshore of the city, causing sudden drops in temperature and an increase in precipitation. The cold Mississippi River water and the surrounding marsh areas increase the occurrence of fog in the late winter and early spring months, particularly when light southerly winds are advecting warm, moist tropical air over the area. A rainy period between mid-December through mid-March occurs annually, with the remaining fall/spring months (e.g., October/November, April/May) being relatively dry. The dominant rainfall event during the summer are thunderstorms. Severe thunderstorms with damaging winds are rare. However, the area is subject to the occasional landfalling hurricane. Waterspouts are common in the offshore area, but rarely cause property damage of loss of life (NOAA, 1992l).

D.2.2.15 Oakland, CA

Oakland, located on the eastern shore of the San Francisco Bay, is directly opposite San Francisco. It is the second largest port on the Bay and is a leading containership terminal on the Pacific Coast. The approach to San Francisco and the transit across the Bay to Oakland is open, however, there are restricted areas such as passing under the Golden Gate and Oakland Bay Bridges. There is considerable traffic in the Bay area. A Federal project channel depth of 10.6 m (35 ft) exists to and in the outer harbor. The same depth is maintained in part of the inner harbor. The width passage from the ocean to San Francisco Bay is reduced to approximately 1,125 m (0.7 mi) at the Golden Gate Bridge pier. The distance from the Golden Gate Bridge to the entrance of Oakland Harbor is less than 16 km (10 mi) (DOC, 1992b). A map of the port is shown in Figure D-46.

Oakland is a huge multi-terminal port complex consisting of Outer, Middle, and Inner Harbor cargo terminals leased to terminal operators and/or container shipping lines. The Port of Oakland is part of the Oakland Municipality. The Port Administration is strictly a "landlord" owner and does not operate any facilities. There is a growing trend for "secondary" use by other shipping lines of privately leased terminals, such as Matson's Outer Harbor 7th Street terminal—Berths 32-34, blurring the distinction between public and private use (Jane's, 1992; AAPA, 1993). The port handled over one million 20-ft equivalent units of containerized cargo in 1992 (AAPA, 1994) Public use container and general cargo facilities include:

Outer Harbor: The Seventh Street Marine Container Terminal, Berths 37 and 38, has three container cranes, 14.4 ha (35.6 acres) of terminal area, and storage for over 2,500 20-ft equivalent units. The Outer Harbor Public Container Terminal, Berth 23, has two container cranes, 16.2 ha (40 acres) of terminal area, and storage for over 3,500 20-ft equivalent units. The Bay Bridge Terminal, Berths 8-10, (a combination general cargo (breakbulk), container, and roll-on/roll-off facility) has 20.6 ha (50.9 acres) of terminal area and 7,072 m^2 (76,130 ft^2) of covered storage. This terminal was inoperative for several years due to earthquake damage sustained in 1989 but is now back in operation (Adams and Renteria, 1994). Quay lengths are as follows: Seventh Street Marine Containers Terminal — 592 m (1,942 ft) of marginal wharf; OHPCT — 274 m (900 ft) of marginal wharf; and Bay Bridge Terminal — 926 m (3,038 ft) of marginal wharf.



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- Interstate, Turnpike
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- Interstate Highway
- Railroad
- Open Water

Scale 1:50,000 (at center)
5000 Feet
1000 Meters

Mag 13.00
Mon Mar 27 09:31:03 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-46 Map of the Port of Oakland, CA

Crane capacities at Seventh Street Marine Containers Terminal include two 40.6 metric ton (45 ton) container cranes and one 30.5 metric ton (34 ton) container crane. Outer Harbor Public Container Terminal has crane capacity of two 40.6 metric ton (45 ton) container cranes (Jane's, 1992; AAPA, 1993).

Inner Harbor: The Charles Howard Terminal, Berths 67 - 69, has three container cranes, 19.8 ha (48.9 acres) of terminal area, and storage for over 3,000 20-ft equivalent units. Quay lengths at Charles Howard Terminal are as follows: two marginal wharves of 501 m each (1,642 ft) and one 173 m (568 ft) wharf. The Charles Howard Terminal has crane capacity consisting of two 40.6 metric ton (45 ton) container cranes and one 50.5 metric ton (56 ton) container crane (Jane's, 1992; AAPA, 1993).

Seventh Street Marine Containers Terminal, Outer Harbor Public Container Terminal, and Charles Howard Terminal have depths alongside at mean low water of 12.2 m (40 ft). Approach channels are currently limited to 10.6 m (35 ft). A dredging program to 12.8 m (42 ft) is scheduled for completion by 1995 (Jane's, 1992; AAPA, 1993).

Located just south of the Oakland Bay Bridge, the Port of Oakland has immediate access to Highway I-80 for shipments to Idaho National Engineering Laboratory and/or transcontinental shipments, and Highways I-580/I-5 for east coast shipments via the southern route I-40. The truck route from Seventh Street Marine Containers Terminal to the interstate appears to be almost entirely within the port complex in an area dedicated to cargo handling and shipping functions. The Port of Oakland is served by the Union Pacific, Southern Pacific, and Santa Fe Railroads (D&B, 1993). The port has an intermodal container transfer facility, but there does not appear to be direct rail service to container berths at the Seventh Street Marine Containers Terminal (Jane's, 1992; AAPA, 1993).

The Port of Oakland is served by many of the world's largest container lines, including American President Lines, Atlantic Container Lines, Australia-New Zealand Container Line, Cho Yang, DSR Senator Line, EAC Lines, Hanjin Shipping Co., Hapag-Lloyd, Hawaiian Marine, Hyundai, Italian Line, "K" Line, Maersk Lines, Matson Navigation Co., Mitsui OSK, Neptune Orient, NYK Lines, OOCL, Sea-Land Service, and Yang Ming Line (Jane's, 1992; AAPA, 1993; D&B, 1993). Four additional lines switched from San Francisco to Oakland in 1994 (Mitchell, 1994; Adams, 1994).

Other Pertinent Information: Security of the port is provided by perimeter fencing and unarmed guards from the International Longshoremen Union who maintain 24-hour patrol and surveillance (Adams, 1993; Adams and Renteria, 1994). Therefore, it is assumed that foreign research reactor spent nuclear fuel shippers using the port would have to provide their own security force.

The Port Commission has an active ban on the handling of spent nuclear fuel in recognition of community anti-nuclear sentiment which led to a citizens legislative initiative banning such shipments (subsequently struck down by a Federal court). The port handles radioactive and other hazardous materials shipments but officials did not know if Oakland has ever handled spent nuclear fuel shipments (Adams, 1993). The available data shows that Oakland has received spent nuclear fuel shipments, with the last shipment in 1988 (NRC, 1993).

Outer Harbor container and general cargo terminals are situated at the entrance to the port and there appears to be little or no conflict with other hazardous cargoes including petroleum products shipped through the port's breakbulk and liquid bulk terminals located within the Inner Harbor (Adams, 1993; Adams and Renteria, 1994).

Emergency response capability is the responsibility of the individual terminal operators. Each terminal operator must have an Emergency Contingency Plan approved by the Port Commission and the U.S. Coast Guard. The Oakland Fire Department has a hazardous materials response team, and the response time for

emergencies is about five minutes (Adams and Renteria, 1994). Beginning in November 1994, the port is increasing its emergency response capabilities. Financed by a new $50 million bond, the port is adding a new fire station, an Emergency Operations Center, new fire boats, a completely equipped hazardous materials van, and a fire-fighting bucket to be lifted in by helicopter. The port also has agreements with neighboring cities (Berkeley, San Leandro, and Alameda) for emergency response (Adams and Renteria, 1994). It is the responsibility of individual terminal organizations and/or the port to provide hazardous materials instruction to the longshoremen (Adams, 1993; Adams and Renteria, 1994).

The Seventh Street Marine Containers Terminal is located in the Outer Harbor terminal complex seaward of the downtown Oakland business district, in an area primarily dedicated to port industrial usage with excellent connections to highways and rail service. However, the port is located in a large urban area in which congestions are to be expected. The San Francisco Bay Area has had only 31 collisions, but 21 fires were reported during the period 1991 to 1993—the worst fire record for major West Coast ports (USCG, 1994b).

There are no known areas of special environmental concern; however, there is strong concern for preservation of the environment, and this area is prone to severe earthquakes. On April 18, 1906, the Bay area was subjected to one of the largest recorded earthquakes in the contiguous United States, a Modified Mercalli Intensity XI (Bolt, 1978), due to movement along the fault line separating the Pacific and Continental tectonic plates (Hamilton, 1976). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Oakland, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). Since the port is located in a very high seismic zone (the highest Uniform Building Code ranking), buildings must be constructed to withstand an acceleration of 0.40 g.

The 1990 population within 16 km (10 mi) of the preferred port terminals was 1,387,611. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,080,000; Oak Ridge Reservation, 786,000; Idaho National Engineering Laboratory, 367,000; Hanford Site, 359,000; and Nevada Test Site, 482,000. Populations along rail routes to these sites are slightly larger for Savannah River Site, Hanford Site and Nevada Test Site, but slightly smaller for Oak Ridge Reservation and Idaho National Engineering Laboratory. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,490 km (2,791 mi); Oak Ridge Reservation, 4,120 km (2,563 mi); Idaho National Engineering Laboratory, 1,550 km (963 mi); Hanford Site, 1,410 km (875 mi); and Nevada Test Site, 1,160 km (719 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The area around the terminal includes the San Francisco Bay to the south and the San Pablo Bay to the north. The Farallon Islands Game Refuge and the Point Reyes-Farallon Islands National Marine Sanctuary lie along the passageway to San Francisco. The San Francisco Bay National Wildlife Refuge and the Redwood Shores Ecological Reserve, both within the San Francisco Bay, are located 33 and 30 km south (20 and 18 mi), respectively from the Port of Oakland. The San Pablo Bay National Wildlife Refuge is located approximately 160 km (99 mi) north of the port.

San Francisco Bay

The Central Bay portion of the San Francisco Bay has several usage classifications, including industrial process supply, industrial service supply, navigation, water contact recreation, noncontact water recreation, ocean commercial and sport fishing, wildlife habitat, preservation of rare and endangered species, fish migration, fish spawning, shellfish harvesting, and estuarine habitat (State of California, 1986).

The San Francisco Bay comprises the largest estuarine ecosystem in California. The estuary encompasses a range of aquatic habitats, from the fresh and brackish waters of the Sacramento-San Joaquin River Delta to the saline waters of the Central and South Bay. The estuary provides habitat for a variety of aquatic species, some of which are important to commercial and recreational fisheries. These waters serve as a nursery area for marine, anadromous, and estuarine species, and provide a migration corridor for several anadromous species. Striped bass, Chinook salmon, steelhead trout, sturgeon, American shad, and English sole support important recreational fisheries in the estuary. Popular recreational fisheries in the Delta also include white catfish, largemouth bass, and sunfish (U.S. Army, 1994). In addition, the area around the port has populations of the common littlenecked clam, the soft-shelled clam, striped bass and flatfish, the California clapper rail, and the salt-marsh harvest mouse (FWS, 1981b).

Historically, marshlands bordering the Bay covered some 300 mi^2; diking for agriculture and filling for development has reduced the marshlands to about 75 mi^2 (U.S. Army, 1994). The marshes and mudflats remaining along the margins of the Bay are very productive and provide habitat for a large number of birds and other wildlife. For instance, the area around the port has populations of the California clapper rail and the salt-marsh harvest mouse (FWS, 1981b). The Bay is a key resting, feeding, and wintering area for birds on the Pacific Flyway. This area annually supports a large number of shorebirds, wintering waterfowl, raptors, seabirds, and passerines. Shorebirds, wading birds, waterfowl, seabirds, songbirds, and other species migrate through the entire coastal zone in the San Francisco area (FWS, 1981b).

Several threatened or endangered species are known to occur or have the potential to occur occasionally or periodically in the San Francisco Bay area. These species include the California least tern, California brown pelican, the American peregrine falcon, and the winter-run chinook salmon (U.S. Army, 1994). The least terns breed in California from mid-May to August and nesting colonies are located on open flat beaches, sand flats, and bare dirt areas with sparse vegetation. The least tern generally migrates from the Bay Area in August and winters south of the United States. The California brown pelican uses the open waters of the central San Francisco Bay for feeding; they roost on rocks, jetties, and piers in the area. Although no brown pelicans breed in the San Francisco area (Bay or offshore), thousands move north and roost on coastal rocks during the June through October nonbreeding season. Several thousand pelicans summer in the San Francisco area. The American peregrine falcon is considered rare in the region. It formerly bred on the Farallon Islands, and though it has yet to breed there again, winter residents have returned and have stabilized in number. The American peregrine falcon is most common to the San Francisco Bay area during the winter, when migrants from farther north concentrate in the estuary. The nesting season is from spring thorough early summer, and several pairs nested on the San Francisco-Oakland Bay Bridge (U.S. Army, 1994). California condors and bald eagles are also found in the coastal zone around San Francisco Bay (FWS, 1981b). The winter-run chinook salmon passes through the Sacramento-San Joaquin Delta, San Pablo Bay, and San Francisco Bay during their upstream and downstream migrations. The adults are present in the Bay area from November to May, and the smelts migrate through the Bay from November through May. The winter-run chinook is fished commercially in North America from Kotzebue Sound, Alaska, to Santa Barbara, California (U.S. Army, 1994).

Open Ocean

Several threatened or endangered species occur either occasionally or periodically in the ocean offshore of the San Francisco area. These include the humpback whale, the blue whale, the sperm whale, and the Stellar sea lion (U.S. Army, 1994). The humpback whale, which has a worldwide range, is typically found in the San Francisco area from March through January. Summer feeding occurs from the Aleutian Islands to the Farallon Islands. The greatest number of blue whales within the Farallon Basin occurs in summer and early fall. The sperm whale regularly occurs in the Gulf of the Farallones in deep oceanic waters, and is rarely reported over the shelf. The Stellar sea lion ranges from California to the Bering Sea. Stellar sea lions have rookeries on Southeast Farallon Islands (as well as other California and Pacific coast sites). The sea lion breeds in the late spring and summer.

Climatic Conditions

The Oakland, CA, area is classified as a marine climate, which is characterized by mild and moderately wet winters, with cool, dry summers. The winter rains, which occur between November and March, account for over 80 percent of the total annual precipitation. Additionally, severe winter storms, with gale-force winds and heavy rains do occur occasionally. The diurnal temperature range is moderated substantially by marine environment. The summer weather is dominated by a cool sea breeze circulation and a sea fog that arrives in the late evening over the area. The fog generally burns-off in the early morning hours, resulting in relatively sunny summer days (NOAA, 1993d).

D.2.2.16 Palm Beach, FL

The Port of Palm Beach is located 2.0 km (1.1 mi) west of the entrance to Lake North Worth Inlet, which consists of a dredged cut, protected by two jetties, through the barrier beach which forms the resort city of Palm Beach. The port borders the communities of Riviera Beach on the north and West Palm Beach on the south, the latter being connected to Palm Beach by highway bridges spanning Lake Worth. The Port of Palm Beach is 110 km (68 mi) north of Miami and 417 km (259 mi) south of Jacksonville. A Federal project provides for a 10.7 m (35 ft) deep entrance channel with a 10.1 m (33 ft) inner channel to a turning basin of the same depth. The 121.9 m (400 ft) wide entrance channel narrows to 91 m (300 ft) and leads into a 442 m by 399 m (1,450 ft by 1,310 ft) turning basin. Port Authority-owned Peanut Island is located between the inlet entrance and Port of Palm Beach terminals. According to the port's 1993 Annual Report, the controlling depth of the entrance and turning basin was 10.1 m (33 ft) to not less than 7.6 m mean low water (25 ft) at the northern terminal extension (POPB, 1994). A map of the port is shown in Figure D-47.

The Port of Palm Beach is a landlord port with 77 ha (190 acres) of land. The Terminal has two slips and four marginal wharves totalling 1,536 m (5,039 ft) of berthing, including six roll-on/roll-off ramps. Pilots limit the maximum size of ships entering the port to 192 m (630 ft) in length. Total tonnage for the fiscal year ending September 1993 was 3,694,034 metric tons (4,071,934 tons), including 158,172 20-ft equivalent units [(1,005,190 metric tons)(1,108021 tons)] of containerized cargo. The port owns one 228 metric ton (251 km) crawler crane, but containers are either handled by ship's gear or with local stevedoring equipment. Primary commodities handled are containerized general cargo, sugar, molasses, and fuel oil for two local power plants (POPB, 1994; AAPA, 1994; Mets, 1994).

Port of Palm Beach Berths 5-6, 7-11, and Berths 12-17 are operated as public terminals for container handling, general cargo, roll-on/roll-off, cruise lines, and heavy lift cargoes. Berths 5 and 6 have 10.1 m (33 ft) depth alongside and have rail service on the pier, which is owned and operated by the port

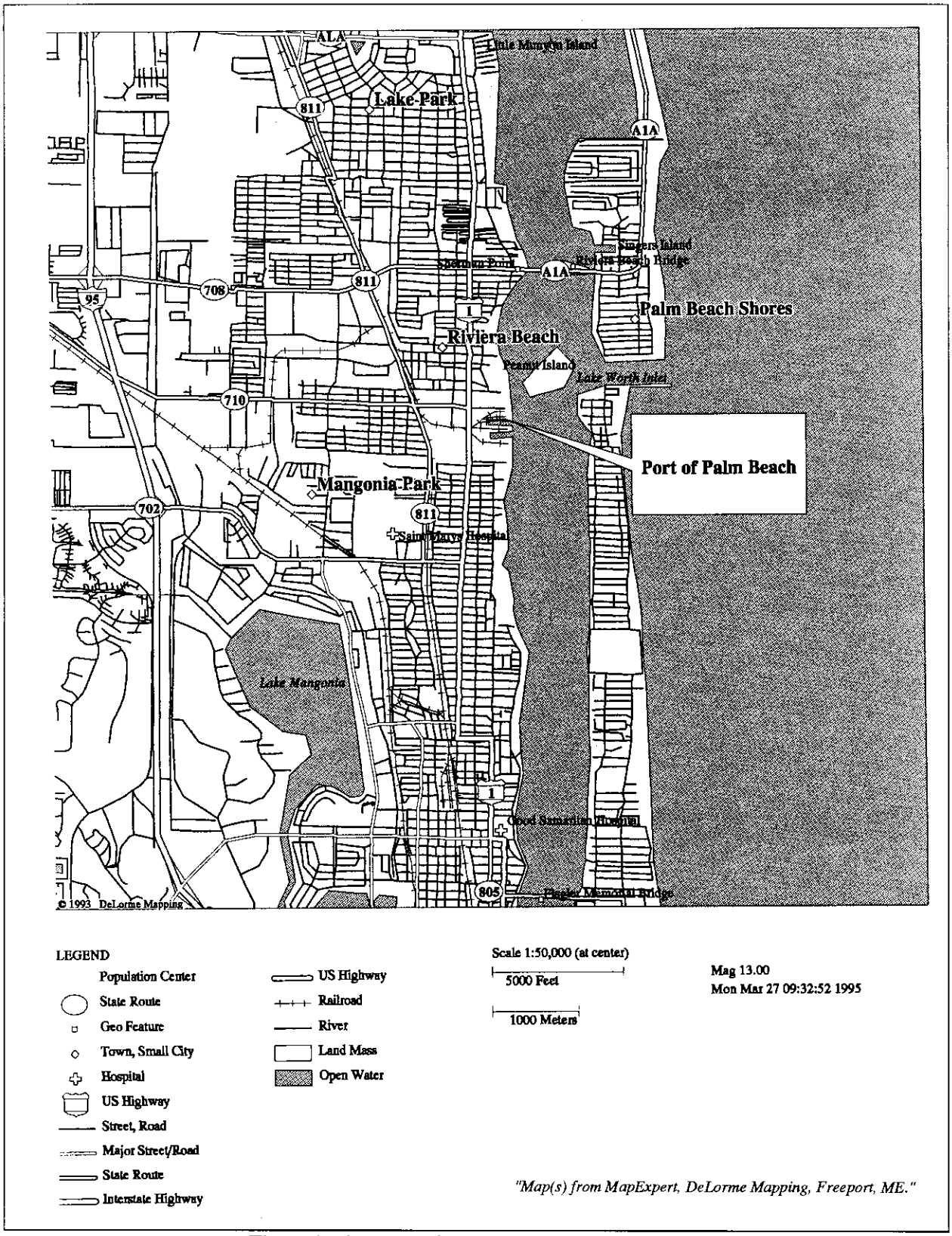


Figure D-47 Map of the Port of Palm Beach, FL

connecting with the Florida East Coast Railroad. The Port of Palm Beach is about 2.4 km (1.5 mi) from I-95 and 8.1 km (5 mi) from the entrance to the Florida Turnpike. The route is through light commercial and residential areas (AAPA, 1994).

Other Pertinent Port Information: Palm Beach has an around-the-clock watchman service, is fenced and lighted and has only one controlled entrance/exit. The port handles explosives and other hazardous goods and according to a port official, the port does not have a prohibition against handling spent nuclear fuel (Mets, 1994). As with other small, multi-use ports, there is some apparent conflict between the handling of petroleum products, cruise ship passengers, and spent nuclear fuel all within the confines of a relatively small, environmentally sensitive harbor complex.

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Palm Beach, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

Negotiations for the sale of the port's Peanut Island, mentioned above, to the Florida Inland Navigation District are currently underway. Use of the island would be permanently limited to a partial dredge spoil area, as well as habitat preservation, and a passive recreation area (POPB, 1994; Mets, 1994).

The 1990 population of the combined port area (Riviera, Palm, and West Palm Beach) was approximately 115,000, and the average are density was about 650 persons/km^2 (1,600 persons/mi^2). The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are approximately (based on data for nearby Port Everglades): Savannah River Site, 240,000; Oak Ridge Reservation, 350,000; Idaho National Engineering Laboratory, 780,000; Hanford Site, 790,000; and Nevada Test Site, 800,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes for nearby Port Everglades are approximately: Savannah River Site, 1,125 km (700 mi); Oak Ridge Reservation, 1,366 km (850 mi); Idaho National Engineering Laboratory, 4,501 km (2,800 mi); Hanford Site, 5,145 km (3,200 mi); and Nevada Test Site, 4,662 km (2,900 mi). Distances along rail routes are slightly longer.

Climatic and environmental conditions are similar to those reported for Port Everglades in Section D.2.2.18.

D.2.2.17 Philadelphia, PA

Philadelphia, one of the chief ports of the United States, is located at the junction of the Delaware and Schuylkill Rivers, approximately 130 km (81 mi) above the entrance to the Delaware Capes. Access to the port is via the Delaware River through the Delaware Bay. Situated directly across the Delaware River from Philadelphia is Camden, NJ, an important shipping center. The shipping activities of the two cities are closely allied; large quantities of general cargo are handled at the Philadelphia port in both domestic and foreign trade. Access to the port is gained via the Delaware Bay and Delaware River (DOC, 1993c). A map of the port (including Camden, NJ immediately opposite) is shown in Figure D-48.

The Delaware Bay has natural depths of 15.4 m (50 ft) or more for a distance of 8 km (5 mi) from the entrance. A Federal project provides depths of 12.2 m (40 ft) from the sea through the Delaware Bay and River to Philadelphia. There are restrictions on the passage through the Delaware Bay and up the Delaware River, such as a traffic separation scheme established off the entrance to the Delaware Bay.

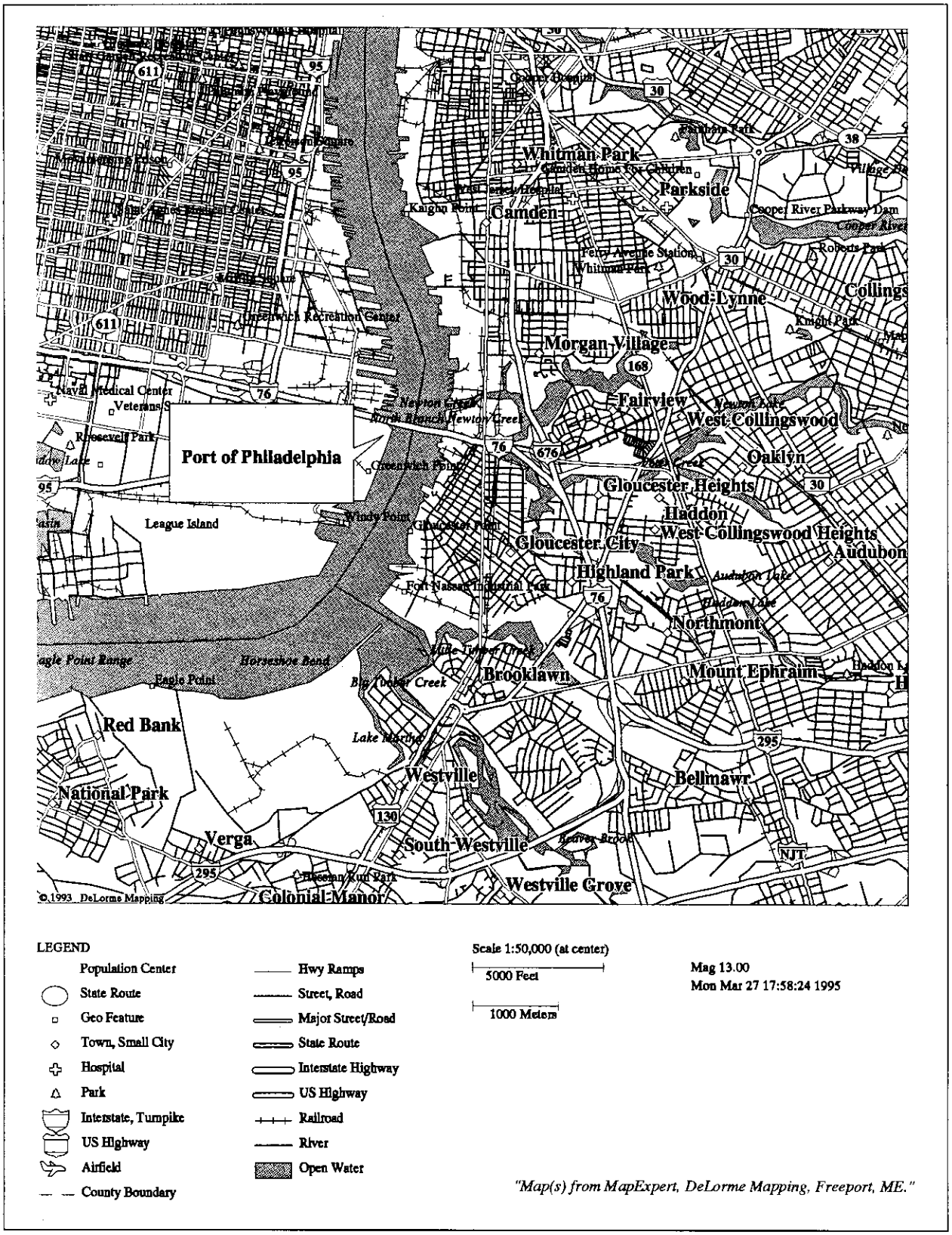


Figure D-48 Map of the Ports of Philadelphia, PA, and Camden, NJ

Ships going to Philadelphia must also pass under the Delaware Memorial Bridge. Roughly 90 percent of the 58,831,000 metric tons (64,849,000 tons) of cargo handled in 1991 were bulk cargo, and a large share of that is known to be crude oil and refinery products (DOC, 1993c; AAPA, 1993).

With the exception of some privately owned terminals, general cargo facilities (including container terminals on the west side of the Delaware River) are owned by the City of Philadelphia. Across the river on the New Jersey side, cargo terminals belong to the South Jersey Port Corporation, a state agency that operates two terminals and leases the remaining ones to private companies. The Philadelphia Regional Port Authority, apparent successor to the Philadelphia Port Corporation, is responsible for City-owned terminals leased to private companies under a landlord-type operation. The Delaware River Port Authority functions as a port planning and economic development division for the facilities controlled by the Philadelphia Regional Port Authority and the South Jersey Port Corporation.

The South Jersey Port Corporation operates two multi-berth terminals, Beckett Street and Broadway. These terminals primarily handle breakbulk general cargoes. Both breakbulk and containers (approximately 4,000 to 5,000 per year) are handled at the Beckett Street terminal during the winter (Castagnola, 1994). Beckett Street terminal has two container cranes, one with a 36.6 metric ton (40 ton) capacity and one with a 77.1 metric ton (85 ton) capacity (Castagnola, 1994). Pier 6, one of the Broadway Berths, is leased to a private company and is equipped with one 72.6 metric ton (80 ton) capacity multi-purpose container crane and one 40.6 metric ton (45 ton) container crane (Jane's, 1992; AAPA, 1993).

Principal container handling facilities owned by Philadelphia Regional Port Authority are the Packer Avenue Terminal, a combination breakbulk/container terminal, and Tioga Container Terminals. The former is located immediately downstream of the Walt Whitman Bridge at the south end of the City's waterfront. The Tioga Terminal is approximately 9.7 km (6 mi) further upstream. Both the Packer Avenue and Tioga Terminals have a depth alongside at mean low water of 12.2 m (40 ft) (Jane's, 1992; AAPA, 1993).

Packer Avenue Terminal: The Packer Terminal is equipped with two 45 metric ton (50 ton) container cranes and one 37.5 metric ton (41 ton) container crane, and has a paved open storage area of 23.5 ha (77 acres). The terminal has 1,184 m (3,885 ft) of marginal wharf (Jane's, 1992; AAPA, 1993).

Tioga Terminal: Tioga has 20.2 ha (50 acres) of paved open storage and is equipped with two 45 metric ton (50 ton) container cranes. The terminal has 796 m (2,612 ft) of marginal container berth; and a 185 m (610 ft) roll-on/roll-off berth.

Both the Packer Avenue and Tioga Terminals are relatively short distances from I-95, which parallels the River and is estimated to be within 0.8 km (0.5 mi) of the container terminals. The Packer Avenue Terminal is served by CSX and Conrail; CSX maintains an intermodal terminal just outside the terminal. On the other hand, the Tioga Terminal has Conrail intermodal service at the terminal. However, neither terminal has ship-side trackage (Jane's, 1992; AAPA, 1993).

A partial list of the diverse liner shipping companies serving these Delaware River terminals include: ABC Container Line, PACE, Atlantik Express Lines, Baltic Shipping Lines, Bangladesh Lines, Barber West Africa, Bottachi, Chilean Line, Colombus, Egyptian National Line, Ellerman, ELMA, Empremar Line, Euro Line, Frota Amazonica, Grandcolumbiana, Hapag-Lloyd, Hoegh, Hyundai, Independent Container Line, Maersk, Netumar, Pakistan National, Shipping Corp of India, SITRAM, Tokai, and Toko (Jane's, 1992).

Other Pertinent Information: The container terminals are fenced with controlled access and 24-hour security. It is not known what arrangements exist for temporary storage of hazardous materials, but it is likely such storage is available in a large port facility (Castagnola, 1993). Spokespersons for the South Jersey Port Corporation (Castagnola, 1993; formerly with the Philadelphia Regional Port Authority) and the Philadelphia Regional Port Authority (Menta, 1993) were unaware of any restrictions on handling spent nuclear fuel, but indicated this was outside their areas of expertise. There are several major oil refineries along the Delaware River below and west of the City of Philadelphia on the Schuylkill River. However, there does not appear to be any serious conflicts in close proximity to the Packer Avenue or Tioga Terminals.

The South Jersey Port Corporation relies on state hazardous materials agencies and the Camden Fire Department for emergency response to the terminals it operates. Private operators are responsible for their own terminals, but basically rely on the fire department. Being a landlord port operator, it is assumed Philadelphia Regional Port Authority terminal operators rely on the Philadelphia Fire Department to respond to hazardous materials incidents. It is not known if there is any hazardous materials training by the Philadelphia Regional Port Authority or the South Jersey Port Corporation (Castagnola, 1993). There are no known special environmental concerns. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Philadelphia, the Uniform Building Code requires buildings to withstand wind speeds up to 120 km/hr (75 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The Philadelphia and Camden waterfronts have become tourist centers due to historical sites and a new aquarium on the Camden waterfront. Veterans Stadium and the Spectrum are located in relatively close proximity to Packer Avenue, as are the Philadelphia Navy Yard and the Philadelphia International Airport.

All terminals in the Greater Philadelphia area are basically located in densely developed and populated industrial/commercial areas. The 1990 population within 16 km (10 mi) of the port terminals (including Camden, NJ) was 1,915,775. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 397,000; Oak Ridge Reservation, 335,000; Idaho National Engineering Laboratory, 513,000; Hanford Site, 622,000; and Nevada Test Site, 756,000. Populations along rail routes to these sites are much larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,190 km (741 mi); Oak Ridge Reservation, 1,090 km (680 mi); Idaho National Engineering Laboratory, 3,950 km (2,452 mi); Hanford Site, 4,610 km (2,868 mi); and Nevada Test Site, 4,220 km (2,623 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The Delaware River at Philadelphia is classified as a low salinity estuarine (generally 0.5 to 5 ppt) and tidal freshwater habitat. Aquatic organisms typically found in the waters of this area include: American shad, Atlantic sturgeon, American eel, blueback herring, shad, alewife, white catfish, brown bullhead, perch, striped bass, bluegill, crappie, pumpkinseed, largemouth bass, carp, and chain pickerel (FWS, 1980f). In addition, the Delaware River is used as a migratory area by the shortnose sturgeon, a Federally listed endangered species. The U.S. Fish and Wildlife Service reported that except for occasional transient species, no federally listed or proposed threatened or endangered species are known to exist in the port's impact area (Perry, 1994).

The Port of Philadelphia is located within Zone 3 (tidal river) of the Delaware River. Protected water used for Zone 3, which encompasses River Mile (RM) 95-108.4, are water supply (agricultural, industry, and public), wildlife, resident fish maintenance, anadromous fish passage, secondary contact, and navigation (DRBC, 1994). However, several uses within Zone 3 are currently impacted, including: 1) fish and other aquatic life due to low dissolved oxygen levels from point source discharges; and 2) fish and shellfish consumption due to chlordane and polychlorinated biphenyl contamination from point and nonpoint source discharges.

Climatic Conditions

The climate of Philadelphia is moderated by the Appalachian Mountains to the west and the Atlantic Ocean to the east. These geographic features cause periods of extreme temperatures to be short-lived in this region (generally, four days). On occasion during the summer months, the area is dominated by maritime tropical air masses, which contribute to elevated local temperature and humidity levels. The average annual precipitation (41.42 in) is relatively evenly distributed throughout the year, with maximum amounts occurring during the late summer months. The summer precipitation regime is dominated by localized thunderstorms and is subject to the influence of the urban heat island effect and local topography, which create varying rainfall amounts across the city for an individual event. Singular snowfall events that generate accumulated totals of greater than 10-in have a 5-year recurrence interval on average. The prevailing wind direction has a bimodal distribution, being southwesterly during summer and northwesterly in the winter months. The annualized average prevailing wind direction is from the west-southwest. Due to Philadelphia's inland location, destructive winds are comparatively rare from tropical cyclones and tornadoes. High winds are generally associated with frontal passages/low pressure systems in winter and thunderstorms in summer months. However, flooding on the Schuylkill River normally occurs twice annually usually associated with strong thunderstorms, with the duration of these events generally lasting less than 12 hrs. The Delaware River is rarely observed at or above flood stage (NOAA, 1992h).

D.2.2.18 Port Everglades, FL

Port Everglades is a major deepwater port located on Florida's southeast coast. It is located immediately off the Atlantic Ocean along the Inland Waterway, within the three cities of Hollywood, Fort Lauderdale, and Dania (DOC, 1993d; D&B, 1993; Southern Shipper, 1993). The major port facilities are immediately inside the harbor entrance, approximately 1.6 km (1 mi) from the south jetty. The approach to Port Everglades is open, and a relatively short 140 m (450 ft)-wide channel leads directly from the Atlantic Ocean to the port facilities. A Federal project provides for depths of 12.8 m (42 ft) to the main port facilities (DOC, 1993d; D&B, 1993; Jane's, 1992; Southern Shipper, 1993; AAPA, 1994; PEA, 1993).

Port Everglades consists of 850 ha (2,100 acres) of land, of which 360 ha (890 acres) are owned by the Port Everglades Authority Commission (Port Everglades Authority). Considerable foreign commerce passes through Port Everglades, in addition to substantial passenger traffic. Many of the world's large passenger vessels call at Port Everglades (it claims to be the world's second-busiest cruise port) (Southern Shipper, 1993). It is a multi-terminal port with more than 3,800 ship calls annually. The port handles over 2,000,000 cruise passengers as well as 17 million tons of cargo, including 1.4 million metric tons (1,600,000 tons) of containerized general cargo (over 100,000 20-ft equivalent units) and over 14 million metric tons (16,000,000 tons) of bulk cargoes (dry/liquid/scrap) in 1991 (FS, 1992). Port Everglades is also a liberty port for the U.S. and North Atlantic Treaty Organization Navies, and is host to facilities operated by the U. S. Naval Surface Warfare Center in Fort Lauderdale. The port is also bordered on the east by a large State Park and seashore recreation area. A map of the port is shown in Figure D-49.

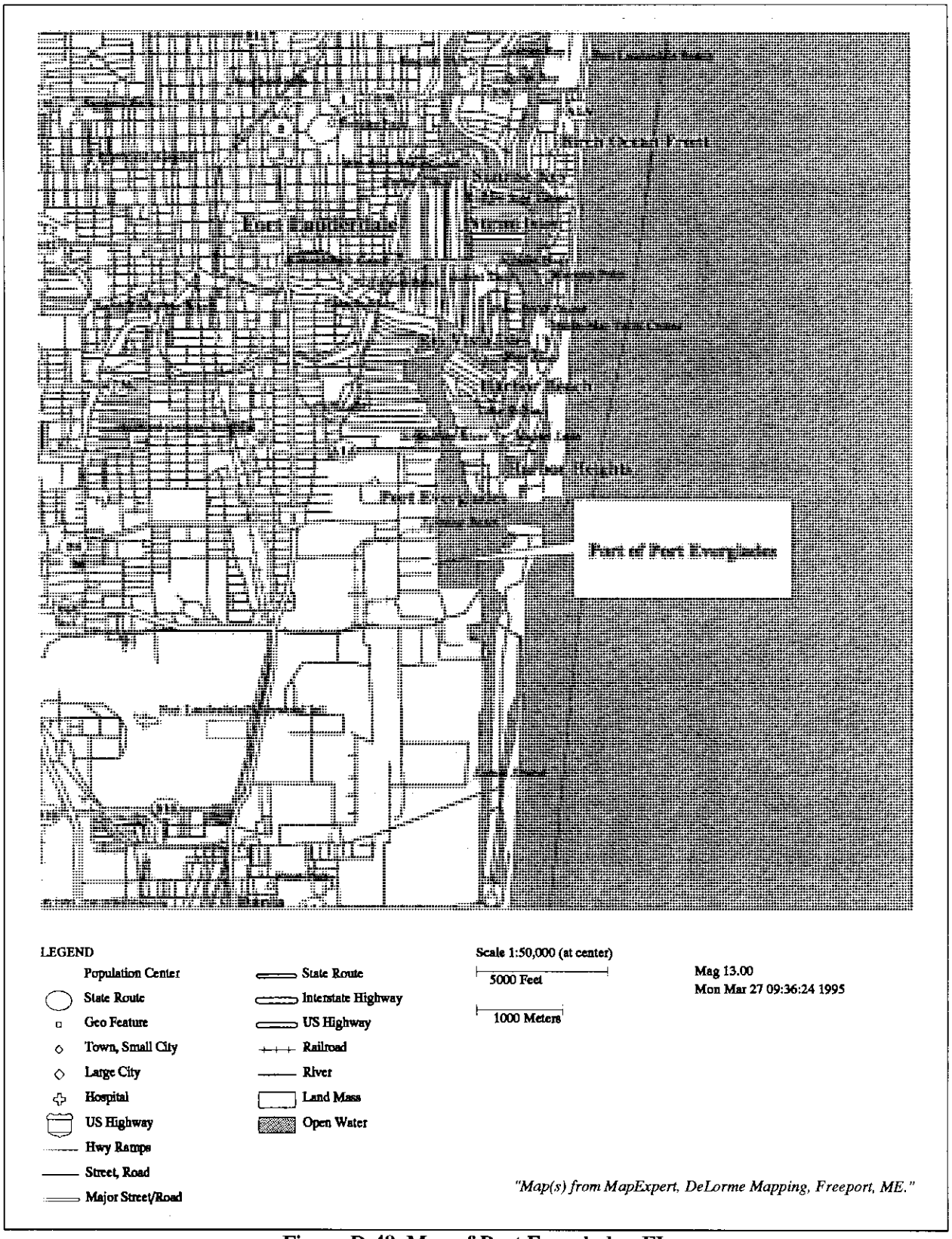


Figure D-49 Map of Port Everglades, FL

The Port Everglades Authority is empowered by the State Legislature to act as the governing entity for the operations, maintenance, and management of port and harbor facilities located within the port's jurisdictional area. The Authority is the governing body responsible for strategic planning and policy setting. In November 1994, governing responsibility for the seaport was transferred to the Broward County Government (PEA, 1993).

Principal container handling facilities at the port include Midport and Southport (there is also a Northport terminal as well).

Midport: Midport has three berths (16, 17, and 18), with three container cranes and 17.8 ha (44 acres) of open storage, and is located just inside the harbor entrance. The berths have a total of 502 m (1,650 ft) of marginal wharf area and depths of 11.6 m (38 ft) alongside at mean low water. Crane capacities at Midport consist of one 50.8 metric ton (56 ton) container crane and two 30.5 metric ton (34 ton) container cranes. Access to Midport is via Port Road, Highway 84, which intersects with U.S. 1, I-95, and all major interstate highways. Highway access appears to be exclusively through port property and/or adjacent industrial-use land. Port-owned trackage is leased to the Florida East Coast Railroad, which maintains an intermodal container yard 1.6 km (1 mi) outside the port on State Road 84. There is trackage to Berth 4, but there does not appear to be any rail lines presently serving the container terminals. There are two roll-on/roll-off ramps at Midport (Jane's, 1992; D&B, 1993; Southern Shipper, 1993; AAPA, 1993 and 1994; PEA, 1993).

Southport: Southport is a new 62.7 ha (155 acre) container/roll-on/roll-off complex at the southern end of Port Everglades Harbor. Southport consists of Berths 31 and 32 with a combined 610 m (2,000 ft) of marginal wharf. Berths 30-33C at Southport have depths of 13.4 m (44 ft) alongside at mean low water. Crane capacities for the berths at Southport include three 40.6 metric ton (45 ton) Post-Panamax container cranes. Access to Southport is via the new Port Everglades Expressway (I-595), which begins just outside the Terminal and connects with I-95, the Florida Turnpike, I-75, and State Highway 84, the cross-Florida Everglades Expressway. Highway access appears to be exclusively through port property and/or adjacent industrial-use land. Southport has a 25.5 ha (63 acre) container yard with storage for up to 5,100 containers on chassis or up to 7,872 grounded and stacked 20-ft equivalent units. There are three roll-on/roll-off ramps at Southport, and an additional three ramps are located at Northport.

Port Everglades is served by over a dozen container and breakbulk Liner Shipping Companies offering sailings to major ports of the world, including South and Central America, Caribbean Islands, North Europe, the United Kingdom, Scandinavia, Spain, the Mediterranean, and the Mid-East. Ship lines include Arawak Caribbean Line, Atlantic Cargo Service, Crowley American Transport, Nedlloyd, Inc., P & O Containers, Ltd., Sea-Land Service, Inc., Tecmarine Lines, and Orient Overseas Container Line.

Other Pertinent Information: Twenty-four-hour security is provided by the Broward Sheriff's Office (BSO). All terminal and container facilities are secured with fencing and controlled gates. The port also has its own 65-member Public Safety Department (fire department), fire fighting equipment and vehicles, and a fireboat for first response to hazardous materials incidents. It is backed up by fire departments from Fort Lauderdale and Hollywood, and Broward County's hazardous materials team. Terminal operators are responsible for their own emergency response arrangements. The Public Safety Department basically functions as an emergency coordinating group. Hazardous materials training is carried out by the municipal agencies responsible for emergency response within the port area. It is not known if the Port Authority itself conducts any training. There appears to be ample space at Southport for temporary storage of hazardous cargoes.

There are no known restrictions to the handling of spent nuclear fuel. However, Item 240 of the Port Everglades Authority Tariff states that explosives, hazardous, or highly flammable commodities or materials may only be handled through special arrangement with the Port Authority. Port Officials indicate that their safety policies, which ban oxidizers such as ammonium nitrate and Class A explosives, would also preclude shipments of spent nuclear fuel. As far as is known, spent nuclear fuel shipments have not been handled by the port (Flint et al., 1993).

Port Everglades is the second-largest petroleum distribution facility in the United States (Southern Shipper, 1993). Major oil companies have more than 86 million barrels of tank space for refined petroleum products inshore of the Midport and Northport terminals. With the possible exception of terminal facilities at Southport, which are remote from the tank farms and other conflicting port users, the potential for conflict between cruise ship operations, tanker traffic, and containerized spent nuclear fuel shipments appears to be great.

Southport Terminal is the preferred terminal, as it is relatively remote from the City of Fort Lauderdale, has direct connection to the Interstate Highway system, and is located in a nonresidential port industrial district. The physical layout and constraints of port waterways, however, combined with its intense use by potentially conflicting types of transport (i.e., cruise ships, tankers and tank barges, container and breakbulk vessels, and recreational traffic, plus a State seashore park on its eastern boundary) detracts from its otherwise superb facilities.

The port is subject to severe hurricanes and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For Port of Everglades, the Uniform Building Code requires buildings to withstand wind speeds up to 150 km/hr (95 mph). Port Everglades is located in a very low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 714,176. The affected populations within 0.8 km (0.5 m) of the interstate routes to the 5 potential DOE management sites are: Savannah River Site, 244,000; Oak Ridge Reservation, 352,000; Idaho National Engineering Laboratory, 754,000; Hanford Site, 803,000; and Nevada Test Site, 817,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,170 km (726 mi); Oak Ridge Reservation, 1,420 km (884 mi); Idaho National Engineering Laboratory, 4,540 km (2,820 mi); Hanford Site, 5,210 km (3,236 mi); and Nevada Test Site, 4,700 km (2,923 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The State of Florida has classified the waters surrounding Port Everglades as Class III. This classification indicates that the waters are suitable for recreation, and propagation and maintenance of a healthy, well balanced population of fish and wildlife (FL DEP, 1994). As previously noted, Outstanding Florida Waters are generally waters located within national parks, state parks, national seashores, marine sanctuaries, or aquatic preserves. Waters located near Port Everglades designated as Outstanding Florida Waters include the waters within the John U. Lloyd Beach State Recreation Area, West Lake, Snake Warrior Island, North Beach, and the Hugh Taylor Birch State Recreation Area (FL DEP, 1994). These waterways are afforded special protection by the State environmental regulations.

The waters surrounding Port Everglades are characterized as high salinity estuarine habitats (generally greater than 20 parts per thousand). Aquatic species of interest in the vicinity of Port Everglades include: crabs, shrimp, lobster, seatrout, croaker, tarpon, sheepshead, spot, kingfish, drum, silver perch, bluefish, mullet, pompano, pinfish, pigfish, Crevalle jack, grunt, ladyfish, permit, grouper, snapper, jewfish, snook, striped mojarra, and Atlantic bottlenose dolphin (FWS, 1982b).

The John U. Lloyd Beach State Recreation Area is located approximately 4 km (2.5 mi) south of Port Everglades. Additional special land use areas located near Port Everglades are Everglades Wildlife Management Area and Hugh Taylor Birch State Recreation Area. Protected animal species in the Port Everglades vicinity include: the West Indian manatee, loggerhead sea turtle, green sea turtle, and the least tern. Protected plant species in the area are the coontis, sea lavender, and the silver palm (FWS, 1982b).

Port Everglades has several ongoing environmental programs, including the creation of a Manatee Refuge and "Nursery" area within the confines of the port, a wetlands program, and a 22.3 ha (55 acre) permanent mangrove forest and manatee reserve deeded to the Florida Department of Environmental Regulation. The port was awarded the 1991 National American Association of Port Authorities Environmental Award of Excellence (PEA, 1993). The U.S. Fish and Wildlife Service reported that the Port Everglades Midport Terminal is located in designated critical habitat for the manatee (Johnson, 1995).

Climatic Conditions

The climate of this region is essentially subtropical marine, featuring long, warm summers with abundant rainfall, generally followed by a mild, dry winter. The influence of the ocean is seen in the small diurnal temperature range (generally less than 10 degrees) and the rapid warming of any cold air masses that invade this portion of the State. The predominant windflow is from the east-southeast, which generates conditions right at the coast that are often different than those encountered further inland, due to land-induced frictional effects. Hurricanes occasionally affect the area, with the months of September and October exhibiting the highest frequencies. However, destructive tornadoes (not associated with tropical systems) are rare. Waterspouts are frequently spotted offshore during the summer months, but rarely cause any loss of life or property damage (NOAA, 1993b).

D.2.2.19 Richmond, VA

The Port of Richmond Terminal is located on the left ascending bank of the James River, approximately 140 km (89 mi) above the City of Newport News. The port is owned by the City of Richmond and operated by Mecham Overseas Terminal, Ltd. A map of the port is provided in Figure D-50.

Drafts of vessels using the river above Newport News generally do not exceed 4.5 m (15 ft). Vessels drawing more than 7.5 m (24 ft) do navigate it occasionally, but the Virginia Pilots Association restricts ship drafts to 6.7 m (22 ft). A Federal Project provides for dredging depths of 7.6 m (25 ft) to the Richmond Terminals. Numerous stakes, pilings, wrecks and other obstructions are on both sides of the main channel. Travel on the upper river is restricted to daytime hours for ships more than 77.7 m (255 ft) in length (DOC, 1993c). For FY 1993-94, the Terminal handled a total of 445,700 metric tons (491,300 tons) including 313,540 metric tons (345,620 tons) of containerized cargoes (41,286 20-ft equivalent units), and 114,890 metric tons of breakbulk freight (AAPA, 1994; PORT, 1994). Major shipping lines connect the port with the Mediterranean, North Europe, South America, the Middle East, and India (PORT, 1994).

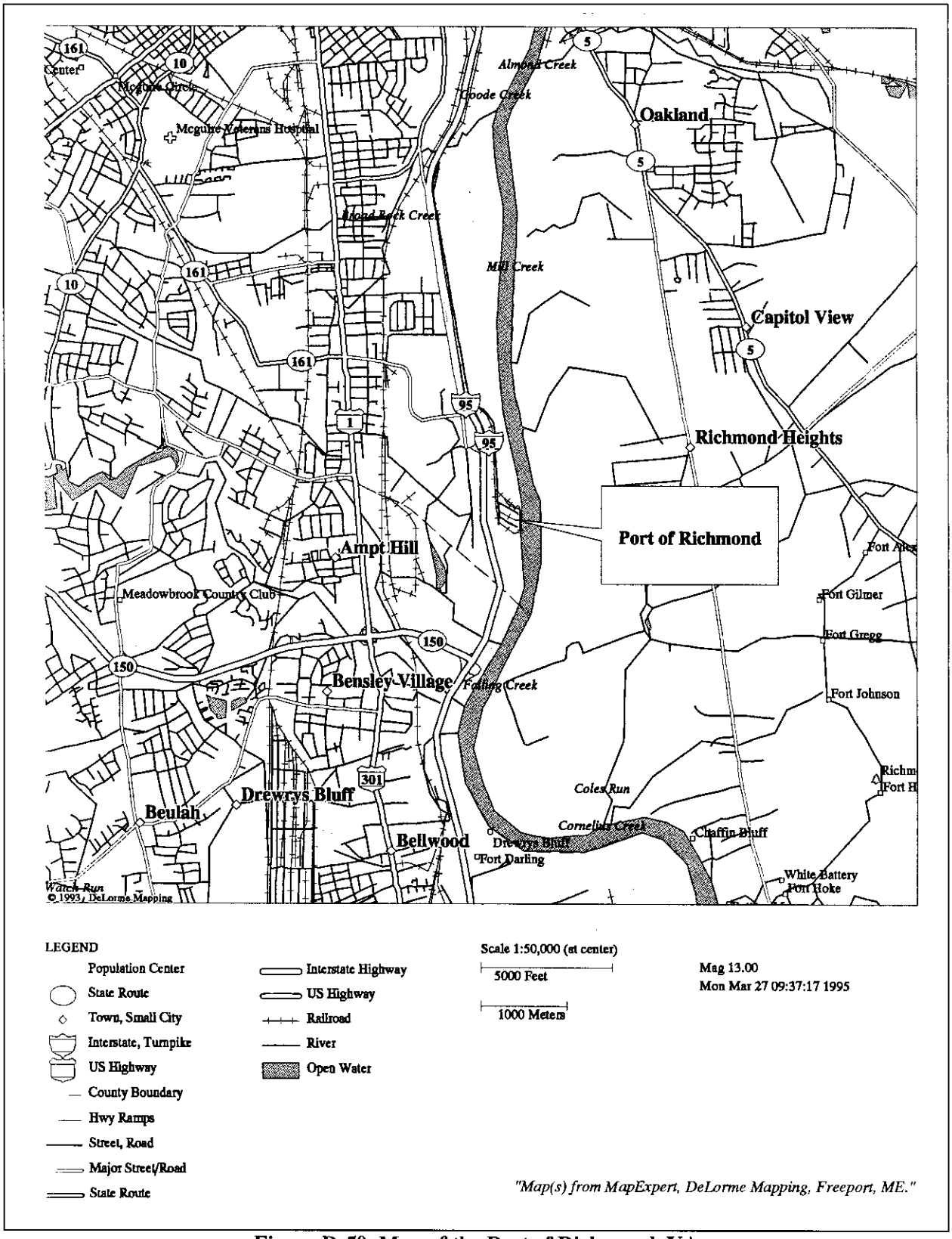


Figure D-50 Map of the Port of Richmond, VA

Richmond Terminal has two marginal berths with a total length of 381 m (1,250 ft) and 7.6 m (25 ft) of water alongside at mean low water. CSX Railroad tracks with multiple sidings serve the port's two warehouses and container storage yards. The Terminal is a container, general cargo, and breakbulk handling facility with roll-on/roll-off vessel and container and trailer on flatcar capabilities. The port has two 209 metric ton (230 ton) and one 319 metric ton (350 ton) capacity crawler cranes outfitted with 22.9 m (75 ft) booms. A new 273 metric ton (300 ton) crane was purchased in April 1994 and set a new accident-free container handling record of 20.43 20-ft equivalent units/hr (PORT, 1994).

The port is about 1.6 km (1 mi) from highway I-95 with travel through an industrial area. It is also served by a trunk railway.

Other Pertinent Port Information: The Port of Richmond has only one entrance which is controlled by a Pinkerton Guard on a 24-hour basis.

The Port of Richmond Terminal is in the midst of an $8-10 million expansion program involving a 96 m (315 ft) wharf extension, new gate entrance and maintenance building, plus upgraded container storage areas (PORT, 1994).

The likelihood of severe natural phenomena such as high winds and earthquakes are reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Richmond, the Uniform Building Code requires buildings to withstand wind speeds up to 120 km/hr (75 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 Census city population was approximately 203,000 and the population density was about 1,304 per km^2 (3,376 per mi^2). The distances to the five management sites on interstate routes are: Savannah River Site, 732 km (455 mi); Oak Ridge Reservation, 784 km (487 mi); Idaho National Engineering Laboratory, 3,900 km (2,426 mi); Hanford Site, 4,570 km (2,842 mi); and Nevada Test Site, 4,070 km (2,529 mi). Distances along rail routes are slightly longer.

Climatic Conditions

Richmond's climate might be classified as modified continental. Summers are warm and humid and winters generally mild. The mountains to the west act as a partial barrier to outbreaks of cold, continental air in winter, the coldest air being delayed long enough to be modified, then further warmed as it subsides in its approach to Richmond. The open waters of the Chesapeake Bay and Atlantic Ocean contribute to the humid summers and mild winters. The coldest weather normally occurs in later December and in January, when low temperatures usually average in the upper twenties and the high temperatures in the upper forties. Temperatures seldom lower to zero.

Precipitation is rather uniformly distributed throughout the year. However, dry periods lasting several weeks do occur, especially in autumn when long periods of pleasant, mild weather are most common. There is considerable variability in total monthly amounts from year to year so that no one month can be depended upon to be normal. Snow has been recorded during seven of the 12 months. Snowfalls of 10 cm (4 in) or more occur on an average of once a year. Snow usually remains on the ground only one or two days at a time. Ice storms (freezing rain or glaze) are not uncommon in winter, but they are seldom severe enough to do any considerable damage. The James River reaches tidewater at Richmond, where flooding has occurred in every month of the year, most frequently in March (28 times in the past 61 years), and only twice in July. Hurricanes and less severe storms of tropical origin have been responsible for most of the

flooding during the summer and early fall. Damaging storms occur mainly from snow and freezing rain in winter, and from hurricanes, tornadoes, and severe thunderstorms in other seasons. Damage may be from wind, flooding, or rain, or from any combination of these (DOC, 1993c).

D.2.2.20 San Francisco, CA

San Francisco, CA, occupies the north portion of the peninsula forming the south entrance to San Francisco Bay. The Port of San Francisco, one of the largest ports on the bay, is the oldest and one of the most important on the Pacific Coast (DOC, 1992b).

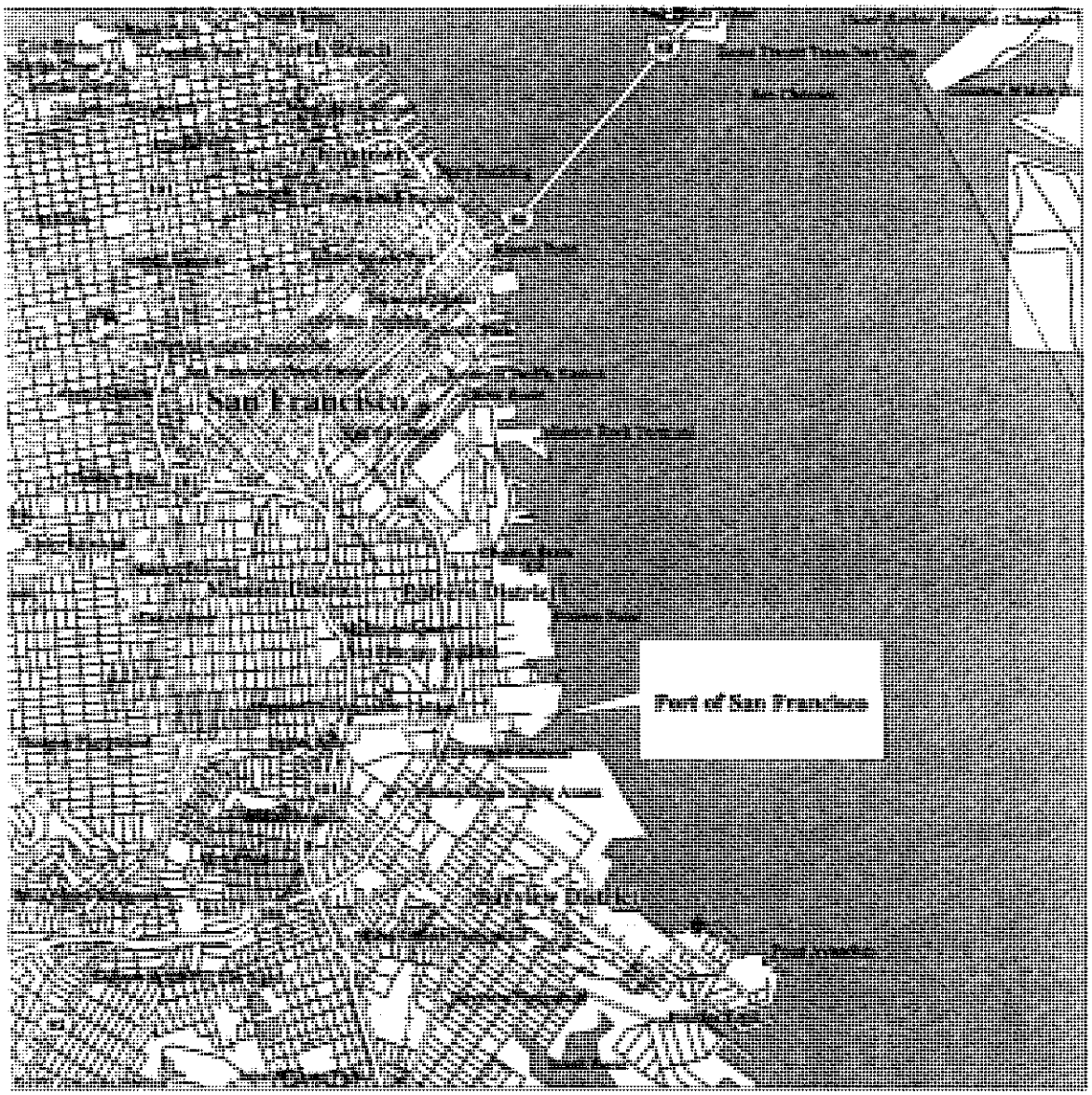
San Francisco is a deepwater port stretching approximately 12 km (7.5 mi) along the southern and western shore of the San Francisco Bay. The approach to San Francisco and down the east side of San Francisco is open; however, there are restricted navigational areas. There is also considerable traffic in the bay area, and there is a traffic separation scheme under U.S. Coast Guard traffic control (Mitchell, 1994). Depths of 13.7 m (45 ft), or more, are available from the Golden Gate Bridge to most of the anchorages. Depths up to 12.2 m (40 ft) are available to most piers, including those at the container facilities in the vicinity of Islais Creek. The wide passage from the ocean to San Francisco Bay is reduced to approximately 1.13 km (0.7 mi) at the Golden Gate Bridge pier. The distance from the Golden Gate Bridge to the entrance of facilities near Islais Creek is approximately 19 km (12 mi)(DOC, 1992b). A map of the port is shown in Figure D-51.

The Port of San Francisco is under the control of the City and County of San Francisco, to which it was transferred by the State in 1969. The "authority" reports to an appointed board of Harbor Commissioners. The port is a multi-terminal, multi-function harbor complex that the Authority operates as a Landlord owner. Services of the port range from cargo handling along the southern waterfront to ferry terminals and tourism services — including a cruise ship terminal, ferry plaza, Embarcadero, excursion boat terminals, Fisherman's Wharf, and aquatic park located on the central, northern, and western sides of the port (POSF, 1993).

Principal container handling facilities are located at North Terminal (Pier 80), operated by Metropolitan Stevedore Co., and South Terminal (Piers 94-96) operated by Stevedoring Services of America. Breakbulk general cargo is handled at Piers 27 - 29 in the northern Embarcadero section of the city. Total tonnage handled in calendar year 1991 amounted to 5,994,000 metric tons (6,607,200 tons) and included 223,676 20-ft equivalent units of containerized cargo (AAPA, 1993). With the 1994 loss of four major container lines to Oakland and the closing of the Naval Supply Center, the Port's Chief Wharfinger expected container traffic for the year to drop to about 50,000 20-ft equivalent units (Mitchell, 1994).

South Container Terminal: This terminal has a total area of 30.6 ha (75.6 acres), one container freight station, and four gantry-type container cranes. Dock/quay lengths for cargo ships at South Terminal are three berths totalling 747 m (2,450 ft) in length. Depths alongside dock/quay are 12.19 m (40 ft) at mean low water. North Terminal has three 40.6 metric ton (45 ton) rail-mounted container cranes and two 30.5 metric ton (34 ton) rail-mounted container cranes (Jane's, 1992; AAPA, 1993).

North Container Terminal: This terminal has a square-shaped, finger-type pier comprised of 27.74 ha (68.6 acres), of which 13 ha (32.1 acres) are laid out for container operations. This terminal has five container cranes and its own CFS. Dock/Quay lengths for cargo ships at this terminal are four berths at a total length of 1,552 m (5,092 ft). Depths alongside dock/quay are 12.2 m (40 ft) at mean low water. Crane capacities at South Terminal include four 30.5 metric ton (34 ton) rail-mounted container cranes (Jane's, 1992; AAPA, 1993; Mitchell, 1994).



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- Hospital
- Park
- Interstate, Turnpike
- US Highway
- Hwy Ramps
- Street, Road
- Major Street/Road
- Interstate Highway
- Railroad
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

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Mon Mar 27 09:38:12 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-51 Map of the Port of San Francisco, CA

Truck access to the container terminals is via Cargo Way and Third Street. Cargo Way connects South Terminal Piers 94 - 96 with 3rd Street. Entrance to the North Container Terminal (Pier 80) is at the intersection of 3rd Street and Army Street, which connects with I-280 and U.S. Highway 101 about 0.8 km (0.5 mi) from the entrance and about 1.6 km (1 mi) from the entrance to the South Container Terminal. These highways link up with the San Francisco/Oakland Bay Bridge (I-80) — the assumed route to Idaho National Engineering Laboratory and/or points east— which is roughly 1.6 to 2.4 km (1 to 1.5 mi) away. The Southern Pacific Railway serves both the North and South Container Terminals, and the Union Pacific also has tracks to the North Terminal's Pier 80. Trackage at South Terminal extends shipside parallel to the berth. Adjacent to the South Container Terminal is a 14.6 ha (36 acre) intermodal container transfer facility (Jane's, 1993; AAPA, 1993).

San Francisco has been served by a number of major container carriers. Lines calling at South Terminal include Grancolombiana and Evergreen. Liner companies using North Terminal include Blue Star Line, Central American Container Line, CSAV (Chilean Line), ELMA, Nedlloyd, NSCP, South Seas Shipping, and Splosna Plovba (Jane's, 1992; AAPA, 1993). However, in 1994, four of its five major container lines moved to Oakland (Adams, 1994; Mitchell, 1994).

Other Pertinent Information: Terminal security is the responsibility of the respective terminal operating companies. Facilities are fenced with controlled access and are patrolled by watchmen supplied by the International Longshoremen Workers Union. There are also City police officers permanently assigned for general port security (Mitchell, 1994). There are places within the container terminals for temporary segregation and storage of hazardous materials (Mitchell, 1993).

There are no regulations prohibiting the handling of containerized spent nuclear fuel. The port handles hazardous cargoes but, as far as known, has not handled spent nuclear fuel. The port allows Class A and B explosives in small amounts only (Mitchell, 1994).

All of San Francisco's marine terminals are located within the densely populated downtown area of the city and the large tourist population. Although there appears to be conflicting use of the Port of San Francisco's marine facilities (primarily attributable to its tourism business, much of which is centered to the north and west of the port's two container terminals) it is not deemed a major consideration. Terminal operators are responsible for accidents within their respective facilities. The Port Authority relies on the City of San Francisco's Fire Department hazardous materials team and the Coast Guard in case of an emergency. The City of San Francisco has a special Engine Company for responding to fires and other dangerous situations within port facilities, with about a five minute response time (Mitchell, 1993 and 1994). The Pacific Maritime Association handles hazardous materials instruction and training, and has just begun a program at the port (Mitchell, 1993). It is noted that U.S. Coast Guard statistics indicate that terminals in the San Francisco Bay have had only 31 reported collisions reported but an unusually high number of fires in recent years (21 fires reported between 1991 and 1993; the worst three-year fire record for major ports on the West Coast) (USCG, 1994b).

There are no known protected habitats or sanctuaries immediately near the terminals that might be affected by an accident in port. However, the predisposition of the City to severe earthquakes, and the high sensitivity of this area to protecting and maintaining environmental quality is considered a basis for concern. The city rests on the edge of the Pacific tectonic plate, while the opposite side of the Bay sits on the Continental plate. This results in the entire Bay area being a highly seismic zone. On April 18, 1906, San Francisco was the site of one of the largest recorded earthquakes in the contiguous United States, a Modified Mercalli Intensity XI (Bolt, 1978) due to movement along the fault line separating the two tectonic plates. The Uniform Building Code requires construction to withstand earthquakes and other severe natural phenomena (UBC, 1991). The Uniform Building Code requires construction for an

acceleration of 0.40 g, the highest seismic ranking in the United States. High winds have not been a problem for the Bay, with a Uniform Building Code minimum basic wind speeds up to 140 km/hr [70 miles per hour (mph)].

The climatic and environmental conditions of the Port of San Francisco are the same as those reported for the Port of Oakland in Section D.2.2.15.

The 1990 population within 16 km (10 mi) of the port terminals was 1,265,529. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,060,000; Oak Ridge Reservation, 766,000; Idaho National Engineering Laboratory, 348,000; Hanford Site, 339,000; and Nevada Test Site, 461,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,570 km (2,803 mi); Oak Ridge Reservation, 4,130 m (2,567 mi); Idaho National Engineering Laboratory, 1,560 m (970 mi); Hanford Site, 1,420 km (882 mi); and Nevada Test Site, 1,160 km (722 mi). Distances along rail routes are slightly longer.

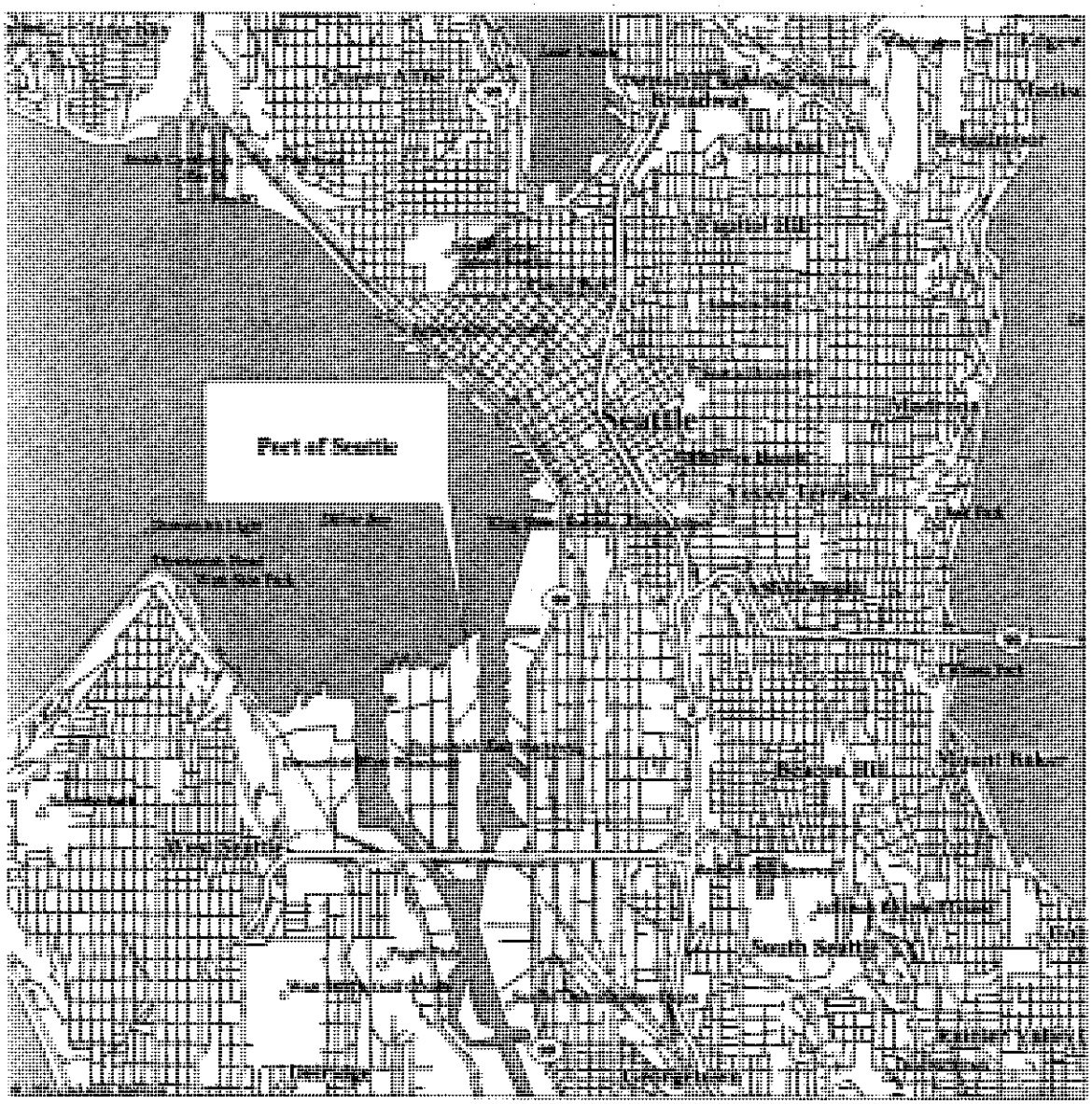
D.2.2.21 Seattle, WA

The Port of Seattle, WA, is located 230 km (143 mi) from the confluence of the Strait of Juan de Fuca and the Pacific Ocean. Seattle is located on Elliott Bay on the eastern shore of the Puget Sound, about 93 km (50 mi) south of the Strait of Juan de Fuca and about 5 km (3 mi) from the Sound. It is the largest and most important city in the Northwest, and one of the major ports on the Pacific Coast. Access from the Pacific Ocean is gained through the Strait of Juan de Fuca and Puget Sound. The transit from the Pacific Ocean to Seattle is open and considered relatively easy, with very deep waters during the entire approach to Seattle (DOC, 1992b). A map of the port is shown in Figure D-52.

The Port of Seattle is a large, diversified, multi-terminal port. Overall container tonnage for 1992 amounted to 7,510,000 metric tons (8,278,300 tons) and 1,155,000 20-ft equivalent units. It is managed by the Managing Director of the Marine Division and staff. Its facilities are municipally owned and leased to tenants (i.e., the Port Authority operates as a Landlord owner) (POS, 1994).

The port has five container terminals, of which two, Terminals 5 and 18, are considered public facilities:

Terminal 5: T5 is located on the West Waterway and is leased to and operated by American President Lines. Terminal 5 has a total area of 36 ha (89 acres), of which 24 ha (59 acres) can be used for container handling and storage. It has three container berths (Berths 4, 5, and 6), is equipped with six 50.8 metric ton (56 ton) Post-Panamax container cranes, and has two container freight stations. Terminal 5 has 760 m (2,500 ft) of marginal wharf, with 12.19 m (40 ft) of water alongside at mean low water. The terminal has good access to Interstate 5; about 3.8 km (2.4 mi) from the ramp to I-5 following a route entirely within the port's industrial district via North Marginal Way and West Seattle Freeway to South Spokane Street. I-5 is the principal north/south roadway linking Seattle with I-84 at Portland, OR (the assumed preferred, year-around route to Idaho National Engineering Laboratory) and/or I-90/82, which also links up with I-84 near Pendleton, OR. Terminal 5 is served by the Burlington Northern Railroad, whose tracks are located at the rear of the Terminal. The port is considering a proposal to provide Union Pacific service (Benham et al., 1994). Terminal 5 is served by major container lines including APL, OOCL, Star Shipping, and Westwood Shipping (Jane's, 1992; AAPA, 1993; D&B, 1993).



LEGEND

- Population Center
- State Route
- Geo Feature
- Major City
- Town, Small City
- Hospital
- Park
- Interstate, Turnpike
- Hwy Ramps
- Street, Road
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- Railroad
- Land Mass
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:39:02 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-52 Map of the Port of Seattle, WA

Terminal 18: T18 is located on the East Waterway (Berths 1 - 4, and 5 - 8), is operated by Stevedoring Services of America, and has a total area of 47 ha (116 acres) devoted to container handling and storage. It is also equipped with six, 40.6 metric ton (45 ton) container cranes and container freight stations. Terminal 18 has 1,844 m (6,049 ft) of marginal wharf, with 15.24 m (50 ft) of water alongside at mean low water. The terminal also has good access to Interstate 5; about 2.9 km (1.8 mi) from the ramp to I-5 following a route that is also entirely within the port's industrial district via South Spokane Street. Terminal 18 is also served by the Burlington Northern as well as the Union Pacific railroads via tracks along the wharf apron (i.e., ship-side).

Terminal 18 is served by several major container lines, including Barber Blue Sea, Grancolombiana Line, COSCO, d'Amico Line, Japan Line, Hyundai, Scindia Line, Chilean Line, ACL/CGM, and P&O Container Line (Jane's, 1992; AAPA, 1993; D&B, 1993; Benham et al., 1994).

Other Pertinent Information: There are potentially conflicting activities near the Terminal; petroleum products are pumped ashore at Terminal 5 (Berths 4 and 5), and across the East Waterway at Terminal 18 (Berths 2 and 3). The terminals are fenced with controlled access and guarded by watchmen on a 24-hour basis. There are areas within the container terminals for segregating hazardous materials cargoes.

The port's Emergency Response Plan relies on the City of Seattle Fire Department for hazardous materials response, with a technical support team including spent nuclear fuel handling experts from the DOE Hanford Site hazardous materials training for port workers is the responsibility of the individual terminal operators (Benham and Schuler, 1993; Benham et al., 1994). As noted in the accident information for Tacoma, the overall ship accident rates in the Puget Sound for the 1991-1993 reporting period are relatively low (USCG, 1994b).

Seattle's container terminals are somewhat separated from the City, which is generally north-east of the terminals. As already noted, these terminals have good access to Interstate highways without passing through congested city streets. However, T5 and T18 are both relatively close to some residential areas in West Seattle.

According to Mr. Schuler, Port Safety Officer, a port Commission resolution banning spent nuclear fuel shipments from the Port of Seattle has been in place for 3 or 4 years (Benham and Schuler, 1993). Reportedly, the Commissioners felt the Federal government was unresponsive to their requests for information concerning material being shipped and decided to ban further spent nuclear fuel shipment. As a result, the port no longer handles spent nuclear fuel and does not want it passing through its facilities. Since discussing this issue with the port safety official, DOE was informed that the Seattle City Council passed a resolution on December 8, 1993, which states the City's position that "high-level nuclear wastes should not be moved through Seattle or the Puget Sound by water or land transportation" (Noland, 1994). This issue is addressed in Section 6.5 of the EIS. Port officials (Benham and Schuler, 1993) thought the port had some prior experience with handling spent nuclear fuel, but this was not confirmed by available data going back to 1979 (NRC, 1993; SNL, 1994).

There are no known particularly environmentally sensitive areas (e.g., such as wildlife sanctuaries) in the immediate area of the terminals, but there is extreme public environmental sensitivity to potential environmental damage to the Puget Sound area.

The entire Puget Sound area is subject to severe earthquakes and volcanism. There have been two major earthquakes in the Puget Sound area this century; a Modified Mercalli Intensity VIII on April 13, 1949 and a Modified Mercalli Intensity VII-VIII on April 29, 1965 (Bott, 1978). On May 18, 1982, Mount Saint Helens suffered a major volcanic eruption (IPA, 1993). All the mountains along the Cascades Range,

from Canada to Northern California, are volcanic in origin and are potentially active (Foster, 1971; Hamilton, 1976; IPA, 1993). The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Seattle, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a high seismic zone with an acceleration of 0.30 g.

The 1990 population within 16 km (10 mi) of the port terminals was 753,296. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 565,000; Oak Ridge Reservation, 395,000; Idaho National Engineering Laboratory, 122,000; Hanford Site, 62,900; and Nevada Test Site, 344,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,670 km (2,900 mi); Oak Ridge Reservation, 4,240 km (2,636 mi); Idaho National Engineering Laboratory, 1,280 km (793 mi); Hanford Site, 360 km (226 mi); and Nevada Test Site, 2,130 km (1,322 mi). Distances along rail routes are slightly longer.

Environmental Conditions

A variety of aquatic species can be found in Puget Sound. Several animal species with special status may also be found in this area. A variety of marine mammals can be found in the central Puget Sound, including the Pacific harbor seal, California sea lion, killer whale, Dall porpoise, and harbor porpoise. In 1991, the U.S. National Marine Fisheries Services reported that the following endangered and/or threatened species may occur in the Puget Sound: the endangered gray whale, the endangered humpback whale, the threatened Stellar sea lion, and the endangered leatherback sea turtle (DOE, 1995). These species are not reported at the port. The U.S. Fish and Wildlife Service reported that the bald eagle and marbled murrelet, both listed protected species, may occur in the vicinity of the port (Frederick, 1994). Bald Eagles can be found throughout this coastal zone and American peregrine falcons are uncommon winter visitors (FWS, 1981a). The FWS's Ecological Inventory for the Puget Sound area indicates that the habitat of Elliott Bay is used by a variety of birds, including: shorebirds, gulls, sandpipers, turnstones, plovers, yellowlegs, herons, rails, great blue heron, waterfowl, loons, grebes, swans, geese, dabbling ducks, diving ducks, mergansers, American widgeon, pintail, mallard, seabirds, cormorants, alcids, common murre, and the pigeon guillemot. Adult concentrations of all of these species may be found in the Bay. Some of these species may also use this area as an overwintering area, a migratory area, and/or a nesting area (FWS, 1981a). It is also indicated that adult concentrations of Chinook salmon, coho salmon, and chum salmon are found in the West Waterway and Duwamish Waterway and use these water bodies and upstream segments as migratory and nursery areas.

According to the State of Washington's Department of Wildlife, the California sea lion uses the waters in the vicinity of Harbor Island as "haulouts" (i.e., areas regularly used by marine mammals for resting). Several seabird colonies also exist in this general area. There is a general lack of wetlands along the southeastern shore of Elliott Bay and along the East and West Waterways and the Duwamish Waterway (WDW, 1994a).

Climatic Conditions

The Strait of Juan de Fuca separates the northern coast of the State of Washington and the southern shore of Vancouver Island, Canada. Also in this general vicinity is the Port of Tacoma, Washington which is located 263 km (142 nautical mi) from the confluence of the ocean and the strait.

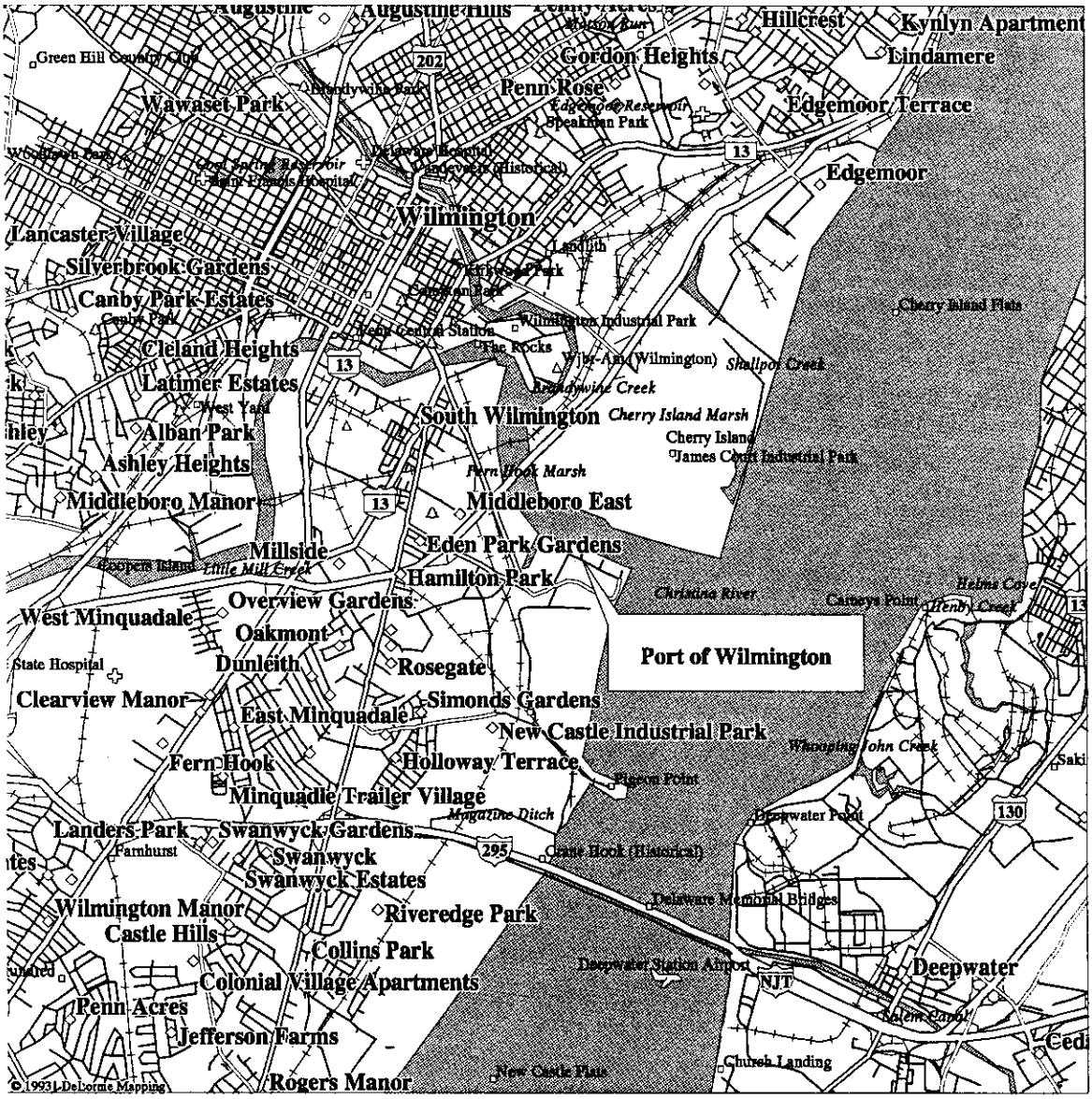
The city of Seattle is situated on a low ridge lying between Puget Sound on the west and the Green River valley on the east. The Olympic Mountains, which rise steeply from the Puget Sound are located approximately 80 km (50 mi) to the northwest. The mild climate of the Pacific Coast is modified by the Cascade Mountains and to a lesser extent by the Olympic Mountains. The climate is characterized by mild temperatures, a well-defined rainy season and prolonged cloud cover, especially during the winter months. The Cascades act as a very effective barrier in both winter and summer, shielding the region from both extreme cold and heat, respectively. The rainy season extends from October through March, with December accounting for the most rainfall. Approximately 75 percent of the annual total precipitation occurs during the winter rainy season. The dry season is centered around July and August. The majority of Seattle's precipitation is associated with normal, mid-latitude disturbances, which are most vigorous during the winter months. During summer, the dominant storm track (e.g., the polar jet) shifts northward into southern Canada, reducing the precipitation in the area. Summer thunderstorms do occur but do not contribute measurably to the annual rainfall budget. Prevailing winds are from the southwest, but occasional severe winter storms will produce strong northerly winds. Summer winds are generally rather light, with the occasional evidence of land-sea breeze effects creating northerly flows. Fog and low-level stratocumulus clouds form over the southern Puget Sound area in the late summer, fall, and early winter months, and often dominate the weather conditions of the early morning hours, reducing surface visibilities. Based on the 1951-1980 climatology, the first occurrence of freezing temperatures should occur around November 11, and the last incidence in spring around March 24 (NOAA, 1992g).

D.2.2.22 Wilmington, DE

The city of Wilmington, DE, sited on the Christina River, has large manufacturing interests. Both sides of the river at the city are lined with wharves that primarily support barge traffic. Deepwater facilities are located at the Port of Wilmington on the south side of the Christina River. The port is located about 3 km (2 mi) north of the Delaware Memorial Bridge on the left ascending bank of the Delaware River, approximately 100 km (62 mi) above the entrance to the Delaware Capes. The port is south of the city of Wilmington and is situated in an area of heavy industrial usage, which appears to be remote from residential, light business, and manufacturing areas (DOC, 1993c). A map of the city is shown in Figure D-53.

Access to the Port of Wilmington is gained via the Delaware Bay and Delaware River. The bay has natural depths of 15.4 m (50 ft) or more for a distance of 8 km (5 mi) from the entrance. A Federal project provides depths of 12.2 m (40 ft) past the entrance to the Christina River where the project depth is 10.6 m (35 ft). A traffic separation scheme has been established off the entrance of the Delaware Bay because of restrictions on passage through the bay and on up the Delaware River. Ships travelling to Wilmington must pass under the Delaware Memorial Bridge (DOC, 1993c).

The port is owned by the City of Wilmington. It is an "operating" port with stevedoring handled by two outside stevedoring companies. Principal cargoes are imported automobiles, dry bulk, roll-on/roll-off and refrigerated containers (primarily bananas and other tropical fruit) (POW, 1994). In 1993, the port handled about 936,000 metric tons (1,026,397 tons) of containerized cargo (about 100,000 20-ft equivalent units; AAPA, 1994). The port has 10,218 m^2 (110,000 ft^2) of chill/heat space and 36,806 m^3 (1,300,000 ft^3) of chill/freeze warehouse space. The terminal has two multi-purpose container cranes and one bulk cargo gantry crane. The marginal wharf area is 1,158 m (3,800 ft) long and there is a 155 m (510 ft) long floating roll-on/roll-off berth. Depth alongside the terminal at mean low water ranges from 11.58 m (38 ft) to 10.67 m (35 ft) due to silting. The port is equipped with one 40.6 metric ton (45 ton) multi-purpose container crane, one 29.1 metric ton (32 ton) multi-purpose container crane, and one 11 m^3 (14 yd^3) Clyde gantry crane (AAPA, 1993; Jane's, 1992; POW, 1994). Approximately half of the cargo going in and out



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- Hill
- Park
- US Highway
- County Boundary
- Hwy Ramps
- Street, Road
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Open Water

Scale 1:50,000 (at center)
5000 Feet
1000 Meters

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"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-53 Map of the Port of Wilmington, DE

of the port is food. Improved tropical fruit comprised 35 percent of the traffic in the port, while manufactured food products and finished perishables make up another 15 percent of the cargo traffic. Forest products handle 15 percent of the cargo traffic and imported steel makes up 7 percent. Several bulk commodities that are nonhazardous are the remaining 28 percent of the cargo handled by the port (Brooks, 1994).

The Port of Wilmington has direct access to I-495, a connector to I-95, which appears to be less than 1.6 km (1 mi) from the port and runs through the industrial district surrounding the Terminal. The Conrail and CSX railroads serve the port; it is not known if direct ship/rail transfer is possible.

Other Pertinent Information: Security of the general cargo terminals is maintained by the port police on a 24-hour basis. The wharves are fenced and truck access is via controlled terminal entrances. There is presently no place within the port for segregation and temporary storage of hazardous cargoes. There are no general cargo container lines currently serving the port, and there is no commercial container facility.

According to the Sandia National Laboratories' Radioactive Materials Postnotification Database, the port has not handled spent nuclear fuel since October 1984, when the database was initiated (SNL, 1994). There are no known conflicts with other hazardous materials in the immediate container terminal area. There are, however, chemical plants near the port, as well as a diversity of marine terminals and heavy tanker traffic (ship and barge) on the Delaware River. Other than increased risk of collision, these conflicts are not considered a major factor.

The port relies on the City of Wilmington's fire department for response in the event of a terminal hazardous materials accident. The port claims there is no hazardous materials training program and avoids handling hazardous materials (Casper, 1993).

There are no known protected habitats or sanctuaries near the port. However, at the mouth of the Christina River near the location of the port, there are extensive wetlands along the banks of the Delaware River.

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Wilmington, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 381,502. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 359,000; Oak Ridge Reservation, 297,000; Idaho National Engineering Laboratory, 535,000; Hanford Site, 584,000; and Nevada Test Site, 718,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,120 km (697 mi); Oak Ridge Reservation, 1,040 km (645 mi); Idaho National Engineering Laboratory, 3,890 km (2,416 mi); Hanford Site, 4,560 km (2,832 mi); and Nevada Test Site, 4,160 km (2,588 mi). Distances along rail routes are about the same.

Environmental Conditions

The Port of Wilmington is located within Zone 5 (Delaware Estuary/Bay) of the Delaware River. Protected water uses for Zone 5, which encompasses River Mi (RM) 48-79, are water supply (industry), wildlife, resident fish propagation and maintenance, anadromous fish passage, primary contact, and navigation (DRBC, 1994). However, within Zone 5, fish and other aquatic life are currently impacted due

to low dissolved oxygen levels from point and nonpoint source discharges. Further south in Delaware Bay (Zone 6), shellfish consumption is an impaired use due to bacterial infestations from local point and nonpoint sources.

The Delaware River at Wilmington is classified as a low salinity estuarine (generally 0.5 to 5 ppt) and tidal freshwater habitat. Aquatic organisms typically found in the waters of this area include: American shad, atlantic sturgeon, American eel, blueback herring, shad, alewife, white catfish, brown bullhead, perch, striped bass, bluegill, crappie, pumpkinseed, largemouth bass, carp, and chain pickerel (FWS, 1980f). In addition, the Delaware River is used as a migratory area by the shortnose sturgeon, a Federally listed endangered species. South of Wilmington, the shoreline of the Delaware River becomes much less developed and numerous fish and wildlife management areas and wetlands are found along the lower Delaware River and Bay. Bald eagles are found throughout these areas. The Delaware Bay supports high densities of geese and ducks along the shores. Waterfowl, particularly loons and grebes, and seabirds, particularly gannet, Wilson's petrel, and greater shearwater are also found in the Bay area. Osprey, peregrine falcon, and Cooper's hawk migrate in fall along the Delaware Bay to Cape May Point. High densities of whitetail deer also occur along the shore of the Bay (FWS, 1980f).

The Delaware Natural Heritage Inventory reported that "there are no Species of Special Concern within 0.8 km (0.5 mi) of the Port of Wilmington" (Dalton, 1994). However, the Inventory reported that the Peregrine falcon (Federally listed endangered) has nested 2.4 km (1.5 mi) from the port, and the short-nosed sturgeon (Federally listed endangered) is found in the Delaware River. It was the opinion of the Inventory that "these species would not be affected by normal operations at the port."

Climatic Conditions

Geographically, Delaware is part of the Atlantic Coastal Plain, which consists mainly of flat lowland and marshes. Small streams and tidal estuaries comprise the major drainage systems for the State. The Delaware River, Delaware Bay, and Atlantic Ocean form the eastern border of the State, while the Chesapeake Bay is the western boundary [approximately 56 km (35 mi) to the west]. These large water bodies contribute significantly to the climate of the Wilmington, DE, region.

Generally, summers are warm and humid and winters are considered rather mild. Summer temperatures rarely exceed 100°F, and average daily temperature during January (the coldest month) is 32°F. The majority of winter precipitation falls as rain, but precipitation during the winter months is often mixed rain, snow, and sleet. However, frozen precipitation rarely remains on the ground more than a few days. The proximity of the water masses causes humidity to remain relatively high year-round, which causes frequent fog events. Light southeasterly winds (e.g., off the Delaware Bay) tend to be most favorable for fog formation, while north-northeast winds tend to transport industrial pollution from the Philadelphia metropolitan area into the region. Rainfall distribution is fairly uniform throughout the year, but the greatest amounts normally come during the summer months in the form of thunderstorms. During the fall, winter, and spring seasons, the majority of rainfall is associated with extratropical and tropical cyclones track along the eastern seaboard of the United States. Hurricane-force winds are rarely experienced in the Wilmington, DE, region. However, strong south and southeasterly winds can cause high tides in Delaware Bay and the Delaware River, causing lowland flooding and damage to bayfront and riverfront properties (NOAA, 1992k).

D.3 Main Routes

The routes selected for potential marine transport are discussed in Appendix C. These routes cover the transport of the spent nuclear fuel from the country of origin to the first port of call in the United States. In the port incident-free and accident analysis it has been assumed that the vessel carrying the spent nuclear fuel would not unload the material at its first port of call. Intermediate port calls have been assumed in the analysis. In the marine impact accident and incident-free analysis, the intermediate port calls result in additional travel time which has been incorporated into both analyses. In the port analysis, this results in additional workers who could be affected by incident-free impacts and additional locations where accidents could occur. Due to the large variability associated with the movement of the vessel between U.S. ports, no specific route has been identified for use in the analysis. With the approach used in this analysis, the specific routes used between the U.S. ports would not affect the results of the risk assessment.

D.4 Accident-Free Impacts: Methods and Results

D.4.1 Introduction

This section of the appendix provides an overview of the approach used to assess the risks associated with port activities involved in transferring the spent nuclear fuel from the vessel to a vehicle for transport to the management site. Included here is a discussion of the incident-free risk assessment methodology and the results of the analyses, including an assessment of the cumulative risk associated with the marine transportation of the foreign research reactor spent nuclear fuel through U.S. ports.

The risk assessment results are presented in terms of a per shipment risk, annual risks from incident-free transport, as well as for the total risks associated with the program.

D.4.2 Scope

All foreign research reactor spent nuclear fuel shipments that would require ocean transport are expected to occur via one of four types of vessels: container ships, roll-on/roll-off vessels, general cargo (breakbulk) vessels, or purpose built vessels. In the incident-free analysis, it has been assumed that all shipments are made on either a breakbulk or a container vessel, an assumption intended to provide bounding assessments of the risks associated with port activities required for the transfer of spent nuclear fuel.

D.4.2.1 Nonradiological Risk of Marine Transportation Related Activities

This portion of the risk assessment is limited to estimating the human health risks incurred during spent nuclear fuel unloading and handling during port operations at U.S. ports and during the vessel's approach to the port and movement within the port. The nonradiological risks from these activities were assessed as resulting in a negligible impact on the health of the public and workers. Approximately 56,000 port calls involving vessels engaged in foreign trade are made at U.S. ports every year (DOC, 1994). As discussed in Appendix C, each of these vessels has the capacity to carry hundreds of pieces of cargo of the size of a container carrying a spent nuclear fuel transportation cask (typically, container vessels carry between 800-1,000 containers, while some carry many more). This translates to millions of pieces of cargo every year. To fulfill the needs of the basic implementation of Management Alternative 1 of the proposed action, less than 60 transportation casks would need to be shipped per year. This is less than 0.001 percent

of the total number of pieces of cargo (originating in foreign countries) to be handled at U.S. ports each year. The limited number of shipments per year should not result in a significant change to the risks to the public including the port workers.

D.4.2.2 Radiological Risks of Marine Transportation

The risks that result from the radioactive nature of the shipments are addressed for both incident-free transportation and accident conditions. The radiological risks associated with the incident-free shipping conditions result from the potential exposure of members of the crew and dock workers to external radiation in the vicinity of the packaged fuel. No other exposure is considered, due to the relative isolation of the material from the general public during all phases of the port activities associated with the transfer of the spent nuclear fuel from the ocean going vessel to the overland transportation mode.

All radiologically-related impacts are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent, which is the sum of the effective dose equivalent (EDE) from the external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure. The EDE is the sum of the tissue and organ weighted dose equivalents for all irradiated tissues and organs. The committed effective dose equivalent considers the initial exposure and the effects of radioactive decay and elimination of the radionuclide through ordinary metabolic processes over the 50-year period. Radiation doses are presented in units of person-rem for collective population and rem for individuals. The impacts are further expressed as health risks, primarily in terms of latent cancer fatalities (LCF). The health risk conversion factors were derived from International Commission of Radiological Protection Publication 60 (ICRP, 1991).

D.4.3 Port Facility Operations

This section describes the principal activities that are performed at a port facility to transfer a radioactive material package ("cask") from an ocean vessel to a surface conveyance, such as a truck trailer or railcar. The purpose of this description is to assist in establishing an estimate of the ionizing radiation dose to personnel that could be associated with the port intermodal transfer. The description of activities, and estimates of durations of specific tasks and personnel requirements is presented later in this section.

The off-load operation would take place at a "facility of particular hazard," as defined in 33 CFR 126.05, that is designated by the Captain of the Port. The Captain of the Port is a U.S. Coast Guard officer that enforces, within his/her respective port, safety, security and marine environmental protection regulations. These include, without limitation, regulations for the protection and security of vessels, harbors, and waterfront facilities; anchorages; security of vessels; waterfront facilities; security zones; safety zones; regulated navigation areas; deepwater ports; water pollution; and ports and waterway safety. The Captain of the Port designates and permits "facilities of particular hazard."

Such a facility is allowed to handle "cargoes of particular hazard" including "highway route controlled quantities of radioactive material," which includes spent nuclear fuel. The Captain of the Port could establish a safety zone or security zone around the vessel, if necessary. These zones would prohibit unauthorized personnel from entering the area. Usually a "facility of particular hazard" will have a secured area onsite for the storage of "cargoes of particular hazard." This facility would be used for the temporary storage of spent nuclear fuel, if necessary. Usually, these cargoes are loaded on a truck or train that departs for its destination soon after being checked by a facility employee and inspected by the proper authorities.

Each "facility of particular hazard" has an operations manual that outlines procedures for handling "cargoes of particular hazard," the personnel used and their qualifications, emergency procedures, and contact numbers. Only the Captain of the Port can approve the required operations manual, and only the Captain of the Port can approve any changes made to the operations manual. The content of the operations manuals can vary by port location and size, and by the type of materials handled. The operations manual of the facility under consideration for off-load operations should be studied prior to receipt of any spent nuclear fuel.

D.4.3.1 Intermodal Transfers

The intermodal transfer of the container (or cask) is largely a mechanical lifting operation with somewhat limited personnel participation. Unloading of vessels is generally performed by members of the International Brotherhood of Longshoremen (East Coast and Gulf Ports), or the International Longshoremen and Warehouseman Union ("Longshoreman") (West Coast ports), sometimes with support from the vessel's crew.

There are various configurations of container (or cask) storage aboard ship that could arise. However, as a preference, containers (or casks) are transported below decks. The following sections describe the principal operations that must occur to achieve both transfer of the container (or cask) from the ship, and to prepare it for departure from the port. It should be noted that as a general rule, departure from the port occurs as soon as is practicable, since the intermodal transfer is merely part of an "in progress" transportation activity, and radioactive materials transport should be expeditious. Infrequently, containers (or casks) may be (temporarily) stored at port facilities for some reason, such as bad weather.

D.4.3.1.1 Container Transfer to Truck Trailer or Railcar

If the port routinely receives containerized freight, it will be equipped with a crane adapted to handle containers. These cranes use a spreader bar equipped with International Standards Organization twistlocks at each of its four corners. The length of the spreader bar is automatically adjustable to accommodate the two International Standards Organization standard container lengths of 6.1 m (20 ft) or 12.2 m (40 ft). Casks are normally shipped in the 6.1-m (20-ft) containers. The twistlocks mate with standard fittings in the corner posts of the container, and are automatically actuated by the crane operator to attach the spreader bar to the container. Typically, no personnel are on the container when the spreader bar is attached. Engagement can be verified by the crane operator or, depending on the container stacking arrangement or port practice, by Longshoremen on the deck. The crane operator is in an enclosed cabin and is usually separated by a considerable distance from the cargo. The procedures described below apply to so-called cellular container ships or combination container/breakbulk ships.

Once engaged, the container is lifted from the hold of the ship, up and over the side to a container trailer, or railcar, on the dock. Engaging the container and moving it to the transporter, takes about 1.5 minutes on average (about 45 containers per hour).

The routine unloading is to install the container on a standard over-the-road container trailer which is pulled by a specially made tractor used at ports. These dock tractors have a single person cab and a hydraulically driven "fifth-wheel" which is used to raise the front end of the container trailer much higher than it would be for regular transport. This allows the Longshoremen to move the container trailer without having to raise and lower the trailer front landing gear at each re-positioning of the trailer. The dock tractor then moves the container to a freight staging area, parks it, connects to an empty container trailer, and re-positions under the container crane. Usually, several dock tractors are used to continuously move containers from "under the hook." Dock tractors are not suitable for over-the-road use.

The receiver (or the agent for the receiver) generally arranges with the Longshoremen to install the cask container directly onto the container trailer, or railcar, which will be used for overland transport, and which has already been inspected. The container trailer will be pulled by the tractor which is to be used for transport.

If the containerized cask is placed on a dock container trailer, sometimes called a "bombcart," then it must be later moved to the trailer which is to be used for transport. This transfer can be made using a large, industrial fork lift, top lift, or a small mobile crane ("forklift") specifically designed to move containers in the port freight staging areas. A bombcart is a special container trailer, used only within the port facility, that does not have twistlocks at its four corners to secure the container being loaded or unloaded.

Spotting the container on its designated trailer (or railcar) and securing it using the trailer mounted International Standards Organization locks, requires two (2), or four (4) longshoremen (at each end of the trailer) and takes about 30 seconds. Four (4) longshoremen have been used for this task at some ports. Once the container has been loaded onto its trailer, it moves immediately away from the container unloading area to a staging area so that ship unloading can continue. The staging area is established by port authorities, but must be approved by the Captain of the Port.

The staging area is usually close to the container unloading area, on the port property, and may be an area where hazardous materials are routinely handled. It may be an indoor location, such as a warehouse. It is used for the conduct of any inspections or surveys that may be desired, to verify documentation received from the ship's captain, to verify marks and labels on the containers, to verify securement of the load, to assemble required documentation for the overland portion of the transport, and install or verify placards. (It should be noted that foreign origin shipments are prepared in accordance with International Atomic Energy Agency standards, which are generally compatible with NRC and the Department of Transportation regulations. In accordance with International Atomic Energy Agency regulations, containers usually are prepared with an oversized label, which is an International Atomic Energy Agency permitted substitute for placards. Even if placarded, the placards usually do not conform to the "Highway Route Controlled Quantity" placard used for these types of shipments in the United States. The overland portion of the transport leaves from this area. Inspections are described in Section D.4.4.

The National Defense Authorization Act for Fiscal Year 1994 requires that, to the extent practicable, casks containing spent nuclear fuel should be moved expeditiously from the port. However, infrequently, continuation of the transport may not occur immediately. This may be due to unplanned events such as severe weather, equipment breakdown or inspection discrepancy, or to planned actions such as queuing of the receipt of individual containers at the receiving site. If one or more containers must remain at the port, they are normally moved to a bonded warehouse, with the container remaining on its transporter. The warehouse is considered a secure area, and it typically meets the requirements of a "safe haven."

Specific handling for rail shipments depends upon the location of rail track with respect to the container handling crane "foot print." If the rail line is within the foot print, then containers are loaded directly onto the railcars and secured using International Standards Organization locks in the deck of the railcar. Typically, two containers are loaded onto each railcar. If the rail line is not in the foot print, then the container is loaded onto a dock container trailer and moved to the rail line. An industrial forklift is used to transfer the container to the railcar. Railcars may be moved by a switch engine, but more commonly, a railcar tugger is used.

For spent fuel shipments, the railcars carrying loaded containers are separated from each other by buffer cars. These cars are usually empty gondolas or flat cars. A caboose is usually provided for escorts and required security equipment. The buffer cars are selected so that the escorts can have a good view of the container cars. Containers mounted on container trailers are not shipped on the railcar in a "piggyback" configuration because of concerns related to the resulting high center of gravity.

D.4.3.1.2 Container Transfer Using Jib-Type Cranes

The port may not have a container crane and instead rely on a dockside, pedestal mounted, or ship installed, jib crane. Containers are moved using this type of crane by attaching a four-legged sling to the crane hook, and extending one leg of the sling to each of the four corners of the container. The sling must be manually attached to (and later removed from) the International Standards Organization fitting at the top of the corner posts of the container. The attachment and removal is done by two longshoremen, who must climb on top of the container.

The attachment of the sling can take as long as three minutes. The reason for this is that, typically, the longshoremen climb onto the container before the crane operator has positioned the crane and lower the sling for attachment. The longshoremen also provide hand signals to direct the positioning for the crane. Disconnecting the sling from the container is done more quickly, and it is usually not necessary to climb onto the top of the container. Two longshoremen usually lock the container to the container trailer and disconnect the sling, but sometimes four are used.

If the ship is equipped with a jib crane, it may also be used to remove containers. The process is the same as with a dock mounted crane, but the crane is operated by a member of the ships crew. Except for the operation of a ship mounted crane, members of the ships crew do not generally have a role in the unloading of the ship.

D.4.3.1.3 Roll-on/Roll-off Operations

In the roll-on/roll-off configuration the casks (either containerized, freestanding, or palletized) are already on the trailer that is used for overland transport. After unlashing, the trailer is moved to the staging area by a longshoreman using a dock tractor.

Unlashing of the trailer may involve up to four longshoremen, and require up to 5 minutes. Transfer of the trailer to the staging area can take as long as 15 minutes depending on the ship's hold and ramp conditions and the distance to the staging area. After the trailer is spotted in the staging area it is connected to the tractor that is used for over-the-road transport.

Since the trailer has not been available for inspection, if an inspection is required [other than that done by the tractor driver(s)], it is performed at the staging area. If the trailer is foreign owned, temporary apportioned motor vehicle tags are provided by the receiver or receivers' agent.

D.4.3.1.4 General Cargo Operations

Breakbulk operations could involve either a containerized or free standing cask. Typically, a free standing cask is mounted on a pallet to facilitate the handling of the cask using the cranes and winches commonly found on ships and at dock side. Handling of a containerized cask would follow the same operation described in Section D.4.3.1.1.

Breakbulk cargo handling of a free standing cask is more labor intensive, since the cask must be unlashed from the deck and may have to be moved using winches to a hatch opening. A crane is used to lift the cask out of the hold and onto the dock. Up to 4 longshoremen may be used to move the cask in the hold and attach crane rigging to the cask or pallet. Two (2) or more longshoremen may be required to complete the transfer to the dock. At the dock, the pallet is typically placed on a standard flat bed trailer and secured with chains or other binders. Total handling time is less if the cask is transported in the center of the hold, as it likely would be if a chartered vessel were used.

In general, breakbulk cargo requires the longest unloading times, compared to containerized freight and roll-on/roll-off configurations. While a good unloading time for general cargo is about 5 minutes per crane load, radioactive materials transfer can take as long as 20 minutes if the cask is not transported on a pallet and must be rigged separately.

Breakbulk shipment of free standing spent nuclear fuel casks is perceived to result in a somewhat less reliable tiedown of the cask to the deck of the vessel. There is also an increased risk of damage to the cask or its pallet due to the variability in lift fixtures on each pallet. For these reasons breakbulk shipments of spent fuel casks have not been routinely made since the mid 1970's. This mode of shipment is not expected to be routinely used for the transport of spent nuclear fuel, except as it would apply to the use of purpose-built ships.

D.4.3.2 Key Intermodal Tasks and Task Durations

This section summarizes the key intermodal handling tasks, and estimates the personnel requirements and task durations for the transfer of the casks from the vessel to the land conveyance. These summaries are based on the narratives presented previously. Actual handling times and resource requirements can be widely variable, depending in large degree upon the cask configuration, transport vessel, intermodal handling equipment, port practice, and specific procedures which could be implemented for a given shipment or shipping program.

Port inspections are described separately in Section D.4.4.

D.4.3.2.1 Intermodal Handling of Containerized Casks

Ports equipped for intermodal handling of containers have achieved average rates of transfer of general cargo containers between the vessel and dock of 45 per hour, or about one container each 80 seconds. This rate may not be achieved for containers carrying spent nuclear fuel. For conservatism, a transfer time of 2 minutes per container is assumed. Longer transfer times would be expected if the port is not equipped with container cranes. A transfer time of 3 minutes is assumed if jib or boom type cranes are used with slings to lift the containers. Containers are assumed to be installed on the container trailer which would be used in over-the-road transport.

Port practices, such as union rules and safety procedures, would dictate the number of personnel used to unlash, transfer, and lash the container to its transporter. Consequently, the number of personnel required for each task could vary slightly between ports.

Each shipment, consisting of one or more containers, is expected to be observed by one or more persons who represent various interests in the shipment. These observers would have no active role in the transfer of the container, and would be expected to be 9.1 m (30 ft) or more away from the container.

Vessel crew members do not normally participate in container transfer operations, except for a member having responsibility for the cargo. Only this individual is considered to be present during transfer, stationed at the vessel hatch.

Table D-8 summarizes the handling of a container on a container ship. All of the distances are assumed to be from the container surface, or the projected container surface if an open container is used. There are no tasks which require contact with the cask surface.

Table D-8 Container Transfer Summary

Task / *Personnel/Location*	*Unlash Cargo*	*Attach to Crane*[a]	*Transfer to Dock*	*Lash to Transporter*	*Move to Staging*
0-.9 m (0-3 ft)	2 - 4[a,b]	1[b]	-	2 - 4[b]	-
duration (min)	0.25	0.5	-	0.25	-
1.5-3 m(5-10 ft)	-	-		-	-
duration (min)		-	-	-	-
3-6 m (10-20 ft)	-	-	-	1[c]	1[c]
duration (min)	-	-	-	0.25	3
6-9 m (20-30 ft)	1[b]	2[d]	1[d]	1[d]	-
duration (min)	0.25	0.5	1	0.1	-
9 m (30 ft)	1[e]	1[e]	1[e]	4[f]	4[f]
duration (min)	0.25	0.5	0.1	0.25	0.25

[a]*Crane attachment to containers is automated.*

[b]*longshoremen*

[c]*truck driver*

[d]*crane operator*

[e]*ships crew*

[f]*observer*

Containerized casks could be shipped aboard container or general cargo vessels. No significant difference in transfer times is expected between these vessel types.

D.4.3.2.2 Intermodal Handling of Roll-on, Roll-off Casks

Casks in a roll-on/roll-off configuration, either containerized or palletized are assumed to be transported on a roll-on/roll-off vessel and received at a port equipped to support roll-on/roll-off operations. Assumptions regarding port practices, observers and crew members are the same as those made for containerized or palletized cask transfer.

Removal of the trailered cask from the vessel is assumed to be done using a port tractor. Attachment of the trailer to the tractor which would be used for over-the-road transport must be done in the freight ready area, or the staging area.

All of the distances are assumed to be from the trailer or personnel barrier surface, or the projected trailer surface if there is no personnel barrier. There are no tasks which require contact with the cask surface.

Table D-9 summarizes the cask unloading and transfer activities for a roll-on/roll-off cargo vessel.

Transfer of roll-on/roll-off configured casks is not expected to occur on vessels not equipped with a ramp. Consequently, lifting of the trailered cask by crane is not expected to occur.

Table D-9 Roll-on/Roll-off Cask Transfer Summary

Task / *Personnel/Location*	*Unlash Cargo*	*Attach to Crane*	*Transfer to Dock*	*Lash to Transporter*	*Move to Staging*
0-.9 m (0-3 ft)	4[a]	2[a]	-	4[a]	-
duration (min)	4	0.5	-	0.5	-
1.5-3 m (5-10 ft)	-	-	1[b]	-	-
duration (min)	-	-	0.25	-	-
3-6 m (10-20 ft)	1[a]	1[b]	1[b]	2[b]	1[b]
duration (min)	4	0.5	2	0.5	3
6-9 m (20-30 ft)	1[c]	1[c]	1[c]	-	-
duration (min)	4	0.5	0.25	-	-
9 m (30 ft)	-	-	-	4[d]	4[d]
duration (min)	-	-	-	0.5	0.25

[a] *longshoremen*
[b] *truck driver*
[c] *ships crew*
[d] *observer*

D.4.3.2.3 Intermodal Handling of Free-Standing (Palletized) Casks

As previously noted, casks are expected to be mounted on a skid, cradle or pallet ("pallet") to facilitate handling, lifting, and stowage. Transfer of these casks is usually somewhat more labor intensive than handling containerized casks, since the pallets are not standardized. The pallets are usually uniquely designed to accommodate a specific cask. Consequently, more effort is usually required to secure the cask in stowage, and to install lift slings for transfer. In addition, some care is needed to ensure that lifting and handling operations do not damage the cask.

Assumptions regarding port practices, observers, and crew members are the same as those made for containerized cask transfer.

It is assumed that the palletized cask would be installed on a flat bed trailer not necessarily having the tiedown fixtures required to secure the pallet. Some additional effort is expected to be required to secure the pallet to a trailer, compared to that required for containerized casks. However, it is assumed that the pallet is placed on the trailer that would be used for over-the-road transport so that no subsequent transfer of the pallet is needed.

Table D-10 summarizes the palletized cask unloading and transfer activities for a breakbulk cargo vessel. Distances are from the edge of the pallet, or its projected edge. There are no tasks which require contact with the cask surface.

D.4.4 Port Inspection Activities

There are several agencies, both Federal and State that could make an inspection of the cargo at any point from when the vessel docked while the cargo is still on board, until the cargo reaches its final resting place in the facility. The U.S. Coast Guard has recently designated personnel to inspect hazardous cargoes, specifically containers laden with hazardous cargo. The U.S. Coast Guard, however, has no current programs in place for the training of inspectors of radioactive materials. This may change in the near future. The U.S. Coast Guard does have an aggressive program for container inspection and compliance. The U.S. Coast Guard would perform an inspection on the vessel, including all documentation (bills of

Table D-10 Palletized Cask Transfer Summary

Task / *Personnel/Location*	*Unlash Cargo*	*Move to Hatch*[a]	*Attach to Crane*	*Transfer to Dock*	*Lash*[b] *to Transporter*	*Move to Staging*
0-.9 m (0-3 ft)	4[c]	0 - 4[c]	2[c]	-	4[c]	-
duration (min)	4	0 - 5	0.5	-	4	-
1.5-3 m (5-10 ft)	-	-	-	-	-	-
duration (min)	-	-	-	-	-	-
3-6 m (10-20 ft)	-	-	-	-	-	1[d]
duration (min)	-	-	-	-	-	3
6-9 m (20-30 ft)	2[c,e]	0 - 2[c,e]	2[c,e]	1[c,e]	4[c,e]	4[f]
duration (min)	4	0 - 5	0.5	0.1	4	0.25
9 m (30 ft)	-	-	1[g]	1[g]	1[g]	-
duration (min)	-	-	0.5	2	0.5	-

[a]*this task is not required if the cask is in the center of the ships hold*

[b]*transporter is to be used for over-the-road transport*

[c]*longshoremen*

[d]*truck driver*

[e]*ships crew*

[f]*observer*

[g]*crane operator*

lading and dangerous cargo manifests) and container placarding. Once the cargo is off-loaded, NRC may require an inspection of the container or cask and perform a radiation survey. Also, state agencies that are designated with such responsibilities as safety and transportation may require an inspection, especially on the tractor and semi-trailer transporting the casks. These latter inspections could take place dockside, at the facility, at a staging area, or at the gate area of the port. It is also possible that there would not be any inspections made by any agency.

The principal kinds of inspections that normally occur are: (1) verification of container (or cask) marks and labels to the accompanying documentation; (2) verification of radiation readings around the container (or cask); and (3) inspection of the transport vehicle, typically a tractor-trailer rig. Other inspections, such as condition of a container, can also be performed. Most of the inspections performed are done at the staging area, although inspection on the ship is also possible.

Port inspections are discretionary in that there is no regulatory requirement that they be performed by any party, with two exceptions. One exception is that a radiation survey map must be prepared for overland transport by truck and rail. This map must show the radiation levels at 2 m (6.6 ft) from the container or cask, and it must show the radiation level in the truck normally occupied by the driver. The agent for the receiver normally completes this map. A second exception is that State laws may require a permit for the transport of the spent fuel. Typically, this permit requires an inspection of the transporter for road worthiness, and sometimes a review of other documents. Inspections of railcars are normally not done by state inspectors. The performance of additional inspections may be established by (local) policy, procedures, or preference. In this context, inspections may occur more than once. The reason for this is that Federal agencies, such as the Department of Transportation and the U.S. Coast Guard, and the States (and the port authority), have a right of inspection. For any given shipment or individual cask, those agencies may not be represented, and even if represented, the right of inspection may not be exercised.

The representative of the receiver normally verifies that the marks and labels of the container conform to the documentation supplied by the shipper, that radiation levels are within U.S. regulatory limits, and that they conform to the radiation survey documents supplied with the shipping papers. These verifications are usually made after the container is removed from the ship and is in place on its transporter. Surveys of the container can also be performed aboard ship. This may be done for example, if there was a belief that actual radiation readings could be higher than those reported in the shipment documentation because of some event that occurred in transit, or for information.

Inspections of the transport equipment may be required by the State. These inspections are normally done prior to loading of the container on the bed of the trailer or railcar. This ensures that the container is loaded on an acceptable transporter. There is no radiation exposure which is attributable to this inspection. Verification of container tiedown is performed by the truck driver, or rail crew, as required by current regulations. Typically, tiedowns are also verified by a representative of the consignee. Tasks and personnel requirements are summarized in Table D-11.

Table D-11 Summary of Inspection Tasks and Personnel Requirements Per Container[a]

		Federal Agencies[b]					
		USCG	*DOT*	*NRC*	*State*	*Local/Port*	*Receiver*
Container	Personnel	1	1	1	1	1	1
	Time (min)	5	2	2	2	5	5
Roll-on/Roll-off	Personnel	1	1[c]	1	1[c]	1	1
	Time (min)	2	15	10	15	5	5
Breakbulk	Personnel	1	1[c]	1	1[c]	1	1
	Time (min)	2	15	10	15	5	5

[a]*Personnel expected to be within 3 m (10 ft) of the container.*

[b]*Discretionary inspections which may be performed; USCG = U.S. Coast Guard, DOT = Department of Transportation.*

[c]*Includes trailer inspection.*

D.4.5 Port Worker Incident-Free Analysis Methodology

Incident-free impacts of the offloading of foreign research reactor spent nuclear fuel have been estimated for port workers, inspectors, and observers of the activity. It has been assumed that no member of the public, other than the above-mentioned workers, would be present at the port during offloading. Ports tend to be relatively large areas with little or no access by the general public. Impacts of the incident-free shipment of foreign research reactor spent nuclear fuel on the general public would not be expected until the shipment leaves the port area. It has also been assumed that all foreign research reactor spent nuclear fuel would be shipped in containers, regardless of whether transport occurs via container or general cargo vessels.

Once a shipment arrives in port, the spent nuclear fuel packages would be inspected by customs officials, U.S. Coast Guard personnel, port officials, etc. Up to six inspections performed by Federal, State, and local agencies, and the shipping agent are assumed to occur for each cask shipment. The durations of these inspections are provided in Table D-11. The assumption is made that the container is opened only for the inspection conducted when the cask is first off-loaded from the vessel.

In addition to the personnel involved in the inspections, there are other port workers (longshoremen, port officials, security personnel, etc.) who would be directly involved in or co-located near the off-loading of the container, its securing to the tractor-trailer, and in the movement of the container to a staging area. (The incident-free impact of offloading operations on the ship's crew were addressed in the marine impact analysis presented in Appendix C). While arrangements are expected to be made for the immediate departure of the spent nuclear fuel from the port of entry, it is recognized that situations could occur where there may be some delay in departing the port. For example, these delays could be caused by weather or road conditions. A delay of up to 24 hours is assumed for all shipments. To account for the impact of these delays, the dose to workers not directly involved in offloading activities was estimated. In addition to workers identified in Tables D-8 through D-9, it was assumed that 50 workers are exposed to the cask for 8 hours at a distance of 50 m (163 ft). This provides a dose estimate for the 24-hour storage period.

These dose estimates are independent of port location or type. Two types of cargo vessels have been addressed in the analysis, encompassing the range of times required for offloading activities. Container vessels required the least amount of time to offload; breakbulk vessels the longest. It has been assumed that offloading operations for both containerized breakbulk cargo and container cargo at all potential ports of entry is similar. These estimates are intended to bound the potential doses associated with port activities. As discussed above, breakbulk transport of the containerized fuel casks are expected to result in the largest dose to workers due to port operations due to the longer times associated with activities that bring workers into proximity of the casks.

External radiation for an intact shipping package must be below specified limits that control the exposure of the handling personnel and general public. These limits are set forth in 49 CFR 173.

The limit of interest established therein is a limit of 10 mrem per hour at any point 2 m (6.6 ft) from the vertical planes projected by the outer lateral surfaces of the transport vehicle. This limit is associated with an "exclusive-use" shipment, that is one in which no other cargo is loaded in the container used for the spent fuel transportation casks, not that the ship is an exclusive use vessel. All shipments within this program would be expected to fall within this category. In general, much of the foreign research reactor spent nuclear fuel potentially to be received would have cooled for a significant amount of time prior to shipment, resulting in external dose rates much less than the regulatory limit. Shipments of research reactor fuel in the past have had doses averaging approximately 2.3 mrem per hour at 1 m (3.3 ft) from the cask surface (see Section F.5 of Appendix F). Due to the scope of this program and the possibility that some of the fuel could be shipped fresher than has been done previously, the above cited regulatory limit has been used to estimate the worker exposures for all shipments. Appendix F, Section F.5, provides exposure rate versus distance for a transportation cask that is loaded with spent fuel that results in a dose rate at 2 m (6.6 ft) of 10 mrem per hour. This relationship was used to assign dose rates for the port activities.

Table D-12 and D-13 describe the types and numbers of personnel involved in the port activities associated with offloading the spent nuclear fuel. The times, distances, and maximum doses associated with these activities are listed for each type of personnel (all doses are simply the product of the dose rate to which the worker is exposed, based upon distance from the transportation cask, and the time the worker is exposed to this dose rate). The total port worker population and the maximally exposed individual doses are also provided. During incident-free port operations, the highest individual exposure would be to handlers and inspectors of the casks. Exposures are port-independent since it is assumed that operations would be similar at any of the potential or alternative ports of entry.

Table D-12 Port Worker Consequences from Shipment of Foreign Research Reactor Spent Nuclear Fuel on Breakbulk Vessels

Exposed Workers	*Exposure Distance (m)*	*Dose Rate (mrem/hr)*	*Exposure Time (minutes/cask)*	*Dose/Person/ Cask (mrem)*	*Exposed Workers*	*Collective Dose (Person-rem)*	*Individual Risk (LCF)*	*Collective Risk (LCF)*
Longshoreman A1	0.50	37[b]	0.25	0.15	2	0.00031	6.2E-08	1.2E-07
Longshoreman A2	0.50	37[b]	3.3	2.0	2	0.0040	8.0E-07	1.6E-06
Longshoreman A3	6.00	6.4[b]	0.25	0.027	1	0.000027	1.1E-08	1.1E-08
Longshoreman B1	0.50	34	1.0	0.57	4	0.0023	2.3E-07	9.1E-07
Maximum				2.0[a]			8.0E-07a	
Subtotal						0.0066		2.6E-06
Crane Operator 1	9.00	1.8	3.0	0.090	1	0.00009	3.6E-08	3.6E-08
Maximum				0.090[a]			3.6E-08[a]	
Subtotal						0.00009		3.6E-08
Truck Driver	3.00	7.1	3.0	0.36	1	0.00036	1.4E-07	1.4E-07
Maximum				0.36[a]			1.4E-07[a]	
Subtotal						0.00036		1.4E-07
Observers	6.00	3.2	0.25	0.013	4	0.000053	5.3E-09	2.1E-08
Observers	50	0.01	480	0.0802	50	0.0040	3.2E-08	1.6E-06
Maximum				0.080[a]			3.2E-08[a]	
Subtotal						0.0041		1.6E-06
USCG Inspector	1.5	15	2.0	0.5	1	0.00050	2.0E-07	2.0E-07
DOT Inspector	1.5	15	15	3.8	1	0.0038	1.5E-06	1.5E-06
NRC Inspector	1.5	15	10	2.5	1	0.0025	1.0E-06	1.0E-06
State Inspector	1.5	15	15	3.8	1	0.0038	1.5E-06	1.5E-06
Local/Port Inspector	1.5	15	5	1.3	1	0.0013	5.0E-07	5.0E-07
Receiver	1.5	15	5	1.3	1	0.0013	5.0E-07	5.0E-07
Maximum				3.8[a]			1.5E-06[a]	
Subtotal						0.013		5.2E-06
Maximum				3.8[a]			1.5E-06[a]	
Total						**0.024**		**9.6E-06**

[a] *Maximum individual exposure/risk.*

[b] *Includes dose from second cask in hold.*

USCG = U.S. Coast Guard, DOT = Department of Transportation

Table D-12 was developed using the information pertaining to the offloading of containerized foreign research reactor spent nuclear fuel from a breakbulk vessel. The exposure times and the distances from the transportation cask used to develop the dose estimates were derived from Table D-8 and assuming the longer transfer times associated with jib or boom cranes. The exposures (worker doses) resulting from the offloading activities associated with this type of vessel are the highest, on a per cask basis, of the three types of vessels considered for transport of the foreign research reactor spent nuclear fuel: breakbulk, container, and roll-on/roll-off (the chartered or purpose-built ship could conceivably be of any of these designs). Therefore, the dose estimates derived from this data provide the upper limit to the doses that could be calculated for the offloading activities.

Alternatively, the worker doses resulting from the offloading of a foreign research reactor spent nuclear fuel cask from a container vessel result in the lowest doses per cask of the types of vessels considered for use in the shipment of the foreign research reactor spent nuclear fuel. Table D-13 was developed using the exposure times and the distances from the transportation cask developed for a container vessel which are provided in Table D-8.

Table D-13 Port Worker Consequences from Shipment of Foreign Research Reactor Spent Nuclear Fuel on Containerized Vessels

Exposed Workers	*Exposure Distance (m)*	*Dose Rate (mrem/hr)*	*Exposure Time (minutes/cask)*	*Dose/Person/ Cask (mrem)*	*Exposed Workers*	*Collective Dose (Person-rem)*	*Individual Risk (LCF)*	*Collective Risk (LCF)*
Longshoreman A1	0.50	37[a]	0.25	0.15	3	0.00046	6.2E-08	1.9E-07
Longshoreman A2	0.50	37[a]	0.75	0.46	1	0.00046	1.9E-07	1.9E-07
Longshoreman A3	6.00	6.4[a]	0.25	0.027	1	0.000027	1.1E-08	1.1E-08
Longshoreman B1	0.50	34	0.25	0.14	4	0.00057	5.7E-08	2.3E-07
Maximum				0.46[a]			1.9E-07[a]	
Subtotal						0.0015		6.1E-07
Crane Operator 1	6.00	32	0.50	0.027	1	0.000027	1.1E-08	1.1E-08
Crane Operator 2	6.00	32	1.6	0.085	1	0.000085	3.4E-08	3.4E-08
Maximum				0.085[a]			3.4E-08a	
Subtotal						0.00011		4.5E-08
Truck Driver	3.00	7.1	3.3	0.38	1	0.00038	1.5E-07	1.5E-07
Maximum				0.38[a]			1.5E-07[a]	
Subtotal						0.00038		1.5E-07
Observers	6.00	3.2	0.5	0.027	4	0.00011	1.1E-08	4.3E-08
Observers	50	0.01	480	0.080	50	0.0040	3.2E-08	1.6E-06
Maximum				0.080[a]			3.2E-08[a]	
Subtotal						0.0041		1.6E-06
USCG Inspector	1.5	15	5.0	1.3	1	0.0013	5.0E-07	5.0E-07
DOT Inspector	1.5	15	2.0	0.5	1	0.00050	2.0E-07	2.0E-07
NRC Inspector	1.5	15	2.0	0.5	1	0.00050	2.0E-07	2.0E-07
State Inspector	1.5	15	2.0	0.5	1	0.00050	2.0E-07	2.0E-07
Local/Port Inspector	1.5	15	5.0	1.3	1	0.0013	5.0E-07	5.0E-07
Receiver	1.5	15	5.0	1.3	1	0.0013	5.0E-07	2.0E-07
Maximum				1.3[a]			5.0E-07[a]	
Subtotal						0.0053		2.1E-06
Maximum				1.3[a]			5.0E-07[a]	
Total						**0.011**		**4.6E-06**

[a]*Maximum individual exposure/risk.*

[b]*Includes dose from second cask in hold.*

USCG = U.S. Coast Guard, DOT = Department of Transportation

In both of these cases it was assumed that two transportation casks were being shipped on a single vessel and the two casks were both in the same hold. By making this assumption, the dose to the workers in the ship's hold is the result of exposure to two radiation fields during the offloading of the first casks. The impact of the presence of the second transportation cask has been included in the dose rates for the longshoremen who are in the ship's hold during the offloading activity. To simplify the analysis, it has been assumed that the dose rates for the offloading of the two casks are the same (i.e., even though when the second cask is being offloaded there is only one transportation cask in the hold, the exposures are calculated assuming that there are two casks in the hold). The total number of transportation casks shipped on a single vessel would not impact the results of this analysis. The per shipment results are for the shipment of a single cask, assuming two casks per hold. Annual exposures and exposures for the entire program do not depend on the number of transportation casks per shipment. Under the assumption that a vessel carrying more than two casks would be loaded two casks per hold, these results are solely dependent on the number of cask shipments per year and the total number of cask shipments.

There is approximately a factor of two difference between the total worker dose resulting from the use of a breakbulk vessel and the use of a container vessel per transportation cask. There is a larger difference between the dose to the maximally exposed individual (MEI). The MEI for the breakbulk vessel receives a dose of 3.8 mrem per transportation cask offloading while for the offloading of a transportation cask from a container vessel the MEI receives a dose of 1.3 mrem.

Another consideration that could affect the total worker exposure is the possibility that the vessel transporting the foreign research reactor spent nuclear fuel could make intermediate port calls between the foreign port at which the transportation cask is loaded and the port of entry for the foreign research reactor spent nuclear fuel. At the intermediate ports of call, it is possible that cargo being shipped on the vessel and in the same hold as the transportation casks could be loaded/offloaded or moved. The analysis was expanded to consider the impacts on port workers at these intermediate ports. Table D-14 provides the information used to estimate the dose to the port workers in intermediate ports. The estimates consider that the hold in which the transportation casks are being stowed have been fully loaded and that all of the cargo in the vicinity of the transportation casks must be moved at one of the intermediate ports of call. The vessel assumed in the intermediate port analysis was a breakbulk vessel. As in the analysis of the impact of the offloading of the transportation casks, this assumption results in calculations based on the type of vessel that will result in the largest estimated impact on the port workers.

Table D-14 Port Worker Exposure - Each Intermediate Port

Exposed Workers	*Distance (m)*	*Dose Rate[a] (mrem/hr)*	*Exposure Time (minutes)*	*Dose/Person (mrem)*	*Number of Workers[b]*	*Collective Dose (person-rem)*	*Individual Risk (LCF)*	*Risk per Port Call (LCF)*
Longshoreman	1.5	18	5	1.5	4	--	--	--
	5	6.4	6	0.64	4	--	--	--
	8	4.6	1	0.08	4	--	--	--
Total				**2.2**	**4**	**0.0089**	**0.00000089**	**0.0000035**

[a]*The dose rate includes the dose rate from two casks stored in the same hold.*

[b]*The same four workers are assumed to receive the entire dose from cargo handling activities in each intermediate port stop.*

The per shipment data provided in Tables D-12 through D-14 was used to develop estimates of the incident-free impact of the marine shipment of 721 transportation casks on port workers. (The number of shipments required is derived in Appendix B. The 721 shipments used in this portion of the analysis exclude all shipments of Canadian origin which are expected to be overland shipments). Table D-15 provides the results of this analysis. Data is provided for two possible shipment conditions. In the first a breakbulk vessel is used to transport all of the foreign research reactor spent nuclear fuel and this vessel is assumed to make two intermediate port calls on every voyage. During these intermediate port calls the cargo in the same hold as the transportation casks is assumed to be moved (loaded and/or offloaded) twice. The impact on port workers, in terms of population exposure and risk, in the intermediate ports is therefore twice the impact presented in Table D-14. The second set of assumptions used is that all shipments are made on a container vessel that does not make intermediate port calls. These assumptions result in a lower estimate of port worker risk since the impact of intermediate port calls is eliminated and the offloading activities for a container vessel result in lower overall doses to the port workers. These two sets of assumptions, therefore, provide estimates of the range of potential impacts on port workers.

In calculating the MEI, it was necessary to estimate the number of shipments to which a single worker could be exposed. Using the information in Table C-1, the shipments of foreign research reactor spent nuclear fuel were divided into eastern and western shipments. The eastern shipments are those that would

Table D-15 Integrated Port Worker Dose for the Basic Implementation of Management Alternative 1

	Breakbulk Vessel with 2 Intermediate Port Calls				*Container Vessel - No Intermediate Port Calls*			
	Maximally Exposed Individual (rem)	*Collective Dose to Workers (person-rem)*	*MEI Risk (LCF)*	*Worker Risk LCF)*	*Maximally Exposed Individual (rem)*	*Collective Dose to Workers (person-rem)*	*MEI Risk (LCF)*	*Worker Risk (LCF)*
Inspectors	2.0[a]	9.4	0.00080	0.0037	0.67	3.8	0.00027	0.0015
Port Handlers - Intermediate Ports	1.2	13	0.00047	0.0051	----	----	----	----
Port Handlers - Port of Entry	1.1	4.8	0.00043	0.009	0.25	1.1	0.00010	0.00044
Port Staging Personnel	0.19	3.2	0.000076	0.0013	0.21	3.3	0.00008	0.0013
Total	----	**30.2**	----	**0.012**	----	**8.2**	----	**0.0033**
Maximum	**2.0[a]**	----	**0.00080**	----	**0.67**	----	**0.00027**	----

[a]*This dose is above the allowed limit of 100 mrem/yr for the general population and would be mitigated to below the limit.*

be expected to be shipped to a port on the East Coast of the United States if the shortest shipping distance were used. Western shipments are those that would be shipped to the West Coast port. From Table C-1, 535 shipments would be considered East Coast shipments; 186 West Coast. In determining the MEI, it was assumed that all of these East Coast shipments were made through the same port, and the same workers were involved in the offloading of the transportation casks for all shipments.

The total impact on the worker population was determined by using the full 721 transportation cask shipments. Both the MEI and the collective dose to the workers have been converted into a risk estimate of LCF resulting from the doses received in offloading the transportation casks loaded with foreign research reactor spent nuclear fuel. The range of impacts for the program is from 8.2 person-rem (0.0033 LCF) (for the use of container vessels with no intermediate port calls) to 30 person-rem (0.012 LCF) (for the use of breakbulk vessels with two intermediate port calls). These risks imply that there is between a three-in-a-thousand and a one-in-a-hundred chance that this program will result in one LCF as a result of the incident-free impact on port workers. The relationship between worker dose and cancer fatalities is that 1 rem is equivalent to 0.0004 LCF.

Under the basic implementation of Management Alternative 1, shipments would be received over a 13-year period, the 10-year period for spent nuclear fuel generation plus 3 additional years to allow for the coordination of available storage, transportation casks, shipping arrangements, etc. Assuming that the shipments were evenly distributed over the 13-year period, the doses to the MEI could be in excess of the DOE and NRC limits for doses to the general public (100 mrem per year). If breakbulk vessels were used, the MEI would receive approximately 150 mrem per year on average, if no mitigation steps were taken. If container vessels were used, no individuals are expected to receive a dose in excess of the public dose limits.

The above calculations were all performed assuming that every transportation cask was shipped with an external dose rate at the selected exclusive use regulatory limit of 10 mrem hour at 2 m (6.6 ft) from the surface of the container. This provides an estimate of the upper limit to what the incident-free impacts of the offloading of the transportation casks could be. To determine a more realistic estimate of these impacts, the analysis was redone using historical data on the external dose rates associated with the transportation of research reactor spent nuclear fuel. This analysis results in an average dose rate of

approximately 2.3 mrem per hour at 1 m (3.3 ft) from the cask surface, which is equivalent to a dose rate of 1 mrem per hour at 2 m (6.6 ft) from the cask surface. If the added distance from the cask surface to the container surface is not credited, this dose rate is one-tenth of the dose rate derived from the "exclusive use" regulatory limit. (See Appendix F, Section F.5)

Tables D-16 through D-19 provide the results of this analysis. No other assumptions were modified between this analysis from those used to develop the data presented earlier in this section. All of the results using the "historical" data are an order-of-magnitude lower than results derived from the use of the regulatory limit dose rates.

Table D-16 Port Worker Consequences from Shipment of Foreign Research Reactor Spent Nuclear Fuel on Breakbulk Vessels (Historical Data)

Exposed Workers	*Exposure Distance (m)*	*Dose Rate (mrem/hr)*	*Exposure Time (minutes/cask)*	*Dose/ Person/Cask (mrem)*	*Exposed Workers*	*Collective Dose (Person-rem)*	*Individual Risk (LCF)*	*Collective Risk (LCF)*
Longshoreman A1	0.50	3.7[b]	0.25	0.015	2	3.1E-05	6.2E-09	1.2E-08
Longshoreman A2	0.50	3.7[b]	3.3	0.20	2	4.0E-04	8.0E-08	1.6E-07
Longshoreman A3	6.00	0.64[b]	0.25	0.0027	1	2.7E-06	1.1E-09	1.1E-09
Longshoreman B1	0.50	34	1.0	0.057	4	2.3E-04	2.3E-08	9.1E-08
Maximum				0.20[a]			8.0E-08a	
Subtotal						6.6E-04		2.6E-07
Crane Operator 1	9.00	0.18	3.0	0.009	1	9.0E-06	3.6E-09	3.6E-09
Maximum				0.009[a]			3.6 E-09[a]	
Subtotal						9.0E-06		3.6E-09
Truck Driver	3.00	0.71	3.0	0.036	1	3.6E-05	1.4E-08	1.4E-08
Maximum				0.036[a]			1.4E-08[a]	
Subtotal						3.6E-05		1.4E-08
Observers	6.00	0.32	0.25	0.0013	4	5.3E-06	5.3E-10	2.1E-09
Observers	50	0.001	480	0.008	50	4.0E-04	3.2E-09	1.6E-07
Maximum				0.008[a]			3.2E-09[a]	
Subtotal						4.1E-04		1.6E-07
USCG Inspector	1.5	1.5	2.0	0.05	1	5.0E-05	2.0E-08	2.0E-08
DOT Inspector	1.5	1.5	15	0.38	1	3.8E-04	1.5E-07	1.5E-07
NRC Inspector	1.5	1.5	10	0.25	1	2.5E-04	1.0E-07	1.0E-07
State Inspector	1.5	1.5	15	0.38	1	3.8E-04	1 .5E-07	1.5E-07
Local/Port Inspector	1.5	1.5	5	0.13	1	1.3E-04	5.0E-08	5.0E-08
Receiver	1.5	1.5	5	0.13	1	1.3E-04	5.0E-08	5.0E-08
Maximum				0.38			1.5E-07	
Subtotal						1.3E-03		5.2E-07
Maximum				0.38[a]			1.5E-07[a]	
Total						**2.4E-03**		**9.6E-07**

[a] *Maximum individual exposure/risk.*

[b] *Includes dose from second cask in hold.*

USCG = U.S. Coast Guard, DOT = Department of Transportation

The total population dose (dose to the port workers) ranges from 3.0 person-rem (breakbulk vessel with two intermediate port calls) and 0.7 person-rem (container vessel with no intermediate port calls). This corresponds to a risk of 0.0012 to 0.00033 LCF, that is, a one-in-a-thousand to a one-in-three thousand chance of incurring one LCF. For a population of workers, the relationship between exposure and LCF is

Table D-17 Port Worker Consequences from Shipment of Foreign Research Reactor Spent Nuclear Fuel on Containerized Vessels (Historical Data)

Exposed Workers	*Exposure Distance (m)*	*Dose Rate (mrem/hr)*	*Exposure Time (minutes/cask)*	*Dose/Person/ Cask (mrem)*	*Exposed Workers*	*Collective Dose (Person-rem)*	*Individual Risk (LCF)*	*Collective Risk (LCF)*
Longshoreman A1	0.50	3.7[a]	0.25	0.015	3	4.6E-05	6.2E-09	1.9E-08
Longshoreman A2	0.50	3.7[b]	0.75	0.046	1	4.6E-05	1.9E-08	1.9E-08
Longshoreman A3	6.00	0.64[b]	0.25	0.0027	1	2.7E-06	1.1E-09	1.1E-09
Longshoreman B1	0.50	340	0.25	0.014	4	5.7E-05	5.7E-08	2.3E-08
Maximum				0.046[a]			1.9E-08[a]	
Subtotal						1.5E-04		6.1E-08
Crane Operator 1	6.00	0.32	0.5	0.0027	1	2.7E-06	1.1E-09	1.1E-09
Crane Operator 2	6.00	0.32	1.6	0.0085	1	8.5E-06	3.4E-09	3.4E-09
Maximum				0.0085[a]			3.4E-09[a]	
Subtotal						1.1E-05		4.5E-09
Truck Driver	3.00	0.71	3.3	0.038	1	3.8E-05	1.5E-08	1.5E-08
Maximum				0.038[a]			1.5E-08[a]	
Subtotal						3.8E-05		1.5E-08
Observers	6.00	0.32	0.5	0.0027	4	1.1E-05	1.1E-09	4.3E-09
Observers	50	0.001	480	0.0080	50	4.0E-04	3.2E-09	1.6E-07
Maximum				0.0080[a]			3.2E-09[a]	
Subtotal						4.1E-04		1.6E-07
USCG Inspector	1.5	1.5	0.5	0.13	1	1.3E-04	5.0E-08	5.0E-08
DOT Inspector	1.5	1.5	0.2	0.050	1	5.0E-05	2.0E-08	2.0E-08
NRC Inspector	1.5	1.5	0.2	0.050	1	5.0E-05	2.0E-08	2.0E-08
State Inspector	1.5	1.5	0.2	0.050	1	5.0E-05	2.0E-08	2.0E-08
Local/Port Inspector	1.5	1.5	0.5	0.013	1	1.3E-04	5.0E-08	5.0E-08
Receiver	1.5	1.5	0.5	0.13	1	1.3E-04	5.0E-08	5.0E-08
Maximum				0.13[a]			5.0E-08[a]	
Subtotal						5.3E-04		2.1E-07
Maximum				0.13[a]			5.0E-08[a]	
Total						**1.1E-03**		**4.5E-07**

[a]*Maximum individual exposure/risk.*

[b]*Includes dose from second cask in hold.*

USCG = U.S. Coast Guard, DOT = Department of Transportation

Table D-18 Port Worker Exposure - Intermediate Ports (Historical Cask External Dose Rate Data)

Exposed Workers	*Distance (m)*	*Dose Rate[a] (mrem/hr)*	*Exposure Time (minutes)*	*Dose/Person (mrem)*	*Number of Workers[b]*	*Collective Dose (person-rem)*	*Individual Risk (LCF)*	*Risk per Port Call (LCF)*
Longshoreman	1.5	1.8	5	0.15	4	--	--	--
	5	0.6	6	0.06	4			--
	8	0.5	1	0.01	4	--	--	--
Total				**0.22**	**4**	**0.00089**	**0.000000089**	**0.00000035**

[a]*The dose rate includes the dose rate from two casks stored in the same hold.*

[b]*The same four workers are assumed to receive the entire dose from cargo handling activities in each intermediate port stop.*

Table D-19 Integrated Port Worker Dose for the Basic Implementation of Management Alternative 1 (Historical Cask Dose Rates)

	Breakbulk Vessel with 2 Intermediate Port Calls				*Container Vessel - No Intermediate Port Calls*			
	Maximally Exposed Individual (rem)	*Collective Dose (person-rem)*	*MEI Risk (LCF)*	*Risk (LCF)*	*Maximally Exposed Individual (rem)*	*Collective Dose (person-rem)*	*MEI Risk (LCF)*	*Risk (LCF)*
Inspectors	0.20	0.94	0.00008	0.00037	0.07	0.38	0.00002	0.00015
Port Handlers - Intermediate Ports	0.12	1.3	0.000047	0.00051	----	----	----	----
Port Handlers - Port of Entry	0.11	0.5	0.000043	0.00019	0.03	0.11	0.000010	0.000044
Port Staging Personnel	0.02	0.3	0.000008	0.00013	0.02	0.33	0.000009	0.00013
Maximum	0.20[a]		0.00008[a]		0.07[a]		0.000027[a]	
Total		**3.0**		**0.0012**		**0.8**		**0.00033**

[a]*Maximally exposed individual.*

1 rem is equivalent to 0.0004 LCF. The MEI would receive a dose of 0.2 rem over the 13-year period of the basic implementation of Management Alternative 1. This is approximately 15 mrem per year, which is well below the NRC and DOE limits for exposure to the public (100 mrem per year).

The results of these analyses indicate that some of the port personnel that handle and inspect foreign research reactor spent nuclear fuel shipping containers could receive doses that exceed public exposure limits established by DOE and the NRC, especially when the dose rate from the casks are assumed to be at the regulatory limit for exclusive use shipments of 10 mrem per hour measured 2 m (6.6 ft) from the surface of the shipping container. The analyses results are conservative due to three factors. First, it is estimated that for most shipments the external dose rate for the loaded transportation cask would be near the historic dose rates, which average a factor of ten below the regulatory limit. Second, the analyses assumed that the same port inspectors and handlers handle all shipments. In reality, most port personnel work on shifts, so the likelihood of all shipments being handled by the same shift is low. Finally, all of the shipments passing through any East Coast port were assumed to pass through the same port. In reality, it is more than likely that the shipments would be made through more than a single port.

However, the existence of some shipments with external dose rates closer to the exclusive use regulatory limit suggests that DOE should provide a means to assure that individual port personnel do not receive doses in excess of the public dose limits. As a minimum, the program should establish administrative procedures that would maintain records of the exposure rates associated with each shipment and the ports of departure and entry. The measurement of interest for the record keeping would be the external dose rates outside the container, which houses the transportation cask since the port personnel do not enter the container. These measurements could be used to identify shipments that would result in port personnel exposures above those calculated based on the historical spent nuclear fuel transportation external dose rate. By tracking this information, DOE would be able to identify if and when additional precautions to reduce individual exposures should be taken.

D.4.6 Cumulative Port Impact Analysis Methodology

Analyses have been carried out to estimate the maximum occupational doses associated with the port activities segment of the transportation of foreign research reactor spent nuclear fuel. Since port workers are expected to be exposed to other shipments of radioactive materials, the cumulative impact of all

radioactive material shipments has been estimated. The cumulative analysis is necessary to determine the impact on port workers from doses received through actions associated with the foreign research reactor spent fuel return program and through other actions, both DOE and commercially initiated.

The maximum exposure for a worker involved in transporting the foreign research reactor fuel is predicted to result from activities associated with the unloading of the spent fuel casks in port, cask inspection, and cask preparation for truck shipment to the management sites. If the same individuals were present for all proposed shipments of foreign research spent nuclear fuel on an annual basis (a conservative assumption), the maximum dose would be approximately 150 mrem, as discussed in the previous section. This estimate is based on the use of the "exclusive use" regulatory external dose rate. Based on historical spent nuclear fuel shipment data, this maximum annual dose would be 15 mrem.

Since commercial ports routinely receive other shipments of radioactive materials under other DOE programs or other commercial activities, the port worker would also be potentially exposed to additional sources of radiation. To estimate the annual exposure rate of port workers resulting from handling of commercial radioactive material shipments, the following must be determined.

- Number of radioactive packages handled per year
- Length of exposure time per package
- Dose rate per package

Records of shipments through the potential ports of entry were used to estimate the annual throughput of packages with radioactive contents. Radioactive materials were identified by the product code listed for each shipment. The radioactive shipments were then grouped into six categories and exposure rates at 1 m (3.3 ft) from the outer surface of the package were assigned for each group as follows:

• enriched uranium hexafluoride	(0.5 mrem per hour)
• normal uranium hexafluoride	(0.2 mrem per hour)
• depleted uranium	(0.2 mrem per hour)
• uranium oxide	(0.2 mrem per hour)
• spent nuclear fuel (foreign research reactor)	10 mrem per hour [at 2 m (6.6 ft) from the container surface]
• other radioactive materials	(0.2 mrem per hour)

Each shipment record lists the weight and number of packages included in the shipment. Since package descriptions were not uniform and included units, containers, cases, boxes, barrels, drums, packages, cartons, etc., the assumption was made that the radioactive shipments would be stacked on skids and the total number of skids per shipment, rather than the number of packages per shipment, would be used to estimate the dose received by workers. The weight and number of individual shipments was examined for each shipment to estimate the number of skids. In most cases, boxes, cartons, barrels, and drums were assumed to be handled four to a skid. When a large number of light packages was included in one shipment, these were assumed to be handled as either eight or 32 packages per skid.

The annual dose to port workers resulting from handling commercial radioactive shipments were estimated based on the number of shipments passing through the port and an estimated handling time of ten minutes per skid or cylinder. Each port typically uses three shifts per day and therefore workers were assumed to be exposed to one-third of the packages passing through the port. This is a conservative assumption given that there are typically many berths and terminals within one port, thus making it unlikely that one individual would be present for even one-third of the shipments of radioactive materials. The estimated dose to the MEI from these commercial shipments is shown in Table D-20.

Table D-20 Estimated Maximum Exposure to Dock Workers from Commercial Shipments of Radioactive Material

Port	*Average No. of Radioactive Shipments per Year*	*Estimated Maximum Exposure per Year (mrem)*	*Port*	*Average No. of Radioactive Shipments per Year*	*Estimated Maximum Exposure per Year (mrem)*
Baltimore, MD	31	3.4	New Orleans, LA	7	3.9
Boston, MA	2	0.2	Norfolk, VA	30	3.9
Charleston, SC	16	3.1	New York, NY	104	16.8
Fernandina Beach, FL	21	less than 0.1	Oakland, CA	39	9.0
Galveston, TX	1	less than 0.1	Philadelphia, PA	1	less than 0.1
Houston, TX	14	4.0	Portland, OR	1	0.6
Jacksonville, FL	4	0.3	Portsmouth, VA	28	5.5
Long Beach, CA	1	less than 0.1	Port Everglades, FL	7	0.1
Los Angeles, CA	6	0.2	Savannah, GA	7	1.5
Miami, FL	1	less than 0.1	Wilmington, NC	2	1.2

As this table shows, yearly exposures for the commercial shipments are typically less than 10 mrem per year, which is well within the regulatory limit of 100 mrem per year established for a member of the general public. New York (at 16.8 mrem per year), which had the most commercial shipments of radioactive material on a yearly basis, was the only port to exceed 10 mrem per year. However, the Port of New York consists of three terminals in Elizabeth (NJ), Brooklyn, and Manhattan. This diversity means that in practice, the average port worker would be involved in only a portion of the shipments through "New York."

Some of the potential ports are being used or have the potential to be used for other DOE-initiated activities. These activities include the purchase of Russian low enriched uranium (LEU) under the agreement Suspending the Antidumping Investigation of Uranium from the Russian Federation and the import of Russian LEU derived from the dismantling of nuclear weapons in Russia. Estimated maximum exposures from these activities are 0.9 mrem and 1.4 mrem per year, respectively.

The impact of all of these shipments can be viewed in two ways. If the foreign research reactor spent nuclear fuel shipments were to have dose rates like the historical data indicate they would, the total maximum worker exposure from all of these activities would be well below the public dose limits (by at least a factor of three). If the foreign research reactor spent nuclear fuel shipments were to be closer to the external dose rate allowed by the "exclusive use" regulatory limit, these other activities do not significantly alter the maximum worker dose. In this case, DOE's response to the worker exposure would be dictated by the exposure resulting from the shipment of foreign research reactor spent nuclear fuel.

D.4.7 Incident-Free Port Impacts of Alternatives to the Basic Implementation of Management Alternative 1

Three alternatives to the basic implementation of Management Alternative 1 were identified that could impact the incident-free port risk calculations that were performed. (Chapter 2 describes the alternatives to the basic implementation of Management Alternative 1.) The implementation subalternative of *accepting spent nuclear fuel only from developing countries*, which are identified as countries other than high-income economies, would result in a reduction in the amount of spent nuclear fuel transported by ship. Table C-12 listed the countries that are considered to be countries other than high-income economies and the number of foreign research reactor spent nuclear fuel shipments that would be required to transport their spent nuclear fuel to the United States. One hundred sixty-eight transportation casks would be shipped to the United States under this implementation subalternative. Under the *foreign research reactor spent nuclear fuel for 5-years only* implementation subalternative, the number of shipments of foreign research reactor spent nuclear fuel would be reduced to 586 shipments requiring ocean transport. (The derivation of the number of shipments required in this alternative is presented in Appendix B.)

The third alternative, with the capability to impact the results of the incident-free port risk analysis, is the *overseas processing of the foreign research reactor spent nuclear fuel with the shipment of the vitrified waste to a storage facility in the United States*. Under this alternative, eight transportation cask shipments of vitrified waste could be made.

In addition to these alternatives, a hybrid alternative was analyzed. In this alternative, those countries that have the capability to store high-level waste would be encouraged to process the aluminum-based research reactor spent nuclear fuel and to accept for storage the resulting high-level waste. (For this alternative these countries are assumed to be Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom). The United States would accept for storage the foreign research reactor spent nuclear fuel from those countries deemed not to have the high-level waste storage capability. In this alternative, this includes all of the countries identified in Table C-1, except for those listed above. Under this hybrid alternative, 452 shipments of foreign research reactor spent nuclear fuel are assumed to be sent to the United States, excluding overland shipments of Canadian origin.

The incident-free port risks associated with these three alternatives are discussed in the following sections.

Implementation Subalternative 1a of Management Alternative I – Acceptance of Foreign Research Reactor Spent Nuclear Fuel Only From Developing Countries: Developing countries are defined as countries other than high-income economies. As stated above, this implementation subalternative would result in the shipment of 168 transportation casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the basic implementation of Management Alternative 1 incident-free port impact have been used in the analysis of this subalternative. To compare this subalternative to the basic implementation of Management Alternative 1, it is only necessary to perform the analysis using one external dose rate, either the regulatory dose limit or the historic dose rate. The regulatory dose rate was chosen for the comparison.

Included in the assumptions that have not changed in this analysis are the following:

- The worker exposure times and distances from the transportation cask are as detailed in Tables D-8 through D-10.
- The intermediate port stops are considered for the breakbulk vessel but not for the container vessel.

- Two transportation casks are being transported in the same hold on each cargo vessel.

The per shipment incident-free impact on the port workers would be identical to that calculated for the basic implementation of Management Alternative 1. None of the assumptions used to generate the per shipment information change. The 168 shipments required to meet the needs of this subalternative would result in a reduction in the total (program) impacts by approximately 77 percent. The total population exposure would range from 7.0 person-rem (for the breakbulk vessel with two intermediate port calls) to 1.9 person-rem (for the container vessel with no intermediate port stops). This corresponds to an incident-free risk of 0.0028 to 0.00076 LCFs (i.e., a chance of between three-in-a-thousand and seven-in-ten thousand of incurring one LCF).

Implementation Subalternative 2a of Management Alternative 1 – Acceptance of Foreign Research Reactor Spent Nuclear Fuel for 5 Year Policy Duration: As stated above, this implementation subalternative would result in the shipment of 586 transportation casks of foreign research reactor spent nuclear fuel. The assumptions used previously for incident-free port impact have been used in the analysis of this subalternative. This implementation subalternative has been analyzed using the "exclusive use" regulatory limit transportation cask external dose rates. To compare this implementation subalternative to the basic implementation of Management Alternative 1, it is only necessary to perform the analysis using one external dose rate.

Included in the assumptions that have not changed in this analysis are the following:

- The worker exposure times and distances from the transportation cask are as detailed in Tables D-8 through D-10.
- The intermediate port stops are considered for the breakbulk vessel but not for the container vessel.
- Two transportation casks are being shipped in the same hold of each cargo vessel.

The per shipment incident-free impact on the port workers would be identical to that calculated for the basic implementation of Management Alternative 1. Therefore, none of the assumptions used to generate the per shipment information change. The 586 shipments required to meet the needs of this implementation subalternative would result in a reduction in the total (program) impacts to approximately 81 percent of the impacts associated with the basic implementation of Management Alternative 1. The total population exposure would be 25 person-rem (for the breakbulk vessel with two intermediate port calls) to 6.7 person-rem (for the container vessel with no intermediate port stops). This corresponds to an incident-free risk of 0.0098 to 0.0027 LCFs (i.e., a chance of between one-in-a-hundred and three-in-a-thousand of incurring one LCF).

Management Alternative 2, Subalternative Ib – Overseas Reprocessing with Shipment of the Vitrified Waste to a U.S. Storage Facility: In this subalternative under Management Alternative 2, the foreign research reactor spent nuclear fuel would be processed overseas (most probably in Great Britain or France) and the waste products are contained within several vitrified waste logs. This high-level waste may be brought to the United States for storage in one of the storage facilities evaluated under this EIS. Under these conditions, up to eight transportation casks containing vitrified waste would be shipped from Europe to the United States. This analysis addresses the incident-free port risks associated with transporting these eight casks of vitrified waste from Europe to the United States.

As with the shipment of foreign research reactor spent nuclear fuel as spent nuclear fuel, the primary incident-free port impacts of shipping vitrified waste would be upon the workers in the ports. The assumptions used in the analysis of the incident-free port impact of the basic implementation of Management Alternative 1 have been used in the analysis of this subalternative. Differences between the foreign research reactor spent nuclear fuel transportation casks and the vitrified waste transportation casks are not expected to significantly alter the work requirements in port. For the purposes of this analysis, it has been assumed that the vitrified waste would be transported on a chartered vessel, and there would be no intermediate port calls.

This alternative has been analyzed using the regulatory limit transportation cask external dose rates. Little information is available on the casks to be used to transport the vitrified waste. No attempt was made to extrapolate limited historical data to determine the port worker incident-free impacts from any other exposure rate other than the limit set forth in NRC and DOE regulations.

Included in the assumptions that have not changed in this analysis are the following:

- The worker exposure times and distances from the transportation cask are as detailed in Tables D-8 through D-10.
- The intermediate port stops are not considered for the container vessel.
- Two transportation casks are being transported in the same hold of the cargo vessels.

The per shipment incident-free impact on the port workers would be identical to that calculated for the basic implementation of Management Alternative 1. None of the assumptions used to generate the per shipment information change. The eight shipments required to meet the needs of this subalternative would result in a reduction in the total (program) impacts by a factor of approximately one hundred. The total population exposure would be 0.0091 person-rem for the container vessel with no intermediate port stops. This corresponds to an incident-free risk of 0.0000036 LCFs (i.e., a chance of approximately four-in-a-million of incurring one LCF).

Hybrid Alternative – Acceptance of Foreign Research Reactor Spent Nuclear Fuel From Countries Without High-Level Waste Disposal Capability: As stated above, this hybrid alternative results in the marine shipment of 452 transportation casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the incident-free port impact of the basic implementation of Management Alternative 1 have been used in the analysis of this alternative. This alternative has been analyzed using external dose rates derived from the exclusive use regulatory limit for a transportation cask.

Included in the assumptions that have not changed in this analysis are the following:

- The worker exposure times and distances from the transportation cask.
- The intermediate port stops are considered for the nonchartered vessel but not for the chartered vessel.
- Two transportation casks are being shipped in the same hold of each cargo vessel.

The per-shipment incident-free impact on the port workers would be identical to that calculated for the basic implementation of Management Alternative 1. None of the assumptions used to generate the per-shipment information changes. The 452 shipments required to meet the needs of this hybrid alternative would result in a reduction in the total (program) impacts to approximately 63 percent of the impacts associated with the basic implementation of Management Alternative 1. Therefore, the total

population exposure would be 19 person-rem (for regularly scheduled commercial vessel with two intermediate port calls) to 5.1 person-rem (for the chartered vessel with no intermediate port calls). This corresponds to an incident-free risk of 0.0076 to 0.0021 LCFs (i.e., a chance of between approximately one-in-five hundred to less than one-in-a-hundred of incurring one LCF).

D.5 Accident Impacts: Methods and Results

D.5.1 Introduction

This section describes the approach used to assess the risks associated with in-port accidents that could result in a release of radioactive material from the transportation cask containing foreign research reactor spent nuclear fuel. The discussion addresses both the accident risk assessment methodology and the results of the analyses. The risk assessment results are presented in terms of a per-shipment accident risk and the total port-accident risks associated with various alternative under the proposed action.

Spent nuclear fuel shipments could occur via any of four types of vessels, container ships, roll-on/roll-off vessels, breakbulk vessels, and purpose-built (dedicated) vessels. In the incident-free analysis, only breakbulk vessels and container vessels were studied, since these two provide a bounding assessment of the risks associated with port activities. Under the assumptions used in the port accident analysis, the type of vessel used to transport the foreign research reactor spent nuclear fuel would not impact the result of the analysis.

All radiologically-related impacts are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent, which is the sum of the effective dose equivalent (EDE) from the external radiation exposure and the 50 year committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of person-rem for collective population and rem or mrem for individuals. The impacts are further expressed as health risks, specifically in terms of LCF. The health risk conversion factors were derived from International Commission on Radiological Protection Publication 60 (ICRP, 1991). See Chapter 4 for a more complete explanation of radiation measurement and health risks.

D.5.1.1 Accident Risks

Risk (R) is the product of the magnitude (M) of an unfavorable consequence and the probability of occurrence (P) of that consequence. Thus,

$$R = PM.$$

For accidents that happen during the transportation of radioactive materials, the unfavorable consequences of the accident may include exposure of people to radiation emitted by the radioactive materials released to the atmosphere by the accident and the occurrence of radiation induced health effects that the exposure may cause. The magnitude of these consequences depends on the amount of radioactivity released to the atmosphere, the degree to which the radioactive materials are diluted during downwind transport, and the size of the population that is exposed to radiation from the passing plume or from materials deposited on the ground or in the water from the plume. The amount of dilution experienced by a plume during downwind transport depends principally on atmospheric stability and windspeed. The size of the exposed population is determined by the direction the wind is blowing at the time of the accident and the number of people in that direction. Thus, the probability that a given consequence occurs is given by the following product,

$$P = P_{st}P_wP_p$$

where P_{st} is the probability of the source term (the amount of radioactive material released), P_w is the probability of the prevailing weather conditions, and P_p is the exposure probability of the population that is exposed to radiation, given the direction that the wind is blowing at the time of the accident.

D.5.1.2 Ship Accident Risks

The total risk caused by transporting foreign research reactor spent fuel to and within the United States is the sum of the risks for transport by land and by ship. Thus,

$$R_{total} = R_{land} + R_{ship}$$

For ships, the risk is given by:

$$R_{ship} = R_{sea} + R_{coast} + R_{port}$$

where R_{sea}, R_{coast}, and R_{port} are the risk while at sea, while sailing in coastal waters, and while in the port (R_{sea} and R_{coast} were addressed in Appendix C). Each risk term has an incident-free and an accident contribution, so

$$R_{port} = R_{port\text{-}incident\text{-}free} + R_{port\text{-}accident}$$

The accident risks associated with the foreign research reactor spent nuclear fuel while it is on a ship in the port, $R_{port\text{-}accident}$, is the subject of this section. $R_{port\text{-}incident\text{-}free}$ was covered in D.4 of this appendix.

The only port accidents considered are those where the ship carrying the spent nuclear fuel is struck by another ship. Accidents where the spent nuclear fuel transport ship rams a fixed structure (a bridge or a dock), rams another ship (a collision where the spent nuclear fuel ship is the striking ship), or runs aground are neglected for the following reasons.

First, ship accident data show that when a ship rams a fixed structure or collides with another ship, damage to the striking ship is confined to its prow and to the forwardmost hold. Even in these cases, the forces exerted on cargo in the forward hold are less than the forces exerted on cargo in the case where a striking ship impacts the cargo hold.

Second, because keel structures are massive and very sturdy, groundings rarely lead to significant damage to cargo, although monetary losses due to sinking of cargo or the ship can be significant. Immersion to the depths of harbor channels is unlikely to damage a spent nuclear fuel cask or pose a significant retrieval problem; therefore, groundings are also neglected in this study.

D.5.2 Risk Analysis Methods

The consequences of ship collisions that occur in ports were estimated using the MELCOR Accident Consequences Code System (MACCS) (Jow et al., 1990, Sprung et al., 1990), originally developed by Sandia National Laboratories and the NRC for use in estimating the consequences of nuclear power plant accidents. The MACCS code was selected for these analyses because it can model an accident that takes place at a specific location and, more importantly, can model the site-specific population distribution around that location including space that is ocean and thus unpopulated.

If a ship transporting spent nuclear fuel is struck by another ship, and the collision leads to the failure of the spent fuel cask, the prevailing winds would transport the radioactive gases and aerosols in the plume released to the atmosphere during the accident away from the accident scene. During transport by the prevailing winds, downwind populations would likely be exposed to radiation, and land, buildings, and crops located below the plume trajectory might be contaminated by the radioactive materials deposited from the plume. Estimation of the range and probability of the health effects induced by the radiation exposures, and of the economic costs and losses that would result from any contamination of land, buildings, and crops is the objective of a MACCS accident consequence analysis.

MACCS calculations require the following accident and site data:

The radioactive inventory of the cask at the time of the accident for those radionuclides important for the calculation of accident consequences.

Release fractions and probability of release for the source term caused by the accident.

Plume characteristics for the radioactivity released to the atmosphere by the accident, the sensible heat content and the release time and duration.

Meteorological data characteristic of the region where the port is located, usually one year of hourly readings of windspeed, atmospheric stability, and rainfall.

The population distribution about the port where the accident occurs.

Emergency response assumptions, such as evacuation time and average speed; building shielding factors and the time when people take shelter if nearby populations are instructed to take shelter.

Land usage (habitable land fractions and farmland fractions) for the region surrounding the port.

Given these data, MACCS predicts:

The downwind transport, dispersion, and deposition of the radioactive materials released from the failed spent fuel cask.

The radiation doses received by the exposed populations via direct (cloudshine, inhalation, groundshine, resuspension) and indirect (ingestion) exposure pathways.

The mitigation of these doses by emergency response actions (evacuation, sheltering, and post-accident relocation of people).

Health effects that might occur in the population exposed to radiation as a result of the accident, both LCF and acute injuries (if short-term exposures are large).

The potential costs of emergency response actions, and of the decontamination, temporary interdiction, and condemnation of milk, crops, land, and buildings located in the region around the port, if necessary.

D.5.3 MACCS Input Data

D.5.3.1 Source Terms

MACCS source terms are specified by five input quantities: the probability (P_{st}) of the accident that leads to the release; the time (t) and duration (delta t) of the release (for ship accidents there may be both a mechanical release following the collision and a later thermal release if the accident progression leads to a fire); and the accident release fraction (f_i) and cask inventory (I_i) of each radionuclide (i) important for the calculation of accident consequences.

D.5.3.1.1 Source Term Probabilities

In the Environmental Assessment for the Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994d), accident risks were estimated using six categories of accident severity. To facilitate comparison of the risk estimates developed for this EIS to those developed for the Environmental Assessment, the EIS retained these six categories of accidents. Table D-21 presents the six categories of accident severity used in the EIS (and Environmental Assessment), the values of the conditional release probabilities (conditional on the occurrence of the specified accident), and the radionuclide release fractions used in the EIS for each severity category.

Inspection of the table shows that no radioactive releases are expected for accidents that fall into severity categories 1 or 2. Accidents that fall into category 3 fail the cask's seal but not the fuel elements contained within the cask. Thus, only radioactivity produced by activation of chemical deposits located on the outside of the fuel elements corrosion deposits can be released. Since research reactor fuel is not significantly plagued by corrosion deposit formation, corrosion deposits are negligible for research reactor spent fuel. Although the accident phenomenology specified for category 6 is more severe than that for category 5, and that for category 5 is more severe than that for category 4, in the Environmental Assessment all three of these categories were assigned a conditional probability of occurrence of 0.0004. Since increasing accident severity should mean decreasing accident probability, the conditional probabilities assigned to these categories should not be identical. Although the Environmental Assessment release fractions given in the table were retained as the base case for analysis, a method to develop new estimates of the conditional probabilities of occurrence for categories 4, 5, and 6 was formulated. That method is presented below.

D.5.3.1.2 EIS Source Term Probability Considerations

Table D-22 presents a sequence of events that encompasses the accident conditions associated with accident severity categories 4, 5, and 6. This sequence of events provides a reasonable description of a severe collision between large ships that leads both to a severe fire and to a release of radioactivity from the violated spent fuel cask.

This construct allows source term probabilities (P_{st}) to be estimated as the product of the probabilities of occurrence for the seven events. Table D-23 shows how values for P_{st} were calculated in this analysis for accident severity categories 4 through 6. $P_{collision}$ and $P_{severe\ fire}$ were estimated from ship accident data. Because data were sparse for some of the ports studied, these probabilities were not developed separately for each port (i.e., dependencies on port traffic were neglected). P_{hold} and $P_{engulfing\ fire}$ were derived from ship specifications (number of cargo holds and the dimensions of these holds for the prototypic breakbulk freighter used in the impact and crush analyses). P_{impact} and P_{crush} were estimated, as is described in

Table D-21 Accident Severity Categories Used in the EIS

Accident Severity Category	*Accident Conditions*	*Conditional Probabilities*	*Radionuclide Release Fractions*[a]
1	Conditions do not exceed those for a Type B package; no release of contents.	0.603	Co 0 Kr 0 Cs 0 Ru 0 Part 0
2	Conditions equal to those for Type B certification tests; no release of contents.	0.395	Co 0 Kr 0 Cs 0 Ru 0 Particulate 0
3	Seal damage creates leak path, but fuel undamaged; only corrosion deposits, if present, released from package.	0.002	Co 0.012 Kr 0 Cs 0 Ru 0 Particulate 0
4	Impact damage great enough to cause damage to spent fuel; fuel particulates and fission gases may be released.	0.0004	Co 0.012 Kr 0.010 Cs 0.00000001 Ru 0.00000001 Particulate 0.00000001
5	Impact damage to seals plus fire severe enough to cause the cask to leak with release of fission gases, volatiles, and particulates.	0.0004	Co 0.012 Kr 0.100 Cs 0.0009 Ru 0.000001 Particulate 0.00000005
6	Severe impact damage plus fire severe enough to oxidize fuel with release of greater amounts of volatiles than Category 5.	0.0004	Co 0.012 Kr 0.100 Cs 0.00098 Ru 0.000042 Particulate 0.00000005

[a]*No credit was taken for the deposition of fission product vapors or aerosols released from a failed cask onto surfaces of the ship or cargo.*

Table D-22 Event Sequence for a Severe Ship Accident

Event	*Event Probability*
Collision between large ships	$P_{collision}$
Foreign research reactor spent nuclear fuel hold struck	P_{hold}
Foreign research reactor spent nuclear fuel hold penetrated (the cask and fuel are subjected to impact forces)	P_{impact}
Cargo compression (the cask is subjected to crush forces)	P_{crush}
Severe fire eusues	$P_{severe\ fire}$
Fire engulfs the cask (heat loads are sufficient to vaporize cesium)	$P_{engulfing\ fire}$
Convective flow of air through cask causes ruthenium to oxidize	$P_{convection}$

Attachment D4, using the methods of Minorsky (Minorsky, 1959) and results from previous studies of ship accidents (ORI, 1981b). $P_{convection}$ was estimated by review of data on fires and on the temperatures required to oxidize ruthenium to RuO_4, which is necessary to yield the higher ruthenium release fractions.

Table D-23 EIS Source Term Probability Expressions

Accident Severity Category	*Probability*
4	$P_{st} = P_{collision}P_{hold}(P_{impact} + P_{crush})$
5	$P_{st} = P_{collision}P_{hold}(P_{impact} + P_{crush})P_{severe\ fire}P_{engulfing\ fire}$
6	$P_{st} = P_{collision}P_{hold}(P_{impact} + P_{crush})P_{severe\ fire}P_{engulfing\ fire}P_{convection}$

D.5.3.1.3 Probabilities Developed From Accident Data

Fifteen years of Lloyd's casualty data (Lloyds, 1991) and previous studies of ship accidents (Warwick, 1976; SRI, 1978; ORI, 1981a; Abkowitz, 1985) were reviewed to develop (1) the probability of a severe collision ($P_{collision}$) between large ships that occurs dockside in ports or while sailing in port channels, and (2) the probability that such a collision leads to a severe fire (P_{fire}).

Collision Probability

Ship accident casualty data for the years 1978 through 1993 and U.S. port call data for the years 1992 and 1993 were obtained from Lloyd's Maritime Information Services, Inc. Searches of the port call data for the 2-year period 1992-1993 identified the number of port calls made in U.S. ports by all ships, all dry cargo ships, and all dry cargo ships of deadweight 10 to 20 thousand long tons (equivalent to approximately 10,160 to 20,321 metric tons or 11,200 to 22,400 tons). The searches were performed twice, once restricting the results to collision that occurred in port waters only and once adding collisions that occurred in restricted approaches (rivers) that lead to the port. The addition of restricted approach waters was done to permit comparison to results from the literature that included or seemed to include collisions in the river that leads to a port.

The collision frequency per port call is based on a relatively small numbers of collisions. The 15 years of Lloyd's data contained only 69 collisions that occurred in U.S. ports or the restricted river waters that lead to them. Because of this, it is inappropriate to select a value for $P_{collision}$ that is significantly more precise than an order-of-magnitude estimate. The Lloyd's data indicate that for all types of commercial vessels in all U.S. ports, the number of collisions per port call is 0.000077. Other studies provide a range of values for collisions per port call (Warwick, 1976; SRI, 1978; ORI, 1981a; Abkowitz, 1985); however, the Lloyd's database is the most inclusive and the largest (based on approximately 900,000 port calls), so the result based on their data was used here. As discussed earlier, only an order-of-magnitude value is warranted, so the 0.000077 collision per port call was rounded up to 0.0001 ($P_{collision} = 0.0001$).

Probability of Severe Fires

The sources of information cited above were examined to determine an estimate of the probability of a severe fire, given a ship collision. Four estimates of this probability were developed. The 15 years of Lloyd's casualty data contains 1,073 ship collisions in ports located anywhere in the world. Eleven of these collisions led to fires, five caused extensive fire damage, and one involved buckling of structures due to thermal loads. Therefore, the Lloyd's data suggest that the chance that a ship collision leads to a severe fire is 5/1073 = 0.0045.

Only one of the 83 collisions identified by Warwick and Anderson (Warwick, 1976) led to a fire. However, that fire consumed one of the ships involved, the *Sea Witch*. Thus, the Warwick and Anderson data suggest that the chance that a collision will lead to a severe fire is 1/83 = 0.012.

Only 17 of the 391 collisions in the Abkowitz and Galarraga study (Abkowitz, 1985) led to fires of any severity. Thus, the probability that a collision leads to a fire of any severity is 17/391 = 0.044. SRI data suggest that about five percent of all ship fires involve an entire hold (SRI, 1978). Thus, the chance that a ship fire on a cargo ship will involve an entire hold is about 0.05. Combining these last two results allows the probability that a cargo ship collision leads to a severe fire to be estimated as follows:

$$(\textit{fires per collision}) \times (\textit{fires involving an entire hold per fire}) =$$

$$(4.4 \times 10^{-2}) \times (0.05) = 0.0022 \textit{ severe fires per collision}$$

Fires on cargo ships were reviewed by several countries for the International Maritime Organization. The French submission (IMO, 1992) to the International Maritime Organization developed data for 599 cargo ship fires that took place during the 11-year period 1978-1988. Only 2 of the 599 fires were caused by ship collisions. Thus, the probability that a collision leads to a fire of any severity is 2/599 = 0.017. Of the 599 fires, 122 led to immediate total loss, and 195 led to damage first thought to be repairable but which later was determined to be beyond repair. Thus, the chance that a fire is severe is greater than 122/599 = 0.20 and less than (122+195)/599 = 0.53. If the average of these two estimates is used, then the probability that a collision leads to a severe fire can again be calculated as was just done above:

$$(\textit{fires per collision}) \times (\textit{fires resulting in total loss per fire}) =$$

$$(1.7 \times 10^{-2}) \times (0.37) = 0.0063 \textit{ severe fires per collision}$$

If these four estimates for severe fires per collision are averaged, a value of 0.0063 results. Rounding to the nearest order-of-magnitude suggests that $P_{\text{severe fire}} = 0.01$ is a reasonable estimate for the chance that a severe fire will be caused by a ship collision.

No credit is taken for fighting of hold fires during accidents that occur in U.S. ports, all of which have fire fighting equipment, even though fighting of hold fires with water should keep fire temperatures well below those assumed in this study.

D.5.3.1.4 Probability of Mechanical Loads That Cause Damage

A severe ship collision could damage a spent nuclear fuel transportation cask and the elements contained in the cask by subjecting the cask to impact forces, crush forces, and/or thermal loads. Because force is the derivative of energy with distance, both impact forces and crush forces at any penetration distance (d) can be estimated by differentiating expressions that give the dependence on distance of the kinetic energy that is dissipated during the collision. Attachment D4 provides the details of this analysis. In Section 1 of Attachment D4, the kinetic energy associated with ship collisions is discussed. Next, in Section 2, the impact forces required to damage a cask and/or the fuel elements inside the cask are estimated. The crush forces required to damage a cask or the fuel elements inside the cask are described in Section 3.

The kinetic energy associated with ship collisions has been studied (Minorsky, 1959) and extended to develop correlations between penetration depth and energy absorbed. The methodology addresses the evaluation of the kinetic energy, impact forces, and crush forces and their relationship to the impact and crush probabilities (P_{impact} and P_{crush}) associated with ship collisions. The results of this evaluation concluded that P_{crush} is equal to 0.40, and P_{impact} is equal to 0.0.

D.5.3.1.5 Probabilities Developed From Ship Design Data

Two probabilities can be derived from the general ship design data, P_{hold} and $P_{engulfing\ fire}$. The first of these probabilities addresses the likelihood that the collision results in damage to the hold in which the spent nuclear fuel cask resides. (If the cask is stowed in an aft hold and the collision results in damage to a forward hold, no cask damage would be expected.) The second probability addresses the likelihood that the severe fire resulting from the accident (see Section D.5.3.1.3) is located in the same hold and on the same deck as the cask of spent nuclear fuel.

If foreign research reactor spent fuel casks were shipped one at a time, as is assumed here, then P_{hold}, the probability that the hold that contains the cask is the hold that is struck, can be approximated by $1/N_{hold}$, where N_{hold} is the number of holds in the ship transporting the spent nuclear fuel cask. The representative breakbulk freighter used in the impact and crush analyses described below has seven holds. Therefore, for this prototypic ship, $P_{hold} = 1/7 = 0.143$.

The total cargo area of this typical breakbulk freighter is about 3,066 m^2 (33,000 ft^2). Each hold includes two, three, or four decks. Together, the seven holds encompass 21 decks. Thus, the area of each deck is about 3,066/21 = 146 m^2 (33,000/21 = 1,600 ft^2). The Pegase cask used as a prototype in this study has a 2.1-m by 3-m (7-ft by 10-ft) base. This cask should be completely engulfed by a pool fire that has a diameter of 9.1 m (30 ft), provided that the fire occurs in the same hold and on the same deck that the cask is stored on. Since a pool fire of diameter 9.1 m (30 ft) occupies about 65 m^2 (700 ft^2), any engulfing fire will probably involve an entire deck in a hold. If a collision can lead to a fire on any deck in the hold, the $P_{engulfing\ fire} = 1/21$. Limiting the location of the fire to the struck hold or an adjacent hold reduces the number of decks on which the fire could occur. In this case, the number of holds of interest is approximately ten, and therefore, $P_{engulfing\ fire} = 1/10$. Using the larger estimate gives $P_{engulfing\ fire} = 0.1$.

D.5.3.1.6 Probability of Convective Flow Through the Failed Cask

Nonuniform heating of the cask during engulfing fires is expected to produce substantial flow of gases through the cask if two or more small holes or one medium hole have been produced in the cask by the ship collision. Because transportation cask bottoms and lid seats are welded to the cylindrical shell of the cask using full-penetration welds that are as strong or stronger than the parent material, when the cask shell is subjected to a severe stress (e.g., high impact or crush forces), the cask shell should yield before the welds fail. In fact, extra-regulatory 97 km/hr (60 mph) drop tests produced large plastic strains in the cylindrical shell of the test cask without failing its welds (Ludwigsen and Ammerman, 1995). Thus, during a ship collision, crush forces should collapse the cask walls inward without producing catastrophic failure of the lid, its seat, or the welds that attach the seat or the bottom of the cask to the cask walls. Therefore, an unusual configuration of cargo and/or deformed ship structures must be produced during the ship collision in order to subject the cask to forces that will produce failures substantially worse than failure of the lid seal. Either the lid seat must be bent significantly, or at least two penetrations must break, or the cask walls must be sheared or punctured. Although data for such failures is lacking, because cask designs normally do not fail by these mechanisms, the probability that a failure substantially worse than seal failure occurs is conservatively assumed to be no larger than 0.1, therefore $P_{convection} = 0.1$.

D.5.3.1.7 Source Term Probability Values

Table D-24 summarizes the estimates developed for the probabilities that enter the EIS source term probability expressions presented in Table D-23.

Table D-24 Estimated Values for Probabilities in Source Term Probability Expressions

Severity Category	*Probability*	*Estimated Value*[a]
	$P_{collision}$	0.0001
	P_{hold}	0.143
	P_{impact}	0.0
	P_{crush}	0.40
	$P_{severe\ fire}$	0.01
	$P_{engulfing\ fire}$	0.1
	$P_{convection}$	0.1
4	$P_{st} = P_{collision}P_{hold}(P_{impact} + P_{crush})$	0.000006
5	$P_{ST} = P_{collision}P_{hold}(P_{impact} + P_{crush})P_{severe\ fire}P_{engulfing\ fire}$	5×10^{-9}
6	$P_{ST} = P_{collision}P_{hold}(P_{impact} + P_{crush})P_{severe\ fire}P_{engulfing\ fire}\ P_{convection}$	6×10^{-10}

[a]*Severity category 6 is a subset of severity category 5, which in turn is a subset of severity category 4. Therefore, the final estimated value for each P was adjusted to account for this.*

D.5.3.1.8 Source Term Magnitudes

In MACCS, source term magnitudes (M_{sti}) are given by the product of the inventory (I_i) of radionuclides (i) available for release and the fraction (f_i) of that inventory that is released during the accident being examined. Therefore,

$$M_{sti} = I_i f_i.$$

Cask radionuclide inventories were developed for three types of research reactor fuel — Training, Research, Isotope, General Atomic (TRIGA), RHF, and BR2 — for use in the port accident analysis (see Appendix B). Table D-25 presents these inventories. Because it is partly metallic, the TRIGA fuel may undergo exothermic oxidation if exposed to air while at elevated temperatures during an accident involving an enveloping fire.

Because of the large number of casks that might be used to transport foreign research reactor spent nuclear fuel, analyses could not be performed for all possible cask/inventory combinations. Since the size of the cask, rather than the details of its construction, determines the size of the cask's inventory, construction details were obtained for one typical spent nuclear fuel transportation cask, and these construction data were the basis for analyses that depended on cask properties. See Appendix B for description and figures of transportation casks.

For base case analyses, the values for the release fractions (f_i) for the five representative elements, cobalt, krypton, cesium, ruthenium, and other (particulate), presented in Table D-21, were taken to be the same as the values presented that were used in the Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994d). During the sensitivity studies described below, MACCS calculations were performed that used release fraction values and an inventory for foreign research reactor spent nuclear fuel that were taken from the DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (Programmatic SNF&INEL Final EIS) (DOE, 1995). Although both the Environmental Assessment and the Programmatic SNF&INEL Final EIS contain release fractions for all six of the severity categories used in the Environmental Assessment, calculations were not performed for the first two categories, because cask failure does not occur for either category, and only a limited number of sensitivity calculations were performed for category 3 because only

Table D-25 Curie Content of Fully Loaded Transportation Casks for Three Representative Fuel Types

Nuclide	Fuel		
	BR-2	RHF	TRIGA
Hydrogen-3	8.6	37	13
Krypton-85	2,470	1,070	363
Strontium-89	40,800	17,600	275
Strontium-Yttrium-90	20,800	8,930	3,160
Yttrium-91	73,000	31,400	4,560
Zirconium-95	107,000	46,300	6,480
Niobium-95	220,000	94,900	12,800
Ruthenium-103, Rh-103m	8,900	3,770	844
Ruthenium-106, Rh-106m	21,500	9,160	2,540
Tin-123	427	184	27
Antimony-125	890	381	119
Tellurium-125m	212	91	29
Tellurium-127m	887	382	56
Tellurium-129m	189	80	23
Cesium-134	16,400	4,000	1,160
Cesium-137	20,600	8,870	3,190
Cerium-141	5,740	2,440	697
Cerium-144	312,000	135,000	25,500
Promethium-147	48,300	24,600	7,020
Promethium-148m	75	29	47
Europium-154	620	163	42
Europium-155	130	46	23
Uranium-234	0.0009	0.0004	0.0001
Uranium-235	0.014	0.01	0.008
Uranium-238	0.0003	0.0002	0.007
Plutonium-238	64	10	3
Plutonium-239	1.8	0.09	0.6
Plutonium-240	1.2	0.4	2
Plutonium-241	284	68	213
Americium-241	0.4	0.1	0.4
Americium-242m	0.001	0.0001	0.009
Americium-243	0.004	0.004	0.0004
Curium-244	1.3	0.009	0.007
Curium-242	1.8	0.1	3

corrosion product are released in a category 3 accident, and only minor amounts of corrosion product deposits form on research reactor spent nuclear fuel. To examine the possible impacts of corrosion products release, during the sensitivity studies, one category 3 accident calculation was performed during which 350 Ci of Co-60 was the only nuclide released, and one calculation was performed that added the same amount of Co-60 to the base case calculation.

D.5.3.1.9 Source Term Timing and Sensible Heat

Ship accident source terms may have both a puff (an immediate release of most material) and a tail (a gradual release of the material over an extended time), where the puff follows the mechanical failure of the cask due to the collision forces, and the tail is produced by the slow heating of the cask contents by an ensuing fire. Because ship collisions are short duration events, if the collision causes a mechanical release,

it should be of relatively short duration and the gases released from the cask should be cold (no significant sensible heat content) and thus not subject to plume rise. Conversely, because a substantial engulfing fire that burns for approximately an hour is required to heat both the cask and the spent nuclear fuel elements in the cask to temperatures where cesium compounds (for example, CsOH) vaporize to a significant extent, thermal releases will be delayed (release won't occur until about one hour after the collision) and will not take place rapidly (release duration of about one hour). Of course, if cask failure is caused by thermal rather than mechanical loads, any radioactivity released inside of the cask by the collision will not be released from the cask until the cask fails due to those thermal loads. Moreover, if heat loads cause the fuel elements in the cask to fail at essentially the same time that the cask seals fail due to thermal stress, a delayed short duration release could occur. Thus, ship accident source terms can have four release patterns: (1) a single short (15 minute) release caused by the mechanical forces engendered by the collision; (2) a single short (15 minute) release caused by the mechanical forces engendered by the collision followed by a longer (60 minute) release caused by the thermal loads produced by an ensuing fire; (3) a single long duration (60 minute) release caused by thermal loads on the cask if the collision does not lead to failure but an ensuing fire does; and (4) a single delayed short (15 minute) duration release if cask failure and burst rupture of fuel elements occur together.

Because a substantial engulfing fire of significant duration is required to cause a thermal release, for such thermal releases the radioactivity released from the failed cask will be assumed to be released into the fire plume, which typically will have a bulk gas temperature of about 1,200°K (1,700°F). Therefore, the sensible heat content of that plume will be 100 kilowatts for severity category 5 releases and 150 kilowatts for severity category 6 releases.

The start time and duration of the four release patterns described above are presented in Table D-26. For base case calculations, the first release pattern will be assumed for severity Category 4 accidents and the second pattern for severity Category 5 and 6 accidents. The third and forth release patterns will be examined by sensitivity studies.

Table D-26 Release Timing Patterns

Pattern	*Puff*		*Tail*	
	Release Start (min)	*Release Duration (min)*	*Release Start (min)*	*Release Duration (min)*
1	0	10		
2	0	10	60	60
3			60	60
4			90	10

D.5.3.2 Population Distributions

MACCS calculations require as input a population distribution and site-specific weather conditions. The populations along each of the sixteen compass sectors (N, NNE, NE, etc.) are used to determine the exposed population for each combination of site weather and wind rose conditions. Depending upon the shape of the plume, the exposed population includes the people along one or more adjacent sectors.

The required population distributions were generated for two locations at each of thirteen ports. Table D-27 lists the ports selected for examination in this study.

Table D-27 Ports Analyzed

Coast	High Population	Medium Population	Low Population
East	Philadelphia, PA New York, NY	Hampton Roads, VA Jacksonville, FL	Charleston, SC MOTSU, NC Savannah, GA Wilmington, NC
West	Long Beach, CA	Concord NWS, CA Portland, OR Tacoma, WA	
Gulf			Galveston, TX

Two accident locations were considered for each port, one at dockside and one channel location near the population center where a major ship collision would be possible. Two exceptions were made for ports able to share the same channel accident location due to their close proximity to each other. These exceptions are the Port of Wilmington and MOTSU, NC, as well as the Wando Terminal and the Charleston NWS in greater Charleston, SC. Population distributions were constructed on a compass-sector polar coordinate grid that has eleven radial interval (1.6, 3.2, 4.8, 6.4, 8.0, 16, 32, 48, 64, and 20 km or 1, 2, 3, 4, 5, 10, 20, 30, 40, and 50 mi). The distributions were constructed from 1990 block census data using Sandia's SECPOP90 code (Humphreys et al., 1994). The coordinates of the midpoint of the compass-sector polar coordinate grid were selected by inspection of navigational maps for the ports examined. Table D-28 gives the coordinates of these dockside and channel locations, which represent the selected locations for possible accidents. The population distributions generated by SECPOP90 represent the population in an 80.5 km (50 mi) radius around each potential accident site.

The 26 population distributions constructed (two per port) using SECPOP90 were entered into the site data file for the dockside or channel accident location at each of the thirteen ports. Examination of these files shows that many of the cells in the 26 population distributions are empty because they are covered by water (ocean, rivers, bays, harbor channels).

At many ports, the work force population is probably much larger than the residential population, at least in the commercial area near to the port. Therefore, the work force population was estimated for one port, Elizabeth, and added to the distribution that has been constructed for that port using SECPOP90. Then, during the sensitivity studies, the effect of the work force population on consequences of accidents at Elizabeth was examined.

D.5.3.3 Meteorological Data

MACCS calculations examine all possible combinations of a representative set of weather sequences and a representative set of population distributions. MACCS calculations require a site wind rose, to give the exposure probability of the compass sector population distributions and one year of hourly readings of wind speed, atmospheric stability, and rainfall rate. These data may be recorded either at the accident site or at some not-too-distant meteorological station that has similar meteorology and topography as the accident site. These data are used to determine dispersion as a function of downwind transport distance. Site wind rose and rainfall data were available for each of the ports studied. One year of hourly meteorological data was available from National Weather Service Stations located in the port city for only two of the 13 ports studied. For the other 11 ports, hourly meteorological data recorded at a nearby National Weather Service station was used during the base case calculations. Table D-29 presents the locations of the National Weather Service Stations where the hourly meteorological data files used in this study were recorded, and indicates the ports with which each file was used.

Table D-28 Accident Location Map Coordinates

Port	Location	Description	Coordinates	
			Latitude	Longitude
Elizabeth, NJ (for New York)	Dock	Marine Terminal, Sealand Pier	40°39'35"N	74°08'52"W
	Channel	Narrows	40° 36'29"N	74° 02'21"W
Philadelphia, PA	Dock	Packer Avenue Marine Terminal(container berths)	39° 53'55"N	75° 08'09"W
	Channel	Commodore Berry Fixed Bridge	39° 49'43"N	75° 22'18"W
Norfolk, VA (for Hampton Roads)	Dock	Portsmouth Marine Terminal	36° 51'25"N	76° 19'45"W
	Channel	Willoughby Bank, Northside	36° 59'57"N	76° 18'43"W
MOTSU, NC	Dock	Sunny Point, Wharf 1	33° 59'39"N	77° 51'21"W
	Channel	Lower Swash Channel	33° 54'39"N	78° 01'12"W
Charleston, SC	Dock	Pier at Wando Terminal	32° 49'51"N	79° 53'34"W
	Dock	Naval Weapons Station	32°56'12"N	79°56'11"W
	Channel	Commercial anchorage area D	32° 47'05"N	79° 55'10"W
Savannah, GA	Dock	Savannah Ocean Terminal	32° 05'00"N	81° 05'18"W
	Channel	Intersection Savannah River and Intracoastal Waterway at Elba Island Cut	32° 04'26"N	80° 58'17"W
Galveston, TX	Dock	Container Terminal, Pier 9	29° 19'00"N	94° 46'53"W
	Channel	Cross of Bolivar Roads Channel and Galveston Channel	29° 20'27"N	95° 46'12"W
Concord NWS, CA	Dock	Naval Weapons Station	38° 03'32"N	122° 00'49"W
	Channel	San Francisco Bay Temporary Anchorage No. 7 Shipping Lane Route	37°49'24"N	122°23'47"W
Tacoma, WA	Dock	Port of Tacoma Pier No. 7	47° 16'03"N	122°24'49"W
	Channel	Intersection of 4 shipping lanes in Puget Sound north of Port Townsend	48° 11'24"N	122°49'48"W
Wilmington, NC	Dock	Main Dock Wilmington Terminal	34° 13'03"N	77°57'09"W
	Channel	Lower Swash Channel	33° 54'39"N	78° 01'12"W
Jacksonville, FL	Dock	Blount Island Terminal	30° 23'16"N	81° 33'00"W
	Channel	St. John's River Ferry crossing to Mayport	30° 23'40"N	81° 26'00"W
Long Beach, CA	Dock	Pier E	33° 45'43"N	118° 12'31"W
	Channel	Breakwater East Side	33° 43'23"N	118° 10'53"W
Portland, OR	Dock	Terminal 2	45° 32'54"N	122° 41'56"W
	Channel	St. Johns Bridge	45° 35'04"N	122° 45'58"W

Table D-29 Locations of National Weather Service Stations

Port	National Weather Service Station
Elizabeth, NJ	New York City, NY
Philadelphia, PA	New York City, NY
Norfolk, VA	Cape Hatteras, NC
MOTSU, NC; Wilmington, NC	Cape Hatteras, NC
Charleston, SC; Savannah, GA; Jacksonville, FL	Charleston, SC
Long Beach, CA; Concord NWS, CA	Santa Maria, CA
Portland, OR; Tacoma, WA	Seattle, WA
Galveston, TX	Lake Charles, LA

Although MACCS calculations can use constant meteorology, one year of hourly meteorological data is preferred because adverse results are often the result of meteorological sequences that involve changing meteorological conditions. MACCS uses an importance sampling method to find these less probable sequences that yield adverse results. The sampling method examines all of the 8,760 weather sequences in one year of hourly data and selects the start times of the approximately 100 weather sequences that are used during a variable meteorology calculation. The impact of using constant versus variable meteorology is the subject of one of the sensitivity calculations.

D.5.4 MACCS Calculations

All of the MACCS calculations performed during this study used a source term probability of one. Thus, the consequence estimates generated and the probabilities associated with those estimates are conditional on the release of the source term (i.e., the estimates are conditional on the accident having occurred).

For any source term, a MACCS calculation generates results for all possible combinations of a representative set of weather sequences and a representative set of exposed downwind populations. Since the probability of occurrence of each weather sequence and the exposure probability of each population distribution is known, the variability of consequences due to weather and population conditional on the accident having occurred can be displayed by plotting the probability that a consequence magnitude will be equaled or exceeded against consequence magnitude. Such a plot is called a Complementary Cumulative Distribution Function.

Two types of MACCS accident consequence calculations were performed, base case calculations and sensitivity calculations. Base case calculations used:

- the inventories given in Table D-25,
- the release fractions presented in Table D-21 for severity categories 4, 5, and 6,
- the release timings specified in Table D-26 (pattern 1 was used for severity category 4 releases and pattern 3 for category 5 and 6 releases),
- one year of hourly meteorological data recorded at the National Weather Service Station listed in Table D-29, and
- population distributions calculated using SECPOP90 for the dockside and channel locations presented in Table D-28.

Population distributions and other site-specific data are input to MACCS via a site data file.

Sensitivity calculations modified the input used in the base case calculations to identify the influence on consequences:

- of using variable meteorology recorded offsite at a nearby National Weather Service station rather than constant meteorology recorded onsite at the harbor,
- of using source terms that contained 17 nuclides for which acute health effect dose conversion factors were not available,
- of neglecting the enhanced shielding likely to be afforded to population near to the harbor by the masonry buildings that typify construction in urban commercial neighborhoods,

- of using release fractions developed for the Programmatic SNF&INEL Final EIS (DOE, 1995) for truck and rail accidents,
- of adding the harbor work force population to the residential population distribution,
- of modeling extremely high temperature effects on aluminum-based and TRIGA fuels release fractions,
- of modeling accidents that lead to severe fires using a puff and a tail (two segment release) rather than only a puff, and
- of adding cobalt-60 to the inventory so that corrosion products release can be calculated.

The results of these sensitivity calculations are presented in Section D.5.4.3.

Both the variable meteorology and the constant meteorology MACCS calculations performed for this study consist of a large number of individual trials (about 1,750 trials for each variable meteorology calculation; about 1,150 trials for each constant meteorology calculation). By accumulating the results of the individual trials, an expected (mean) result and a Complementary Cumulative Distribution Function are generated for each output quantity (result) calculated. In addition, for each output calculated, the value of the largest result obtained for any individual trial, the probability of occurrence of that trial, and the weather sequence used in that trial are saved by MACCS.

D.5.4.1 Acute Health Effects

The MACCS code can calculate the numbers of fatalities and injuries that are caused by acute exposures that occur over time periods of a few days (due to dose to the stomach or the intestines) to one year (due to internal dose to the lungs). Of the seven acute injuries that MACCS can examine, prodromal vomiting is the acute injury most likely to appear at low doses and dose rates. Because the occurrence of acute health effects would be cause for considerable concern, acute fatalities and cases of prodromal vomiting were calculated during every MACCS run made during this study; and the output of every run was inspected to see if either acute effect had occurred. Inspection of all of the MACCS output generated showed that no acute fatalities and no cases of prodromal vomiting were ever predicted to occur for any output quantile (i.e., the mean result, all quantile values on the Complementary Cumulative Distribution Function, and the result obtained for the least favorable weather sequence were all zero for acute fatalities and cases of prodromal vomiting).

D.5.4.2 Base Case Calculations

The base case calculations estimated the consequences that might result if any one of nine ship accidents (the combination of three cask inventories presented in Table D-25, with the release fractions for accident severity categories 4, 5, and 6) occurred at any of the 25 accident locations examined (one dockside and one channel location at each of the 13 ports, except MOTSU and Wilmington, which share a channel accident location). Thus, 3 x 3 x 25 = 225 base case MACCS calculations were performed and are presented in this assessment.

D.5.4.2.1 Typical Output

Table D-30 presents MACCS output for one base case calculation, the calculation for the channel accident location at Elizabeth performed with the BR-2 source term and severity categories 4, 5, and 6 release fractions. Using as an example the severity category 5 results, the first group of results in this table are

Table D-30 Sample Output from MACCS

SITE=NEW LOC=CHANNEL INV=BR-2 ST=EA4 VAR MET=NYC

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.9TH	PEAK CONS	PEAK PROB	PEAK TRIAL
EARLY + CHRONIC RESULTS											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.5675	4.16E-05	2.43E-07	1.30E-04	2.15E-04	4.38E-04	8.02E-04	1.13E-03	2.50E-04	73
CAN FAT/TOTAL	0-80.5 KM	1.0000	1.64E-04	7.15E-05	4.38E-04	6.29E-04	9.89E-04	1.29E-03	1.50E-03	2.50E-04	73
POPULATION DOSE (SV)											
EDEWBODY TOT LIF	0-1.6 KM	0.5675	9.45E-04	5.56E-06	2.90E-03	4.96E-03	1.03E-02	1.56E-02	2.56E-02	2.50E-04	73
EDEWBODY TOT LIF	0-8.1 KM	0.8016	2.42E-03	1.01E-03	7.02E-03	9.14E-03	1.60E-02	2.50E-02	3.30E-02	2.50E-04	73
EDEWBODY TOT LIF	0-16.1 KM	0.8848	3.04E-03	1.18E-03	8.37E-03	1.16E-02	2.04E-02	2.59E-02	3.37E-02	2.50E-04	73
EDEWBODY TOT LIF	0-80.5 KM	1.0000	3.76E-03	1.65E-03	9.67E-03	1.33E-02	2.27E-02	3.13E-02	3.41E-02	2.50E-04	73
CENTERLINE DOSE AT SOME DISTANCES (SV)											
EDEWBODY TOT LIF	0-1.6 KM	1.0000	5.98E-07	4.09E-07	1.13E-06	1.48E-06	2.39E-06	NOT-FOUND	3.66E-06	3.45E-03	73
CHRONIC RESULTS ONLY											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.5675	3.50E-05	2.28E-07	1.11E-04	1.86E-04	3.60E-04	6.54E-04	9.36E-04	2.50E-04	73
CAN FAT/TOTAL	0-80.5 KM	1.0000	1.40E-04	6.32E-05	3.67E-04	5.20E-04	8.26E-04	1.11E-03	1.24E-03	2.50E-04	73
EDEWBODY POP. DOSE (SV)	0-80.5 KM										
TOTAL LONG-TERM PATHWAYS DOSE		1.0000	3.35E-03	1.46E-03	9.03E-03	1.21E-02	2.05E-02	2.58E-02	2.96E-02	2.50E-04	73
TOTAL INGESTION PATHWAYS DOSE		1.0000	5.65E-05	4.28E-05	1.18E-04	1.60E-04	2.50E-04	3.14E-04	3.57E-04	2.24E-05	28
LONG-TERM GROUNDSHINE DOSE		1.0000	2.66E-03	1.15E-03	6.82E-03	9.90E-03	1.43E-02	2.13E-02	2.39E-02	2.50E-04	73
ECONOMIC COST MEASURES ($)	0-80.5 KM										
TOTAL ECONOMIC COSTS		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CROP DISPOSAL COST		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
MAXIMUM LONG-TERM ACTION DISTANCE (KM)											
CROP DISPOSAL DIST.		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

SITE=NEW LOC=CHANNEL INV=BR-2 ST=EA5 VAR MET=NYC

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.9TH	PEAK CONS	PEAK PROB	PEAK TRIAL
EARLY + CHRONIC RESULTS											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.6818	9.85E-02	2.32E-06	2.71E-02	2.60E-01	2.60E+00	6.86E+00	1.75E+01	7.65E-05	45
CAN FAT/TOTAL	0-80.5 KM	1.0000	2.90E+00	1.22E+00	7.38E+00	1.17E+01	2.91E+01	3.89E+01	5.53E+01	7.65E-05	45
POPULATION DOSE (SV)											
EDEWBODY TOT LIF	0-1.6 KM	0.6818	2.36E+00	5.22E-05	6.05E-01	6.11E+00	6.21E+01	1.74E+02	4.21E+02	7.65E-05	45
EDEWBODY TOT LIF	0-8.1 KM	0.8854	1.57E+01	2.08E-01	3.35E+01	6.23E+01	3.23E+02	5.13E+02	9.27E+02	9.81E-05	45
EDEWBODY TOT LIF	0-16.1 KM	0.9686	3.30E+01	3.13E+00	9.40E+01	1.48E+02	5.48E+02	8.50E+02	1.25E+03	1.16E-05	44
EDEWBODY TOT LIF	0-80.5 KM	1.0000	6.93E+01	2.84E+01	1.80E+02	2.61E+02	7.09E+02	9.41E+02	1.33E+03	7.65E-05	45
CENTERLINE DOSE AT SOME DISTANCES (SV)											
EDEWBODY TOT LIF	0-1.6 KM	1.0000	1.17E-03	1.60E-06	5.29E-03	6.69E-03	1.52E-02	NOT-FOUND	4.12E-02	1.06E-03	45
CHRONIC RESULTS ONLY											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.6371	9.83E-02	2.14E-07	2.71E-02	2.60E-01	2.60E+00	6.86E+00	1.75E+01	7.65E-05	45
CAN FAT/TOTAL	0-80.5 KM	0.9943	2.90E+00	1.22E+00	7.38E+00	1.17E+01	2.88E+01	3.89E+01	5.53E+01	7.65E-05	45
EDEWBODY POP. DOSE (SV)	0-80.5 KM										
TOTAL LONG-TERM PATHWAYS DOSE		0.9943	6.92E+01	2.84E+01	1.80E+02	2.61E+02	7.09E+02	9.41E+02	1.33E+03	7.65E-05	45
TOTAL INGESTION PATHWAYS DOSE		0.9943	1.70E+00	2.50E-01	6.02E+00	8.37E+00	1.07E+01	1.24E+01	1.66E+01	9.94E-06	25
LONG-TERM GROUNDSHINE DOSE		0.9942	6.72E+01	2.61E+01	1.77E+02	2.57E+02	7.09E+02	9.41E+02	1.32E+03	7.65E-05	45
ECONOMIC COST MEASURES ($)	0-80.5 KM										
TOTAL ECONOMIC COSTS		0.0038	1.80E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	5.64E+03	2.44E-03	16
CROP DISPOSAL COST		0.0038	1.33E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	4.15E+03	2.44E-03	16
MAXIMUM LONG-TERM ACTION DISTANCE (KM)											
CROP DISPOSAL DIST.		0.0038	6.12E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	1.61E+00	3.80E-03	16

SITE=NEW LOC=CHANNEL INV=BR-2 ST=EA6 VAR MET=NYC

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.9TH	PEAK CONS	PEAK PROB	PEAK TRIAL
EARLY + CHRONIC RESULTS											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.6713	8.02E-02	7.79E-07	1.48E-02	1.73E-01	2.59E+00	7.25E+00	1.91E+01	7.65E-05	45
CAN FAT/TOTAL	0-80.5 KM	1.0000	2.84E+00	1.14E+00	6.92E+00	1.14E+01	3.15E+01	4.36E+01	6.04E+01	7.65E-05	45
POPULATION DOSE (SV)											
EDEWBODY TOT LIF	0-1.6 KM	0.6713	1.92E+00	1.98E-05	3.56E-01	3.03E+00	6.20E+01	1.88E+02	4.59E+02	7.65E-05	45
EDEWBODY TOT LIF	0-8.1 KM	0.8854	1.43E+01	2.64E-02	2.23E+01	4.25E+01	3.31E+02	5.41E+02	1.01E+03	7.65E-05	45
EDEWBODY TOT LIF	0-16.1 KM	0.9686	3.02E+01	5.66E-01	7.75E+01	1.32E+02	5.82E+02	8.62E+02	1.37E+03	1.16E-05	44
EDEWBODY TOT LIF	0-80.5 KM	1.0000	6.77E+01	2.71E+01	1.61E+02	2.57E+02	7.46E+02	1.08E+03	1.45E+03	7.65E-05	45
CENTERLINE DOSE AT SOME DISTANCES (SV)											
EDEWBODY TOT LIF	0-1.6 KM	1.0000	9.53E-04	3.45E-07	3.32E-03	6.25E-03	1.75E-02	NOT-FOUND	4.50E-02	1.06E-03	45
CHRONIC RESULTS ONLY											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.5752	8.01E-02	0.00E+00	1.48E-02	1.73E-01	2.59E+00	7.25E+00	1.93E+01	7.65E-05	45
CAN FAT/TOTAL	0-80.5 KM	0.9879	2.83E+00	1.14E+00	6.92E+00	1.13E+01	3.15E+01	4.36E+01	6.04E+01	7.65E-05	45
EDEWBODY POP. DOSE (SV)	0-80.5 KM										
TOTAL LONG-TERM PATHWAYS DOSE		0.9879	6.76E+01	2.71E+01	1.61E+02	2.57E+02	7.46E+02	1.08E+03	1.45E+03	7.65E-05	45
TOTAL INGESTION PATHWAYS DOSE		0.9879	1.83E+00	2.61E-01	6.23E+00	8.54E+00	1.09E+01	1.29E+01	1.81E+01	9.94E-06	25
LONG-TERM GROUNDSHINE DOSE		0.9879	6.54E+01	2.46E+01	1.59E+02	2.55E+02	7.42E+02	1.06E+03	1.44E+03	7.65E-05	45
ECONOMIC COST MEASURES ($)	0-80.5 KM										
TOTAL ECONOMIC COSTS		0.0038	1.80E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	5.64E+01	2.44E-03	16
CROP DISPOSAL COST		0.0038	1.33E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	4.15E+03	2.44E-03	16
MAXIMUM LONG-TERM ACTION DISTANCE (KM)											
CROP DISPOSAL DIST.		0.0038	6.12E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	1.61E+00	3.80E-03	16

Health Effect Cases. The first health effect considered is the number of cancer fatalities expected to occur among the population located within 1.6 km (1 mi) of the accident location. For this population group, the table shows:

- that the probability of getting a nonzero result is 0.6818 which means that not even a fractional cancer fatality was predicted to occur in this population group for 31.82 percent of the approximately 1,750 trials run during this calculation (conversely, at least a fractional cancer death was predicted to occur in 68.18 percent of the trials);
- that the expected (mean) number of cancer fatalities for this population group is 0.098;
- that the 90th and 99th quantiles of the Complementary Cumulative Distribution Function of cancer fatalities for this population group have values of 0.0271 and 2.60; and
- that the largest number of cancer fatalities predicted for this population group for any weather trial was 17.5, that this result had a probability of occurrence of 0.000077, and that the 45th weather sequence selected by the importance sampling scheme led to this result. While the number of LCF is two orders of magnitude higher than the mean, the probability of occurrence of this peak value is four orders of magnitude lower than the mean value.

Figure D-54 presents the Complementary Cumulative Distribution Function for cancer fatalities among the population located within 1.6 km (1 mi) of the channel accident location at Elizabeth, for a severity category 5 accident release fraction. Figure D-54 shows that there is one chance in a thousand (probability = 0.001) that an accident that leads to a severity category 5 release from a cask that contains the BR-2 inventory will produce at least seven cancer deaths. Thus, the 99.9th quantile of the Complementary Cumulative Distribution Function has a value of about seven. Inspection of the Complementary Cumulative Distribution Function also shows that the tail of the distribution has a probability of occurrence of 0.0001 and a magnitude of about 17. These are the values produced by the weather trial that led to the largest result among the full set of weather trials.

The results presented in Table D-30 illustrate a pattern that is general over all of the calculations performed: population dose increases monotonically as distance range increases (e.g., 0-1.6 km, 0-8.1 km, ..., 0-80.5 km). Although not shown in Table D-30, this applies to cancer deaths also. Note that all doses are in Sieverts.

The centerline dose to an individual standing under the plume decreases monotonically with increasing distance, as it should, until it reaches the last computational interval (64.4-80.5 km or 40-50 mi, not shown on D-30) where counter-intuitively it increases. It increases because, during all calculations, rain was artificially forced to occur when the radioactive plume entered this computational interval in order to ensure that all radioactive particulates in the plume deposit onto the ground before the plume exits the computational grid at 80.5 km (50 mi) from the accident location. Deposition of all remaining radioactive particulates onto the ground within the last computational interval ensures that all radioactivity that might enter food pathways at some time after the accident does enter those pathways.

Another pattern that can be seen from Table D-30 is that total population dose is caused almost entirely by long-term groundshine exposures (external direct exposure to radiation emitted by radionuclides deposited on the ground).

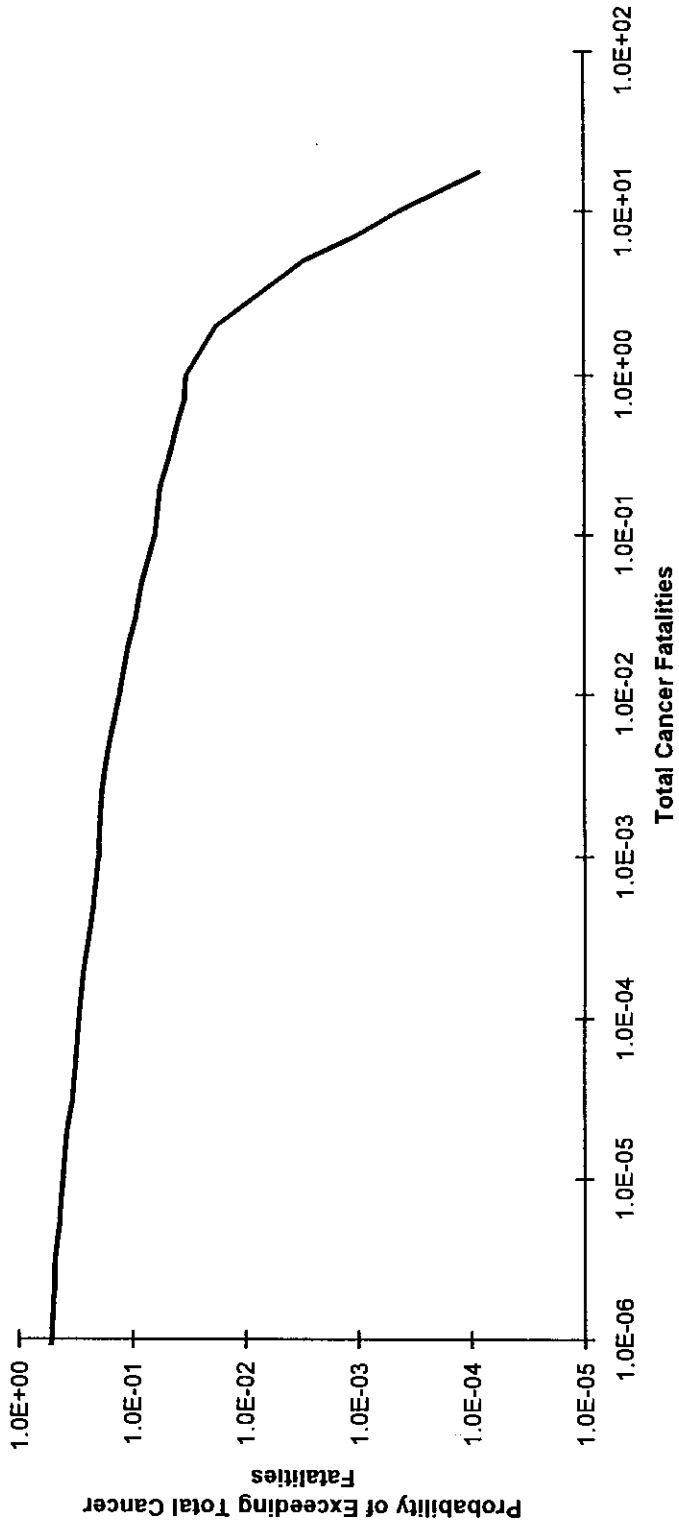

Figure D-54 Total Cancer Fatalities, 0-1.6 km (0-1 mi), Elizabeth Channel, Variable Meteorology, BR-2 Inventory, Severity Category 5

D-30 also shows that the economic losses (costs) caused by the accident are very small (expected value of $18.00; peak value of $5,640) and are entirely attributable to the disposal of contaminated crops and milk by farms located close to the accident site (the largest disposal distance found was 1.6 km or 1 mi). This also is typical of the MACCS output for all accidents analyzed.

The values of mean (expected) centerline dose (D_{cl}) (not shown in Table D-30) for severity category 5 release fractions are plotted versus distance (d) in Figure D-55. The figure shows that on a log-log plot dose decreases linearly with distance with a slope very close to minus one. Therefore, as one would expect, individual centerline dose is inversely proportional to distance ($D_{C1} \propto 1/d$).

Table D-30 presents a breakdown of long-term population dose (calculated as a wholebody dose by the Effective Dose Equivalent method and thus labeled EDEWBODY POP. DOSE) by exposure pathways. Inspection of this breakdown and comparison of the total long-term pathway dose to the total population dose for release category 5, mean results, in the 0-80.5 km (0-50 mi) ranges shows:

- that the total population dose 6,930 rem (69.3 Sv), is almost entirely due to the 6,920 rem (69.2 Sv) dose delivered by long-term exposure pathways;
- that short-term (acute) pathways deliver only a minor dose of 10 rem (0.1 Sv), which is the difference between the 69.3 Sv and the 69.2 Sv;
- that the long-term dose of 6,920 rem (69.2 Sv) is caused mainly by direct exposure pathways [6,750 rem (67.5 Sv)] and only secondarily by ingestion pathways [170 rem (1.7 Sv)];
- that groundshine [6,720 rem (67.2 Sv)] causes almost all of the long-term direct dose; resuspension (external direct exposure to radiation emitted by radionuclides resuspended from the ground) causes the rest of the long-term pathway dose, 30 rem (0.3 Sv);
- that the dose from radioactivity deposited directly on food crops [125 rem (1.25 Sv)] or on grass consumed by milk cows [30 rem (0.30 Sv)] accounts for most ingestion dose; and
- that the rest of the ingestion dose is caused by root uptake [to food crops, 10 rem (0.10 Sv); to grass and fodder crops, 4 rem (0.04 Sv)] with drinking of contaminated water causing only a very small dose of 1 rem (0.01 Sv).

D.5.4.2.2 Principal Base Case Consequence Results

Accident consequence mean (expected) values for whole body population dose and total cancer fatalities for the distance range 0-80.5 km (0-50 mi), and individual centerline dose and individual centerline cancer risk for the distance range 0-1.6 km (0-1 mi) are presented in Table D-31. Table D-32 provides 99.9th quantile values for whole body population dose and total cancer fatalities for the range 0-80.5 km (0-50 mi). Table D-33 presents the largest (peak) result calculated for individual centerline dose and cancer risk in the range 0-1.6 km or 0-1 mi. Table D-34 presents probabilities of the largest results calculated.

Table D-31 shows that the expected total population dose within 80.5 km (50 mi) of the accident location varies from a low of 0.000972 person-rem (0.00000972 person-Sv) for the MOTSU dock calculation that used the TRIGA inventory, severity category 4 (EA4) release fractions, and Cape Hatteras weather to a high of 6,930 person-rem (69.3 person-Sv) for the Elizabeth channel calculation that used the BR-2 inventory, severity category 5 (EA5) release fractions, and New York City weather. Since the total

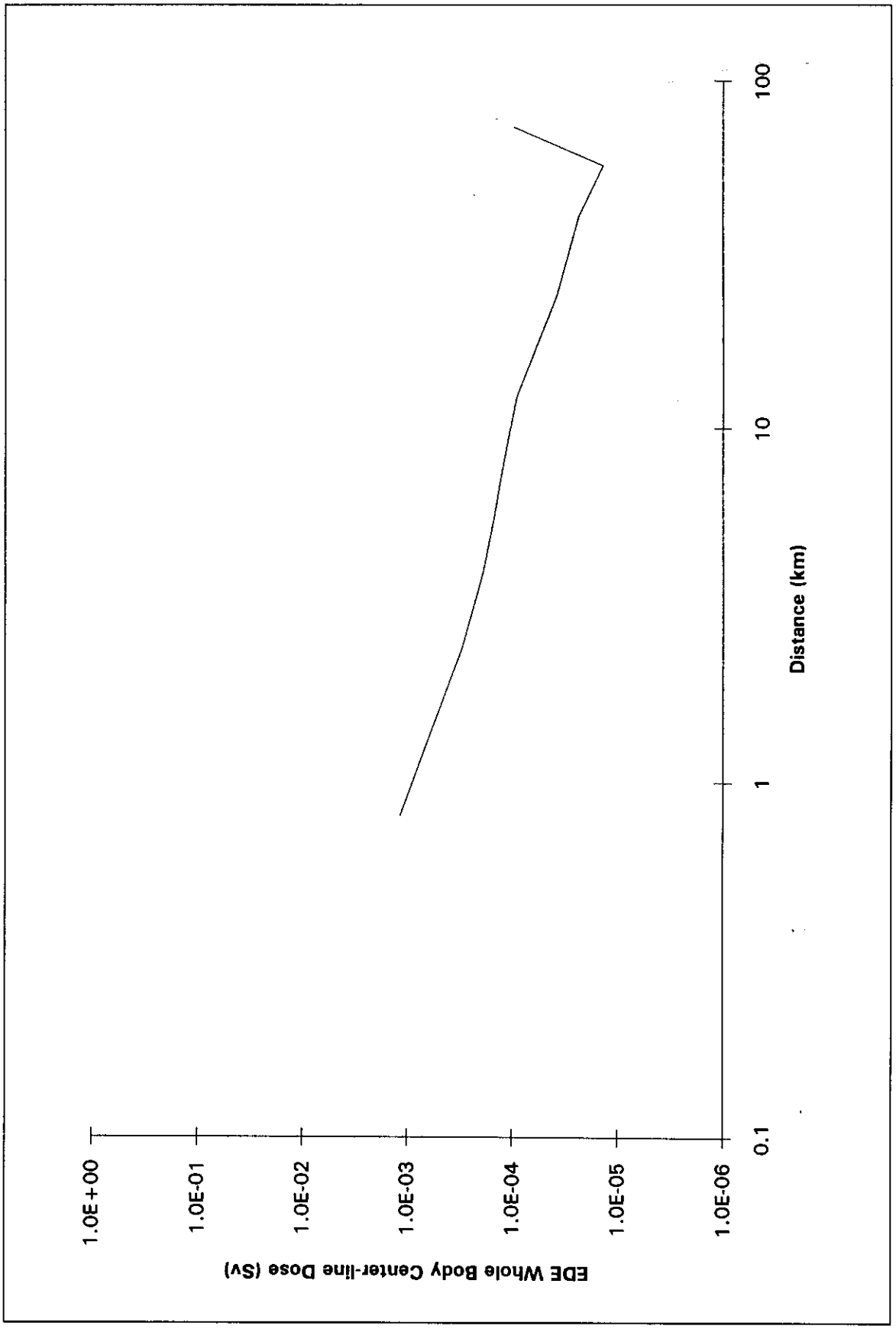

Figure D-55 Mean Effective Dose Equivalent Whole Body Center-Line Dose (Sv) vs Distance, Elizabeth Channel, Variable Meteorology, BR-2 Inventory, Severity Category 5

Table D-31 Mean Results, Variable Meteorology

EDE Whole Body Population Dose, 0-80 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	2.40E-04	4.15E+00	4.13E+00	9.55E-05	1.54E+00	1.53E+00	2.97E-05	5.32E-01	5.26E-01
CHA-C	3.78E-04	4.18E+00	4.21E+00	1.51E-04	1.55E+00	1.56E+00	4.58E-05	5.35E-01	5.37E-01
CNC-D	4.40E-04	2.07E+01	2.21E+01	1.76E-04	7.97E+00	8.51E+00	5.43E-05	2.78E+00	2.97E+00
CNC-C	9.44E-04	3.31E+01	3.40E+01	3.77E-04	1.29E+01	1.32E+01	1.13E-04	4.52E+00	4.63E+00
GAL-D	7.26E-04	1.44E+01	1.58E+01	2.90E-04	5.45E+00	6.00E+00	8.94E-05	1.90E+00	2.08E+00
GAL-C	3.23E-04	1.42E+01	1.55E+01	1.29E-04	5.36E+00	5.89E+00	4.13E-05	1.86E+00	2.04E+00
JAC-D	2.79E-04	6.82E+00	6.76E+00	1.11E-04	2.55E+00	2.52E+00	3.48E-05	8.84E-01	8.71E-01
JAC-C	2.58E-04	5.33E+00	5.45E+00	1.03E-04	1.99E+00	2.03E+00	3.22E-05	6.87E-01	6.99E-01
LOS-D	2.13E-03	4.71E+01	4.82E+01	8.52E-04	1.85E+01	1.89E+01	2.54E-04	6.49E+00	6.62E+00
LOS-C	8.09E-04	4.26E+01	4.40E+01	3.23E-04	1.67E+01	1.73E+01	9.72E-05	5.86E+00	6.05E+00
MOT-D	7.24E-05	2.08E+00	2.21E+00	2.88E-05	7.45E-01	7.91E-01	9.72E-06	2.54E-01	2.70E-01
NEW-D	2.33E-03	6.55E+01	6.51E+01	9.30E-04	2.58E+01	2.56E+01	2.77E-04	9.07E+00	9.00E+00
NEW-C	3.76E-03	6.93E+01	6.77E+01	1.50E-03	2.73E+01	2.67E+01	4.46E-04	9.60E+00	9.37E+00
NOR-D	5.52E-04	8.54E+00	8.32E+00	2.20E-04	3.25E+00	3.15E+00	6.69E-05	1.13E+00	1.09E+00
NOR-C	3.02E-04	6.65E+00	6.64E+00	1.21E-04	2.51E+00	2.50E+00	3.70E-05	8.73E-01	8.67E-01
PHI-D	1.77E-03	2.81E+01	2.78E+01	7.08E-04	1.10E+01	1.08E+01	2.11E-04	3.84E+00	3.79E+00
PHI-C	8.48E-04	2.74E+01	2.81E+01	3.39E-04	1.07E+01	1.09E+01	1.02E-04	3.74E+00	3.83E+00
POR-D	7.70E-04	1.17E+01	1.19E+01	3.07E-04	4.45E+00	4.50E+00	9.32E-05	1.55E+00	1.56E+00
POR-C	5.33E-04	1.12E+01	1.15E+01	2.13E-04	4.26E+00	4.36E+00	6.52E-05	1.48E+00	1.51E+00
SAV-D	5.60E-04	4.91E+00	5.01E+00	2.23E-04	1.80E+00	1.83E+00	6.82E-05	6.18E-01	6.28E-01
SAV-C	1.34E-04	3.82E+00	3.93E+00	5.32E-05	1.38E+00	1.42E+00	1.75E-05	4.74E-01	4.86E-01
SEA-C	1.31E-04	7.54E+00	8.29E+00	5.21E-05	2.84E+00	3.12E+00	1.68E-05	9.86E-01	1.08E+00
TAC-D	5.55E-04	1.73E+01	1.83E+01	2.21E-04	6.67E+00	7.02E+00	6.81E-05	2.33E+00	2.45E+00
TAC-C	3.87E-04	1.43E+01	1.50E+01	1.55E-04	5.50E+00	5.73E+00	4.75E-05	1.92E+00	2.00E+00
WIL-D	3.80E-04	4.82E+00	5.02E+00	1.51E-04	1.79E+00	1.86E+00	4.66E-05	6.19E-01	6.43E-01
WIL-C	9.65E-05	2.07E+00	2.20E+00	3.84E-05	7.47E-01	7.96E-01	1.24E-05	2.56E-01	2.72E-01
CHN-D	1.67E-04	4.76E+00	4.74E+00	6.63E-05	1.76E+00	1.77E+00	2.13E-05	6.09E-01	6.08E-01

Total Cancer Fatalities, 0-80 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.05E-05	1.89E-01	1.90E-01	4.20E-06	6.97E-02	6.95E-02	1.24E-06	2.40E-02	2.39E-02
CHA-C	1.66E-05	1.90E-01	1.93E-01	6.65E-06	7.01E-02	7.08E-02	1.90E-06	2.41E-02	2.43E-02
CNC-D	1.91E-05	8.96E-01	9.57E-01	7.63E-06	3.44E-01	3.67E-01	2.23E-06	1.20E-01	1.28E-01
CNC-C	4.10E-05	1.41E+00	1.45E+00	1.65E-05	5.48E-01	5.62E-01	4.63E-06	1.92E-01	1.97E-01
GAL-D	3.17E-05	6.39E-01	7.02E-01	1.27E-05	2.41E-01	2.65E-01	3.70E-06	8.35E-02	9.17E-02
GAL-C	1.39E-05	6.30E-01	6.92E-01	5.57E-06	2.37E-01	2.60E-01	1.71E-06	8.20E-02	9.01E-02
JAC-D	1.22E-05	3.07E-01	3.06E-01	4.88E-06	1.14E-01	1.13E-01	1.45E-06	3.94E-02	3.91E-02
JAC-C	1.13E-05	2.42E-01	2.49E-01	4.51E-06	8.95E-02	9.16E-02	1.34E-06	3.09E-02	3.15E-02
LOS-D	9.32E-05	1.99E+00	2.04E+00	3.75E-05	7.79E-01	7.97E-01	1.04E-05	2.73E-01	2.79E-01
LOS-C	3.51E-05	1.80E+00	1.86E+00	1.41E-05	7.05E-01	7.28E-01	3.96E-06	2.47E-01	2.55E-01
MOT-D	3.16E-06	9.94E-02	1.06E-01	1.25E-06	3.53E-02	3.76E-02	4.13E-07	1.20E-02	1.28E-02
NEW-D	1.02E-04	2.75E+00	2.73E+00	4.09E-05	1.08E+00	1.07E+00	1.13E-05	3.80E-01	3.77E-01
NEW-C	1.64E-04	2.90E+00	2.84E+00	6.62E-05	1.14E+00	1.12E+00	1.83E-05	4.01E-01	3.92E-01
NOR-D	2.42E-05	3.77E-01	3.70E-01	9.71E-06	1.42E-01	1.39E-01	2.76E-06	4.94E-02	4.82E-02
NOR-C	1.32E-05	2.96E-01	2.97E-01	5.30E-06	1.11E-01	1.11E-01	1.53E-06	3.85E-02	3.84E-02
PHI-D	7.75E-05	1.20E+00	1.19E+00	3.12E-05	4.66E-01	4.61E-01	8.67E-06	1.63E-01	1.61E-01
PHI-C	3.70E-05	1.17E+00	1.20E+00	1.49E-05	4.53E-01	4.66E-01	4.19E-06	1.59E-01	1.63E-01
POR-D	3.37E-05	5.18E-01	5.27E-01	1.35E-05	1.95E-01	1.98E-01	3.85E-06	6.78E-02	6.87E-02
POR-C	2.33E-05	4.97E-01	5.12E-01	9.34E-06	1.87E-01	1.92E-01	2.70E-06	6.50E-02	6.66E-02
SAV-D	2.46E-05	2.28E-01	2.34E-01	9.87E-06	8.28E-02	8.46E-02	2.83E-06	2.84E-02	2.90E-02
SAV-C	5.88E-06	1.79E-01	1.85E-01	2.32E-06	6.45E-02	6.65E-02	7.42E-07	2.20E-02	2.27E-02
SEA-C	5.61E-06	3.37E-01	3.70E-01	2.23E-06	1.26E-01	1.39E-01	6.93E-07	4.36E-02	4.79E-02
TAC-D	2.41E-05	7.54E-01	7.95E-01	9.66E-06	2.88E-01	3.04E-01	2.81E-06	1.00E-01	1.06E-01
TAC-C	1.68E-05	6.26E-01	6.55E-01	6.75E-06	2.38E-01	2.49E-01	1.95E-06	8.30E-02	8.66E-02
WIL-D	1.66E-05	2.19E-01	2.29E-01	6.66E-06	8.09E-02	8.43E-02	1.93E-06	2.79E-02	2.90E-02
WIL-C	4.22E-06	9.76E-02	1.04E-01	1.68E-06	3.50E-02	3.74E-02	5.22E-07	1.20E-02	1.28E-02
CHN-D	6.76E-06	2.17E-01	2.19E-01	2.67E-06	7.98E-02	8.03E-02	8.44E-07	2.75E-02	2.76E-02

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-31 Mean Results, Variable Meteorology (Continued)

Individual Center-line EDE Whole Body Dose, 0-1.6 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
CHA-C	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
CNC-D	1.07E-06	2.28E-04	2.17E-04	4.29E-07	9.01E-05	8.59E-05	1.21E-07	3.17E-05	3.02E-05
CNC-C	1.07E-06	2.28E-04	2.17E-04	4.29E-07	9.01E-05	8.59E-05	1.21E-07	3.17E-05	3.02E-05
GAL-D	9.29E-07	6.52E-04	6.91E-04	3.71E-07	2.58E-04	2.74E-04	1.05E-07	9.08E-05	9.62E-05
GAL-C	9.29E-07	6.52E-04	6.91E-04	3.71E-07	2.58E-04	2.74E-04	1.05E-07	9.08E-05	9.62E-05
JAC-D	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
JAC-C	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
LOS-D	1.07E-06	2.28E-04	2.17E-04	4.29E-07	9.01E-05	8.59E-05	1.21E-07	3.17E-05	3.02E-05
LOS-C	1.07E-06	2.28E-04	2.17E-04	4.29E-07	9.01E-05	8.59E-05	1.21E-07	3.17E-05	3.02E-05
MOT-D	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
NEW-D	5.98E-07	1.17E-03	9.53E-04	2.39E-07	4.63E-04	3.77E-04	6.81E-08	1.63E-04	1.33E-04
NEW-C	5.98E-07	1.17E-03	9.53E-04	2.39E-07	4.63E-04	3.77E-04	6.81E-08	1.63E-04	1.33E-04
NOR-D	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
NOR-C	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
PHI-D	1.01E-06	6.31E-04	6.59E-04	4.02E-07	2.50E-04	2.61E-04	1.14E-07	8.78E-05	9.16E-05
PHI-C	1.01E-06	6.31E-04	6.59E-04	4.02E-07	2.50E-04	2.61E-04	1.14E-07	8.78E-05	9.16E-05
POR-D	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
POR-C	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
SAV-D	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
SAV-C	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
SEA-C	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
TAC-D	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
TAC-C	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
WIL-D	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
WIL-C	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
CHN-D	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05

Individual Center-line Cancer Risk, 0-1.6 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
CHA-C	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
CNC-D	5.12E-08	9.50E-06	9.06E-06	2.08E-08	3.75E-06	3.58E-06	5.36E-09	1.32E-06	1.26E-06
CNC-C	5.12E-08	9.50E-06	9.06E-06	2.08E-08	3.75E-06	3.58E-06	5.36E-09	1.32E-06	1.26E-06
GAL-D	4.43E-08	2.72E-05	2.88E-05	1.80E-08	1.08E-05	1.14E-05	4.63E-09	3.78E-06	4.01E-06
GAL-C	4.43E-08	2.72E-05	2.88E-05	1.80E-08	1.08E-05	1.14E-05	4.63E-09	3.78E-06	4.01E-06
JAC-D	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
JAC-C	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
LOS-D	5.12E-08	9.50E-06	9.06E-06	2.08E-08	3.75E-06	3.58E-06	5.36E-09	1.32E-06	1.26E-06
LOS-C	5.12E-08	9.50E-06	9.06E-06	2.08E-08	3.75E-06	3.58E-06	5.36E-09	1.32E-06	1.26E-06
MOT-D	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
NEW-D	2.80E-08	4.88E-05	3.97E-05	1.14E-08	1.93E-05	1.57E-05	2.96E-09	6.78E-06	5.52E-06
NEW-C	2.80E-08	4.88E-05	3.97E-05	1.14E-08	1.93E-05	1.57E-05	2.96E-09	6.78E-06	5.52E-06
NOR-D	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
NOR-C	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
PHI-D	4.80E-08	2.63E-05	2.75E-05	1.95E-08	1.04E-05	1.09E-05	5.01E-09	3.66E-06	3.82E-06
PHI-C	4.80E-08	2.63E-05	2.75E-05	1.95E-08	1.04E-05	1.09E-05	5.01E-09	3.66E-06	3.82E-06
POR-D	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
POR-C	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
SAV-D	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
SAV-C	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
SEA-C	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
TAC-D	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
TAC-C	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
WIL-D	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
WIL-C	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
CHN-D	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-32 99.9th Quantile Results, Variable Meteorology

EDE Whole Body Population Dose, 0-80 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.20E-03	4.63E+01	5.26E+01	3.82E-04	1.71E+01	2.04E+01	1.30E-04	6.21E+00	7.13E+00
CHA-C	3.40E-03	9.03E+01	9.75E+01	1.23E-03	3.70E+01	4.13E+01	3.98E-04	1.22E+01	1.37E+01
CNC-D	3.31E-03	9.47E+01	1.02E+02	1.29E-03	3.41E+01	3.56E+01	3.87E-04	1.17E+01	1.22E+01
CNC-C	1.06E-02	1.55E+02	1.73E+02	3.77E-03	5.55E+01	5.20E+01	1.17E-03	1.91E+01	1.97E+01
GAL-D	5.03E-03	1.33E+02	1.38E+02	2.01E-03	4.90E+01	4.92E+01	5.35E-04	1.66E+01	1.92E+01
GAL-C	1.37E-03	8.70E+01	1.16E+02	6.77E-04	3.72E+01	3.93E+01	2.01E-04	1.30E+01	1.38E+01
JAC-D	1.29E-03	7.23E+01	7.86E+01	5.59E-04	2.78E+01	3.07E+01	1.45E-04	1.01E+01	1.05E+01
JAC-C	1.26E-03	6.40E+01	6.77E+01	5.28E-04	2.45E+01	2.69E+01	1.55E-04	8.55E+00	9.22E+00
LOS-D	NOT-FOUND	2.67E+02	2.70E+02	NOT-FOUND	1.02E+02	1.04E+02	NOT-FOUND	3.32E+01	3.64E+01
LOS-C	3.66E-03	2.19E+02	2.41E+02	1.38E-03	9.72E+01	9.84E+01	5.13E-04	3.14E+01	3.12E+01
MOT-D	5.03E-04	2.46E+01	2.63E+01	1.80E-04	8.87E+00	9.48E+00	6.25E-05	2.90E+00	2.95E+00
NEW-D	1.20E-02	5.87E+02	6.37E+02	4.24E-03	2.42E+02	2.52E+02	1.34E-03	8.29E+01	8.68E+01
NEW-C	3.13E-02	9.41E+02	1.08E+03	1.19E-02	3.86E+02	3.99E+02	3.52E-03	1.33E+02	1.42E+02
NOR-D	3.40E-03	1.03E+02	1.10E+02	1.40E-03	4.04E+01	4.41E+01	4.59E-04	1.30E+01	1.39E+01
NOR-C	1.62E-03	9.02E+01	1.00E+02	7.13E-04	3.35E+01	3.56E+01	2.06E-04	1.16E+01	1.24E+01
PHI-D	8.45E-03	3.10E+02	3.32E+02	3.45E-03	1.15E+02	1.18E+02	1.00E-03	4.65E+01	5.03E+01
PHI-C	5.07E-03	2.86E+02	2.98E+02	2.03E-03	1.05E+02	1.08E+02	6.95E-04	4.13E+01	4.63E+01
POR-D	3.16E-03	1.09E+02	1.13E+02	1.18E-03	4.62E+01	4.90E+01	3.55E-04	1.40E+01	1.59E+01
POR-C	3.48E-03	1.01E+02	1.04E+02	1.35E-03	3.55E+01	4.37E+01	4.02E-04	1.19E+01	1.46E+01
SAV-D	5.26E-03	1.26E+02	1.28E+02	2.10E-03	5.56E+01	5.98E+01	6.05E-04	2.04E+01	2.16E+01
SAV-C	8.01E-04	4.37E+01	4.44E+01	3.33E-04	1.59E+01	1.72E+01	9.87E-05	5.65E+00	5.42E+00
SEA-C	8.69E-04	5.53E+01	5.99E+01	3.54E-04	2.10E+01	2.29E+01	1.12E-04	7.43E+00	8.11E+00
TAC-D	1.55E-03	8.36E+01	8.67E+01	7.01E-04	3.51E+01	3.60E+01	2.09E-04	1.19E+01	1.23E+01
TAC-C	1.87E-03	9.69E+01	1.01E+02	7.23E-04	3.65E+01	4.21E+01	2.20E-04	1.29E+01	1.52E+01
WIL-D	4.47E-03	7.09E+01	7.65E+01	1.79E-03	2.75E+01	2.98E+01	5.10E-04	9.33E+00	1.08E+01
WIL-C	1.03E-03	2.25E+01	2.79E+01	3.89E-04	8.70E+00	9.96E+00	1.18E-04	3.00E+00	3.11E+00
CHN-D	7.70E-04	3.69E+01	3.91E+01	3.04E-04	1.36E+01	1.42E+01	9.88E-05	5.22E+00	5.46E+00

Total Cancer Fatalities, 0-80 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	4.33E-05	1.98E+00	2.24E+00	1.63E-05	7.81E-01	8.41E-01	5.50E-06	2.67E-01	2.89E-01
CHA-C	1.30E-04	3.96E+00	4.27E+00	7.14E-05	1.61E+00	1.69E+00	1.60E-05	4.94E-01	5.80E-01
CNC-D	1.43E-04	4.07E+00	4.38E+00	5.84E-05	1.28E+00	1.35E+00	1.60E-05	5.30E-01	5.55E-01
CNC-C	4.60E-04	6.20E+00	6.23E+00	1.80E-04	2.55E+00	2.59E+00	5.25E-05	8.80E-01	8.78E-01
GAL-D	2.07E-04	4.93E+00	5.12E+00	7.64E-05	1.96E+00	2.07E+00	2.18E-05	7.00E-01	7.31E-01
GAL-C	7.04E-05	3.88E+00	4.09E+00	2.89E-05	1.49E+00	1.54E+00	7.95E-06	5.73E-01	5.88E-01
JAC-D	5.80E-05	3.10E+00	3.21E+00	2.38E-05	1.12E+00	1.15E+00	6.67E-06	3.91E-01	4.34E-01
JAC-C	6.04E-05	2.71E+00	2.87E+00	2.29E-05	1.08E+00	1.14E+00	6.50E-06	3.53E-01	3.75E-01
LOS-D	NOT-FOUND	1.03E+01	1.05E+01	NOT-FOUND	4.49E+00	4.52E+00	NOT-FOUND	1.45E+00	1.58E+00
LOS-C	1.98E-04	9.81E+00	1.01E+01	7.39E-05	3.39E+00	3.90E+00	2.08E-05	1.18E+00	1.19E+00
MOT-D	2.32E-05	1.22E+00	1.33E+00	8.61E-06	4.08E-01	4.42E-01	2.67E-06	1.43E-01	1.47E-01
NEW-D	4.40E-04	2.46E+01	2.58E+01	1.72E-04	1.10E+01	1.15E+01	5.49E-05	3.48E+00	3.83E+00
NEW-C	1.29E-03	3.89E+01	4.36E+01	5.34E-04	1.47E+01	1.67E+01	1.40E-04	5.75E+00	6.28E+00
NOR-D	1.55E-04	4.23E+00	4.63E+00	6.51E-05	1.62E+00	1.81E+00	1.85E-05	5.99E-01	6.50E-01
NOR-C	7.23E-05	3.59E+00	3.97E+00	3.08E-05	1.39E+00	1.43E+00	8.46E-06	5.25E-01	5.47E-01
PHI-D	3.54E-04	1.18E+01	1.23E+01	1.29E-04	5.18E+00	5.55E+00	3.88E-05	1.90E+00	2.05E+00
PHI-C	2.03E-04	1.22E+01	1.12E+01	9.95E-05	4.74E+00	4.93E+00	2.85E-05	1.78E+00	1.85E+00
POR-D	1.47E-04	4.90E+00	5.16E+00	6.01E-05	1.96E+00	2.03E+00	1.61E-05	6.89E-01	7.13E-01
POR-C	1.48E-04	3.78E+00	4.71E+00	6.08E-05	1.56E+00	1.84E+00	1.63E-05	5.45E-01	6.19E-01
SAV-D	2.23E-04	5.68E+00	6.22E+00	1.01E-04	2.20E+00	2.45E+00	2.41E-05	7.91E-01	8.82E-01
SAV-C	3.52E-05	1.79E+00	1.93E+00	1.28E-05	6.84E-01	7.04E-01	3.71E-06	2.45E-01	2.62E-01
SEA-C	3.73E-05	2.41E+00	2.61E+00	1.40E-05	9.06E-01	9.68E-01	4.16E-06	3.17E-01	3.47E-01
TAC-D	7.55E-05	3.60E+00	3.74E+00	3.03E-05	1.32E+00	1.42E+00	8.41E-06	5.55E-01	5.74E-01
TAC-C	8.26E-05	3.94E+00	4.73E+00	3.19E-05	1.59E+00	1.87E+00	9.47E-06	5.25E-01	6.20E-01
WIL-D	1.99E-04	2.90E+00	3.25E+00	7.68E-05	1.17E+00	1.21E+00	2.05E-05	3.98E-01	4.27E-01
WIL-C	4.89E-05	1.04E+00	1.16E+00	1.96E-05	4.13E-01	4.66E-01	5.15E-06	1.45E-01	1.55E-01
CHN-D	3.33E-05	1.55E+00	1.70E+00	1.32E-05	6.28E-01	7.15E-01	3.67E-06	2.19E-01	2.32E-01

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-33 Peak Results, Variable Meteorology

EDE Whole Body Population Dose, 0-80 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	2.82E-03	1.04E+02	1.10E+02	1.12E-03	4.07E+01	4.32E+01	3.41E-04	1.43E+01	1.51E+01
CHA-C	4.41E-03	2.12E+02	2.31E+02	1.76E-03	8.56E+01	9.34E+01	5.23E-04	3.00E+01	3.27E+01
CNC-D	3.35E-03	1.02E+02	1.11E+02	1.33E-03	4.03E+01	4.40E+01	4.00E-04	1.42E+01	1.55E+01
CNC-C	1.42E-02	6.71E+02	7.27E+02	5.70E-03	2.66E+02	2.88E+02	1.72E-03	9.34E+01	1.01E+02
GAL-D	5.60E-03	2.33E+02	2.54E+02	2.22E-03	9.59E+01	1.05E+02	6.69E-04	3.36E+01	3.66E+01
GAL-C	2.09E-03	1.85E+02	2.02E+02	8.28E-04	7.23E+01	7.89E+01	2.56E-04	2.53E+01	2.76E+01
JAC-D	2.15E-03	1.10E+02	1.20E+02	8.53E-04	4.28E+01	4.66E+01	2.61E-04	1.50E+01	1.63E+01
JAC-C	1.90E-03	1.12E+02	1.23E+02	7.52E-04	4.59E+01	5.01E+01	2.31E-04	1.61E+01	1.76E+01
LOS-D	1.30E-02	3.94E+02	4.29E+02	5.20E-03	1.56E+02	1.70E+02	1.54E-03	5.49E+01	5.98E+01
LOS-C	5.97E-03	3.47E+02	3.76E+02	2.39E-03	1.37E+02	1.49E+02	7.21E-04	4.83E+01	5.23E+01
MOT-D	7.72E-04	4.61E+01	5.03E+01	2.98E-04	1.71E+01	1.86E+01	9.74E-05	5.88E+00	6.42E+00
NEW-D	1.86E-02	1.13E+03	1.24E+03	7.44E-03	4.48E+02	4.89E+02	2.24E-03	1.58E+02	1.72E+02
NEW-C	3.41E-02	1.33E+03	1.45E+03	1.36E-02	5.25E+02	5.73E+02	4.01E-03	1.85E+02	2.02E+02
NOR-D	4.29E-03	1.66E+02	1.81E+02	1.71E-03	6.49E+01	7.06E+01	5.08E-04	2.27E+01	2.47E+01
NOR-C	2.78E-03	1.64E+02	1.78E+02	1.11E-03	6.44E+01	7.00E+01	3.36E-04	2.26E+01	2.45E+01
PHI-D	1.13E-02	7.15E+02	7.80E+02	4.52E-03	2.82E+02	3.08E+02	1.36E-03	9.91E+01	1.08E+02
PHI-C	8.77E-03	4.87E+02	5.31E+02	3.50E-03	1.92E+02	2.10E+02	1.06E-03	6.75E+01	7.36E+01
POR-D	3.85E-03	1.69E+02	1.85E+02	1.54E-03	6.69E+01	7.30E+01	4.64E-04	2.35E+01	2.57E+01
POR-C	3.94E-03	1.70E+02	1.85E+02	1.57E-03	6.66E+01	7.27E+01	4.67E-04	2.34E+01	2.55E+01
SAV-D	6.18E-03	2.39E+02	2.60E+02	2.46E-03	9.64E+01	1.05E+02	7.32E-04	3.38E+01	3.69E+01
SAV-C	1.43E-03	7.65E+01	8.00E+01	5.66E-04	2.94E+01	3.07E+01	1.75E-04	1.03E+01	1.07E+01
SEA-C	9.93E-04	5.62E+01	6.13E+01	3.93E-04	2.14E+01	2.33E+01	1.22E-04	7.44E+00	8.11E+00
TAC-D	2.67E-03	1.18E+02	1.29E+02	1.06E-03	4.66E+01	5.09E+01	3.23E-04	1.64E+01	1.79E+01
TAC-C	3.13E-03	1.70E+02	1.85E+02	1.25E-03	6.64E+01	7.25E+01	3.78E-04	2.33E+01	2.54E+01
WIL-D	5.50E-03	1.49E+02	1.63E+02	2.20E-03	5.91E+01	6.44E+01	6.51E-04	2.08E+01	2.27E+01
WIL-C	1.18E-03	3.28E+01	3.58E+01	4.71E-04	1.29E+01	1.41E+01	1.41E-04	4.57E+00	4.99E+00
CHN-D	1.76E-03	6.83E+01	7.43E+01	6.97E-04	2.64E+01	2.87E+01	2.14E-04	9.20E+00	1.00E+01

Total Cancer Fatalities, 0-80 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.20E-04	4.41E+00	4.69E+00	4.80E-05	1.72E+00	1.83E+00	1.37E-05	6.02E-01	6.39E-01
CHA-C	1.95E-04	8.90E+00	9.71E+00	7.83E-05	3.60E+00	3.93E+00	2.17E-05	1.26E+00	1.38E+00
CNC-D	1.49E-04	4.31E+00	4.68E+00	5.96E-05	1.68E+00	1.83E+00	1.68E-05	5.90E-01	6.44E-01
CNC-C	6.01E-04	2.80E+01	3.03E+01	2.42E-04	1.11E+01	1.20E+01	6.84E-05	3.89E+00	4.22E+00
GAL-D	2.48E-04	9.97E+00	1.09E+01	9.90E-05	4.08E+00	4.45E+00	2.79E-05	1.43E+00	1.55E+00
GAL-C	9.02E-05	7.89E+00	8.61E+00	3.57E-05	3.07E+00	3.35E+00	1.05E-05	1.07E+00	1.17E+00
JAC-D	9.25E-05	4.71E+00	5.12E+00	3.68E-05	1.82E+00	1.98E+00	1.07E-05	6.37E-01	6.93E-01
JAC-C	8.20E-05	4.71E+00	5.13E+00	3.26E-05	1.93E+00	2.11E+00	9.47E-06	6.77E-01	7.38E-01
LOS-D	5.75E-04	1.64E+01	1.79E+01	2.31E-04	6.50E+00	7.08E+00	6.37E-05	2.29E+00	2.49E+00
LOS-C	2.52E-04	1.45E+01	1.57E+01	1.01E-04	5.73E+00	6.20E+00	2.87E-05	2.01E+00	2.18E+00
MOT-D	3.49E-05	2.10E+00	2.29E+00	1.34E-05	7.69E-01	8.39E-01	4.24E-06	2.65E-01	2.88E-01
NEW-D	7.88E-04	4.72E+01	5.16E+01	3.17E-04	1.87E+01	2.04E+01	8.94E-05	6.56E+00	7.16E+00
NEW-C	1.50E-03	5.53E+01	6.04E+01	6.06E-04	2.19E+01	2.39E+01	1.66E-04	7.69E+00	8.39E+00
NOR-D	1.89E-04	7.09E+00	7.71E+00	7.62E-05	2.75E+00	2.99E+00	2.10E-05	9.63E-01	1.05E+00
NOR-C	1.19E-04	6.92E+00	7.52E+00	4.75E-05	2.71E+00	2.94E+00	1.35E-05	9.50E-01	1.03E+00
PHI-D	4.79E-04	2.99E+01	3.26E+01	1.93E-04	1.18E+01	1.29E+01	5.44E-05	4.14E+00	4.52E+00
PHI-C	3.71E-04	2.04E+01	2.23E+01	1.49E-04	8.03E+00	8.77E+00	4.23E-05	2.82E+00	3.08E+00
POR-D	1.68E-04	7.05E+00	7.70E+00	6.74E-05	2.79E+00	3.04E+00	1.87E-05	9.80E-01	1.07E+00
POR-C	1.74E-04	7.18E+00	7.84E+00	7.00E-05	2.81E+00	3.06E+00	1.94E-05	9.84E-01	1.07E+00
SAV-D	2.74E-04	1.00E+01	1.09E+01	1.10E-04	4.07E+00	4.44E+00	3.04E-05	1.43E+00	1.55E+00
SAV-C	6.15E-05	3.32E+00	3.48E+00	2.44E-05	1.27E+00	1.33E+00	7.14E-06	4.42E-01	4.62E-01
SEA-C	4.27E-05	2.47E+00	2.69E+00	1.69E-05	9.33E-01	1.02E+00	4.99E-06	3.24E-01	3.53E-01
TAC-D	1.14E-04	5.03E+00	5.49E+00	4.54E-05	1.95E+00	2.12E+00	1.31E-05	6.83E-01	7.45E-01
TAC-C	1.33E-04	7.18E+00	7.84E+00	5.34E-05	2.80E+00	3.06E+00	1.51E-05	9.82E-01	1.07E+00
WIL-D	2.43E-04	6.24E+00	6.80E+00	9.80E-05	2.46E+00	2.69E+00	2.69E-05	8.70E-01	9.48E-01
WIL-C	5.22E-05	1.41E+00	1.54E+00	2.10E-05	5.42E-01	5.92E-01	5.94E-06	1.93E-01	2.10E-01
CHN-D	7.30E-05	2.95E+00	3.21E+00	2.90E-05	1.13E+00	1.23E+00	8.51E-06	3.95E-01	4.29E-01

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-33 Peak Results, Variable Meteorology (Continued)

Individual Center-line EDE Whole Body Dose, 0-1.6 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
CHA-C	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
CNC-D	3.66E-06	2.51E-02	2.74E-02	1.46E-06	9.94E-03	1.08E-02	4.06E-07	3.50E-03	3.81E-03
CNC-C	3.66E-06	2.51E-02	2.74E-02	1.46E-06	9.94E-03	1.08E-02	4.06E-07	3.50E-03	3.81E-03
GAL-D	3.66E-06	5.34E-02	5.82E-02	1.46E-06	2.11E-02	2.31E-02	4.06E-07	7.43E-03	8.10E-03
GAL-C	3.66E-06	5.34E-02	5.82E-02	1.46E-06	2.11E-02	2.31E-02	4.06E-07	7.43E-03	8.10E-03
JAC-D	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
JAC-C	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
LOS-D	3.66E-06	2.51E-02	2.74E-02	1.46E-06	9.94E-03	1.08E-02	4.06E-07	3.50E-03	3.81E-03
LOS-C	3.66E-06	2.51E-02	2.74E-02	1.46E-06	9.94E-03	1.08E-02	4.06E-07	3.50E-03	3.81E-03
MOT-D	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
NEW-D	3.66E-06	4.12E-02	4.50E-02	1.46E-06	1.63E-02	1.78E-02	4.06E-07	5.74E-03	6.26E-03
NEW-C	3.66E-06	4.12E-02	4.50E-02	1.46E-06	1.63E-02	1.78E-02	4.06E-07	5.74E-03	6.26E-03
NOR-D	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
NOR-C	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
PHI-D	3.66E-06	6.06E-02	6.61E-02	1.46E-06	2.40E-02	2.62E-02	4.06E-07	8.43E-03	9.20E-03
PHI-C	3.66E-06	6.06E-02	6.61E-02	1.46E-06	2.40E-02	2.62E-02	4.06E-07	8.43E-03	9.20E-03
POR-D	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
POR-C	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
SAV-D	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
SAV-C	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
SEA-C	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
TAC-D	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
TAC-C	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
WIL-D	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
WIL-C	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
CHN-D	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03

Individual Center-line Cancer Risk, 0-1.6 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
CHA-C	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
CNC-D	1.79E-07	1.05E-03	1.14E-03	7.29E-08	4.14E-04	4.52E-04	1.84E-08	1.46E-04	1.59E-04
CNC-C	1.79E-07	1.05E-03	1.14E-03	7.29E-08	4.14E-04	4.52E-04	1.84E-08	1.46E-04	1.59E-04
GAL-D	1.79E-07	2.23E-03	2.43E-03	7.29E-08	8.80E-04	9.60E-04	1.84E-08	3.09E-04	3.37E-04
GAL-C	1.79E-07	2.23E-03	2.43E-03	7.29E-08	8.80E-04	9.60E-04	1.84E-08	3.09E-04	3.37E-04
JAC-D	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
JAC-C	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
LOS-D	1.79E-07	1.05E-03	1.14E-03	7.29E-08	4.14E-04	4.52E-04	1.84E-08	1.46E-04	1.59E-04
LOS-C	1.79E-07	1.05E-03	1.14E-03	7.29E-08	4.14E-04	4.52E-04	1.84E-08	1.46E-04	1.59E-04
MOT-D	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
NEW-D	1.79E-07	1.72E-03	1.88E-03	7.29E-08	6.80E-04	7.42E-04	1.84E-08	2.39E-04	2.61E-04
NEW-C	1.79E-07	1.72E-03	1.88E-03	7.29E-08	6.80E-04	7.42E-04	1.84E-08	2.39E-04	2.61E-04
NOR-D	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
NOR-C	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
PHI-D	1.79E-07	2.53E-03	2.76E-03	7.29E-08	9.98E-04	1.09E-03	1.84E-08	3.51E-04	3.83E-04
PHI-C	1.79E-07	2.53E-03	2.76E-03	7.29E-08	9.98E-04	1.09E-03	1.84E-08	3.51E-04	3.83E-04
POR-D	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
POR-C	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
SAV-D	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
SAV-C	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
SEA-C	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
TAC-D	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
TAC-C	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
WIL-D	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
WIL-C	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
CHN-D	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-34 Probability of Peak Results, Variable Meteorology

EDE Whole Body Population Dose, 0-80 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05
CHA-C	7.77E-06	4.44E-06	4.44E-06	7.77E-06	4.44E-06	4.44E-06	7.77E-06	4.44E-06	4.44E-06
CNC-D	8.47E-04	1.18E-05	1.18E-05	8.47E-04	1.18E-05	1.18E-05	8.47E-04	1.18E-05	1.18E-05
CNC-C	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06
GAL-D	7.80E-06	9.75E-06	9.75E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06
GAL-C	7.80E-06	9.75E-06	9.75E-06	7.80E-06	9.75E-06	9.75E-06	7.80E-06	9.75E-06	9.75E-06
JAC-D	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06
JAC-C	6.08E-06	8.06E-06	8.06E-06	6.08E-06	8.06E-06	8.06E-06	6.08E-06	8.06E-06	8.06E-06
LOS-D	1.83E-03	3.42E-06	3.42E-06	1.83E-03	3.42E-06	3.42E-06	1.83E-03	3.42E-06	3.42E-06
LOS-C	4.91E-06	2.81E-06	2.81E-06	4.91E-06	2.81E-06	2.81E-06	4.91E-06	2.81E-06	2.81E-06
MOT-D	1.67E-05	2.76E-05	2.76E-05	1.67E-05	2.76E-05	2.76E-05	1.67E-05	2.76E-05	2.76E-05
NEW-D	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05
NEW-C	2.50E-04	7.65E-05	7.65E-05	2.50E-04	7.65E-05	7.65E-05	2.50E-04	7.65E-05	7.65E-05
NOR-D	3.51E-04	9.62E-06	9.62E-06	3.51E-04	9.62E-06	9.62E-06	3.51E-04	9.62E-06	9.62E-06
NOR-C	1.07E-05	1.61E-05	1.61E-05	1.07E-05	1.61E-05	1.61E-05	1.07E-05	1.61E-05	1.61E-05
PHI-D	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06
PHI-C	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05
POR-D	1.26E-05	1.09E-05	1.09E-05	1.26E-05	1.09E-05	1.09E-05	1.26E-05	1.09E-05	1.09E-05
POR-C	3.19E-04	1.09E-05	1.09E-05	3.19E-04	1.09E-05	1.09E-05	3.19E-04	1.09E-05	1.09E-05
SAV-D	1.13E-05	6.22E-06	6.22E-06	1.13E-05	6.22E-06	6.22E-06	1.13E-05	6.22E-06	6.22E-06
SAV-C	1.30E-05	1.14E-05	1.14E-05	1.30E-05	1.14E-05	1.14E-05	1.30E-05	1.14E-05	1.14E-05
SEA-C	1.36E-04	6.79E-04	6.79E-04	1.36E-04	6.79E-04	6.79E-04	1.36E-04	9.80E-04	9.80E-04
TAC-D	5.33E-06	1.14E-05	1.14E-05	5.33E-06	1.14E-05	1.14E-05	5.33E-06	1.14E-05	1.14E-05
TAC-C	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05
WIL-D	3.26E-04	1.16E-04	1.16E-04	3.26E-04	1.16E-04	1.16E-04	3.26E-04	1.16E-04	1.16E-04
WIL-C	3.04E-04	1.08E-04	1.08E-04	3.04E-04	1.08E-04	1.08E-04	3.04E-04	1.08E-04	1.08E-04
CHN-D	6.27E-06	3.89E-06	3.89E-06	6.27E-06	3.89E-06	3.89E-06	6.27E-06	3.89E-06	3.89E-06

Total Cancer Fatalities, 0-80 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05
CHA-C	7.77E-06	4.44E-06	4.44E-06	7.77E-06	4.44E-06	4.44E-06	7.77E-06	4.44E-06	4.44E-06
CNC-D	8.47E-04	5.08E-04	5.08E-04	8.47E-04	1.18E-05	1.18E-05	8.47E-04	1.18E-05	1.18E-05
CNC-C	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06
GAL-D	7.80E-06	9.75E-06	9.75E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06
GAL-C	7.80E-06	9.75E-06	9.75E-06	7.80E-06	9.75E-06	9.75E-06	7.80E-06	9.75E-06	9.75E-06
JAC-D	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06
JAC-C	6.08E-06	8.06E-06	8.06E-06	6.08E-06	8.06E-06	8.06E-06	6.08E-06	8.06E-06	8.06E-06
LOS-D	1.83E-03	3.42E-06	3.42E-06	1.83E-03	3.42E-06	3.42E-06	1.83E-03	3.42E-06	3.42E-06
LOS-C	4.91E-06	2.81E-06	2.81E-06	4.91E-06	2.81E-06	2.81E-06	4.91E-06	2.81E-06	2.81E-06
MOT-D	1.67E-05	2.76E-05	2.76E-05	1.67E-05	2.76E-05	2.76E-05	1.67E-05	2.76E-05	2.76E-05
NEW-D	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05
NEW-C	2.50E-04	7.65E-05	7.65E-05	2.50E-04	7.65E-05	7.65E-05	2.50E-04	7.65E-05	7.65E-05
NOR-D	3.51E-04	9.62E-06	9.62E-06	3.51E-04	9.62E-06	9.62E-06	3.51E-04	9.62E-06	9.62E-06
NOR-C	1.07E-05	1.61E-05	1.61E-05	1.07E-05	1.61E-05	1.61E-05	1.07E-05	1.61E-05	1.61E-05
PHI-D	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06
PHI-C	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05
POR-D	2.98E-04	1.09E-05	1.09E-05	2.98E-04	1.09E-05	1.09E-05	2.98E-04	1.09E-05	1.09E-05
POR-C	3.19E-04	1.09E-05	1.09E-05	3.19E-04	1.09E-05	1.09E-05	3.19E-04	1.09E-05	1.09E-05
SAV-D	1.13E-05	6.22E-06	6.22E-06	1.13E-05	6.22E-06	6.22E-06	1.13E-05	6.22E-06	6.22E-06
SAV-C	1.30E-05	1.14E-05	1.14E-05	1.30E-05	1.14E-05	1.14E-05	1.30E-05	1.14E-05	1.14E-05
SEA-C	1.36E-04	6.79E-04	6.79E-04	1.36E-04	6.79E-04	6.79E-04	1.36E-04	6.79E-04	6.79E-04
TAC-D	5.33E-06	1.08E-05	1.08E-05	5.33E-06	1.08E-05	1.08E-05	5.33E-06	1.14E-05	1.14E-05
TAC-C	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05
WIL-D	3.26E-04	1.16E-04	1.16E-04	3.26E-04	1.16E-04	1.16E-04	3.26E-04	1.16E-04	1.16E-04
WIL-C	3.04E-04	1.27E-05	1.27E-05	3.04E-04	1.08E-04	1.08E-04	3.04E-04	1.08E-04	1.08E-04
CHN-D	6.27E-06	3.89E-06	3.89E-06	6.27E-06	3.89E-06	3.89E-06	6.27E-06	3.89E-06	3.89E-06

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-34 Probability of Peak Results, Variable Meteorology (Continued)

Individual Center-line EDE Whole Body Dose, 0-1.6 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
CHA-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
CNC-D	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
CNC-C	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
GAL-D	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03
GAL-C	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03
JAC-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
JAC-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
LOS-D	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
LOS-C	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
MOT-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
NEW-D	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03
NEW-C	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03
NOR-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
NOR-C	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
PHI-D	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04
PHI-C	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04
POR-D	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
POR-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
SAV-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
SAV-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
SEA-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
TAC-D	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
TAC-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
WIL-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
WIL-C	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
CHN-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03

Individual Center-line Cancer Risk, 0-1.6 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
CHA-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
CNC-D	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
CNC-C	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
GAL-D	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03
GAL-C	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03
JAC-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
JAC-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
LOS-D	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
LOS-C	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
MOT-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
NEW-D	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03
NEW-C	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03
NOR-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
NOR-C	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
PHI-D	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04
PHI-C	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04
POR-D	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
POR-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
SAV-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
SAV-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
SEA-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
TAC-D	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
TAC-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
WIL-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
WIL-C	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
CHN-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

population within 80.5 km (50 mi) of the Elizabeth channel accident location is about 16 million people and typical plumes are about two compass sectors wide, a typical accident plume might expose about two million people to radiation. Thus, for the largest mean result obtained, an average 50-year individual dose over the total exposed populations is about 6,900 person-rem/2,000,000 people = 0.0035 rem per person, which is 5,300 times smaller than the average dose (15 rem) people normally receive from natural, medical, and occupational exposures during the same period of time (BEIR, 1990).

Due to variable weather conditions, the calculated accident consequences vary over a range of values of approximately two orders of magnitude. Quantile values are one means used to indicate how much variation exists among the quantified consequences. The 99.9th quantile values presented in Table D-32 represent the accident consequences that are expected no more than 0.1 percent of the time, that is 99.9 percent of the time the accident consequences will be less than the values presented here. The 99.9th quantile values range from 0.00625 rem (at the MOTSU dock, TRIGA fuel, release category 4) to 108,000 rem (at the Elizabeth channel, BR-2 fuel, release category 6). These results are about three orders of magnitude less likely than the mean, but are less than two orders of magnitude higher than the mean results. (In some cases a 99.9th quantile value is listed as "NOT FOUND." In these instances the peak values, discussed in the following paragraph, occur with a probability of greater than 0.001).

Table D-33 shows that the largest value (peak result) calculated for population dose within 80.5 km (50 mi) of the accident location was 145,000 person-rem (1,450 person-Sv) and that this result was obtained for the Elizabeth channel calculation that used the BR-2 inventory, severity category 6 (EA6) release fractions, and New York City weather. Dividing by the two million people exposed by the accident gives an average 50-year individual dose over the exposed population of about 73 mrem, which is still 250 times smaller than a normal annual individual dose from background and medical exposure over the same period of time. In addition, Table D-34 shows that the probability of this result was 0.0000765 conditional on the accident having occurred. Since the probability of this accident occurring is about 6 x 10^{-10} per port call, the chance of having this result is much less than 1 x 10^{-10} per port call.

Table D-31 also shows that mean (expected) 50-year individual centerline doses at a distance of 0.8 km (0.5 mi) from the accident location [the midpoint of the 0-1.6 km or 0-1 mi computational interval] range from a low of 0.000006 rem (0.00000006 Sv) for the Norfolk and MOTSU calculations that used the TRIGA inventory, severity category 4 (EA4) release fractions, and Cape Hatteras weather to a high of 117 mrem (0.00117 Sv) for the Elizabeth calculations that used the BR-2 inventory, severity category 5 (EA5) release fractions, and New York City weather. Thus, the largest expected individual dose is 190 times smaller than a normal background medical and occupational individual dose during the same period (50 years), which suggests that the mean risk to a maximally exposed member of the general population is not of great concern. Note that the channel and dock values for centerline doses are the same for each port. This is because MACCS, in calculating centerline doses, develops the dose for a hypothetical person and so does not take into account population distribution. Therefore, the usually minor difference in position between the dock and channel does not result in different values. Table D-33 shows that the largest value (peak result) calculated for 50-year individual centerline dose for a person located 0.8 km (0.5 mi) from the accident location was 6.6 rem (0.066 Sv) and that this result was obtained for the Philadelphia calculations that used the BR-2 inventory, severity category 6 (EA6) release fractions, and Washington, DC, weather. This dose of 6.6 rem is less than half of the dose received due to background radiation over the same 50-year period. Table D-34 shows that the probability of this result is 0.00051 conditional on the accident having occurred. Thus, the chance per port call of the MEI receiving this 50-year dose is significantly less than 1 x 10^{-10}.

Table D-31 shows that the mean number of cancer deaths predicted to occur during the decades after the accident, among the populations located within 80.5 km (50 mi) of the accident site at the time of the accident, ranges from 0.00000041 for the MOTSU dock calculation that used the TRIGA inventory, severity category 4 (EA4) release fractions, and mean Cape Hatteras weather to 2.9 for the Elizabeth channel calculation that used the BR-2 inventory, severity category 5 (EA5) release fractions, and mean New York City weather. If all three of the cancer deaths predicted to occur as a result of the accident at the Elizabeth site should happen to occur in the same year, then the death rate among the two million people exposed to radiation by this accident would be 3/2,000,000 = 0.0000015 deaths per person year. Since the normal death rate due to all types of cancer is about 150 deaths per 100,000 people per year (World Almanac, 1992) or 0.0015 deaths per person year, the largest mean (expected) death rate for any base case calculation is 1,000 times smaller than the normal death rate due to cancer. Table D-33 shows that the largest number of cancer deaths obtained for any weather trial in any base case calculation was 60 and that this result was obtained for the Elizabeth channel calculation that used the BR-2 inventory, severity category 6 (EA6) release fractions, and New York City weather. Again, if all of these deaths were to occur in the same year in the future (a very improbable outcome), the death rate during that year among the population exposed to radiation by the accident would be 0.00003 or 50 times lower than the normal death rate due to cancer among this population. Table D-34 shows that the probability of this result is 0.000077 conditional on the occurrence of the accident or less than 1×10^{-10} per port call. Thus, even the worst case number of cancer deaths would be wholly undetectable in the exposed population by the best of epidemiological studies.

Figures D-56 and D-57 present Complementary Cumulative Distribution Functions for population dose and cancer fatalities among the population located within 80.5 km (50 mi) of the accident site for seven of the thirteen ports studied. Only seven were plotted to simplify the figure; these seven provide the full range of results. The figures display the range and probability (conditional on the occurrence of the accident) of these two consequence measures. Figure D-56 shows that any large accident (severity category 5 with the BR-2 inventory is a severe ship collision and fire accident) will lead to a population dose of 10 person-rem, that the values of the 99th quantile (probability of 0.01) range from about 2,000 person-rem to about 40,000 person-rem, and that the largest (peak) result calculated ranges from about 4,600 rem (MOTSU) to about 110,000 rem (Elizabeth). Figure D-57 shows that a large accident has about one chance in 10 (range of 0.002 to 0.6) of causing at least one cancer death among the exposed population in future years, that the values of the 99.9th quantile range from 1 cancer fatality to about 25 cancer deaths, and that the largest (peak) result calculated ranges from 2.1 to 47 deaths due to cancer during the years after the accident.

Figure D-58 presents an example of Complementary Cumulative Distribution Functions for population dose and cancer fatalities for the distance range 0 to 80.5 km (0 to 50 mi) for both the dock and channel locations at Charleston. This figure shows that the dock and channel Complementary Cumulative Distribution Functions for both population dose and cancer fatalities are quite similar, which is typical for all of the ports examined. This suggests that moving the coordinates of the origin of a population distribution a small distance (a few kilometers) has little effect on population dose or cancer fatalities among population located within 80.5 km (50 mi) of the accident location for severe accidents (Table D-28 lists the coordinates of the origins of the polar coordinate population distributions used in these calculations).

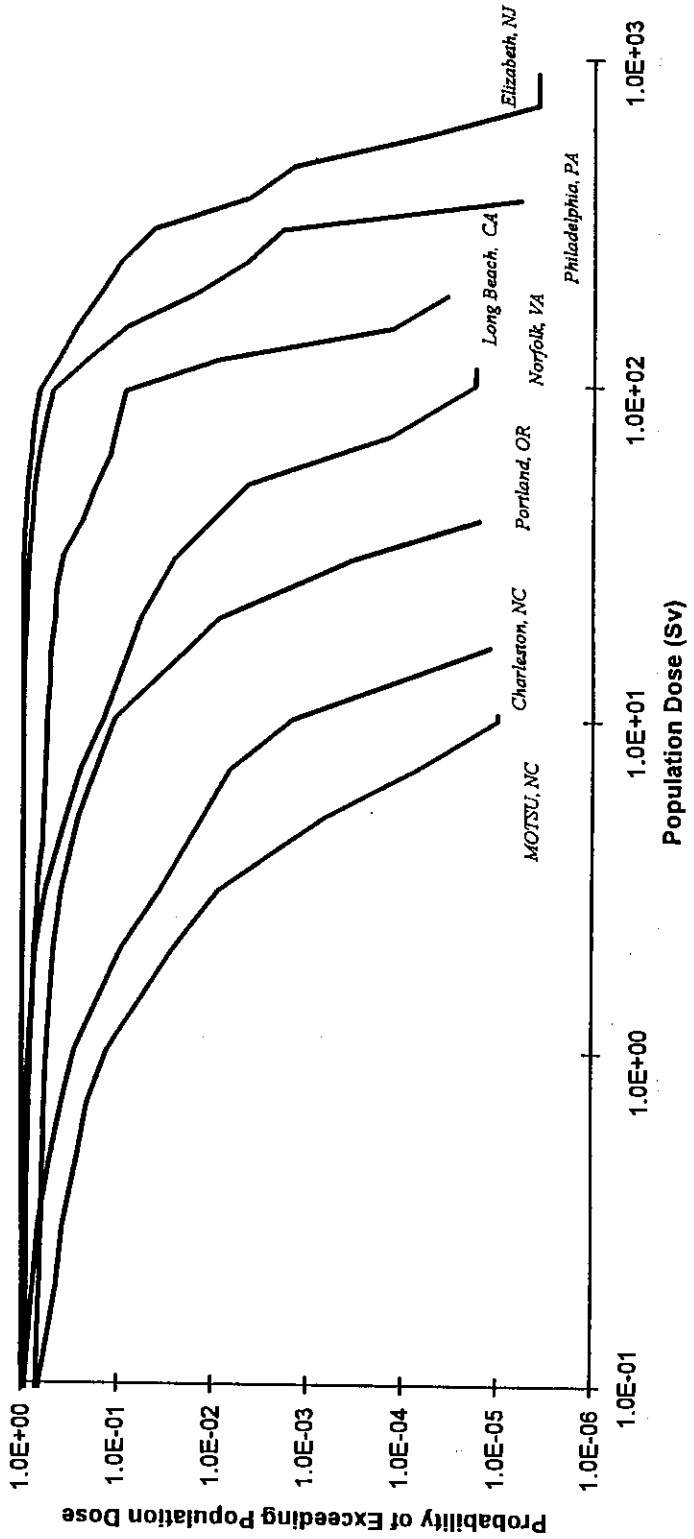

Figure D-56 Effective Dose Equivalent Whole Body Population Dose, 0-80 km (0-50 mi), Select Ports (at the Dock), Variable Meteorology, BR-2 Inventory, Severity Category 5 Releases

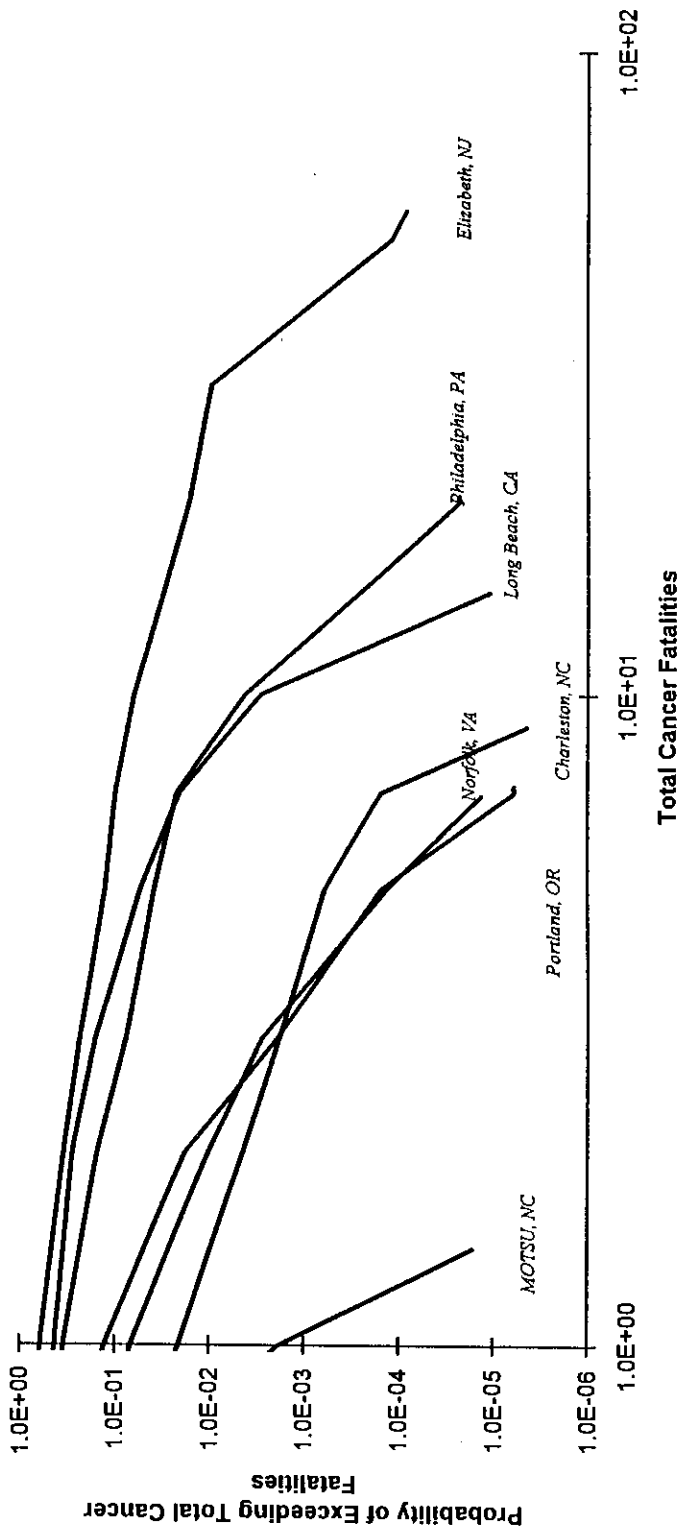

Figure D-57 Total Latent Cancer Fatalities, 0-80 km (0-50 mi), Select Ports (in the Channel), Variable Meteorology, BR-2 Inventory, Severity Category 5 Release

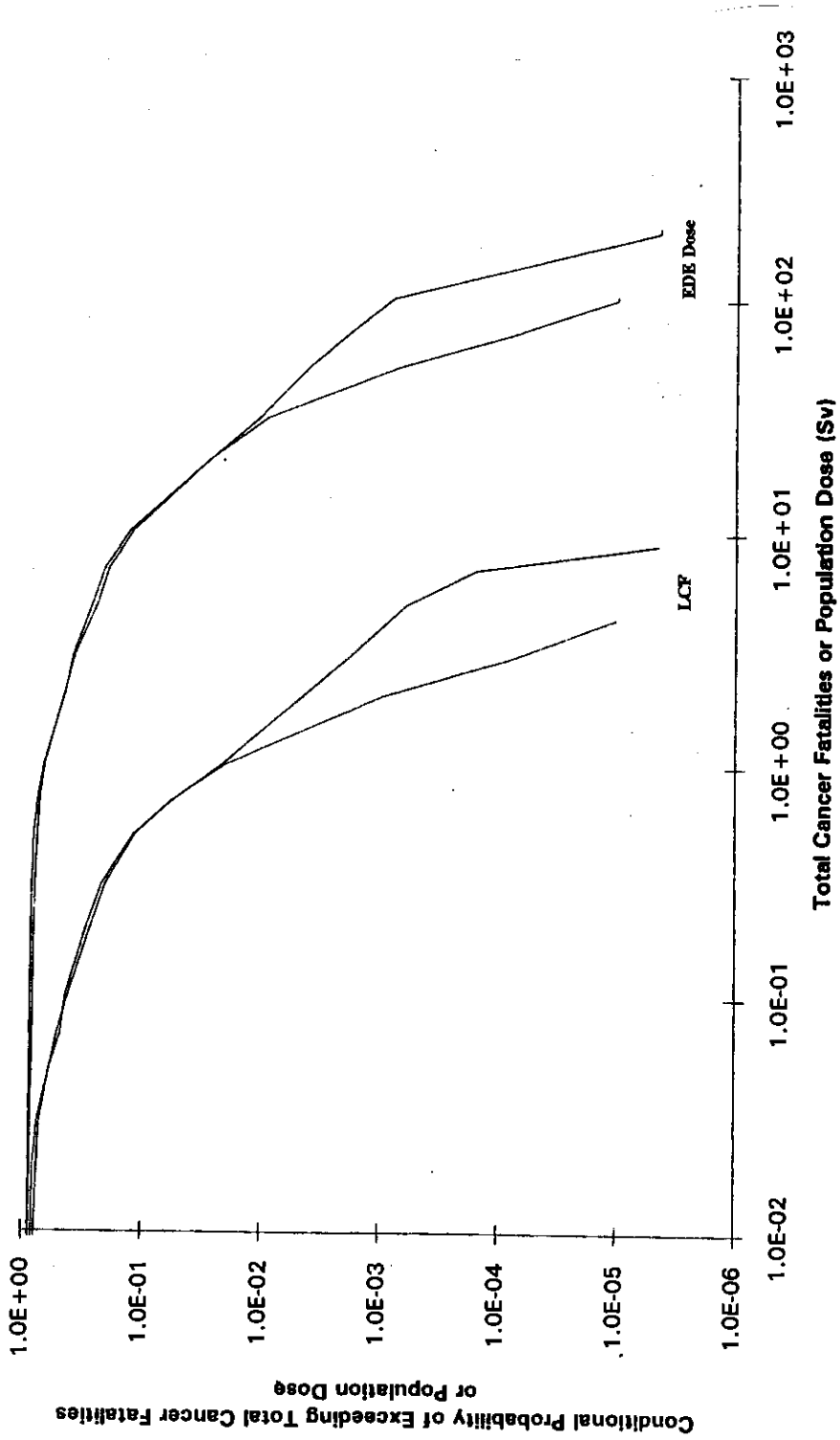

Figure D-58 Effective Equivalent Dose Whole Body Population Dose (Person-Sv) and Total Latent Cancer Fatalities, 0-80 km (0-50 mi), Charleston Dock and Channel Locations, Variable Meteorology, BR-2 Inventory, Severity Category 5 Release

D.5.4.3 Sensitivity Calculations

Two principal sensitivity calculations were performed to determine the sensitivity of the results to key parameters. First, the effect of using local less detailed meteorological data versus meteorological data recorded at a National Weather Service station located some distance from the port was evaluated. Second, the results of exceptionally high spent nuclear fuel temperatures were examined. Additionally, the sensitivity of changes in plume buoyancy, the size of the nuclide set, modal study release fractions, corrosion products release, and work force population were examined. The meteorological sensitivity calculations compared results obtained using variable meteorology recorded at a National Weather Service station away from the port to results obtained using constant meteorology recorded at the port. All other sensitivity calculations except the work force calculations were performed by modifying the Elizabeth base case channel calculation as was appropriate in order to examine the parameter of interest. The work force calculations were based on the Elizabeth dock site. All of the sensitivity calculations used the BR-2 inventory and all, except those that examined release fractions, used severity category 5 release fractions.

D.5.4.3.1 Variable vs. Constant Meteorology

Variable meteorology, which takes into account hourly changes of wind direction and speed, was used in the calculations that led to the results presented in this EIS. However, the detailed weather data required to support these calculations are not available in most ports, so detailed data from the most appropriate National Weather Service Station location possible were used. A sensitivity study was performed to better understand the effect of using detailed but not local weather data versus using local less-detailed port weather data. The local weather is called constant meteorology, to reflect the fact that the weather remains constant during the course of the accident, not varying on an hourly basis.

This study performed, for each port, a large number of constant meteorology calculations for each port, using the conditions and probabilities specified in the joint frequency distributions that were available for each port. Since joint frequency distributions specify for each compass sector the probability of occurrence of each of the six Pasquil-Gifford atmospheric stability classes with each of six windspeed ranges, 16 x 6 x 6 = 576 constant meteorology calculations could be performed, once assuming that it was raining and once assuming that it was not. Then, by cumulating the results of each set of approximately 1,150 constant meteorology single weather trial calculations (rain does not occur for all of the sets of conditions in the joint frequency distribution), a Complementary Cumulative Distribution Function could be constructed to compare with the complementary cumulative distribution function obtained using variable meteorology recorded at the nearest National Weather Service Station.

Table D-35 presents a sample joint frequency table for one of the ports examined during this EIS (Charleston). Tables D-36 and D-37 present the port wind rose and a probability of rain by stability class respectively for selected ports.

Constant meteorology calculations were performed as follows. For each port examined, two sets of constant meteorology calculations were performed. Both used the joint frequency distribution of windspeed (6 windspeed ranges) and stability class (6 stability classes) by wind direction (16 compass sectors) for the port being analyzed as the meteorological input data for MACCS. Each calculation was run two times, once for no rain and once assuming that it was raining throughout the entire simulation. Therefore, each MACCS constant meteorology calculation consisted of 6 x 6 x 16 x 2 = 1,152 constant meteorology trials. From these 1,152 trials, a complementary cumulative Distribution Function and a mean (expected result) was constructed for each consequence measure calculated.

Table D-35 1988-92 Summary Joint Frequency Table for Charleston, SC Port

A Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0003	.0002	.0002	.0002	.0002	.0002	.0002	.0002	.0003	.0003	.0002	.0003	.0001	.0002	.0001	.0001
4- 7	.0005	.0003	.0005	.0004	.0004	.0006	.0003	.0003	.0006	.0004	.0004	.0002	.0002	.0001	.0002	.0003
8-12	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
13-18	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
19-24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

B Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0008	.0003	.0006	.0007	.0005	.0007	.0008	.0005	.0011	.0006	.0010	.0007	.0007	.0006	.0004	.0005
4- 7	.0018	.0013	.0017	.0019	.0023	.0020	.0017	.0013	.0031	.0016	.0023	.0017	.0018	.0011	.0013	.0007
8-12	.0021	.0013	.0021	.0025	.0025	.0014	.0013	.0008	.0016	.0013	.0013	.0010	.0013	.0012	.0016	.0012
13-18	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
19-24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

C Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0002	.0002	.0003	.0003	.0003	.0003	.0003	.0003	.0004	.0003	.0005	.0005	.0003	.0002	.0001	.0001
4- 7	.0015	.0014	.0016	.0022	.0026	.0021	.0019	.0020	.0026	.0021	.0035	.0021	.0014	.0010	.0014	.0012
8-12	.0081	.0038	.0061	.0072	.0090	.0049	.0037	.0031	.0057	.0051	.0062	.0042	.0034	.0043	.0042	.0049
13-18	.0017	.0012	.0021	.0020	.0021	.0014	.0006	.0005	.0006	.0005	.0005	.0004	.0008	.0005	.0008	.0007
19-24	.0000	.0000	.0000	.0001	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

D Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0009	.0005	.0005	.0007	.0006	.0005	.0004	.0008	.0013	.0012	.0013	.0012	.0007	.0007	.0004	.0004
4- 7	.0045	.0030	.0037	.0027	.0028	.0026	.0022	.0044	.0094	.0079	.0093	.0056	.0042	.0039	.0026	.0022
8-12	.0196	.0151	.0165	.0112	.0108	.0070	.0047	.0061	.0168	.0216	.0201	.0124	.0108	.0075	.0066	.0083
13-18	.0140	.0179	.0145	.0082	.0133	.0096	.0058	.0058	.0084	.0080	.0057	.0047	.0051	.0035	.0034	.0036
19-24	.0009	.0019	.0020	.0007	.0020	.0023	.0010	.0004	.0003	.0000	.0000	.0001	.0001	.0001	.0001	.0000
>24	.0002	.0005	.0002	.0001	.0003	.0003	.0001	.0000	.0000	.0001	.0000	.0000	.0000	.0000	.0000	.0000

E Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
4- 7	.0127	.0071	.0088	.0047	.0033	.0023	.0021	.0025	.0050	.0061	.0094	.0066	.0052	.0049	.0036	.0051
8-12	.0063	.0072	.0085	.0059	.0067	.0058	.0032	.0030	.0038	.0066	.0043	.0021	.0019	.0013	.0011	.0016
13-18	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
19-24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

F Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0090	.0076	.0081	.0059	.0052	.0039	.0035	.0057	.0086	.0076	.0101	.0054	.0054	.0036	.0032	.0035
4- 7	.0122	.0088	.0112	.0074	.0063	.0051	.0044	.0059	.0095	.0104	.0122	.0057	.0044	.0044	.0029	.0039
8-12	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
13-18	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
19-24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

Table D-36 Wind Rose Table for Select Ports

1988-92 Summary Wind Rose Table For Charleston, SC Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0974	.0796	.0892	.0650	.0712	.0530	.0380	.0436	.0790	.0817	.0882	.0549	.0479	.0389	.0341	.0385

1988-92 Summary Wind Rose Table For Long Beach, CA Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0246	.0171	.0602	.3093	.1804	.0157	.0177	.0229	.0331	.0227	.0271	.0475	.1115	.0601	.0348	.0154

1988-92 Summary Wind Rose Table For Newark, NJ Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0784	.0725	.1015	.0871	.0854	.0639	.0788	.0559	.0832	.0786	.0442	.0273	.0231	.0304	.0452	.0447

1988-92 Summary Wind Rose Table For Norfolk, VA Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.1078	.0963	.1021	.0647	.0562	.0456	.0344	.0285	.0940	.0665	.0860	.0573	.0470	.0321	.0358	.0458

1988-92 Summary Wind Rose Table For Philadelphia, PA Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0682	.0440	.0950	.1118	.1281	.0913	.0715	.0568	.0669	.0266	.0275	.0639	.0545	.0284	.0278	.0378

1988-92 Summary Wind Rose Table For Portland, OR Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0936	.0551	.0337	.0315	.0756	.0956	.1163	.1054	.0704	.0187	.0171	.0225	.0638	.1126	.0576	.0304

1988-92 Summary Wind Rose Table For Wilmington, NC Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0744	.0804	.0994	.0798	.0747	.0378	.0417	.0435	.0955	.0780	.0699	.0488	.0549	.0351	.0411	.0451

Table D-37 Rainfall Data, Select Ports

Rainfall Data for the Charleston, SC Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.05000	0.00264
B	0.22400	0.00371
C	0.16322	0.00771
D	0.13860	0.11099
E	0.14740	0.01099
F	0.07941	0.00125

Rainfall Data for the Long Beach, CA Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.00000	0.00000
C	0.14375	0.00233
D	0.07837	0.03648
E	0.06596	0.00809
F	0.07083	0.00115

Rainfall Data for the Newark, NJ Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.05000	0.00059
C	0.08571	0.00648
D	0.08577	0.12139
E	0.08968	0.00971
F	0.05000	0.00153

Rainfall Data for the Norfolk, VA Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.09167	0.00371
C	0.10921	0.00771
D	0.47136	0.11099
E	0.12574	0.01099
F	0.05000	0.00125

Rainfall Data for the Philadelphia, PA Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.17500	0.00089
C	0.11250	0.00431
D	0.07520	0.12101
E	0.10682	0.00649
F	0.17500	0.00035

Rainfall Data for the Portland, OR Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.11250	0.00139
C	0.08125	0.01245
D	0.06172	0.15220
E	0.06493	0.01428
F	0.05000	0.00087

Rainfall Data for the Wilmington, NC Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
8	0.18235	0.00718
C	0.17500	0.01937
D	0.15048	0.12490
E	0.16295	0.02310
F	0.08571	0.00244

Table D-38 Comparison of Population Dose and Selected Ports Using Variable vs. Constant Meteorology for Category Accident of a BR-2 Fuel Cask

Site/Loc	EDE Whole Body Population Dose, 0-80 KM (Sv)				Total Cancer Fatalities, 0-80 KM			
	Mean		99.9th Quantile		Mean		99.9th Quantile	
	Var	Const	Var	Const	Var	Const	Var	Const
CHA-D	4.15E+00	3.06E+00	4.63E+01	2.13E+01	1.89E-01	1.29E-01	1.98E+00	8.43E-01
CHA-C	4.18E+00	3.41E+00	9.03E+01	3.71E+01	1.90E-01	1.43E-01	3.96E+00	2.01E+00
LOS-D	4.71E+01	3.44E+01	2.67E+02	1.19E+02	1.99E+00	1.44E+00	1.03E+01	5.35E+00
LOS-C	4.26E+01	3.31E+01	2.19E+02	8.16E+01	1.80E+00	1.38E+00	9.81E+00	3.43E+00
NEW-D	6.55E+01	5.47E+01	5.87E+02	2.32E+02	2.75E+00	2.28E+00	2.46E+01	9.54E+00
NEW-C	6.93E+01	5.89E+01	9.41E+02	NOT-FOUND	2.90E+00	2.46E+00	3.89E+01	NOT-FOUND
NOR-D	8.54E+00	8.88E+00	1.03E+02	7.26E+01	3.77E-01	3.72E-01	4.23E+00	3.07E+00
NOR-C	6.65E+00	6.76E+00	9.02E+01	3.51E+01	2.96E-01	2.83E-01	3.59E+00	1.34E+00
PHI-D	2.81E+01	2.53E+01	3.10E+02	NOT-FOUND	1.20E+00	1.06E+00	1.18E+01	1.34E+00
PHI-C	2.74E+01	2.01E+01	2.86E+02	5.91E+01	1.17E+00	8.40E-01	1.22E+01	2.45E+00
POR-D	1.17E+01	1.08E+01	1.09E+02	7.76E+01	5.18E-01	4.54E-01	4.90E+00	3.29E+00
POR-C	1.12E+01	8.76E+00	1.01E+02	3.88E+01	4.97E-01	3.68E-01	3.78E+00	1.51E+00
MOT-D	2.08E+00	1.02E+00	2.46E+01	NOT-FOUND	9.94E-02	4.37E-02	1.22E+00	1.51E+00
WIL-C	2.07E+00	1.05E+00	2.25E+01	5.47E+00	9.76E-02	4.49E-02	1.04E+00	2.22E-01

Table D-38 compares for seven ports the expected (mean) and 99.9th quantile values of population dose and cancer fatalities for the distance range 0 to 80.5 km (0-50 mi) obtained using variable meteorology to the values obtained using constant meteorology. Inspection of the table shows that the mean values for constant meteorology are quite similar to mean values for variable meteorology. For example, for population dose, the ratio of the variable meteorology result to the constant meteorology result has an average value and standard deviation of 1.31±0.31 for population dose and 1.34±0.41 for cancer fatalities. The MOTSU dock calculation yielded the largest values for these ratios, 2.04 for mean population dose and 2.27 for cancer fatalities. Thus, the use of meteorological data recorded at a nearby National Weather Service station yields expected (mean) values for population dose and cancer fatalities that are on average about 30 to 40 percent larger than the values obtained using constant meteorological conditions for each of the six Pasquil-Gifford atmospheric stability classes that were derived from data recorded at the harbor.

The 99.9th quantile values of population dose and cancer fatalities among the population that resides within 80.5 km (50 mi) of the harbor results agree less well for constant and variable meteorology. For several ports, the 99.9th quantile population dose is missing, ("not found"), for the constant meteorology calculation. This means that the probability of the largest result obtained for any of the 1,152 trials run during each constant meteorology calculation was larger than 0.001 for that particular calculation. For the locations that yielded a 99.9th quantile value for both the variable and constant meteorology calculation, the ratio of the 99.9th quantile variable meteorology result to the 99.9th quantile constant meteorology result has a value of 2.64±0.98 for population dose and 3.00±2.02 for cancer fatalities. The fact that the 99.9th quantile values obtained using variable meteorology are on average 2.5 to 3.0 times larger than the 99.9th quantile values obtained using constant meteorology suggests that the importance sampling scheme, used by MACCS to select weather sequences from a year of variable meteorological data, is able to find weather sequences that lead to adverse results that are not represented in the sets of constant meteorological conditions found in the joint frequency distributions of windspeed and atmospheric stability by wind direction that were recorded at the harbors. This is so because the occurrence of rain is usually the cause of peak results at some later time when the plume is passing over some downwind highly populated region. Thus, because rain at some downwind location was not modeled by the constant meteorology calculations, these results should differ significantly from those obtained using variable meteorology, especially for the higher quantiles of result distributions.

Figure D-59 presents, as an example, complementary cumulative distribution functions for Long Beach of the 50-year population dose and lifetime LCFs over the distance range from 0 to 80.5 km (0 to 50 mi) obtained using both variable and constant meteorology. All four calculations used the BR-2 inventory and severity category 5 release fractions. The dose calculation was performed for the dock location at Long Beach; and the LCF calculation was performed for the channel location. Inspection of the figures shows that the constant and variable meteorology complementary cumulative distribution functions are quite similar until the 90th quantile of the distributions are reached, and diverge increasingly as higher quantiles are passed, with the constant meteorology complementary cumulative distribution function generally falling off faster than the variable meteorology complementary cumulative distribution function (smaller consequence value at any consequence probability). Thus, the figures confirm the conclusion reached by inspection of Table D-38, that variable and constant meteorology yield quite similar estimates for mean results and that adverse meteorological conditions are more likely to be modeled if weather sequences are selected by importance sampling from a year of variable data than if constant meteorological conditions are used.

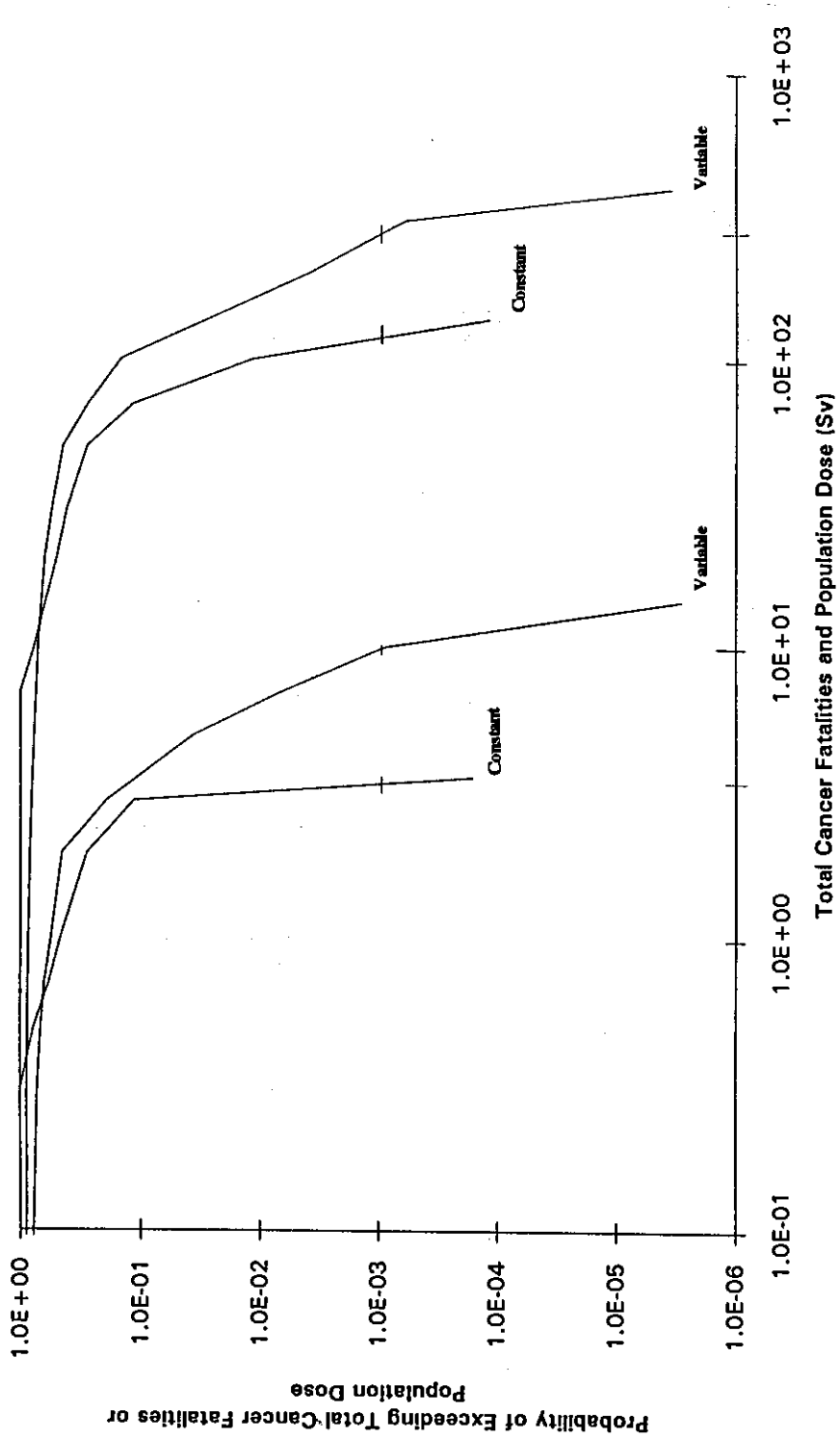

Figure D-59 Effective Dose Equivalent Whole Body Population Dose (Sv) (Dock) and Total Cancer Fatalities (Channel), 0-80 km (0-50 mi), Long Beach, Variable and Constant Meteorology, BR-2 Inventory, Severity Category 5 Releases

D.5.4.3.2 High-Temperature Sensitivity Calculations

As previously discussed, releases of radioactive material from spent nuclear fuel transportation casks are categorized by severity. Severity category 6, which results in the largest release, is assumed for the marine transportation portion of this EIS to be caused by a severe ship collision that results in damage to the transportation cask and a severe fire that engulfs the cask. Only around one in five severe ship fires reach temperatures above approximately 700°K or 800°F (see Attachment D5, Section 4). As discussed below, extremely high temperatures, above 900°K (1,160°F), result in phenomena that could significantly alter the release fraction for aluminum-based and TRIGA fuel (previous studies have not specifically addressed the impact of these phenomena). Therefore, the release fractions assumed for severity category 6 (Table D-21) are for temperatures of the spent nuclear fuel above 700°K (800°F) but below 900°K (1,160°F).

Section D.5.3.1 of this appendix developed probabilities of the more severe marine accidents. Table D-24 stated that the probability of a severity category 6 accident is 6×10^{-10}, or less than one chance in a billion per cask shipment. This very low probability is made even lower if the probability of the severe fire causing the spent nuclear fuel temperature to exceed 900°K (1,160°F) is considered. Appendix D Attachment D5 concludes that the probability of a severe ship fire exceeding spent nuclear fuel temperatures of 900°K (1,160°F) is 0.1. Multiplying the probability of a severity category 6 accident (6×10^{-10}) by the probability of a severe fire on the ship (0.1) results in the probability of a severity category 6 accident that includes a severe ship fire, 6×10^{-11}. This exceedingly small probability indicates that the occurrence of this condition is not a creditable accident. However, for completeness, an evaluation of the consequences of such an accident is presented below as a sensitivity calculation.

The review of the behavior of aluminum-uranium (Al-U) alloy and TRIGA fuels at temperatures above 900°K (1,160°F), presented in Attachment D5, found that at these temperatures Al-U fuels melt, and if exposed to air, TRIGA fuel burns. Table D5-2 (in Attachment D5) compares the release fractions estimated for these high-temperature scenarios to those used in the base case calculations. These data show the high-temperature events (the category 5B and 6B events) increase release from these fuels significantly.

Since both processes (melting and burning) are expected to produce fission product release fractions that are significantly larger than those used during base case calculations for severity category 6 accidents, sensitivity calculations were performed so that the consequences and risks associated with these larger releases could be compared to the consequences and risks of the base case results. Again the Elizabeth channel location was used to perform the sensitivity calculations. Three calculations were performed, two BR-2 aluminum-uranium alloy fuel calculations and one TRIGA fuel calculation. All of these calculations used the release fractions specified in Table D-39 for high-temperature scenarios. The first aluminum-uranium alloy fuel sensitivity calculation used severity category 5B and the second category 6B release fractions. The single TRIGA sensitivity calculation used category 6B release fractions. Calculations were not performed with any of the other sets of release fractions presented in Table D-39, because each of the other sets is smaller than the set used in the base case calculations that it would replace; and would thus yield smaller consequences and risks.

Table D-39 presents the results of these high-temperature sensitivity calculations and compares them to the base case results obtained using the same inventories but using the severity category 5 or 6 release fractions given in Table D-21. Table D-39 shows that, as expected, the larger severity category 5B and severity category 6B release fractions lead to consequences significantly larger than those obtained for the base case calculations that used severity category 5 and severity category 6 release fractions. Inspection of the table shows that the larger release fractions increase consequence estimates by factors of ten to 100.

Table D-39 High-Temperature Sensitivity Calculation Results

	BR-2				TRIGA	
Accident Severity Category	5	5B	6	6B	6	6B
Accident Probability	5×10^{-9}	5×10^{-10}	6×10^{-10}	6×10^{-11}	6×10^{-10}	6×10^{-11}
Peak Probability [0-1.6 km [0-1 mi)]	8.41×10^{-5}	7.65×10^{-5}	8.41×10^{-5}	7.03×10^{-5}	8.41×10^{-5}	2.17×10^{-4}
Peak Probability [0.80.5 km (0-50 mi)]	8.41×10^{-5}	1.16×10^{-5}	8.41×10^{-5}	1.45×10^{-5}	8.41×10^{-5}	1.45×10^{-5}
EDE Whole Body Population Dose (person-rem)						
0-1.6 km (0-1 mi)						
Mean	236	1,490	192	3,810	26.8	3,980
Peak	42,100	203,000	45,900	271,000	6,390	297,000
0-80.5 km (0-50 mi)						
Mean	6,930	68,400	6,770	639,000	937	298,000
Peak	133,000	1,450,000	145,000	14,400,000	20,200	6,390,000
Total Cancer Fatalities						
0-1.6 km (0-1 mi)						
Mean	0.098	0.622	0.0802	1.59	0.0112	1.66
Peak	17.5	84.5	19.1	113	2.66	123
0-80.5 km (0-50 mi)						
Mean	2.90	28.7	2.84	268	0.392	125
Peak	55.3	603	60.4	6,000	8.39	2,660
Impact Distances (km)						
Decontamination						
Mean	0.0	0.0156	0.0	0.302	0.0	0.0993
Peak	0.0	1.61	0.0	8.05	0.0	6.44
Cond. Peak Prob.	---	0.00969	---	0.00116	---	7.53×10^{-5}
Interdiction						
Mean	0.0	0.0156	0.0	0.302	0.0	0.0993
Peak	0.0	1.61	0.0	8.05	0.0	6.44
Cond. Peak Prob.	---	0.00969	---	0.00116	---	7.53×10^{-5}
Condemnation						
Mean	0.0	0.0	0.0	0.0292	0.0	0.00263
Peak	0.0	0.0	0.0	3.22	0.0	1.61
Cond. Peak Prob.	---	---	---	0.000648	---	0.00163
Population Dose Risk						
0-1.6 km (0-1 mi)						
Mean	1.2×10^{-6}	7.5×10^{-7}	1.2×10^{-7}	2.3×10^{-7}	1.6×10^{-8}	2.4×10^{-7}
Peak	1.8×10^{-8}	7.8×10^{-9}	2.3×10^{-9}	1.1×10^{-9}	3.2×10^{-10}	3.9×10^{-9}
0-80.5 km (0-50 mi)						
Mean	3.5×10^{-5}	3.4×10^{-6}	4.1×10^{-6}	3.8×10^{-5}	5.6×10^{-7}	1.8×10^{-5}
Peak	5.6×10^{-8}	8.4×10^{-9}	7.3×10^{-9}	1.3×10^{-8}	1.0×10^{-9}	5.5×10^{-9}
Cancer Fatality Risk						
0-1.6 km						
Mean	4.9×10^{-10}	4.4×10^{-10}	4.8×10^{-11}	9.5×10^{-11}	6.7×10^{-12}	1.0×10^{-10}
Peak	7.4×10^{-12}	3.2×10^{-12}	9.6×10^{-13}	4.8×10^{-13}	1.3×10^{-13}	1.6×10^{-12}
0-80.5 km						
Mean	1.5×10^{-7}	1.6×10^{-7}	1.7×10^{-9}	1.6×10^{-8}	2.4×10^{-10}	7.5×10^{-9}
Peak	2.3×10^{-11}	3.5×10^{-12}	3.0×10^{-12}	5.2×10^{-12}	4.2×10^{-13}	2.3×10^{-12}

However, because the probabilities of occurrence of these high-temperature release fractions (see Attachment D-5) for BR-2 aluminum uranium alloy fuel inventories are generally ten times smaller than those associated with the severity category 5 and severity category 6 accident categories, the risks associated with these larger releases are comparable to or smaller than those predicted for base case BR-2 calculations. For TRIGA fuel, severity category 6B release fractions are much larger than the severity category 6 release fractions. The probability of the severity category 6B release fractions is only ten times smaller than that of the severity category 6 release fractions. Therefore, the risks associated with a TRIGA fuel category 6B release are significantly larger than those obtained for the base case accident severity category 6 calculation. But, because the TRIGA inventory is substantially smaller than the BR-2 inventory, the TRIGA severity category 6B risks are still smaller than the risks obtained for base case calculations using the BR-2 inventory and the severity category 5 set of release fractions.

Other environmental impacts in addition to the public health consequences are presented in Table D-39. These impacts were determined as part of the MACCS calculations. MACCS calculated land impacts based on a habitability dose criterion and cost effectiveness of mitigative actions such as evacuation, temporary relocation, and land decontamination and interdiction. The habitability criterion is based on the need to take action to ensure that the dose to a person remains below 4 rem[1] over a 5-year period. MACCS code determines the mitigative actions in a predetermined sequence in order to select the least stringent action which will allow the habitability dose criterion to be satisfied. The order of actions is: 1) decontamination alone (minimum decontamination process, three levels of decontamination process can be specified), 2) maximum level of decontamination followed by an interdiction period, and 3) permanent interdiction (condemnation) of the land. The decontamination distance is that distance from the accident location that requires post-accident clean-up to ensure this dose level is not achieved. The land is usable, that is, people may live and work in the area, within a relatively short period after the accident. The interdiction distance is that distance from the accident that even after decontamination would require some time, typically seven years, before the land area would be useable. The condemnation distance characterizes the land area that even after decontamination would remain unusable for at least 30 years.

MACCS code calculates both the affected population in the urban areas and the affected farmlands in the rural areas. The affected distances, (i.e., decontamination, interdiction, and condemnation distances), in the rural areas are generally larger than those of the urban area. Since one of the principal uses of rural land is agricultural, the consumption of contaminated food produced in these areas would result in larger doses to some members of the public.

Table D-39 provides the land impact distances for an accident that occurs in the Port of Elizabeth for the most severe accident severity categories of both the base case calculations (category 5 and 6 for the BR-2 fuel and category 6 for the TRIGA fuel) and for the most severe of the high temperature accident scenarios (categories 5B and 6B for BR-2 fuel and category 6B for the TRIGA fuel). Since the ports are located primarily in urban areas, the impact distances presented are those based on the urban (population) impact calculations. For the base case accident scenarios, MACCS predicted no impact on the usability of the land. However, when temperatures reaching the melting point of the aluminum based fuel and the combustion temperature of the TRIGA fuel are realized, some land-use impacts are calculated. All mean impact distances are well under 1 km (0.6 mi), with the largest distance being approximately 300 m

1 *This arises from 2 rem in first year and 0.5 rem per year for the years 2 to 5. This criterion is consistent with the Environmental Protection Agency's long-term objectives of the Protective Action Guide, (Section 4.2.1 of "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA 1991).*

(1000 ft). The peak values quoted in Table D-39 represent the worst possible consequences, driven by meteorological conditions that create the maximum potential damage. The occurrences of these meteorological conditions are of low probabilities which are ranging from approximately one-in-one hundred to less than one-in-ten thousand.

In addition to the Port of Elizabeth, the land impact analysis was performed for several of the candidate ports, including Concord NWS, CA; Galveston, TX; MOTSU, NC; and Tacoma, WA. For these four ports, the mean values for the land impacts resulting from the category 6B accidents, the most severe of all accident categories, were of the same order of magnitude as, and slightly smaller than, the results presented in Table D-39 for the Port of Elizabeth.

D.5.4.3.3 Other Sensitivity Calculations

In addition to the two sensitivity calculations discussed above, sensitivity calculations were also performed that examined the effect on consequences of (1) plume buoyancy, (2) the size of the set of nuclides used to specify inventories, (3) Modal Study release fractions, (4) corrosion deposits release, and (5) work force population. Table D-40 summarizes the calculations performed. For all of these calculations, the reference calculation was the base case Elizabeth dock or channel calculation that used the BR-2 inventory, severity category 5 release fractions, and variable meteorology recorded at the New York City National Weather Service station. Work force sensitivity calculations used the Elizabeth dock population distribution. All of the other sensitivity calculations used the Elizabeth channel population distribution. Table D-41 presents mean and peak population doses and cancer fatalities for two distance ranges, 0-1.6 km and 0-80.5 km, (0-1 and 0-50 mi) for all of the "other" sensitivity calculations, and also for the reference Elizabeth base case calculations to which sensitivity calculation results should be compared.

D.5.4.3.3.1 Plume Buoyancy

As Table D-21 showed, a severity category 5 release scenario results from a collision and a severe fire. Thus, the first sensitivity calculation performed examined the effect of plume buoyancy (i.e, of plume rise) on accident consequences. This was done by repeating the Elizabeth channel reference calculation setting the sensible heat content of the release to zero. This change produces a cold plume that is not subject to plume rise and thus is not lofted over the population located close to the release point (the accident location). The results of this sensitivity calculation are presented in Table D-41.

Table D-41 shows that changing the reference Elizabeth channel calculation to a cold release not subject to plume rise causes mean and peak population doses and cancer fatalities to increase somewhat for the 0-80.5 km (0-50 mi) distance range and substantially for the 0-1.6 km (0-1 mi) distance range. For the 0-80.5 km (0-50 mi) distance range, mean population dose and cancer fatalities both increase by a factor of 2.4, and peak population dose and cancer fatalities increase by a factor of 1.1. For the 0-1.6 km (0-1 mi) distance range, mean population dose and cancer fatalities both increase by a factor of 17, and peak population dose and cancer fatalities both increase by a factor of 2.7. Thus, if engulfing fires increase release magnitudes, consequence magnitudes will not increase proportionately because the fire will produce a hot plume that will be lofted over nearby populations decreasing radiation exposures and thus health effects among those populations. It should be mentioned that the releases assumed here (category 5) are not considered possible *without* the fire. This calculation was done to show the sensitivity of the results to the presence of a fire.

Table D-40 Other Sensitivity Calculations

Run No.	Meteorology[a]		Nuclides[b]		Release Fractions[c]			Heat[d]		Shielding[e]	
	Variable	Constant	MACCS	EIS	5	MS/nM5	MS/M5	H	C	N	C
BC	x			x	x			x		x	
Buoyancy Calculations											
1a.	x			x	x				x	x	
Nuclide Sensitivity Calculations											
2a.	x		x		x			x			x
2b.	x		x		x				x		x
Modal Study Release Fraction Calculations											
3a.	x			x		x		x		x	
3b.	x			x			x	x		x	
Corrosion Products Calculations											
4a.[f]	x			x	EA3				x	x	
4b.[g]	x			x	x			x		x	
Work Force Calculations											
5a.	x			x	x			x		x	
5b.	x			x	x				x	x	
5c.	x			x	x				x		x
5d.[h]	x			x	x				x	x	
5e.[h]	x			x	x				x		x
5f.[i]	x			x	x				x	x	

[a]*Meteorology: Variable = hourly National Weather Service data, Constant = Joint Frequency Data.*

[b]*Nuclides: MACCS = 22 MACCS nuclides, EIS = 34 EIS nuclides.*

[c]*Release Fractions: 5 = severity category 5 release fractions; MS/nM5 = release fractions for nonmetallic (TRIGA) spent nuclear fuel for Modal study cask response region roughly corresponding to severity category 5; MS/M5 = release fraction for metallic (aluminum-based) spent nuclear fuel for Modal study cask response regions roughly corresponding to severity category 5.*

[d]*Heat: H = hot plume, C = cold plume.*

[e]*Shielding: N = normal shielding factors; C = sheltering shielding factors from 0-8 km (0-5 mi) for one day and normal shielding factors at all other times and distances.*

[f]*Only Corrosion Products released*

[g]*With Corrosion Products release added to the reference release.*

[h]*With puff and tail*

[i]*With puff and tail, and evacuation from 0-1.6 km (0-1 mi.)*

D.5.4.3.3.2 Size of Nuclide Set

Table D-25 presented the three inventories used in the base case analyses. Each inventory contains 34 radionuclides. The default set of radionuclides used by MACCS does not contain dose conversion factors for 13 of these 34 radionuclides. These 13 radionuclides are hydrogen-3, tin-123, antimony-125, tellurium-125m, promethium-147, promethium-148m, europium-154, europium-155m, uranium-234, uranium-235, uranium-238, americium-242m, and americium-243. *Chronic* health effect dose conversion factors for all 13 of these radionuclides were available (DOE, 1988a; DOE, 1988b) and were added to the MACCS dose conversion factor library for this study. However, because generally accepted *acute* health effect dose conversion factors were not available, all calculations performed for this study were run not including acute health effects for these 13 radionuclides.

Table D-41 Sensitivity Study Results, Elizabeth Dock and Channel, Inventory BR-2, Severity Category 5

Run	EDE Whole Body Population Dose (person-rem)				Total Cancer Fatalities			
	0-1.6 km (0-1 mi)		0-80.5 km (0-50 mi)		0-1.6 km (0-1 mi)		0-80.5 km (0-50 mi)	
	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak
Base Case (Channel)	236	42,100	6,930	133,000	0.099	17.5	2.90	55.3
Buoyancy								
1	4,200	114,000	16,900	151,000	1.75	47.6	7.07	62.9
Nuclide Sensitivity								
2a	236	42,100	6,930	133,000	0.0985	17.5	2.90	55.3
2b	4,200	114,000	16,900	151,000	1.75	47.6	7.07	62.9
Modal Study Release Fraction								
3a	53.7	9,540	1,570	30,100	0.0224	3.98	0.661	12.6
3b	0.3	47.7	7.9	151	0.000112	0.0199	0.00331	0.0628
Corrosion Products Calculations								
4a	739	20,100	2,950	26,600	0.319	8.70	1.27	11.5
4b	278	49,400	8,120	156,000	0.116	20.7	3.42	65.4
Base Case (Dock)	71.3	13,300	6,550	113,000	0.0298	5.56	2.75	47.2
Work Force								
5a	105	14,400	6,600	113,000	0.0438	6.02	2.77	47.2
5b	1,870	40,400	11,200	84,600	0.780	16.8	4.69	35.3
5c	1,860	40,300	11,200	84,500	0.778	16.8	4.68	35.3
5d	1,940	40,400	11,600	72,500	0.810	16.9	4.84	30.2
5e	1,940	40,400	11,500	72,500	0.808	16.8	4.83	30.2
5f	1,940	40,300	11,500	72,500	0.808	16.8	4.83	30.2

The effect of not including acute impacts for 13 of the radionuclides in the inventories was examined by two sensitivity calculations. For these calculations, the reference Elizabeth channel calculation was performed with and without the chronic effects of the 13 radionuclides for two situations, once assuming a hot release, and once assuming a cold release. Table D-41 shows that removing these 13 radionuclides from the BR-2 inventory had no significant impact on either mean or peak values of population dose or cancer fatalities over the distance ranges 0-1.6 km (0-1 mi) and 0-80.5 km (0-50 mi) for either calculation. The cold release results and the hot release peak results are identical to those obtained using all 34 radionuclides in the full BR-2 inventory. The hot release mean values obtained with the 13 radionuclides removed differ by no more than 5 percent from the results obtained using 34 radionuclides. Thus, the 13 radionuclides for which acute dose conversion factors were not available do not contribute significantly to chronic dose or health effects, which suggests that none should have a significant impact on acute health effects.

The relative contributions to radiation exposures of the nuclides in an inventory can be estimated by normalizing the ratio of each nuclide's curie amount and the run 2a value by the sum of those ratios. A run 2a value is the curie amount of the radionuclide that produces significant radiation doses (IAEA, 1961; IAEA, 1990). The RADSEL code was used to perform this calculation for the set of 34 nuclides in the inventories used in this study. The RADSEL calculation showed that only one radionuclide, promethium-147, in the set of 13 nuclides for which acute health effect dose conversion factors were lacking, contributes significantly to dose at the 99.9 percent level. More importantly, the calculation also showed that promethium-147 accounts for only 0.5 percent of the total dose produced by the full set of 34 radionuclides. Thus, the 21 nuclides in the inventories for which acute health effect conversion factors were available account for all significant contributions to dose. Therefore, not including acute health effects for 13 of the 34 radionuclides in the inventories used in this study is not believed to have had a

significant impact on the estimation of acute health effects, especially since none of these nuclides contributes significantly to chronic dose or health effects and since no acute effects were observed at any level including peak results for any calculation performed during this study.

D.5.4.3.3.3 Modal Study Cask Response Regions Release Fractions

The Modal Study (Fischer et al., 1987) developed release fractions for truck and rail accidents involving transportation cask containing commercial spent nuclear fuel. DOE as part of the preparation of the Programmatic SNF&INEL EIS, developed representative release fractions for metallic (aluminum-based) and nonmetallic (TRIGA) fuel for each of the Modal Study's cask response regions (DOE, 1995). Although there is not a direct relationship between the accident classification used in this EIS for ship accidents and that developed in the Modal Study, attempts were made to establish a meaningful comparison based on the definition of accidents and their consequences. Based on the accident definitions, one can approximate the severity category 5 ship accidents to the Modal Study's cask response region resulting from a medium impact mechanical force with a medium intensity thermal load. Table D-42 provides the values of release fractions used in this EIS for severity category 5 accident and that used for metallic and nonmetallic fuel in the Programmatic SNF&INEL EIS for a similar accident category. For ease of comparison, the EIS release fractions that were used in all of the base case calculations performed for this study are repeated in this table.

Table D-42 Programmatic SNF&INEL EIS Release Fractions

Element Group	*Release Fraction*		
	EIS (Base Case Category 5)	*Programmatic SNF&INEL EIS*	
		Metallic	*Nonmetallic*
Krypton	0.1	0.39	0.39
Cesium	9.0×10^{-4}	1.0×10^{-6}	0.00020
Ruthenium	1.0×10^{-6}	2.4×10^{-7}	0.000048
Particulate	5.0×10^{-8}	1.0×10^{-8}	0.0000020

Source: DOE, 1995

Inspection of the table shows that, except for the krypton element group, the base case EIS release fraction values for severity category 5 are somewhat larger than the values for nonmetallic fuel and are quite a bit larger than the values for metallic fuel. Thus, as would be expected, Table D-41 shows that mean and peak population doses and cancer fatalities for the distance ranges 0-1.6 and 0-80.5 km (0-1 and 0-50 mi) obtained using EIS release fractions are about five times larger than those obtained using nonmetallic fuel release fractions, which in turn are about 200 times larger than those obtained using metallic fuel release fractions. Therefore, since severity category 5 largely determines risk, use of EIS release fractions is conservative even if metallic and nonmetallic release fractions better represent releases during ship collisions.

D.5.4.3.3.4 Corrosion Products Release

During the operation of power reactors, radioactive cobalt is formed by neutron activation of chemical deposits on the outer surfaces of fuel rods. Thus, during transportation accidents, release of these radioactive deposits, usually referred to as corrosion products, can be a significant contributor to the size of the accident source term.

Because corrosion products formation is usually not a problem for research reactors, radioactive cobalt is not present in the inventories used in this study, and the sets of source terms input to MACCS do not contain fractions for corrosion products release. The potential impact of corrosion products release on foreign research reactor spent nuclear fuel accident source terms was examined by performing two sensitivity calculations. For these calculations, after scaling to match the size of the BR-2 inventory used in this study, the cobalt-60 content of the spent nuclear fuel inventory for a DOE test reactor (DOE, 1995) was added to the BR-2 inventory that was used in these sensitivity calculations (cobalt-58 was ignored as it should largely have decayed away before the fuel is shipped). Then, two sensitivity calculations were performed. Both calculations added 360 Ci of cobalt-60 to the BR-2 inventory and both used a value of 0.012 for the release fraction for the corrosion products chemical element group, as had been done in earlier studies. The first calculation examined the consequences of an accident that releases only corrosion products. Because corrosion products are not volatile, this release was assumed to be cold, that is driven by mechanical forces generated by the ship collision. The second calculation added the corrosion products release to the severity category 5 release used in the reference calculation. Because this release postulates a severe engulfing fire, the second calculation assumed that the release was hot.

Table D-41 shows that the first calculation, the cold release that contained only corrosion products (run 4a), leads to consequences that differ from those produced by the reference calculation as follows: for the 0-1.6 km (0-1 mi) distance range, mean values of population dose and cancer fatalities are about three times larger and peak values about two times smaller; for the 0-80.5 km (0-50 mi) distance range, mean and peak values for these two consequences are both smaller than the reference calculation results by factors of about 2.5 and 5 respectively. Mean and peak results for the 0-80.5 km (0-50 mi) distance range and peak results for the 0-1.6 km (0-1 mi) range are smaller because the curie content of the corrosion products release is smaller than the total curie content of the release used in the reference calculation (the release produced by severity category 5 release fractions and the BR-2 inventory). Mean results for the 0-1.6 km (0-1 mi) distance range are larger because the release is cold and therefore not lofted over nearby populations. Table D-41 also shows that adding the corrosion products release to the reference calculation (run 4b) increases consequence predictions only slightly (by about 20 percent), as would be expected given the small curie content of the corrosion products release compared to the reference release.

D.5.4.3.3.5 Work Force Population

Approximately 7,000 people work in Port Elizabeth in Newark, NJ. Thus, at least for accidents that occur during the workweek, these workers could be exposed to radiation as a result of a ship collision that involves a ship carrying foreign research reactor spent nuclear fuel. Inspection of maps showed that these workers should be added to the residential populations in the first distance intervals of the north sector of the Newark dock population distribution. Since the division of workers between these two distance intervals was not known, 3,500 workers were added to each interval for these sensitivity calculations.

Work force sensitivity calculations were performed first assuming, as was done for the reference calculation, a hot release, the BR-2 inventory, and severity category 5 release fractions. Then, this calculation was repeated two times assuming a cold release. The first of these two cold release calculations used the same shielding factors that had been used in the reference calculation. For the second cold release calculation, larger shielding factors were used during the first 24 hours after the accident over the distance range 0-8 km (0-5 mi) because the commercial buildings near the port are likely to provide better shielding than is provided by the mix of buildings located within 80.5 km (50 mi) of the port. Next, these two cold release calculations were repeated assuming that the release consists of a puff caused by the collision impact and a tail caused by the ensuing fire. Severity category 4 release fractions were used for the puff, and the release fractions for the tail were obtained by subtracting the severity

category 4 release fractions from the severity category 5 release fractions. The puff was released when the collision occurred and lasted for 10 minutes; the tail was released one hour later and had a one hour release duration. Finally, the puff and tail calculation that did not use increased shielding factor values was repeated assuming that an evacuation would be called for should a severe accident lead to a fire that engulfed a radioactive material transportation cask, that the evacuation would begin about one hour after the accident took place (i.e., at about the time the tail release begins), and that the average evacuation speed would be slow because of city congestion.

Inspection of Table D-41 shows that, when a hot release is assumed (run 5a), adding a work force population increases mean population dose and cancer fatalities by less than a factor of 2 in the 0-1.6 km (0-1 mi) distance range, but has little effect on peak values in this distance range or on either mean or peak values in the 0-80.5 km (0-50 mi) distance range. When the release is cold (run 5b), 0-1.6 km (0-1 mi) mean population doses and cancer fatalities are increased by factors of about 26 and 2 respectively, and peak doses and cancer fatalities are increased by factors of about 3. For the 0-80.5 km (0-50 mi) distance range mean results are increased by factors of about 2 and peak results actually decrease by a factor of about 0.7. Moreover, these results are little changed by using increased shielding factors for commercial buildings, by assuming a puff and tail release, or by assuming a slow delayed evacuation.

The insensitivity to short-term shielding factor values, to release timing, and to evacuation is easy to understand when one remembers that population dose and cancer fatalities in these calculations are determined almost entirely by long-term groundshine exposures, which are of course little influenced by variation of any of these three short-term effects. Thus, as was shown above, elimination of lofting by assuming a cold release increases consequences, especially those that occur at short distances, but little else has much effect because only recovery actions (decontamination, temporary interdiction, condemnation) not examined by these sensitivity calculations can significantly affect long-term groundshine dose.

D.5.5 Port Accident Risk

The port accident risk analysis combines the results of the analysis of the frequency of ship accidents in the port area with the results of the consequence analysis of each of these accidents. Each of the accident severity categories contributes to the overall risk of accidents in the port. The total risk is the sum of the risk for each severity category. The specific methodology used to evaluate port accident risks and the results of that analysis are presented in this section.

The port accident risk analysis was performed based on 721 individual shipments of foreign research reactor spent nuclear fuel. Unlike the incident-free analyses, where the shipment of two or more casks on the same vessel results in an increase in the worker risk, the number of casks shipped on a single vessel does not affect the results of the analysis. The larger the number of casks on a single vessel, the fewer the number of shipments required to ship all 721 casks. Accident data is generated on a per transit basis. Assuming a single cask per shipment maximizes the number of shipments and maximizes the probability of an accident involving a ship carrying foreign research reactor spent nuclear fuel. If it is assumed that an accident that results in damage to a foreign research reactor spent nuclear fuel cask results in damage to all of the casks on a single vessel, the risks from the shipment of multiple casks on a single vessel would be identical to the risks associated with the shipment of the same number of casks individually. From the analysis performed in Appendix D Attachment D4, it is apparent that the probability of damage to all casks given that one is damaged in an accident is less than one. Therefore, performing the port accident risk analysis assuming that one cask is shipped per voyage results in an estimate of risk that is maximized for number of transportation casks shipped per voyage.

The accident risks have been evaluated for 13 ports: Elizabeth, NJ; the Hampton Roads, VA, ports of Portsmouth, Norfolk, and Newport News (using Portsmouth as the representative port); MOTSU, NC; Charleston, SC; Philadelphia. PA; Long Beach, CA; Savannah, GA; Galveston, TX; Concord NWS, CA; Tacoma, WA; Wilmington, NC; Jacksonville, FL; and Portland, OR. Although high population density ports do not meet the port selection screening criteria, the three high population ports of Elizabeth, Long Beach, and Philadelphia were included in the analysis for two purposes. First, it is possible that the shipments of foreign research reactor spent nuclear fuel could be made on vessels that make intermediate port calls, which could include these high population ports. Additionally, by evaluating these high population ports as ports of entry it was possible to estimate the maximum port accident risks resulting from the shipment of foreign research reactor spent nuclear fuel into the United States.

As discussed in the port accident consequence analysis (Section D.5.4), the accident analysis has evaluated the impact of accidents at two locations within each of the ports considered in the risk analysis. The two locations represent the possibility of: (1) an accident involving the ship transporting the foreign research reactor spent nuclear fuel while at the dock and (2) an accident at some point in the approach to the dock. Two locations were selected to address the possibility that the terminal (pier at which the cargo vessel is docked) may not be the location within the port that would yield the highest consequences for an accident. The key consideration is that in approaching the terminal, at some ports, the cargo vessel would pass through areas with a higher nearby population than the area around the terminal. To ensure that the accident consequence analysis did not underestimate the potential consequences, this second accident location was selected. It was selected by identifying the point in the approach to the terminal which had characteristics most likely to result in consequences representative of the largest consequences associated with an accident within the port facility. This generally meant a location near a population center. Accident locations were identified earlier in Table D-28.

Because two locations were selected for the accident analysis in each port, the total risk associated with a port call at the port of entry is the sum of the risks at these two locations. Accidents may occur either at the terminal (dock) or in the channel as the vessel approaches the dock This risk can be expressed as:

$$R_{PE} = \sum \left(M_D P_D + M_C P_C \right)$$

where:

R_{PE}=Risks from accidents in the port of entry,

M_D=Magnitude of the consequences for a severity category 4, 5, 6 accident at the dock,

P_D=Probability of an accident of severity category 4, 5, 6 at the dock,

M_C=Magnitude of the consequences for an accident in the approach to the dock (in the channel), and

P_C=Probability of an accident of severity category 4, 5, 6 in the approach to the dock (in the channel).

One of the assumptions made in the port risk analysis is that the vessel carrying the foreign research reactor spent nuclear fuel may make intermediate port calls at up to two different ports before arriving at the port of entry. In the event that these intermediate port calls are made, the risks associated with each of these port calls can be expressed as follows:

$$R_{IP} = \sum \left(M_D P_D + 2 M_C P_C \right)$$

where R_{IP} is the risk from an accident in one of the intermediate ports of call. All other parameters have the same definitions as in the equation defining R_{PE}. The risks associated with accidents in the channel of the port is considered twice for the intermediate ports because the vessel must enter the harbor and approach the dock and, with the foreign research reactor spent nuclear fuel still on board, must depart the harbor. The accident frequency data is derived as a per transit frequency. For this risk analysis the approach to the dock has been considered to be part of one transit, the departure as part of a second transit.

From Section D.5.3.1.7, the probabilities per transit for the three accident severity categories evaluated are provided in Table D-43. These accident frequencies were used to develop the per transit probabilities for the accidents at the dock and in the channel for each of the intermediate ports and the ports of entry for the foreign research reactor spent nuclear fuel. The port accident data collected was not detailed enough to determine the percentage of accidents that occurred at the dock versus the percentage that occurred in the channel. For the purposes of this analysis, it was assumed that the accidents were evenly distributed between the dock and the approach to the dock. Table D-43 presents the per transit probabilities used in the port accident analysis for accidents at the dock and in the channel.

Table D-43 Port Accident Probabilities

Accident Severity Category	P	P_d	P_c
4	0.000006	0.000003	0.000003
5	0.000000005	0.0000000027	0.0000000027
6	0.0000000006	0.0000000003	0.0000000003

Accident consequences (mean results) for each of the accident severity categories are reproduced in Tables D-44 and D-45, in terms of total population dose and LCF, respectively. The consequences vary depending on the type of fuel involved in the accident, the port at which the accident occurs, the severity category, and the location of the accident within the port environs. The largest differences are between the different release categories and is the result of the smaller release fractions for a severity category 4 accident than for the severity category 5 and 6 accidents. Between the different ports assessed in the analysis, the consequences vary by a factor of approximately 30 [i.e., the consequences of an accident in Elizabeth (the location of the highest consequences) are approximately 30 times greater than the consequences of the same accident at MOTSU (the location with the lowest consequences)].

Using the equations presented previously in this section, the probability and consequence data were combined to generate the risk data presented in Table D-46. This table presents data on a per shipment basis and for the shipment of all 721 foreign research reactor spent nuclear fuel casks. Data is presented for shipments that are made with no intermediate port stops (identified as direct shipments in the table) and for shipments that are made with intermediate port stops. The direct shipments are quantified using the relationship developed for R_{PE}. For example, the risks in terms of person-rem associated with a single direct shipment consisting of a single cask of BR-2 fuel into the port of Elizabeth are the sum of the severity category 4 risks, severity category 5 risks, and severity category 6 risks associated with accidents at the Elizabeth dock (0.00000069, 0.000018, 0.0000020) and in the approach to the port of Elizabeth (0.0000011, 0.000019, 0.0000020), which is 0.000042 as shown in the table.

In developing the risk estimates for shipments that pass through intermediate ports, several combinations of intermediate ports were considered for each ultimate port of entry. The ports selected for use in this analysis represent the range of populations found in ports around the United States. As stated previously; Elizabeth, Philadelphia, and Long Beach are considered high population ports; Portland, Jacksonville, Tacoma, Concord NWS, and the Hampton Roads ports are considered to be intermediate population ports; and Charleston, Savannah, Wilmington, Galveston, and MOTSU are considered low population ports. Each possible combination of populations was considered for the intermediate ports. The risks associated

Table D-44 Port Accident Analysis—Total Effective Dose Equivalent Population Dose (Person-Rem)

Location	*BR-2 Spent Nuclear Fuel*			*RHF Spent Nuclear Fuel*			*TRIGA Spent Nuclear Fuel*		
	Severity Category			*Severity Category*			*Severity Category*		
	4	*5*	*6*	*4*	*5*	*6*	*4*	*5*	*6*
Elizabeth (D)[1]	0.23	6600	6500	0.093	2600	2600	0.028	910	900
Elizabeth (C)[2]	0.38	6900	6800	0.15	2700	2700	0.045	960	940
Long Beach (D)[1]	0.21	4700	4800	0.085	1900	1900	0.025	650	660
Long Beach (C)[2]	0.081	4300	4400	0.032	1700	1700	0.0097	590	610
Philadelphia (D)[1]	0.18	2800	2800	0.071	1100	1100	0.021	380	380
Philadelphia (C)[2]	0.085	2700	2800	0.034	1100	1100	0.010	370	380
Portland (D)[1]	0.077	1200	1200	0.031	450	450	0.0093	160	160
Portland (C)[2]	0.053	1100	1200	0.021	430	440	0.0065	150	150
Norfolk (D)[1]	0.055	850	830	0.022	330	320	0.0067	110	110
Norfolk (C)[2]	0.030	670	660	0.012	250	250	0.0037	87	87
Charleston Wando Terminal (D)[1]	0.024	420	410	0.0096	150	150	0.003	53	53
Charleston NWS (D)[1]	0.016	480	480	0.0066	180	180	0.0021	61	61
Charleston (C)[2]	0.038	420	420	0.015	160	160	0.0046	54	54
Tacoma (D)[1]	0.056	1,700	1,800	0.022	670	700	0.0068	230	250
Tacoma (C)[2]	0.039	1,400	1,500	0.016	550	570	0.0048	190	200
Concord NWS (D)[1]	0.044	2,100	2,200	0.018	800	850	0.0054	280	300
Concord NWS (C)[2]	0.094	3,300	3,400	0.038	1,300	1,300	0.011	450	460
Jacksonville (D)[1]	0.028	680	680	0.011	260	250	0.0035	88	87
Jacksonville (C)[2]	0.026	530	550	0.010	200	200	0.0032	69	70
Savannah (D)[1]	0.056	490	500	0.022	180	180	0.0068	62	63
Savannah (C)[2]	0.013	380	390	0.005	140	140	0.0018	47	49
Wilmington (D)[1]	0.038	480	500	0.015	180	190	0.0047	62	64
Wilmington (C)[2]	0.0097	210	220	0.0038	75	80	0.0012	26	27
Galveston (D)[1]	0.073	1,400	1,600	0.029	550	600	0.0089	190	210
Galveston (C)[2]	0.032	1,400	1,600	0.013	540	590	0.0041	190	200
MOTSU (D)[1]	0.0073	210	220	0.0029	75	80	0.0010	25	27
MOTSU (C)[2]	0.0097	210	220	0.0038	75	80	0.0012	26	27

[1]*Accident is at the Dock*

[2]*Accident is in the Channel, the approach to the dock*

with a shipment that passed through two U.S. ports before arriving at the port of entry for the foreign research reactor spent nuclear fuel were calculated using the relationships for R_{PE} and R_{IP}. The risks were calculated for each intermediate port stop and added to the risks associated with operations within the port of entry, i.e., the risks associated with a direct shipment.

The per shipment data was used to calculate the risks associated with the basic implementation of Management Alternative 1 of proposed action. The values shown in the two rightmost columns of Table D-46 represent the risks associated with the shipment of all of the foreign research reactor spent nuclear fuel through a single port of entry via the same intermediate ports. Using the shipments through Elizabeth as an example, the value given for the program risks for the shipment of the foreign research reactor spent nuclear fuel through one intermediate and one low population port (0.027 person-rem or 0.000011 LCF) assumes that all 721 foreign research reactor spent nuclear fuel casks are shipped through these same three ports. The number of shipments of each type of fuel (473 BR-2, 86 RHF, and 162 TRIGA) were incorporated into the development of the risks.

Table D-45 Port Accident Analysis—Accident Consequences (LCF)

Location	BR-2 Spent Nuclear Fuel			RHF Spent Nuclear Fuel			TRIGA Spent Nuclear Fuel		
	Severity Category			Severity Category			Severity Category		
	4	5	6	4	5	6	4	5	6
Elizabeth (D)[1]	0.00010	2.8	2.7	0.000041	1.1	1.1	0.000011	0.38	0.38
Elizabeth (C)[2]	0.00016	2.9	2.8	0.000066	1.1	1.1	0.000018	0.40	0.39
Long Beach (D)[1]	0.000093	2.0	2.0	0.000038	0.78	0.80	0.000010	0.27	0.28
Long Beach (C)[2]	0.000035	1.8	1.9	0.000014	0.71	0.73	0.0000040	0.25	0.26
Philadelphia (D)[1]	0.000078	1.2	1.2	0.000031	0.47	0.46	0.0000087	0.16	0.16
Philadelphia (C)[2]	0.000037	1.2	1.2	0.000015	0.45	0.47	0.0000042	0.16	0.16
Portland (D)[1]	0.000034	0.52	0.53	0.000014	0.20	0.20	0.0000039	0.068	0.069
Portland (C)[2]	0.000023	0.50	0.51	0.0000093	0.19	0.19	0.0000027	0.065	0.067
Norfolk (D)[1]	0.000024	0.38	0.37	0.0000097	0.14	0.14	0.0000028	0.049	0.048
Norfolk (C)[2]	0.000013	0.30	0.30	0.0000053	0.11	0.11	0.0000015	0.039	0.039
Charleston Wando Terminal (D)[1]	0.000011	0.19	0.19	0.0000042	0.070	0.070	0.0000012	0.024	0.024
Charleston NWS (D)[1]	0.0000068	0.22	0.22	0.0000027	0.080	0.080	0.00000084	0.028	0.028
Charleston(C)[2]	0.000017	0.19	0.19	0.0000067	0.070	0.071	0.0000019	0.024	0.024
Tacoma (D)[1]	0.000024	0.75	0.80	0.0000097	0.29	0.30	0.0000028	0.10	0.11
Tacoma (C)[2]	0.000017	0.63	0.66	0.0000068	0.24	0.25	0.0000020	0.083	0.087
Concord NWS (D)[1]	0.000019	0.90	0.96	0.0000076	0.34	0.37	0.0000022	0.12	0.13
Concord NWS (C)[2]	0.000041	1.4	1.5	0.000017	0.55	0.56	0.0000046	0.19	0.20
Jacksonville (D)[1]	0.000012	0.31	0.31	0.0000049	0.11	0.11	0.0000015	0.039	0.039
Jacksonville (C)[2]	0.000011	0.24	0.25	0.0000045	0.090	0.092	0.0000013	0.031	0.032
Savannah (D)[1]	0.000025	0.23	0.23	0.0000099	0.083	0.085	0.0000028	0.028	0.029
Savannah (C)[2]	0.0000059	0.18	0.19	0.0000023	0.065	0.067	0.00000074	0.022	0.023
Wilmington (D)[1]	0.000017	0.22	0.23	0.0000067	0.081	0.084	0.0000019	0.028	0.029
Wilmington (C)[2]	0.0000042	0.098	0.10	0.0000017	0.035	0.037	0.0000005	0.012	0.013
Galveston (D)[1]	0.000032	0.64	0.70	0.000013	0.24	0.27	0.0000037	0.084	0.092
Galveston (C)[2]	0.000014	0.63	0.69	0.0000056	0.24	0.26	0.0000017	0.082	0.090
MOTSU (D)[1]	0.0000032	0.099	0.11	0.0000013	0.035	0.038	0.00000041	0.012	0.013
MOTSU (C)[2]	0.0000042	0.098	0.10	0.0000017	0.035	0.037	0.00000052	0.012	0.013

[1] *Accident is at the Dock*

[2] *Accident is in the Channel, the approach to the dock*

Table D-46 Summary of Latent Cancer Fatalities and Population Exposure Risk—Per Shipment and for the Entire Program (Basic Implementation)

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
Elizabeth via:								
Two High Population Ports	0.00013	0.000052	0.000018	0.000000056	0.000000022	0.0000000075	0.070	0.000029
One High and One Intermediate Population Port	0.00011	0.000044	0.000016	0.000000048	0.000000019	0.0000000065	0.060	0.000025
One High and One Low Population Port	0.00011	0.000043	0.000015	0.000000045	0.000000018	0.0000000062	0.057	0.000024

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
Two Intermediate Population Ports	0.000056	0.000022	0.0000076	0.000000024	0.0000000093	0.0000000032	0.030	0.000013
One Intermediate and One Low Population Port	0.000051	0.000020	0.0000070	0.000000022	0.0000000085	0.0000000029	0.027	0.000011
Two Low Population Ports	0.000046	0.000018	0.0000063	0.000000020	0.0000000077	0.0000000026	0.024	0.000010
Direct	0.000042	0.000017	0.0000058	0.000000018	0.0000000070	0.0000000024	0.022	0.0000094
Long Beach via:								
Two High Population Ports	0.000011	0.000044	0.000015	0.000000047	0.000000018	0.0000000064	0.058	0.000025
One High and One Intermediate Population Port	0.000080	0.000032	0.0000011	0.000000034	0.000000013	0.0000000043	0.042	0.000018
One High and One Low Population Port	0.000071	0.000028	0.0000097	0.000000030	0.0000000012	0.0000000041	0.038	0.000016
Two Intermediate Population Ports	0.000050	0.000019	0.0000067	0.000000021	0.0000000083	0.0000000022	0.026	0.000011
One Intermediate and One Low Population Port	0.000041	0.000016	0.0000055	0.000000018	0.0000000068	0.0000000020	0.022	0.0000092
Two Low Population Ports	0.000032	0.0000013	0.0000043	0.000000014	0.0000000053	0.0000000018	0.017	0.0000072
Direct	0.000028	0.0000011	0.0000038	0.000000012	0.0000000046	0.0000000016	0.015	0.0000062
Philadelphia via:								
Two High Population Ports	0.00011	0.000042	0.000015	0.000000045	0.000000018	0.0000000061	0.057	0.000024
One High and One Intermediate Population Port	0.000088	0.000035	0.000012	0.000000037	0.000000015	0.0000000050	0.047	0.000020
One High and One Low Population Port	0.000083	0.000033	0.000011	0.000000035	0.000000014	0.0000000048	0.044	0.000019
Two Intermediate Population Ports	0.000031	0.000012	0.0000041	0.000000014	0.0000000052	0.0000000018	0.016	0.0000072

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
One Intermediate and One Low Population Port	0.000026	0.000010	0.0000035	0.000000011	0.0000000044	0.0000000015	0.014	0.0000061
Two Low Population Ports	0.000021	0.0000083	0.0000028	0.0000000093	0.0000000036	0.0000000012	0.011	0.0000049
Direct	0.000017	0.0000069	0.0000023	0.0000000075	0.0000000029	0.0000000099	0.0092	0.0000040
Portland via:								
Two High Population Ports	0.000090	0.000035	0.000012	0.000000038	0.000000015	0.0000000050	0.047	0.000020
One High and One Intermediate Population Port	0.000059	0.000023	0.0000080	0.000000025	0.0000000099	0.0000000029	0.031	0.000013
One High and One Low Population Port	0.000050	0.000020	0.0000068	0.000000022	0.0000000084	0.0000000027	0.027	0.000011
Two Intermediate Population Ports	0.000029	0.000011	0.0000039	0.000000013	0.0000000049	0.00000000088	0.015	0.0000066
One Intermediate and One Low Population Port	0.000020	0.0000077	0.0000027	0.0000000090	0.0000000034	0.00000000068	0.011	0.0000048
Two Low Population Ports	0.000011	0.0000042	0.0000015	0.0000000051	0.0000000019	0.00000000049	0.0059	0.0000026
Direct	0.0000073	0.0000028	0.00000098	0.0000000032	0.0000000012	0.00000000026	0.0039	0.0000017
Norfolk via:								
Two High Population Ports	0.000095	0.000037	0.000013	0.000000040	0.000000016	0.0000000054	0.050	0.000021
One High and One Intermediate Population Port	0.000076	0.000030	0.000010	0.000000032	0.000000013	0.0000000043	0.040	0.000017
One High and One Low Population Port	0.000071	0.000028	0.0000097	0.000000030	0.000000012	0.0000000040	0.037	0.000016
Two Intermediate Population Ports	0.000019	0.0000071	0.0000024	0.0000000083	0.0000000031	0.0000000011	0.0098	0.0000044
One Intermediate and One Low Population Port	0.000014	0.0000052	0.0000018	0.0000000061	0.0000000023	0.00000000078	0.0072	0.0000032
Two Low Population Ports	0.0000088	0.0000033	0.0000011	0.0000000040	0.0000000015	0.00000000050	0.0046	0.0000021

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
Direct	0.0000048	0.0000018	0.00000062	0.0000000021	0.0000000081	0.00000000028	0.0025	0.0000011
Charleston (Wando Terminal) via:								
Two High Population Ports	0.000092	0.000036	0.000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000074	0.000029	0.000010	0.000000031	0.000000012	0.0000000042	0.039	0.000016
One High and One Low Population Port	0.000069	0.000027	0.0000094	0.000000029	0.000000011	0.0000000039	0.036	0.000015
Two Intermediate Population Ports	0.000016	0.0000063	0.0000021	0.0000000074	0.0000000028	0.00000000095	0.0087	0.0000039
One Intermediate and One Low Population Port	0.000012	0.0000043	0.0000015	0.0000000052	0.0000000019	0.00000000066	0.0061	0.0000027
Two Low Population Ports	0.0000066	0.0000024	0.00000082	0.0000000031	0.0000000011	0.00000000038	0.0035	0.0000016
Direct	0.0000027	0.000001	0.00000034	0.0000000012	0.00000000045	0.00000000015	0.0014	0.00000064
Charleston NWS via:								
Two High Population Ports	0.000093	0.000033	0.000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000074	0.000029	0.000010	0.000000031	0.000000012	0.0000000042	0.039	0.000016
One High and One Low Population Port	0.000069	0.000027	0.0000094	0.000000029	0.000000011	0.0000000039	0.036	0.000015
Two Intermediate Population Ports	0.000017	0.0000063	0.0000022	0.0000000075	0.0000000028	0.00000000096	0.0084	0.0000039
One Intermediate and One Low Population Port	0.000012	0.0000044	0.0000015	0.0000000053	0.0000000020	0.00000000067	0.0058	0.0000028
Two Low Population Ports	0.0000068	0.0000025	0.00000084	0.0000000032	0.0000000011	0.00000000039	0.0032	0.0000017
Direct	0.0000028	0.0000011	0.00000036	0.0000000013	0.00000000048	0.00000000016	0.0011	0.00000068
MOTSU via:								
Two High Population Ports	0.000091	0.000036	0.000012	0.000000039	0.000000015	0.0000000052	0.048	0.000020

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
One High and One Intermediate Population Port	0.000072	0.000028	0.0000099	0.000000031	0.000000012	0.0000000041	0.038	0.000016
One High and One Low Population Port	0.000067	0.000026	0.0000092	0.0000000028	0.0000000011	0.0000000038	0.036	0.000015
Two Intermediate Population Ports	0.000015	0.0000057	0.0000019	0.0000000068	0.0000000025	0.00000000087	0.0080	0.0000036
One Intermediate and One Low Population Port	0.000010	0.0000038	0.0000013	0.0000000046	0.0000000017	0.00000000058	0.0054	0.0000024
Two Low Population Ports	0.0000053	0.0000019	0.00000064	0.0000000025	0.00000000088	0.0000000003	0.0028	0.0000013
Direct	0.0000013	0.00000047	0.00000016	0.00000000062	0.00000000022	0.000000000075	0.00069	0.00000032
Galveston via:								
Two High Population Ports	0.000099	0.000039	0.000013	0.000000042	0.000000016	0.0000000056	0.052	0.000022
One High and One Intermediate Population Port	0.000080	0.000031	0.000011	0.000000034	0.000000013	0.0000000046	0.042	0.000018
One High and One Low Population Port	0.000075	0.000029	0.0000010	0.000000032	0.0000000012	0.0000000043	0.040	0.000017
Two Intermediate Population Ports	0.000023	0.0000087	0.0000030	0.000000010	0.0000000038	0.0000000013	0.012	0.0000053
One Intermediate and One Low Population Port	0.000018	0.0000067	0.0000023	0.0000000080	0.000000003	0.000000001	0.0094	0.0000042
Two Low Population Ports	0.000013	0.0000048	0.0000017	0.0000000058	0.0000000022	0.00000000074	0.0068	0.0000031
Direct	0.0000090	0.0000034	0.0000012	0.0000000040	0.0000000015	0.00000000052	0.0047	0.0000021
Jacksonville via:								
Two High Population Ports	0.000094	0.000037	0.000013	0.000000040	0.000000016	0.0000000053	0.050	0.000021
One High and One Intermediate Population Port	0.000075	0.000029	0.00001	0.000000032	0.000000012	0.0000000043	0.040	0.000017

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
One High and One Low Population Port	0.000070	0.000027	0.0000096	0.000000029	0.000000012	0.0000000040	0.037	0.000016
Two Intermediate Population Ports	0.000018	0.0000067	0.0000023	0.0000000079	0.0000000030	0.0000000010	0.0093	0.0000041
One Intermediate and One Low Population Port	0.000013	0.0000048	0.0000016	0.0000000057	0.0000000021	0.00000000073	0.0067	0.000003
Two Low Population Ports	0.0000078	0.0000028	0.00000097	0.0000000036	0.0000000013	0.00000000045	0.0041	0.0000019
Direct	0.0000038	0.0000014	0.00000049	0.0000000017	0.00000000064	0.00000000022	0.0020	0.00000090
Savannah via:								
Two High Population Ports	0.000093	0.000037	0.000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000074	0.000029	0.00001	0.000000031	0.000000012	0.0000000042	0.039	0.000016
One High and One Low Population Port	0.000069	0.000027	0.0000094	0.000000029	0.000000011	0.0000000039	0.036	0.000015
Two Intermediate Population Ports	0.000017	0.0000063	0.0000021	0.0000000075	0.0000000028	0.00000000095	0.0088	0.0000039
One Intermediate and One Low Population Port	0.000012	0.0000044	0.0000015	0.0000000053	0.000000002	0.00000000067	0.0062	0.0000028
Two Low Population Ports	0.0000068	0.0000025	0.00000083	0.0000000032	0.0000000011	0.00000000039	0.0036	0.0000017
Direct	0.0000028	0.000001	0.00000035	0.0000000013	0.00000000048	0.00000000016	0.0015	0.00000069
Wilmington via:								
Two High Population Ports	0.000092	0.000036	0.000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000073	0.000029	0.00001	0.000000031	0.000000012	0.0000000042	0.039	0.000016
One High and One Low Population Port	0.000068	0.000027	0.0000094	0.000000029	0.000000011	0.0000000039	0.036	0.000015
Two Intermediate Population Ports	0.000016	0.0000061	0.0000021	0.0000000072	0.0000000027	0.00000000092	0.0084	0.0000038

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
One Intermediate and One Low Population Port	0.000011	0.0000042	0.0000014	0.0000000050	0.0000000019	0.00000000064	0.0058	0.0000026
Two Low Population Ports	0.0000062	0.0000022	0.00000076	0.0000000029	0.0000000010	0.00000000035	0.0032	0.0000015
Direct	0.0000022	0.00000082	0.00000028	0.0000000010	0.00000000037	0.00000000013	0.0012	0.00000053
Tacoma via:								
Two High Population Ports	0.000092	0.000036	0.0000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000062	0.000024	0.0000084	0.000000026	0.000000010	0.0000000032	0.033	0.000014
One High and One Low Population Port	0.000053	0.000021	0.0000072	0.000000023	0.0000000088	0.0000000031	0.028	0.000012
Two Intermediate Population Ports	0.000031	0.000012	0.0000042	0.000000014	0.0000000053	0.0000000012	0.017	0.0000072
One Intermediate and One Low Population Port	0.000023	0.0000086	0.0000030	0.000000010	0.0000000038	0.00000000099	0.012	0.0000053
Two Low Population Ports	0.000014	0.0000052	0.0000018	0.0000000061	0.0000000023	0.00000000079	0.0072	0.0000032
Direct	0.0000097	0.0000038	0.0000013	0.0000000043	0.0000000016	0.00000000057	0.0051	0.0000023
Concord NWS via:								
Two High Population Ports	0.000099	0.000039	0.000013	0.000000042	0.000000016	0.0000000057	0.052	0.000022
One High and One Intermediate Population Port	0.000069	0.000027	0.0000093	0.000000029	0.000000011	0.0000000036	0.036	0.000015
One High and One Low Population Port	0.000060	0.000024	0.0000081	0.000000025	0.0000000099	0.0000000034	0.032	0.000013
Two Intermediate Population Ports	0.000038	0.000015	0.0000051	0.000000017	0.0000000064	0.0000000016	0.020	0.0000087
One Intermediate and One Low Population Port	0.000029	0.000011	0.0000039	0.000000013	0.0000000049	0.0000000014	0.016	0.0000067

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
Two Low Population Ports	0.000021	0.0000079	0.0000027	0.0000000090	0.0000000034	0.0000000012	0.011	0.0000047
Direct	0.000017	0.0000065	0.0000022	0.0000000071	0.0000000028	0.00000000096	0.0088	0.0000038

These risk estimates provide an estimate of the range of the port accident risks that would result from the basic implementation of Management Alternative 1 via the use of a wide range of ports. The ports of Elizabeth, Philadelphia, and Long Beach were included in the analysis as ports of entry even though they did not survive the port screening criteria. However, because of the high populations around these ports, their use provides an estimate of the highest risks associated with the shipment of foreign research reactor spent nuclear fuel into the United States. These risks can be contrasted with the risks associated with the shipment of the foreign research reactor spent nuclear fuel through MOTSU, which has an extremely low population around the port.

The port accident risks associated with the entire program range from a high of 0.070 person-rem and 0.000029 LCF, which assumes that all shipments would be made through two high population intermediate ports into Elizabeth, to a low of 0.0007 person-rem and 0.00000032 LCF, which assumes that the shipments are made directly into MOTSU. In the worst case analyzed the mean risks associated with port accidents results in an approximately one-in-a-thousand chance of a single LCF. The highest risks associated with a port that did meet the port selection criteria (assuming no restrictions on the selection of intermediate ports) is 0.000022 LCF. If, in addition, all intermediate port calls are restricted to port cities of similar size to those that meet the selection criteria, the highest calculated risk is reduced to 0.000009 LCF, approximately a one-in-a-hundred thousand chance of a single LCF.

D.5.6 Port Accident Impacts for Implementation Alternatives

Two implementation alternatives to Management Alternative 1 were identified that could impact the results of the port accident risk analysis that was developed for the basic implementation case. They are: 1a, Accepting Fuel from Developing Countries Only, and 2a, Accepting Fuel for Only Five Years. Developing countries are countries other than high income economies. Both of the implementation alternatives change the number of transportation casks containing foreign research reactor spent nuclear fuel that would be shipped to the United States.

The difference in the number of shipments does not affect the per-transit probability of an accident. The conditional probabilities of a severity category 4, 5, or 6 accident also do not change. On a per-shipment basis, the probability an accident of each of these severity categories is identical to the estimates used in the analysis of the basic implementation.

The consequences associated with each of the three accident severity categories also do not change just due to the change in the number of shipments. Since neither the probability nor the consequences of the accidents change, the per-shipment risks are identical to those of the basic implementation.

These alternatives are discussed in the following paragraphs.

Acceptance of Foreign Research Reactor Spent Nuclear Fuel from Developing Countries Only: Developing countries are defined as countries other than high-income economies. Under this alternative 168 transportation casks of foreign research reactor spent nuclear fuel would be shipped to the United States (see Appendix C.4.2 for details). All of these shipments would be shipped by ocean vessel and, therefore, would enter the United States through ports.

In addition to a reduced number of shipments associated with this alternative, the mix of fuel types changes. In the basic implementation of Management Alternative 1, most of the foreign research reactor spent nuclear fuel shipments would be BR-2 type fuel. Only about 20 percent of the shipments would be of the TRIGA fuel type. From the information provided in Appendix B, most of the shipments from countries other than high-income economies would be TRIGA fuel. Of the 168 shipments under this implementation alternative, 109 are TRIGA shipments. The remaining 59 shipments are BR-2 fuel shipments.

The risks of the basic implementation of Management Alternative 1, provided in Table D-46, have been recalculated to incorporate the change in the number and makeup of the shipments associated with this implementation alternative. These results are presented in Table D-47. The highest calculated port accident risks are associated with the shipment of all of the foreign research reactor spent nuclear fuel through the port of Elizabeth via two high population intermediate ports. The port accident risks for this implementation alternative for this route are 0.0098 person-rem and 0.000004 LCF. The lowest calculated impacts are for the shipment of all of the material directly into MOTSU (no intermediate port calls) which results in port accident risks of 0.000095 person-rem and 0.000000045 LCF.

Table D-47 Summary of Risk and Population Exposure—For the Implementation Alternative of Acceptance of Foreign Research Reactor Spent Nuclear Fuel Only From Countries Other than High-Income Economies

Port	*Exposure (person-rem)*	*Risk (LCF)*
Elizabeth via:		
Two High Population Ports	0.0098	0.0000041
One High and One Intermediate Population Port	0.0084	0.0000035
One High and One Low Population Port	0.0080	0.0000034
Two Intermediate Population Ports	0.0041	0.0000018
One Intermediate and One Low Population Port	0.0038	0.0000016
Two Low Population Ports	0.0034	0.0000014
Direct	0.0031	0.0000013
Long Beach via:		
Two High Population Ports	0.0081	0.0000034
One High and One Intermediate Population Port	0.0059	0.0000025
One High and One Low Population Port	0.0052	0.0000022
Two Intermediate Population Ports	0.0037	0.0000015
One Intermediate and One Low Population Port	0.0030	0.0000013
Two Low Population Ports	0.0023	0.0000010
Direct	0.0021	0.00000087
Philadelphia via:		
Two High Population Ports	0.0079	0.0000033
One High and One Intermediate Population Port	0.0065	0.0000028
One High and One Low Population Port	0.0062	0.0000026
Two Intermediate Population Ports	0.0023	0.00000010
One Intermediate and One Low Population Port	0.0019	0.00000084

Port	Exposure (person-rem)	Risk (LCF)
Two Low Population Ports	0.0016	0.00000068
Direct	0.0013	0.00000055
Portland via:		
Two High Population Ports	0.0066	0.0000028
One High and One Intermediate Population Port	0.0044	0.0000018
One High and One Low Population Port	0.0037	0.0000016
Two Intermediate Population Ports	0.0021	0.00000085
One Intermediate and One Low Population Port	0.0015	0.00000060
Two Low Population Ports	0.00082	0.00000035
Direct	0.00054	0.00000022
Norfolk via:		
Two High Population Ports	0.0070	0.0000030
One High and One Intermediate Population Port	0.0056	0.0000024
One High and One Low Population Port	0.0052	0.0000022
Two Intermediate Population Ports	0.0014	0.00000061
One Intermediate and One Low Population Port	0.0010	0.00000045
Two Low Population Ports	0.00064	0.00000029
Direct	0.00035	0.00000016
Charleston (Wando Terminal) via:		
Two High Population Ports	0.0069	0.0000029
One High and One Intermediate Population Port	0.0054	0.0000023
One High and One Low Population Port	0.0051	0.0000021
Two Intermediate Population Ports	0.0012	0.00000054
One Intermediate and One Low Population Port	0.00084	0.00000038
Two Low Population Ports	0.00048	0.00000022
Direct	0.00020	0.000000089
Charleston NWS via:		
Two High Population Ports	0.0068	0.0000029
One High and One Intermediate Population Port	0.0054	0.0000023
One High and One Low Population Port	0.0051	0.0000021
Two Intermediate Population Ports	0.0012	0.00000054
One Intermediate and One Low Population Port	0.00084	0.00000038
Two Low Population Ports	0.00048	0.00000022
Direct	0.00020	0.000000089
MOTSU via:		
Two High Population Ports	0.0067	0.0000028
One High and One Intermediate Population Port	0.0053	0.0000022
One High and One Low Population Port	0.0050	0.0000021
Two Intermediate Population Ports	0.0011	0.00000049
One Intermediate and One Low Population Port	0.00074	0.00000034
Two Low Population Ports	0.00038	0.00000018
Direct	0.000095	0.000000045
Galveston via:		
Two High Population Ports	0.0073	0.0000031
One High and One Intermediate Population Port	0.0059	0.0000025
One High and One Low Population Port	0.0055	0.0000023
Two Intermediate Population Ports	0.0017	0.00000074
One Intermediate and One Low Population Port	0.0013	0.00000058
Two Low Population Ports	0.00094	0.00000043
Direct	0.00066	0.00000029
Jacksonville via:		
Two High Population Ports	0.0069	0.0000029

Port	Exposure (person-rem)	Risk (LCF)
One High and One Intermediate Population Port	0.0055	0.0000023
One High and One Low Population Port	0.0052	0.00000022
Two Intermediate Population Ports	0.0013	0.00000057
One Intermediate and One Low Population Port	0.00093	0.00000042
Two Low Population Ports	0.00056	0.00000026
Direct	0.00028	0.00000013
Savannah via:		
Two High Population Ports	0.0068	0.0000029
One High and One Intermediate Population Port	0.0055	0.0000023
One High and One Low Population Port	0.0051	0.0000021
Two Intermediate Population Ports	0.0012	0.00000054
One Intermediate and One Low Population Port	0.00085	0.00000039
Two Low Population Ports	0.00049	0.00000023
Direct	0.00021	0.000000095
Wilmington via:		
Two High Population Ports	0.0068	0.0000029
One High and One Intermediate Population Port	0.0054	0.0000023
One High and One Low Population Port	0.0050	0.0000021
Two Intermediate Population Ports	0.0012	0.00000052
One Intermediate and One Low Population Port	0.00081	0.00000037
Two Low Population Ports	0.00045	0.00000021
Direct	0.00016	0.000000074
Tacoma via:		
Two High Population Ports	0.0068	0.0000029
One High and One Intermediate Population Port	0.0045	0.0000019
One High and One Low Population Port	0.0039	0.0000017
Two Intermediate Population Ports	0.0023	0.00000095
One Intermediate and One Low Population Port	0.0017	0.00000070
Two Low Population Ports	0.00010	0.00000045
Direct	0.00072	0.00000032
Concord NWS via:		
Two High Population Ports	0.0073	0.0000031
One High and One Intermediate Population Port	0.0051	0.0000021
One High and One Low Population Port	0.0044	0.0000019
Two Intermediate Population Ports	0.0028	0.0000012
One Intermediate and One Low Population Port	0.0022	0.00000091
Two Low Population Ports	0.0015	0.00000066
Direct	0.0012	0.00000053

Acceptance of Foreign Research Reactor Spent Nuclear Fuel for 5 Years Only: Under this implementation alternative, 586 transportation casks of foreign research reactor spent nuclear fuel would be shipped to the United States. All of these shipments would be shipped by ocean vessel and would enter the United States through ports.

In addition to a reduced number of shipments associated with this implementation alternative, the mix of fuel types changes slightly. From the information provided in Appendix B, 376 of the 586 shipments in this alternative are BR-2 spent fuel shipments, 56 are RHF, and 154 are TRIGA.

The risks of the basic implementation of Management Alternative 1, provided in Table D-46, have been recalculated to incorporate the change in the number and makeup of the shipments associated with this implementation alternative. These results are presented in Table D-48. The highest calculated port accident risks are associated with the shipment of all of the foreign research reactor spent nuclear fuel

through the port of Elizabeth via two high population intermediate ports. The port accident risks for the implementation alternative for this route are 0.055 person-rem and 0.000023 LCF. The lowest calculated impacts are for the shipment of all of the material directly into MOTSU (no intermediate port calls) which results in port accident risks of 0.00055 person-rem and 0.00000026 LCF.

Table D-48 Summary of Risk and Population Exposure—For the Implementation Alternative of a 5-Year Acceptance Duration

Port	*Exposure (person-rem)*	*Risk (LCF)*
Elizabeth via:		
Two High Population Ports	0.055	0.000023
One High and One Intermediate Population Port	0.047	0.000020
One High and One Low Population Port	0.045	0.000019
Two Intermediate Population Ports	0.023	0.000010
One Intermediate and One Low Population Port	0.021	0.0000091
Two Low Population Ports	0.019	0.0000082
Direct	0.018	0.0000074
Long Beach via:		
Two High Population Ports	0.046	0.000019
One High and One Intermediate Population Port	0.033	0.000014
One High and One Low Population Port	0.030	0.000013
Two Intermediate Population Ports	0.021	0.0000089
One Intermediate and One Low Population Port	0.017	0.0000073
Two Low Population Ports	0.013	0.0000057
Direct	0.012	0.0000049
Philadelphia via:		
Two High Population Ports	0.045	0.000019
One High and One Intermediate Population Port	0.037	0.000016
One High and One Low Population Port	0.035	0.000015
Two Intermediate Population Ports	0.013	0.0000057
One Intermediate and One Low Population Port	0.011	0.0000048
Two Low Population Ports	0.0089	0.0000039
Direct	0.0073	0.0000031
Portland via:		
Two High Population Ports	0.038	0.000016
One High and One Intermediate Population Port	0.025	0.000011
One High and One Low Population Port	0.021	0.0000090
Two Intermediate Population Ports	0.012	0.0000052
One Intermediate and One Low Population Port	0.0084	0.0000037
Two Low Population Ports	0.0047	0.0000021
Direct	0.0031	0.0000013
Norfolk via:		
Two High Population Ports	0.040	0.000017
One High and One Intermediate Population Port	0.032	0.000013
One High and One Low Population Port	0.030	0.000013
Two Intermediate Population Ports	0.0078	0.0000035
One Intermediate and One Low Population Port	0.0057	0.0000026
Two Low Population Ports	0.0036	0.0000017
Direct	0.0020	0.00000089
Charleston (Wando Terminal) via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0069	0.0000031

Port	Exposure (person-rem)	Risk (LCF)
One Intermediate and One Low Population Port	0.0048	0.0000022
Two Low Population Ports	0.0028	0.0000013
Direct	0.0011	0.00000051
Charleston NWS via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0066	0.0000031
One Intermediate and One Low Population Port	0.0048	0.0000022
Two Low Population Ports	0.0025	0.0000013
Direct	0.00087	0.00000054
MOTSU via:		
Two High Population Ports	0.038	0.000016
One High and One Intermediate Population Port	0.030	0.000013
One High and One Low Population Port	0.028	0.000012
Two Intermediate Population Ports	0.0063	0.0000028
One Intermediate and One Low Population Port	0.0042	0.0000019
Two Low Population Ports	0.0022	0.0000010
Direct	0.00055	0.00000028
Galveston via:		
Two High Population Ports	0.041	0.000018
One High and One Intermediate Population Port	0.033	0.000014
One High and One Low Population Port	0.031	0.000013
Two Intermediate Population Ports	0.0095	0.0000042
One Intermediate and One Low Population Port	0.0074	0.0000033
Two Low Population Ports	0.0054	0.0000024
Direct	0.0037	0.0000017
Jacksonville via:		
Two High Population Ports	0.039	0.000017
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0073	0.0000033
One Intermediate and One Low Population Port	0.0053	0.0000024
Two Low Population Ports	0.0032	0.0000015
Direct	0.0016	0.00000072
Savannah via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0069	0.0000031
One Intermediate and One Low Population Port	0.0049	0.0000022
Two Low Population Ports	0.0028	0.0000013
Direct	0.0012	0.00000055
Wilmington via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0067	0.0000030
One Intermediate and One Low Population Port	0.0046	0.0000021
Two Low Population Ports	0.0026	0.0000012
Direct	0.00093	0.00000042

Port	Exposure (person-rem)	Risk (LCF)
Tacoma via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.026	0.0000011
One High and One Low Population Port	0.022	0.0000094
Two Intermediate Population Ports	0.013	0.0000057
One Intermediate and One Low Population Port	0.0094	0.0000041
Two Low Population Ports	0.0057	0.0000026
Direct	0.0041	0.0000018
Concord NWS via:		
Two High Population Ports	0.041	0.000017
One High and One Intermediate Population Port	0.029	0.000012
One High and One Low Population Port	0.025	0.000011
Two Intermediate Population Ports	0.016	0.0000070
One Intermediate and One Low Population Port	0.012	0.0000054
Two Low Population Ports	0.0086	0.0000037
Direct	0.0070	0.0000030

D.5.7 Port Accident Impacts Associated with Management Alternative 2

Of the two subalternatives under Management Alternative 2, only subalternative 1b requires assessment of the impacts of accidents in port. This subalternative involves overseas reprocessing of foreign research reactor spent nuclear fuel. Under this subalternative, which is explained in detail in Chapter 2, up to eight transportation casks of vitrified high-level waste might pass through U.S. ports on their way to storage sites in the United States. The port accident impacts associated with this subalternative are evaluated below.

Foreign Reprocessing with Shipment of Vitrified Waste to a U.S. Storage Facility: In this subalternative to Management Alternative 2, all of the foreign research reactor spent nuclear fuel (including that generated in Canada) would be sent to either Great Britain or France for reprocessing and part or all of the vitrified high-level waste generated in the process could be shipped to the United States. Based on the reprocessing of approximately 23 metric tons of spent nuclear fuel (all of the fuel considered by the basic implementation of Management Alternative 1), enough vitrified high-level waste would be generated to require the transportation of up to eight transportation casks carrying logs of vitrified high-level waste to the United States.

The consequences of an accident in port involving a cask of vitrified high-level waste could not be derived from the analysis of the port accidents for the foreign research reactor spent nuclear fuel. Two significant differences in the contents of the cask carrying vitrified high-level waste and the casks carrying foreign research reactor spent nuclear fuel dictate that revised source terms be calculated for the vitrified high-level waste case. The release fractions associated with the accident severity categories are different for the vitrified high-level waste than they are for the foreign research reactor spent nuclear fuel. Based on previous DOE efforts (DOE, 1994b) the release fractions for vitrified high-level waste are the same for all three release categories (categories 4, 5, and 6). Vitrified waste release fractions are relatively insensitive to the affects of the fires that differentiate the category 5 and 6 accidents from the category 4 accidents. The release fractions used in this analysis are a factor of 0.05 higher than those used in the referenced analysis because the use of the MACCS code eliminates the need to describe a respirable fraction of the release. In the referenced analysis, the release fraction was determined and then modified by the respirable fraction (0.05) to use the value of 0.00000005 (5.0E-08) used in that analysis. Without the respirable fraction modification the release fraction is 0.000001 (1.0E-06). This is the release fraction used in the analysis of the vitrified high-level waste shipment port accident analysis.

These release fractions apply to all material in the vitrified high-level waste. Each isotope contained in the glassified waste has been assigned the same release fraction.

All of the wastes generated in reprocessing the foreign research reactor spent nuclear fuel would be transported in no more than eight casks, compared to the 837 marine and overland shipments of spent nuclear fuel required under the basic implementation of Management Alternative 1. This means that the curie content of the vitrified high-level waste could be approximately 100 times the content of a single transportation cask of foreign research reactor spent nuclear fuel.

In this analysis no credit has been taken for the reduction in the curie content of the vitrified high-level waste due to the natural decay that would result during the temporary storage of the vitrified high-level waste at the reprocessing facility. One of the options considered for this subalternative includes the storage of this material at the reprocessing facility until a permanent U.S. facility is ready to receive it for storage. Even if the material is not held until a permanent facility is available, some temporary storage at the reprocessing facility would probably be necessary. In either case, the reduction in the curie content of the waste logs has conservatively not been incorporated into this analysis. The risks associated with the shipment of aged vitrified high-level waste would be less, proportional to the reduction in the curie content, than the risks associated with the shipment of recently reprocessed material of the same volume. Therefore, while the risks calculated in this analysis are more appropriate for the shipment of recently reprocessed waste, the analysis bounds the risks associated with both options.

The isotopic content of the material shipped in one transportation cask of vitrified high-level waste is presented in Table D-49. This estimate was developed by combining the isotopic inventory of every assembly being shipped in the basic implementation of Management Alternative 1 and equally dividing these inventories into eight shipments. This inventory of material was developed from an earlier estimate of the number of foreign research reactor spent nuclear fuel shipments than that analyzed as the basic implementation of Management Alternative 1. The isotopic content of the earlier estimate of the number of shipments is slightly higher than results from the shipments in the basic implementation of Management Alternative 1, for every isotope found in the vitrified high-level waste. Therefore, the estimate used to generate the data in Table D-49 is slightly conservative.

The consequence analysis was performed using the MACCS code utilizing the inventory and release fraction data presented above. Since it is anticipated that the vitrified high-level waste would be stored, temporarily, at the Savannah River Site and the shipments are originating in Europe, only selected East Coast sites were analyzed. Port accident risks were analyzed for the ports of Philadelphia, Charleston, and MOTSU. Also, it has been assumed that the vitrified high-level waste shipments would be made on vessels that would not make intermediate port calls, i.e., on a chartered vessel. The results of these consequences analyses are presented in Table D-50. The highest mean value for an exposure to the MEI is 740 mrem for a 50-year dose to that individual. This corresponds to a LCF consequence of 0.00035.

The probability of an accident in port has been modeled using the data generated for the analysis of the basic implementation of Management Alternative 1. Although the use of a chartered (especially a purpose-built) ship could result in somewhat lower accident frequencies for each of the severity categories, these differences were judged to be minor and were not incorporated into the analysis. The port accident risks associated with the shipment of a single cask and of all eight casks containing the entire inventory of vitrified high-level waste generated in the reprocessing of all of the foreign research reactor spent nuclear fuel considered in the basic implementation of Management Alternative 1 are presented in Table D-51.

Table D-49 Radionuclide Inventory for Each of Eight Vitrified High-Level Waste Shipments

Radionuclide	*Vitrified High-Level Waste Inventory (Ci)*	*Radionuclide*	*Vitrified High-Level Waste Inventory (Ci)*
Hydrogen-3	7,302	Cerium-141	559,300
Krypton-85	207,000	Cerium-144 Promethium-144	24,890,000
Strontium-89	3,072,000	Promethium-147	3,703,000
Strontium-90 Yttrium-90	1,743,000	Promethium-148m	7,133
Yttrium-91	5,477,000	Europium-154	62,390
Zirconium-95	8,079,000	Europium-155	12,900
Niobium-95	16,540,000	Plutonium-238	8,484
Ruthenium-103 Rh-103m	716,000	Plutonium-239	405
Ruthenium-106 Rh-106m	1,882,000	Plutonium-240	326
Tin-123	33,340	Plutonium-241	78,440
Antimony-125	75,700	Americium-241	98
Tellurium-125m	18,060	Americium-242m	0.67
Tellurium-127M	69,720	Americium-243	1.4
Tellurium-129M	15,870	Curium-244	122
Cesium-134	1,413,000	Curium-242	990
Cesium-137	1,743,000		

Table D-50 Port Accident Consequences for Vitrified High-Level Waste

Location	*Mean Accident Consequences*		*99th Percentile Consequences*	
	Population Exposure (person-rem)	*LCF*	*Population Exposure (person-rem)*	*LCF*
MOTSU at the Dock	93.1	0.04	572	0.25
MOTSU in the Channel	66.1	0.029	332	0.13
Charleston at the Dock	202	0.088	747	0.32
Charleston in the Channel	293	0.13	2450	1.02
Philadelphia at the Dock	1250	0.54	5110	2.12
Philadelphia in the Channel	733	0.32	2990	1.21

The port accident risks associated with the implementation of this subalternative to Management Alternative 2 results in a negligible risk to the public. The highest mean port accident risk results in a less than one-in-ten thousand chance of a single LCF.

D.5.8 Port Accident Impacts Associated with a Combination of Returning Foreign Research Reactor Spent Nuclear Fuel and Overseas Management

In addition to evaluating the port accident impacts for the various alternatives associated with bringing all of the foreign research reactor spent nuclear fuel to the United States (Management Alternative 1) and managing all of the spent nuclear fuel overseas (Management Alternative 2), a hybrid scenario was analyzed. In this scenario, those countries that have the capability to store high-level waste would be encouraged to process aluminum-based foreign research reactor spent nuclear fuel and to accept the resulting high-level waste. For this scenario, those countries are assumed to be Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom. The United States would accept the foreign

Table D-51 Port Accident Risks for the Acceptance of Vitrified High-Level Waste

Port	Risk per Shipment of One Cask of Waste		Risk of the Entire Waste Acceptance Option	
	Population Dose (person-rem)	LCF	Population Dose (person-rem)	LCF
Philadelphia	0.006	0.000003	0.05	0.00002
Charleston	0.001	0.0000007	0.01	0.000005
MOTSU	0.0005	0.0000002	0.004	0.000002

research reactor spent nuclear fuel from those countries deemed not to have the high-level waste storage capability. In this option, this includes all of the countries identified in Table C-1, except for those listed above. Under the hybrid scenario, 452 shipments of spent nuclear fuel are assumed to be sent to the United States through U.S. ports, excluding shipments of Canadian origin, which are assumed to be transported overland. Of these, 290 are of the BR-2 fuel type and 162 are of the TRIGA type.

In analyzing the exposure and risk associated with this scenario, much of the information that was developed for Management Alternative 1 can be used. Both the per-transit probability of an accident and the conditional probabilities of severity category 4, 5, and 6 accidents are valid for this hybrid scenario. The consequences associated with each of the three accident severity categories also do not change, because the only thing that is changing is the number of shipments. Since neither the probability nor the consequences of the accidents change, the per-shipment risks are identical to those of the basic implementation of Management Alternative 1.

The risks associated with the basic implementation of Management Alternative 1 (Table D-46) have been recalculated to incorporate the change in the number and makeup of the shipments associated with the hybrid scenario. These results are presented in Table D-52. The highest calculated port accident risks are associated with the shipment of all of the foreign research reactor spent nuclear fuel through the port of Elizabeth via two high population intermediate ports. The port accident risks for the Management Alternative for this route are 0.041 person-rem and 0.000017 LCF. The lowest calculated impacts are for the shipment of all the material directly into MOTSU (no intermediate port calls), which results in port accident risk of 0.0004 person-rem and 1.9×10^{-7} LCF.

Table D-52 Summary of Risk and Population Exposure—For the Hybrid Scenario

Port	Exposure (person-rem)	Risk (LCF)
Elizabeth via:		
Two High Population Ports	0.041	1.7×10^{-5}
One High and One Intermediate Population Port	0.035	1.5×10^{-5}
One High and One Low Population Port	0.034	1.4×10^{-5}
Two Intermediate Population Ports	0.017	7.5×10^{-6}
One Intermediate and One Low Population Port	0.016	6.8×10^{-6}
Two Low Population Ports	0.014	6.1×10^{-6}
Direct	0.013	5.5×10^{-6}
Long Beach via:		
Two High Population Ports	0.034	1.5×10^{-5}
One High and One Intermediate Population Port	0.025	1.1×10^{-5}
One High and One Low Population Port	0.022	9.4 x 10-6
Two Intermediate Population Ports	0.015	6.6 x 10-6
One Intermediate and One Low Population Port	0.013	5.4 x 10-6
Two Low Population Ports	0.0099	4.3×10^{-6}
Direct	0.0087	3.7×10^{-6}
Philadelphia via:		
Two High Population Ports	0.033	1.4×10^{-5}

Port	Exposure (person-rem)	Risk (LCF)
Two High Population Ports	0.033	1.4×10^{-5}
One High and One Intermediate Population Port	0.028	1.2×10^{-5}
One High and One Low Population Port	0.026	1.1×10^{-5}
Two Intermediate Population Ports	0.0097	4.2×10^{-6}
One Intermediate and One Low Population Port	0.0082	3.6×10^{-6}
Two Low Population Ports	0.0066	2.9×10^{-6}
Direct	0.0054	2.3×10^{-6}
Portland via:		
Two High Population Ports	0.028	1.2×10^{-5}
One High and One Intermediate Population Port	0.018	7.8×10^{-6}
One High and One Low Population Port	0.016	6.7×10^{-6}
Two Intermediate Population Ports	0.0090	3.9×10^{-6}
One Intermediate and One Low Population Port	0.0063	2.7×10^{-6}
Two Low Population Ports	0.0035	1.6×10^{-6}
Direct	0.0023	9.8×10^{-7}
Norfolk via:		
Two High Population Ports	0.030	1.2×10^{-5}
One High and One Intermediate Population Port	0.024	1.0×10^{-5}
One High and One Low Population Port	0.022	9.3×10^{-6}
Two Intermediate Population Ports	0.0058	2.6×10^{-6}
One Intermediate and One Low Population Port	0.0043	1.9×10^{-6}
Two Low Population Ports	0.0027	1.2×10^{-6}
Direct	0.0015	6.7×10^{-7}
Charleston (Wando Terminal) via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.7×10^{-6}
One High and One Low Population Port	0.021	9.0×10^{-6}
Two Intermediate Population Ports	0.0051	2.3×10^{-6}
One Intermediate and One Low Population Port	0.0036	1.6×10^{-6}
Two Low Population Ports	0.0021	9.5×10^{-7}
Direct	0.00083	3.8×10^{-7}
Charleston NWS via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.7×10^{-6}
One High and One Low Population Port	0.021	9.1×10^{-6}
Two Intermediate Population Ports	0.0049	2.3×10^{-6}
One Intermediate and One Low Population Port	0.0034	1.6×10^{-6}
Two Low Population Ports	0.0019	9.8×10^{-7}
Direct	0.00041	4.0×10^{-7}
MOTSU via:		
Two High Population Ports	0.028	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.5×10^{-6}
One High and One Low Population Port	0.0021	8.8×10^{-6}
Two Intermediate Population Ports	0.0047	2.1×10^{-6}
One Intermediate and One Low Population Port	0.0032	1.4×10^{-6}
Two Low Population Ports	0.0016	7.6×10^{-7}
Direct	0.00041	1.9×10^{-7}
Galveston via:		
Two High Population Ports	0.031	1.3×10^{-5}
One High and One Intermediate Population Port	0.025	1.1×10^{-5}
One High and One Low Population Port	0.023	9.9×10^{-6}
Two Intermediate Population Ports	0.0071	3.2×10^{-6}

Port	Exposure (person-rem)	Risk (LCF)
One Intermediate and One Low Population Port	0.0056	2.5×10^{-6}
Two Low Population Ports	0.0040	1.8×10^{-6}
Direct	0.0028	1.2×10^{-6}
Jacksonville via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.9×10^{-6}
One High and One Low Population Port	0.022	9.2×10^{-6}
Two Intermediate Population Ports	0.0055	2.4×10^{-6}
One Intermediate and One Low Population Port	0.0039	1.8×10^{-6}
Two Low Population Ports	0.0024	1.1×10^{-6}
Direct	0.0012	5.3×10^{-7}
Savannah via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.7×10^{-6}
One High and One Low Population Port	0.022	9.1×10^{-6}
Two Intermediate Population Ports	0.0052	2.3×10^{-6}
One Intermediate and One Low Population Port	0.0036	1.7×10^{-6}
Two Low Population Ports	0.0021	9.8×10^{-7}
Direct	0.00088	4.1×10^{-7}
Wilmington via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.6×10^{-6}
One High and One Low Population Port	0.021	9.0×10^{-6}
Two Intermediate Population Ports	0.0050	2.2×10^{-6}
One Intermediate and One Low Population Port	0.0035	1.6×10^{-6}
Two Low Population Ports	0.0019	8.9×10^{-7}
Direct	0.00069	3.2×10^{-7}
Tacoma via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.019	8.2×10^{-6}
One High and One Low Population Port	0.016	7.0×10^{-6}
Two Intermediate Population Ports	0.0098	4.2×10^{-6}
One Intermediate and One Low Population Port	0.0070	3.1×10^{-6}
Two Low Population Ports	0.0043	1.9×10^{-6}
Direct	0.0030	1.3×10^{-6}
Concord NWS via:		
Two High Population Ports	0.031	1.3×10^{-5}
One High and One Intermediate Population Port	0.021	9.1×10^{-6}
One High and One Low Population Port	0.019	7.9×10^{-6}
Two Intermediate Population Ports	0.012	5.1×10^{-6}
One Intermediate and One Low Population Port	0.0092	4.0×10^{-6}
Two Low Population Ports	0.0064	2.8×10^{-6}
Direct	0.0052	2.2×10^{-6}

D.5.9 Consequences of Sabotage or Terrorist Attack

This section provides an evaluation of impacts that could potentially result from a malicious act on a shipment of foreign research reactor spent nuclear fuel. In no instance, even in severe cases such as those discussed below, could a nuclear explosion or permanent contamination of the environment leading to condemnation of land occur. Furthermore, DOE considers that, due to the security measures that would be in place for any spent nuclear fuel shipments, such attacks would be unlikely to occur. At a minimum, the extent or effects of any such attacks, would be mitigated by the security measures.

Since it is impossible to determine with certainty the probability of a deliberate act of sabotage or terrorist attack, this section presents an analysis of potential consequences of sabotage or terrorist attack on a spent nuclear fuel shipping cask, and does not attempt to estimate the risk of such an activity. Although judged very unlikely to actually occur, a malicious attack on a foreign research reactor spent nuclear fuel shipping cask has been postulated to occur at a U.S. port or during transportation from the port to the management site, for purposes of illustrating the effects that might result from such an event.

The spectrum of attacks that can be postulated is broad, falling into three categories or scenarios: (1) exploding a bomb near a shipping cask, (2) attacking a cask with a shaped charge, or an armor-piercing weapon (i.e., an anti-tank weapon), and (3) hijacking (stealing) a shipping cask. None of the scenarios considered would lead to a criticality accident.

D.5.9.1 Exploding a Bomb Near a Shipping Cask

This sabotage/terrorist attack scenario assumes that a large bomb, similar to that detonated in Oklahoma City in April of 1995, is detonated in the immediate vicinity of a spent nuclear fuel shipping cask. The primary threats to the cask integrity would arise from: (1) direct blast forces (shock wave) from the bomb, (2) impact forces from fragments (e.g., motor vehicle parts) generated by the bomb, and (3) other dynamic forces such as a roll-over of the cask transport vehicle in response to the blast forces. The casks are rugged structures that would be expected to survive the effects of a nearby bomb explosion with no significant loss of integrity. At worst, the blast might produce a crack in the wall of the cask. In any case, all spent nuclear fuel elements would remain inside the cask. Blast-related damage might, however, reduce the effectiveness of cask shielding and/or cause locally higher dose rates outside the cask (e.g., from damaged shielding areas and radiation streaming through a crack in the cask wall).

Although no mechanism has been postulated that could cause such an event, an analysis of a total loss of cask shielding has been performed for the purposes of demonstrating limiting case effects of an attack on a spent nuclear fuel shipping cask, such as that discussed above. The analysis scenario assumes that the cask was full of a highly irradiated foreign research reactor spent nuclear fuel, and that the spent nuclear fuel elements were spread on the ground producing the highest possible direct dose rate. For the calculation of direct dose, no credit was taken for self-shielding of the spent fuel, and it was assumed that no other obstacle would exist between the spent nuclear fuel and individual members of the public. Since the spent nuclear fuel would be a solid metal structure, this analysis assumes that no spent nuclear fuel damage occurs, therefore, no radioactive materials would be dispersed. The results of this unrealistically conservative analysis are shown in Figure D-60. This figure provides a conservative estimate of the direct dose rate (rem per hour) to an individual member of the public versus distance from a spent nuclear fuel pile consisting of 30 highly irradiated fuel elements. Based on the results of this hypothetical, conservative analysis, an evacuation distance of about 900 meters (3000 ft) would be sufficient to maintain a dose rate of less than 10 mrem per hour, (or 0.01 rem per hour). This is a very conservative evacuation distance, but it would provide a good measure for consideration by an emergency response team. This scenario would result in minimal or no contamination of the area where it occurred and once the spent nuclear fuel was shielded, the evacuation zone would be greatly reduced. Once the spent nuclear fuel was removed from the site, the area would be decontaminated, if necessary, before it returned to normal.

D.5.9.2 Attacking a Cask with a Shaped Charge or Armor-Piercing Weapon

If a cask were attacked by an armor-piercing weapon or a shaped charge, the cask would be penetrated and spent nuclear fuel elements inside the cask could be damaged. An analysis of a hypothetical attack on a spent nuclear fuel shipping cask using a shaped charge was performed using the MACCS code. The

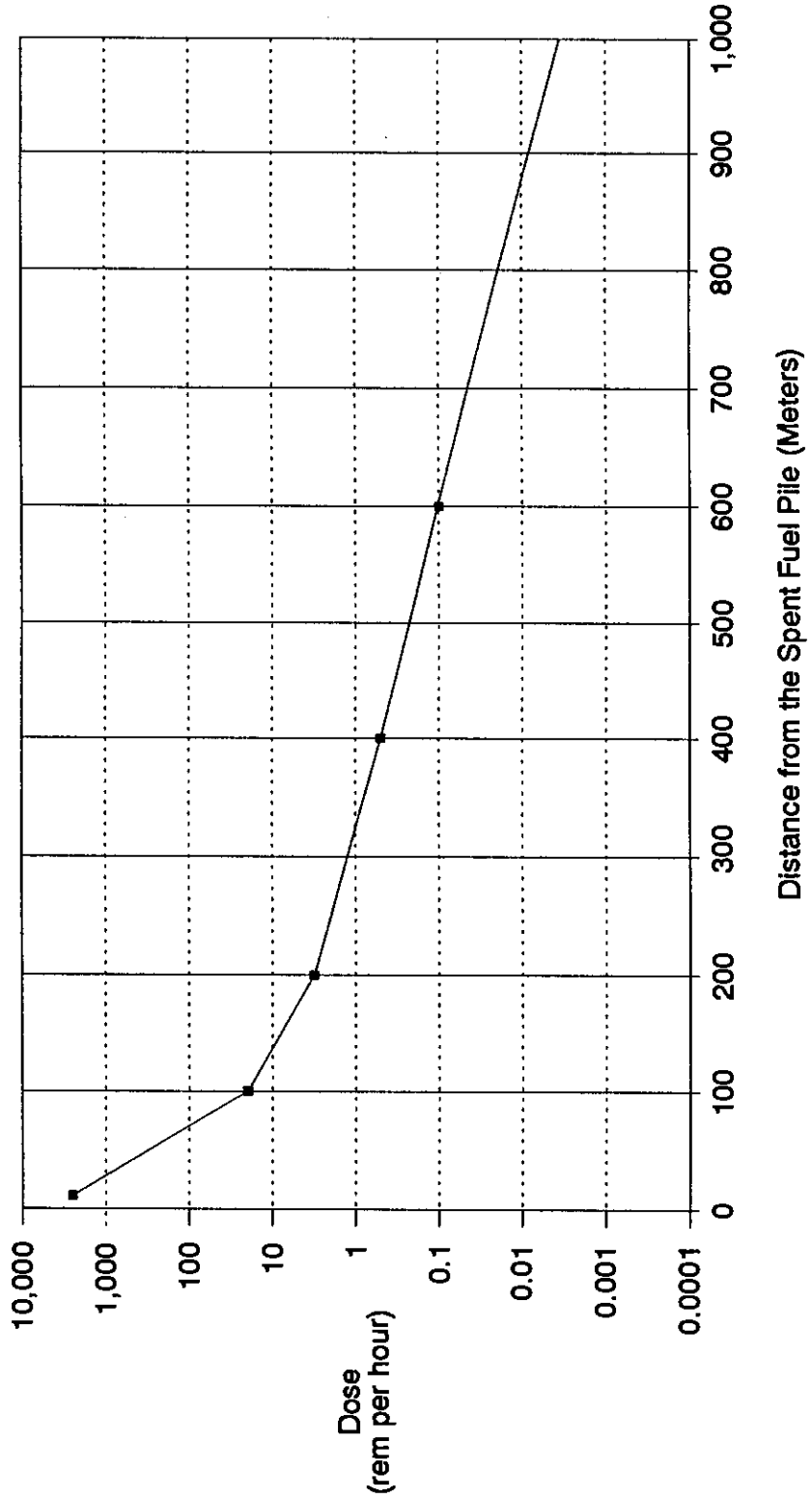

Figure D-60 Direct Dose vs Distance to an Individual Member of the Public

accident was assumed to occur on a city street in a highly populated area near the harbor where the spent nuclear fuel cask was transferred to a truck after trans oceanic shipment from overseas. The analysis assumed that the cask contained the highest radionuclide inventory, and the blast released all of the noble gases and one percent of the bulk of the spent nuclear fuel as airborne aerosols. The one percent of bulk spent nuclear fuel release assumption was based on measurements of aerosols released during tests where spent nuclear fuel was explosively disrupted. These tests yielded spent nuclear fuel release mass fractions that ranged from 0.05 to 2.5 percent (Sanders, et al., 1992). The blast energy would be quickly dissipated and the released fission products and gases and aerosols were assumed to be relatively cool; thus no plume rise was assumed to occur. These assumptions are very conservative and the results provide an enveloping estimate of consequences on the environmental and health effects. The MACCS calculations estimated a population dose of 208,000 person-rem with no acute fatalities or short-term adverse health effects among the exposed population. The MACCS results estimated that 91 latent cancer fatalities could occur among the 16 million persons living within 80 kilometers (50 miles) of the attack. The average individual lifetime radiation dose among the one to two million people who would be exposed is estimated to be about 200 mrem. This is less than one percent of a person's lifetime natural background radiation dose. This evaluation did not consider any evacuation and/or sheltering activities after the attack. MACCS also estimated a contamination distance of about 1 kilometer (0.6 miles) down wind from the attack. This distance, though conservative, could be used by an emergency response team for evacuation purposes. Of course, any actual evacuation distance would be determined on a case-by-case basis, if such an event were ever to occur. Mitigation activities in the aftermath of such an explosion, as required by law (EPA), would reduce the size of the contaminated area drastically and the area could become rehabitable in a short period of time. It is important to bear in mind that the explosion itself would be likely to produce fatalities, injuries and property damage that far exceed that caused by any release of radioactive material from the spent nuclear fuel.

In a terrorist attack using an anti-tank weapon, any cask damage and resulting consequences would be less severe than the accidents analyzed elsewhere in the EIS. This is because (1) there would be no explosive material inside the cask so the cask would not explode. Therefore, no additional radioactivity, other than that released directly by the projectile, would be forced out of the cask, and (2) there would be no fire to disperse the radioactivity that would be released when the cask was breached. At worst, the consequences of a terrorist attack on a spent nuclear fuel shipping cask with an anti-tank weapon would be similar to that analyzed above for a hypothetical terrorist attack on a cask with a high explosive shaped charge.

D.5.9.3 Hijacking a Shipping Cask

The discret theft of a spent nuclear fuel transportation cask is considered to be very unlikely, due to security measures that would be in place during transportation activities, especially the guarding of the cask, and communication and tracking systems (see Section 2.8 and Appendix H). In addition, the large size and weight of these casks (20 to 30 metric tons) and the inherent radioactivity of the spent nuclear fuel (which could kill a person upon contact) would deter most would-be hijackers. In the event of a hijack attempt, required communications systems would ensure timely notification of authorities who would mobilize response forces. The installed tracking system would allow the location of the cask to be determined in real time, thereby aiding timely interception of hijackers by response forces.

No release of radioactive material or increase in radiation level would be expected during a hijack scenario unless the hijacker could blow up the cask using explosive material (e.g., a shaped charge), or open the cask. In case of a cask explosion using a shaped charge, the consequences would be the same as, or smaller than (depending on the location of the accident), the case described in Section D.5.9.2. If the cask were opened (a lengthy process requiring special tooling), shielding would be decreased and the radiation

level in the immediate vicinity of the cask would increase. The cask opening could only be accomplished at great personal risk to hijackers due to large (possibly immediately lethal) radiation exposures that they would receive while handling the unshielded fuel elements.

Should such an attempt be made, the hijackers would not be able to alter the fuel configuration inside the cask to make it critical. Criticality analyses that have been performed in support of the cask certification process consider various fuel and moderation configurations. These analyses are performed to ensure that none of potential configurations that could occur during loading and transport of the cask would lead to a criticality condition. Changing moderating material to achieve criticality, would require special materials that are not readily available (safeguard materials). Based on the time available to the hijackers, and tooling and materials that are needed, DOE considers that the potential for achieving criticality in a hijacked spent nuclear fuel cask is beyond credibility. If the hijackers were to dump the unshielded spent nuclear fuel, the resulting consequences to the public from the bare spent nuclear fuel radiation exposure would be less severe than those already analyzed for other hypothetical scenarios in this appendix.

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Attachment D1
Capital Improvement Plans and Other Significant Port Developments Anticipated During the Period of the Proposed Action

Port	*Improvements and Other Significant Developments*
Baltimore, MD	The Board of Public Works approved the purchase of a new $7.4 million container crane to be installed at the Dundalk Marine Terminal, a 231 ha (570 acre) terminal complex with 13 deepwater berths and 9 container cranes. The new crane is expected to be operational by early 1995 (Governors Press Office, State of Maryland, May 18, 1994). Governor William Schaefer announced Board of Public Works approval of a contract to modernize and improve (up-grade to post-Panamax capacity) three container cranes located at the Dundalk Marine Terminal (Ibid, June 22, 1994).
Boston, MA	Officials of the Massachusetts Port Authority (Massport) have submitted a draft environmental impact report to Federal and State officials that calls for dredging the harbor and access channels to 12.2 m (40 ft) from 9.75-10.97 m (32-36 ft) (*American Shipper*, "Boston Seeks Direct Calls From Asia," October 1994, Pg. 94). Ralph Cox, Marine Director, and other port officials claim that the deeper water is critical to the Port's viability. Massport is also seeking support of the State Legislature for road and rail clearances to permit double-stack train service to the City of Boston and its marine terminals. A $50 million expansion and modernization of Boston's Conley Terminal is approximately 80 percent complete. When completed, Conley Terminal will have 40.5 ha (100 acres) of container storage and handling area, 4 post-Panamax container cranes, 304.8 m (1,000 ft) of berth, and a new gate complex. Reportedly, container tonnage is up for 1994 over 1993 tonnage when Boston handled 152,240 twenty-foot equivalent units for the year.
Charleston, SC	As of late 1993, the $90 million Wando terminal expansion project was nearing completion. When completed, the project will add an additional 418.5 m (1,373 ft) of berthing space, 26.3 ha (65 acres) of container storage area, and two $5.4 million post-Panamax container cranes. The entire project is scheduled for completion by Fall 1994. Planning is progressing for development of the approximately 323.8 ha (800-plus acre) Daniel Island terminal site in Charleston Harbor. The container terminal is being designed to meet demand at the port well into the 21st century. The massive project is expected to take 15 to 20 years to complete and will ultimately consist of 323.8 ha (800 acres) of paved container storage and 2,438 m (8,000 lineal ft) of berthing space (*American Shipper* - Southern Ports, January 1994). At NWS Charleston, the U.S. Army is planning to expand Wharf Alpha and upgrade the railroad in support of the Army Strategic Mobility Logistics Base. This upgrade is scheduled for completion in 1998.

Concord NWS, CA	Currently authorized improvements expected to be completed by 1997 include an upgrade of Pier 3 to withstand greater loadings, and will also include two new 36 metric ton (40 ton) container cranes, container storage pads, and support facilities and equipment. These improvements are projected to permit average load rates of about 20 containers per hour. The improvements will also permit increasing the channel depth and depth alongside to 12.7 m (42 ft) in the future if necessary. The facility will be the designated West Coast container facility for military shipments (personal communications from Karl Yocum, Concord NWS Office of Business Development, September 1, 1994, and 1994 Fact Sheets received during port visit).
Eddystone, PA	No immediate improvements identified.
Fernandina Beach, FL	U.S. Army Corps of Engineers is expected to award a contract in October 1994 for deepening of the harbor channel to 11 m (36 ft), and to construct a 366 m (1,200 ft) turning basin (Personal communication from Mr. Stubbs, Port of Fernandina, September, 1994).
Gulfport, MS	A consulting firm has recommended a 15-year, $160 million terminal expansion for the Port of Gulfport to a projected tripling of port business by 2010. The proposed expansion would add 34 ha (84 acres) of land to West Pier at a cost of $81 million, replace or reconfigure existing warehouses ($13 million), and include the purchase of $11 million in additional container handling equipment (but not necessarily a container gantry crane) (*American Shipper*, September 1994, p. 102; *Containerization International*, September 1994, p. 11). A $41 million dredging project that deepened the harbor from 9.14 m to 10.97 m (30-36 ft) was completed in April 1994. In August 1993, the Port Authority issued $15 million in bonds to pay for three development projects that included expansion of East Pier warehouse facilities and the addition of 11.7 ha (29 acres) of land—through diking and pumping sand, at West Pier. The latter to be used for a new container terminal (*American Shipper* - Southern Ports, January 1994).
Jacksonville, FL	In anticipation of continued strong growth in cargo demand over the next 20 years, JAXPORT adopted a 20-year, $934 million development plan designed to prepare its facilities for 2010. In addition to recommendations for immediate construction of a third container terminal at Dames Point, consultants recommended expansion and reconfiguration of the Authority's Blount Island and Talleyrand Terminals, projected to cost about $274 million over 7 years. A $1 million feasibility cost sharing agreement was signed this year with the Corps of Engineers to develop a dredging study to deepen the harbor from 11.6 m (38 ft) to 12.8 m (42 ft). Design of a second roll-on/roll-off dock plus 2 ha (5 acres) more container storage and a 450 m (1,500 ft) extension of marginal wharf is scheduled for Blount Island in fiscal 1995 [The Jacksonville Port Authority (JAXPORT), Marketing Department, October 3, 1994].

The Port Authority anticipated breaking ground in 1994 for a new 203 ha (500 acre) container and general cargo terminal complex at Dames Point, immediately adjacent and upstream of its existing Blount Island terminal. The new facility is expected to cost $160 million when completed in the future. Other improvements scheduled include a $2.5 million investment in increased intermodal rail capacity at Blount Island and Talleyrand Terminals, and widening of Hecksher Drive to four lanes from the entrance to Blount Island to State Road 9A, which connects with I-95. I-295 is also being widened to four lanes (American Shipper - Southern Ports, January 1994).

Long Beach, CA

The Long Beach Port Commission and City Council have approved a 1994-95 budget of $417 million, which includes $236.5 million for port construction, land acquisition, and environmental mitigation. Last year's budget included $405 million to purchase land owned by Union Pacific Resources Company in the north harbor, which the Port plans to convert to a marine cargo terminal. The property is comprised of 117 ha (289 acres) north of the Cerritos Channel, 143 ha (354 acres) south of the Channel, and 33 ha (82 acres) within the Channel. The new budget provides allocations of $60 million for street overpasses to cross rail lines in the port area, $22 million for other street and road improvements, $78 million for continuing container terminal improvements at Pier J, $25 million for other construction projects, and $40 million for land acquisitions and environmental mitigation. These land acquisitions will increase the Port's operating area by 35 percent (American Shipper, September 1994, p. 94; "Long Beach to Spend $417 million").

Los Angeles, CA

Los Angeles' "2020 Program" represents the Port's comprehensive long-term development plan, which is designed to accommodate a doubling of cargo throughput through the next decade and a forecast California population of 20 million people. The major components of the 2020 Program include:

a. Construction of Pier 300 on landfill completed in 1983. When completed, Pier 300 will include the American President Lines container terminal, an intermodal container/rail/truck transfer facility and a coal export terminal;

b. Landfill and construction of Pier 400, with three container terminals, an intermodal container transfer facility, and liquid bulk terminals;

c. The Alameda Corridor, a road and rail improvement program linking the Port to rail facilities in downtown Los Angeles with a fully grade-separated trackage (Port of Los Angeles, Property Management Division, October 3, 1994).

Implementation of the 2020 Program is well underway and will involve expenditures of approximately $600 million over the next three years. Work has begun on the new 91.5 ha (226 acres) American President Lines Container Terminal on Pier 300 which, when completed in 1997, will be the largest container terminal in the United States. Costing about $270 million, the terminal will have 1,219 m (4,000 ft) of wharf capable of handling four of American President Lines' largest ships at one time, and an adjacent 19 ha

(47 acre) intermodal rail yard that will also serve the coal export yard being constructed next to the facility. The American President Lines complex will be equipped with six to eight new-generation container cranes. The Port has also embarked on a mammoth $148.6 million dredging project that will create 4.8 km (3 mi) of new channels 13.6 m to 19.1 m (45-63 ft) deep, providing access to Pier 300, a turning basin, and 1,520 m (5,000 ft) of berthing space south of Pier 300. Dredge spoil will be used to create about 91 ha (225 acres) of new land to be called Pier 400, which will be located south (seaward) of the new American President Lines Terminal. Plans for Pier 400, call for the construction of three container terminals on the north side of the terminal, each with two berths and five container gantry cranes, and a large bulk liquid/petroleum terminal complex on the south (ocean) side. Other on-going improvement projects include replacement of the Badger Avenue Bridge providing rail and road access to Pier 300 and Terminal Island, construction of a $200 million coal export terminal on Pier 300 and a $20 million rail yard to serve Terminal Island container terminals [*Long Beach Press Telegram* (Business), "Port Builds for Future," September 26, 1994].

Miami, FL

Phase I of Miami's $100 million port deepening project (begun in April 1991), was completed July 1993 and included deepening of the harbor channel to 12.8 m (42 ft) from the sea buoy to the Lummus Island Container Terminal. Phase II (now underway) extends the 12.8 m (42 ft) channel from container berths on Lummus Island to a new south channel turning basin between Dodge and Lummus Islands. Completion of dredging is expected by mid-1995. The dredging project has already added 24.3 ha (60 acres) of land to Lummus Island and current dredging is expected to add another 16.2 ha (40 acres) for additional container and roll-on/roll-off ship berths. The Port also plans to add two 49 metric ton (54 ton) post-Panamax size container cranes to the existing three 49 mt and three 39.2 metric ton (43 ton) gantry cranes already installed (American Shipper-Southern Ports, January 1994).

Mobile, AL

The Port has just completed about $80 million in improvements through 1993. No new immediate improvements have been identified (Alabama State Docks System, "Port of Mobile Handbook," 1993).

New Orleans, LA

The newest terminal to be added to the Port of New Orleans is the Nashville "B" multi-purpose facility, which marked the completion of the first phase in the ongoing $200 million capital improvements program which, as part of the Mississippi River Terminal Complex, will take New Orleans into the 21st century. When complete, it will feature two miles of continuous modern wharves and state-of-the-art facilities. A full array of multipurpose and ocean-going container ships will be able to discharge cargo quickly, take on new cargo and sail for the next port without delay. A newly paved marshalling yard will eliminate trucking congestion and tie-ups, and an increased shedded area will allow stevedores to operate more efficiently. Flood protection barriers are being raised to eliminate the possibility of flooding. Two ship berths have been added and three more are scheduled to open by the end of 1995. The Napoleon Avenue Wharf C apron width will be replaced to increase the load capacity to 36 kg sm (850 psf), along with other

	improvements. The Tchoupitoulas Corridor Project will provide a new, high-speed dedicated roadway from the port through the city (Annual Directory, Port of New Orleans, 1993-1994; Board of Commissioners of the Port of New Orleans, "Mississippi River Terminal Complex," 1993).
Newport News, VA	No immediate improvements identified.
Norfolk, VA	In 1991, the Virginia Ports Authority began a $40 million expansion of the Norfolk International Terminal that will double the size and cargo handling capacity of the terminal. When completed in 2004, improvements include adding 1,300 m (4,300 ft) of new berthing space and 120 ha (300 acres) of backup cargo handling area, creating a massive (819 acre) intermodal terminal with 27,000 m (89,000 ft) of onsite rail, connecting the terminal with Norfolk Southern's bullet train and providing double stack service to major U.S. markets (Virginia Port Authority, "Promises, Results," 1993; Financial World, "The Ports of Virginia: Destiny Controlled," p. 63, New York, NY, July 20, 1993).
Oakland, CA	The $50 million reconstruction of Oakland's 22.7 ha (56 acre) Seventh Street Terminal is nearing completion. Severely damaged in the 1989 Loma Prieta earthquake, three new post-Panamax cranes have been added and the entire wharf structure and upland areas have been rebuilt. The final phase of the redevelopment program is a $5 million gate relocation and construction project providing six entry and four exit lanes. Truck queues outside the terminal will be avoided by the addition of 46 inbound and 44 outbound queue spaces plus six "trouble" lanes for trucker paperwork problems within the gate area. The gate complex will use computer and video technology to speed container movements through the Port (American Shipper, August 1994, "Rebirth for Oakland Terminal," p. 77).
Philadelphia, PA	A new bi-state agency, *The Port of Philadelphia and Camden, Inc.*, has been created to assume responsibility for regional port operations previously directed by the Philadelphia Regional Port Authority (ports of Philadelphia), the South Jersey Port Corporation (terminals in Camden), and the World Trade Division of the Delaware River Port Authority—a regional economic development agency. The new agency will begin operation in 1995 (WWS/World Wide Shipping, June 1994, p. 35).
Port Everglades, FL	Completion of the Port Everglades Authority's new $100 million, 62.7 ha (155-acre) container complex at Southport, and the development of 6.7 ha (15 acres) of expanded container storage area at Midport, both scheduled for 1994, culminates years of planning and construction by Port Everglades. Southport is equipped with three 39.2 metric ton (43 ton) low-profile, post-Panamax container cranes designed to avoid interference with nearby airport operations. Design planning studies are underway for lift-on/lift-off support facilities at the new 26 ha (63 acre) lift-on/lift-off container yard located immediately adjacent to Southport's cranes. These include a container freight station, electrical outlets for reefer containers, gatehouse with scales, inspection shed, automated facilities, and a feasibility study for developing an intermodal container transfer facility nearer to the Southport complex. The

Fiscal 1993-94 budget provides $9.6 million for a tenth cruise line terminal and enhancements to the two facilities described above (FS, 1992; Southern Ports, January 1994, Pg. 33).

Port of New York, NY

The Port Authority of New York/New Jersey's 1993 capital spending budget totaled $57 million, largely for terminal improvements such as wharf rehabilitation, berth deepening, paving, etc.

Port of Elizabeth, NJ

Funds were also included for deepening Federal channels in the Kill Van Kull and into Newark Bay to the Elizabeth Marine Terminal. The total project, scheduled for completion in 1995, will provide a 12.2 m (40 ft) channel from Upper New York Bay through the Kill Van Kull into Newark Bay. The lack of adequate channel depths has resulted in the diversion of ships to other ports. Three and a half years of wrangling over permits for maintenance dredging and ocean spoil disposal have reportedly increased the cost of dredging from $1 million to $15 million, in part due to court-ordered dredging requirements. The Port Authority has previously announced that it will construct a new $8.5 million on-dock rail terminal at its Port Elizabeth container facilities, which is scheduled for completion in the first quarter of 1995. Initial capacity of the facility will be 100,000 containers annually (WWS/World Wide Shipping, June 1994, p. 33).

Red Hook Container Terminal - Brooklyn, Howland Hook Container Terminal - Staten Island, NY. Red Hook terminal is the only marine cargo terminal still operating on the East side of the Harbor. It was reactivated in January 1994 under the management of American Stevedoring Ltd. The NY/NJ Port Authority is in the process of dredging the approach channel to its project depth of 11.6 m (38 ft). American Stevedoring anticipates handling 20,000 twenty-foot equivalent units in 1994 and as many as 70,000 twenty-foot equivalent units by 1995. The terminal also benefits from a Port Authority subsidized container-on-barge service connecting Red Hook with New Jersey railheads. Terminal facilities include 920 m (3,030 ft) of berthing, containers, roll-on/roll-off and breakbulk cargoes, rail service, four 36.3 metric ton (40 tons) container cranes, and one 63.7 metric ton (70 tons) container crane. While seeking an operator to revitalize the 58.7 ha (145 acre) Howland Hook container terminal—the former base of U.S. Lines idled since 1991, the Port Authority is completing a $25 million renovation of the terminal. Work includes replacement of electrical and distribution systems and resurfacing of a 762 m (2,500 ft) wharf. The Port Authority is also seeking a dredging permit to increase the depth of the berths from the original 10.1 m (33 ft) to the authorized depth of 12.2 m (40 ft). The terminal has a capacity of more than 300,000 containers a year. Its facilities include 762 m (2,500 ft) of lineal berthing space, four 36.3 metric ton (40 ton) and two 45.5 metric ton (50 ton) container cranes, and rail service (American Shipper, August 1994, Pages 73-74).

The City of New York – owner of the Terminal (Howland Hook), and the State of New Jersey are negotiating for the purchase of the Staten Island Railroad tracks between the Terminal and Cranford, NJ, where the short line

connects with Conrail. CSX owns the Staten Island line, but was granted approval in 1991 to abandon the route, so a new owner is needed to reactivate the rail line. City officials and the prospective operator of the Howland Hook facility predicted that the future of this terminal as a viable facility may hinge on the acquisition of the trackage and the installation of on-dock rail service (American Shipper, August 1994, p. 84).

Portland, OR

The Port Commission has approved a $60 million container terminal upgrade program for Terminal 6 to increase throughput capacity to 510,000 twenty-foot equivalent units over the next 10 years, nearly double its present capacity. The Terminal currently handles 314,500 twenty-foot equivalent units a year. Improvement plans include a new $16 million post-Panamax size container crane scheduled to come on stream by late 1995. The Port Commission has also hired an engineering consulting firm to recommend a development strategy and 20-year development program for a new marine terminal complex on West Hayden Island (*American Shipper*, October 1994, "Port of Portland Builds for the Future")

In July, the Port Of Portland Commission contracted with IBM and Stevedoring Services of America to provide the hardware and software for a new $1.0 million computerized terminal management system for its Terminal 6 container facility. The Port presently handles 600 trucks a day with a cargo inventory system developed in 1980. Portland is the fastest growing port on the West Coast (Containerization International, September 1994, "Portland Buys SSA System").

Portsmouth, VA

No immediate improvements identified.

San Francisco, CA

San Francisco's future as a leading West Coast container port is in jeopardy following the decision of Evergreen line to leave the port when its lease expires in June 1995. Evergreen's move follows the departure of Cosco, National Shipping Co. of the Philippines, Nedlloyd Line, Blue Star Lines, and South Seas Steamship. The anticipated reduction in revenues caused by these defections to the Port of Oakland may effect San Francisco's Port capital expenditure programs, including the $10 million rail tunnel improvement project designed to accommodate double-stack train services south of the City. Delays in executing this project are cited as the reason for the loss of these lines. The Port's North Container Terminal is presently dormant and the South Terminal is significantly under-utilized. As reported in WWS/World Wide Shipping, July/August 1994, Pg. 41: The Mayor of San Francisco announced a plan for a New Age entertainment center, incorporating a ballpark and a sports area to be built in space formerly used for cargo handling and Southern Pacific trackage—underscoring the trend to convert prime commercial waterfront land into resort and entertainment areas—an industry-wide problem (Containerization International, "San Franciscos Latest Setback," September 1994, p. 27).

Savannah, GA

Completion of a new 12.8 m (42 ft) shipping channel was completed this Spring. The 1.22 m (4 ft) deepening of the channel makes the Savannah terminals accessible to 98 percent of ships currently in the trade. 1994 is the

third year of Savannah's $319 million development program called *Focus 222*, which is designed to provide the facilities and infrastructure needed to maintain growth into the year 2000. Remaining elements of the Program include steps to help restore the freshwater habitat in the Savannah National Wildlife Refuge, completion of upgrading the 1,680 m (5,500 ft) of contiguous berth at Garden City's Container Berth 6, the addition of 12 ha (30 acres) of container storage and delivery of four new post-Panamax container cranes, two of which were scheduled to arrive late in 1994, and upgrading of existing container cranes, making a total of 13 container cranes at the Garden City port complex (WWS/World Wide Shipping, May 1994, p. 27).

Seattle, WA

The ports of Seattle and Tacoma use the findings of a 1990 econometric study sponsored by the Washington Public Ports Association as an integral part of their planning strategies. In the case of Seattle, this means being capable of handling 2.1-2.5 million twenty-foot equivalent units annually, 15 years hence. The port's Container Terminal Development Plan, adopted by the Seattle Port Commission in May 1991, called for another 97 ha (240 acres) of land to be developed by the end of the century. A further 41 ha (100 acres) has been scheduled for possible acquisition by the year 2010. Seattle currently has about 140 ha (350 acres) of land that is dedicated to container handling activities. The initial phase of the Program involves adding parking space, extending certain piers and upgrading shipside cargo handling gear. Additionally, the Container Terminal Development Plan calls for expansion of existing, and construction of new on-dock rail yards, and improving overall access to/from the port area. A summary of Seattle's current expansion/development programs includes:

a. Expansion of Terminal 5, operated by an affiliate of American President Lines, from 33.6 has to 64 ha (83 to 158 acres) and a 122 m (400 ft) extension of the berth. Work is scheduled for completion in 2 to 3 years;

b. An on-dock intermodal rail facility at Terminal 5 capable of handling two full-length double-stack rail cars simultaneously plus capacity for storing two more, and an overpass to segregate rail and truck traffic;

c. A 36.4 ha (90 acre) expansion to the 44.5 ha (110 acre) Terminal 18 located on the eastern side of Harbour Island. The expansion will permit doubling of the existing intermodal on-dock rail yard from 28 to 56 double-stack rail cars. The new south intermodal rail yard will have separate rail access to avoid conflict with Terminal 5 rail traffic. Container aprons will be upgraded, and the terminal's seven container cranes will be upgraded, and/or replaced by post-Panamax capacity gantry cranes. Additional plans call for an addition of 4 ha (10 acres) to the northernmost extremity of the Terminal, increasing its size to 18.2 ha (45 acres) and the lengthening of the ship berth by 122 m (400 ft). Terminal 18 is the Port's largest common-user facility, and will be able to handle two post-Panamax vessels at the same time (Containerization International, July 1994, pages 87-90).

Tacoma, WA

Tacoma's 20-year, $450 million 2010 Blair Waterway terminal expansion program is equally ambitious, but its implementation will be geared to customer demand. Major elements of the 2010 Blair Waterway program, which is designed to enable the waterway to handle the largest containerships afloat include:

a. The addition of approximately 125 ha (309 acres) of new container terminal area, 11 berths, and 30 ha (75 acres) of new intermodal rail facilities at the Port;

b. Dredging of the main access channel to a depth of 13.7 m (45 ft), and construction of a new city bypass road with subsequent dismantling of the Blair Road Bridge. The bridge is slated to be removed by the end of 1995 and the entire West Blair terminal project is to be completed by the end of 1996;

Additional planned port improvements include the construction of two new container terminals on the north side of the Blair waterway and the new terminals have two berths and 20.2 ha (50 acres) of land. The second new terminal will be built at the existing Terminal 7 and will consist of a one-berth 20.2 ha (50 acre) facility. Spoil from dredging work is being used to fill in the Milwaukee Channel and increase the Sea Land terminal by 9.7 ha (24 acres). According to the econometric study cited above, Tacoma will need to be able to handle between 2.5 and 2.8 million twenty-foot equivalent units in the year 2010 (Containerization International, July 1994, pages 87-90).

Wilmington, DE

No immediate improvements identified.

Wilmington, NC

Long term development plans by the North Carolina State Ports Authority include studies for the deepening of the outer bar channel to 14 m (46 ft), the river and harbor channel to 13.4 m (44 ft), and development of a new marine terminal upstream of the existing port complex. Dredging was expected to begin in early summer 1994 and site development work for the new terminal is slated for fiscal year 1996 provided funding is available. Similar planning for a new marine terminal on Radio Island, adjacent to existing port facilities at Morehead City, is underway. The recently completed channel and harbor dredging to 13.7 m (45 ft) makes Morehead City one of the deepest ports on the East Coast (WWS/World Wide Shipping, May 1994, p. 26).

Attachment D2
Port Population Growth Factors (1990 - 2010)

U.S. Ports	*Counties*	*1990*	*2010*	*Growth Factor*
		East Coast		
Boston, Massachusetts	Suffolk	663,906	792,200	
	Norfolk	616,087	631,300	
		1,279,993	1,423,500	1.11
Elizabeth, New Jersey	Essex	778,206	757,200	
	Kings, NY	2,369,966	2,364,992	
	Hudson	553,099	566,600	
	Richmond, NY	385,224	463,529	
	Union	493,819	502,300	
		4,580,314	4,654,621	1.02
Philadelphia, Pennsylvania	Philadelphia	1,585,577	1,513,674	
	Camden	502,824	550,500	
	Gloucester	230,082	269,300	
		2,318,483	2,333,474	1.01
Eddystone, Pennsylvania	Delaware	547,651	508,557	
	Philadelphia	1,585,577	1,434,694	
		2,133,228	1,943,251	0.91
Wilmington, Delaware	New Castle	441,946	513,750	
		441,946	513,750	1.16
Baltimore, Maryland	Baltimore	692,134	728,898	
	Anne Arundel	427,239	499,204	
	Howard	187,328	288,701	
		1,306,701	1,516,803	1.16
Newport News, Virginia	Isle of Wight	25,053	34,283	
	Norfolk City	261,229	253,809	
	Hampton City	133,793	146,648	
	York	42,422	56,000	
		462,497	490,740	1.06
Norfolk, Virginia	Isle of Wight	25,053	34,283	
	Norfolk City	261,229	253,809	
	Portsmouth City	103,907	101,965	
	Hampton City	133,793	146,648	
	York	42,422	56,000	
		566,404	592,705	1.05
Portsmouth, Virginia	Isle of Wight	25,053	34,283	
	Portsmouth City	103,907	101,965	
	Norfolk City	261,229	253,809	
		390,189	390,057	1.00
Wilmington, North Carolina	New Hanover	120,284	150,936	
	Brunswick	50,985	79,644	
		171,269	230,580	1.35
Charleston, South Carolina	Charleston	295,039	339,400	
	Berkeley	128,776	252,800	
		423,815	592,200	1.40
Savannah, Georgia	Chatham	216,935	273,391	
	Byran	15,438	23,610	
		232,373	297,001	1.28

U.S. Ports	Counties	1990	2010	Growth Factor
Fernandina Beach, Florida	Nassau	43,941	79,800	
		43,941	79,800	1.82
Jacksonville, Florida	Nassau	43,941	79,800	
	Duval	672,971	1,014,100	
		716,912	1,093,900	1.53
Port Everglades, Florida	Broward	1,255,488	1,980,900	
		1,255,488	1,980,900	1.58
Miami, Florida	Dade	1,937,094	2,809,700	
		1,937,094	2,809,700	1.45
		Gulf Coast		
Mobile, Alabama	Mobile	378,643	408,600	
	Baldwin	98,280	110,300	
		476,923	518,900	1.09
Gulfport, Mississippi	Harrison	165,365	175,291	
		165,365	175,291	1.06
Galveston, Texas	Galveston	217,399	245,820	
	Brazoria	191,707	249,644	
	Chambers	20,088	21,200	
		429,194	516,663	1.20
New Orleans, Louisiana	Jefferson	448,306	513,980	
	Orleans	496,938	514,740	
	St. Bernard	66,631	79,950	
	Plaquemines	25,575	29,820	
		1,037,450	1,138,490	1.10
		West Coast		
Seattle, Washington	King	1,507,319	1,833,133	
	Kitsap	189,731	261,970	
		1,697,050	2,095,103	1.23
Tacoma, Washington	Pierce	586,203	792,179	
		586,203	792,179	1.35
San Francisco, California	Marin	231,200	245,500	
	San Mateo	652,100	787,300	
	San Francisco	723,900	781,700	
		1,607,200	1,814,500	1.13
Concord Naval Weapons, California	Contra Costa	810,300	1,096,300	
	Solano	345,700	557,400	
		1,156,000	1,653,700	1.43
Oakland, California	Alameda	1,279,182	1,561,900	
	San Francisco	723,959	781,700	
		2,003,141	2,343,600	1.17
Los Angeles, California	Orange	2,424,100	3,104,100	
	Los Angeles	8,897,500	11,441,900	
		11,321,600	14,546,000	1.28
Long Beach, California	Orange	2,424,100	3,104,100	
	Los Angeles	8,897,500	11,441,900	
		11,321,600	14,546,000	1.28

+1990 Census taken from Rand McNally/The New Cosmopolitan World Atlas Census/Environmental Edition, 1992.

Alabama	Alabama Population Projections 1990-2015, Alabama State Data Center Center for Business and Economic Research, University of Alabama, Tuscaloosa, AL, January 1994.
California	Population Projections by Race/Ethnicity for California and its Counties, Report 93 P-1, Demographic Research Unit, Sacramento, CA, (916) 322-4651, April 1993.
Delaware	Census info and projection numbers through Evelyn Pearson, Delaware Development Office, Business Research Section, Dover, DE, Consortium Series, (302) 739-4271, June 30, 1994.
Florida	Projected from Florida Population Studies (by county) by Stanley K. Smith, Director, Bureau of Economic and Business Research, University of Florida, Volume 27/Number 2/Bulletin No. 108, February, 1994.
Georgia	Census info and projection numbers through Marty Sik, Governor's Office of Planning and Budget, Atlanta, GA, (404) 656-0911.
Louisiana	Census info and projections provided by Division of Administration, Baton Rouge, LA, Department of Budget, ATTN: ARL, (504) 342-7410.
Maryland	Department of State Planning, Office of State Planning Data, Office of Michael Lettre, , Baltimore, MD, (410) 225-4452, September 29, 1994.
Massachusetts	Ms. Alice Rarig, Massachusetts Inst. for Social & Econ Research (MISER), University of Mass., Amherst, MA, (413) 545-6660, September 30, 1994. **Calculations are only preliminary numbers. Final reports will be made available by end of October 1994.
Mississippi	Projections given by phone through the Office of Dr. Barbara Logue (EPA) on 9/29/94 from Center Policy Research & Planning, MS Institute of Higher Learning, Jackson, MS, (601) 982-6576, September 29, 1994.
New Jersey	Census info and projection numbers provided by Sen-Juan Wu, New Jersey Dept of Labor, Labor Market & Demographic Rsr, Trenton, NJ, (609) 292-0076.
New York	Census info and projections provided by New York State Bureau of Economic and Demographic Info, Albany, NY, (518) 474-6005.
North Carolina	Census info and projection numbers through Bill Tillman, Office of State Planning, Raleigh, NC, (919) 733-4131, Prepared April 1994.
Pennsylvania	Projections given by David Gordner, Bureau of Water Management, Department of Environmental Resources, Harrisburg, PA, (717) 772-4048, September 30, 1994.
South Carolina	Census info and projection numbers through Diana Tester, South Carolina Budget and Control Board, Office of Research & Statistical Services, Columbia, SC, (803) 734-3619, Published October 29, 1993.
Texas	Census info and projection numbers obtained through Texas State Data Center, Texas A & M University System, College Station, TX, 77843-2125, (409) 845-5115. Contact: Hazel Dolar.

Virginia	Projection given by Jeanne Brown, Center for Public Service University of Virginia, Charlottesville, VA, (804) 982-5580, September 28, 1994.
Washington	Census info and projections from Washington State County Population Projections, Office of Financial Management, Forecasting Division, Olympia, WA, January 31, 1992.

Attachment D3
Background Discussion of Alternative Analytical Models for Evaluation of Potentially Impacted Port Populations

In the Fall of 1993, the Department of Energy (DOE) began to collect and analyze information required for the list of port criteria included in the Notice of Intent (DOE, 1993) for this environmental impact statement (EIS). DOE recognized that there would be public concern associated with consideration of potential ports of entry for the foreign research reactor spent nuclear fuel. Therefore, DOE decided to develop a sound technical basis for the identification of potential ports of entry.

As a result (concurrently with the independent evolution of the Urgent Relief Environmental Assessment), a list of 28 potential commercial ports was established based on the recommendations of independent maritime consultants. The database included information in the following categories:

1. *Geophysical Factors*, harbor and channel water depths [a port would fail if it had less than 7 m (23 ft) of water, but receive the maximum score if it had more than the 12 m (40 ft) of water required for all but the largest cellular container vessels]; the nautical distance from the open ocean to the port [ports greater than 40 km (25 mi) from open ocean received no points, but were not disqualified from further consideration]; and navigational factors that might increase public risks (narrow, winding channels with currents or other factors seriously affecting safe navigation were given no points, but a weighting factor was applied to channels with good characteristics to account for the relatively greater importance of this factor for maritime safety).

2. *Port facilities*, which included the capabilities of cargo terminals for handling containerized foreign research reactor spent nuclear fuel, wharves and depths alongside, crane capacities, terminal access (truck and rail), terminal security, and the liner services available.

3. *Factors related to spent nuclear fuel handling and transport*, including past experience with spent nuclear fuel or other hazardous cargoes, whether there were local restrictions on the receipt of foreign research reactor spent nuclear fuel, emergency response capabilities, hazardous material handling training, locations of terminals relative to nearby populations with a doubleweighted score for ports that were remote from urban populations (e.g., heart of a city), 1990 census statistics for port city populations and population densities, environmental factors (whether the immediate port vicinity had sensitive populations of animals), and distance from the port to Savannah River Site and Idaho National Engineering Laboratory (at that time these were the preferred storage sites due to historical experience and facilities; the other three sites were added later as the result of the Programmatic Spent Fuel Draft EIS decision to consider them).

Using the database developed, a semi-quantitative analysis of the port criteria was prepared that summed the "score" assigned to each port attribute by the maritime experts, and the ports were ranked from best to the least acceptable (this list of ports is, for the most part, a subset of the set of over 40 ports that were subsequently analyzed in detail in Section D.2 of this appendix).

DOE determined that a semi-quantitative analysis of all ports for all of the noticed criteria was unacceptably subjective, especially concerning the assignment and weighting of the numerical scores. Furthermore, it did not differentiate well between ports, and when weighting factors were applied to better discriminate between criteria that were very important to safety versus those that were "desirable attributes," the methodology became very difficult to justify.

Attachment D4
Derivation of Ship Collision Damage Probabilities

Derivation of the accident severity category probabilities requires that a probability of damage to the transportation cask, given a collision between two vessels, be calculated. In Appendix D, this probability has been characterized by two values, P_{Impact} and P_{Crush}. The first is a probability that the cask is damaged due to impact forces associated with the collision. The second represents the probability that crush forces result in damage to the cask. This attachment describes how these probabilities were derived.

D4.1 Kinetic Energy

V.U. Minorsky developed a method for analyzing the collision of ships that provides a correlation between resistance to penetration and the energy absorbed in the collision (Minorsky, 1959). The absorbed energy was determined for actual collisions by assuming the impact was nearly transverse, the hydrodynamic forces due to water entrained by the hull of the struck ship could be treated as a virtual increase in mass, and the collision was perfectly inelastic. The resistance to penetration was quantified through a resistance factor, R_t, which was computed from accident and ship design information. He found, for higher energy incidents, that there is a linear correlation between R_t and the absorbed energy.

ORI Inc., in a draft report on accident severities associated with water transport of radioactive materials, extended Minorsky's method to develop correlations between penetration depth and the energy absorbed in ship collisions (ORI, 1981b). By considering empirical probability distributions for displacement of the striking ship, its speed, and the angle of impact, bounding case curves were developed for the probability of occurrence of force levels at selected penetration depths. The force value referred to is the collision force acting between the two ships.

Only a fraction the collision force would be seen by a spent fuel transportation cask on board the struck ship. ORI gave a qualitative discussion of this aspect of the collision, together with some limiting case values based on assumptions about stowage and the presence and type of other cargo.

The present analysis depends, to a large extent, on the Minorsky and ORI analyses. It does add an approximate treatment of accelerations experienced by the spent fuel package and includes the effects due to cargo in determining the maximum penetration depth in collision events. The dynamics of inelastic collisions are treated through conservation laws for momentum and energy. Following Minorsky, the transverse hydrodynamic forces on the hull of the struck ship are accounted for by a virtual increase in mass, hence kinetic energy. This is a conventional method used by naval architects, but has limitations when applied to collisions. M.J. Petersen pointed out that experiments and calculations by Motora et al., have shown that the added mass treatment is not always a good approximation (Petersen, 1982; Motora, 1971). Here we accept the limitations imposed by the added mass method, because a more rigorous treatment of the collision is not warranted due to other uncertainties in the analysis, particularly in the modeling of cargo effects.

It should be noted that the ORI/Minorsky method of calculating hull penetration probably does not take account of the massive keel structures in the struck transport ships. Therefore, they most likely significantly overestimate the probability of penetration further than one-fifth of the beam of the struck ship, since penetration to this distance would mean that the keel structures had been encountered. Note that historic experience (rule-of-thumb experience) indicates that few ship collisions lead to penetration more than one-fifth of the beam of the struck ship.

Parameters and Assumptions

The target ship in the following calculations is assumed to have a beam of 24.99 m (82 ft) and a displacement, 'm', of 25,310 metric tons (27,841 tons). The virtual mass, 'dm', due to hydrodynamic forces is 0.4 m = 10,120 metric tons (11,132 tons). Eight cases are considered for the displacement, 'm', of the striking ship: 5,600; 16,800; 28,000; 39,200; 50,400; 61,600; 72,800; and 84,000 metric tons (6,160; 18,480; 30,800; 43,120; 55,440; 67,760; 80,080; and 92,400 tons. The normal component of the striking speed at impact ranges from 1 to 10 meters per second (1.9 to 19 knots or 2.2 to 22 statute miles per hour).

A full distribution of sailing speeds (0-22 knots) was used in the penetration calculations even though speeds in port channels are likely to be no greater than 10-15 knots and speeds at dockside only a few knots (minimum required to maintain steerage). In addition, large ships (e.g., tankers) are likely to be pushed/towed by tugs near docks.

The models for energy absorption by the ship and its cargo follow the methods of ORI. The work, 'W', due to cargo compression is the product of the crush strength of the cargo, the cross sectional area of the blunted bow of the striking ship, and the difference between the penetration distance and the cargo closeup distance. ORI gave examples of this calculation, which are reproduced in the formula

$$W_{cargo} = 19.44 f \sigma (x - f(\,beam\,))$$

where f is the fraction of open space on the hold floor, σ is the crush strength of the cargo in MPa (mega pascals), 'x' is the penetration depth and beam is the width of the struck ship, both in meters. This formula follows ORI in assuming the vertical size of the damage zone is 7.62 m (25 ft), and one third of the blunted bow is the effective area.

Prior to the initiation of cargo compression, energy is absorbed solely by deformation of the ship structure; this effect is modeled using the Minorsky value of 32 'mj' (mega joules) for the energy to penetrate the hull, together with the semi-empirical curves in Figure 6.2 of the ORI report. Table D4-1 gives coefficients for a quadratic fit used to represent the ORI curves below 15 m (49.2 ft) penetration, while a second fit for greater penetration distances is given in Table D4-2.

Table D4-1 Quadratic Coefficients for Energy Absorbed Due to Ship Structures <15m

$$W_{ship} = a + bx + cx^2 \quad (x < 15m)$$

Metric ton	*a (mj)*	*b (mj/m)*	*c (mj/m²)*
5,600	9.551	0.6836	0.0405
16,800	8.709	0.8118	0.2984
28,000	8.056	1.2030	0.4558
39,200	8.121	1.0850	0.5296
50,400	9.234	0.6555	0.6217
61,600	8.956	0.8639	0.6698
72,800	8.574	1.1790	0.6906
84,000	8.204	1.5290	0.7154

Table D4-2 Quadratic Coefficients for Energy Absorbed due to Ship Structures >15m

$$W_{ship} = a + bx + cx^2 \quad (x > 15m)$$

Metric ton	*a (mj)*	*b (mj/m)*	*c (mj/m²)*
5,600	58.13	-4.837	0.1919
16,800	14.72	-0.057	0.3337
28,000	15.76	-0.304	0.5179
39,200	64.87	-5.622	0.7306
50,400	189.5	-19.21	1.162
61,600	264.4	-29.02	1.531
72,800	303.8	-36.36	1.878
84,000	412.2	-50.50	2.393

Distribution of Ship Displacements, Speeds and Angles

Analysis of two years of shipping accident data allowed ORI to develop probability distributions for 'M' (mass of the striking ship), 'V' (transverse speed of the striking ship), and θ (angle of incidence), which are presented here in Table D4-3 through D4-5. The ORI tables originally contained eleven intervals for displacement of the striking ship. Four cargo loadings were examined in the analysis: no cargo, light cargo, medium cargo, and heavy cargo (light, medium, and heavy refer to the amount of cargo on board). For the present work, the two lowest intervals were combined as were the three highest, yielding eight intervals to match the eight ORI curves for 'W'. There were also 11 values of 'V' in the ORI tables, with speeds ranging up to 11.3 meters per second (21.5 knots or 24.9 statute miles per hour), and 9 values of the collision angle. Thus 968 different combinations of these values are treated in determining transportation cask failure.

Table D4-3 Probabilities for Striking Ship Displacement

Displacement (metric ton)	*Probability of Occurrence*
0 - 10,160	0.15
10,161 - 20,321	0.25
20,322 - 30,481	0.25
30,482 - 40,642	0.05
40,643 - 50,802	0.05
50,803 - 60,963	0.05
60,964 - 71,123	0.10
71,124 - 152,407+	0.10

Table D4-4 Probabilities of Striking Ship Speeds

Speed (meters/second)[a]	*Probability of Occurrence*
0.0 - 1.028	0.0448
1.028 - 2.058	0.2538
2.058 - 3.087	0.1045
3.087 - 4.115	0.1343
4.115 - 5.144	0.1343
5.144 - 6.173	0.0896
6.173 - 7.202	0.0746
7.202 - 8.231	0.0597
8.231 - 9.260	0.0746
9.260 - 10.29	0.0149
10.29 - 11.32	0.0149

[a] *1 meters per second = 1.9 knots = 2.2 miles per hour*

Table D4-5 Probabilities of Striking Ship Angles of Incidence

Angle From the Normal (degrees)	*Probability of Occurrence*
0 - 10	0.2754
10 - 20	0.1305
20 - 30	0.0725
30 - 40	0.1305
40 - 50	0.1015
50 - 60	0.0724
60 - 70	0.1303
70 - 80	0.0435
80 - 90	0.0434

Speed During a Collision

In the following, 'M' and 'V' are the mass and transverse speed of the striking ship, while 'm' and 'v' denote the mass and transverse speed of the struck ship. Theta (θ) is the angle of impact, measured from the normal to the direction of the struck ship (this is the angle used by ORI, Minorsky and Petersen use its complement). The amount of virtual mass attributed to the struck ship to account for transverse hydrodynamic forces is 'dm'. W(x) denotes the work done in deforming the ships and compressing the cargo during a penetration to a depth 'x', and E_0 is the initial kinetic energy in the motion of the striking ship transverse to the struck ship.

The total energy in the transverse motion of the striking ship is:

$$E = MV^2 \cos^2(\theta)/2$$

Because energy is conserved during the collision, and neglecting turning effects,

$$E = \frac{MV^2}{2} + \frac{(m+dm)\,v^2}{2} + W(x)$$

Because momentum is conserved,

$$MV\cos(\theta) = MV + (m+dm)\,v$$

Together these equations yield a quadratic expression of the velocity of the struck ship:

$$\frac{Av^2}{2} - V\cos(\theta)\,v + \frac{W(x)}{m+dm} = 0$$

where A = (1+(m+dm)/M).

The value of the struck ship's transverse speed during the collision is, therefore,

$$v = \frac{V\cos(\theta)}{A} - \frac{1}{A}\sqrt{V^2\cos^2(\theta) - \frac{2AW(x)}{m+dm}}$$

The second term in this equation decreases to zero during the collision, yielding a terminal speed of $V\cos(\theta)/A$. This is also the terminal speed component of the striking ship in the same direction. The change in kinetic energy is (1-1/A)E = (m+dm)/(M+m+dm)E, in agreement with Minorsky.

Maximum Penetration Distance

The maximum penetration of the bow of the striking ship into the target ship was computed by finding, using Newton's method, the position at which the ships reached their terminal speed. From the conservation laws for energy and momentum, the condition for this to occur is:

$$0.5\,\mu\,V^2\cos^2(\theta) = W(x)$$

where $\mu = M(m+dm)/(M+m+dm)$, and $V_{\cos(\theta)}$ is the initial normal speed of the striking ship. For the no cargo case, it was found that for each of the striking ship displacements considered, initial normal speeds of 8 meters per second (15.2 knots or 17.6 statute miles per hour) and 10 meters per second (19.0 knots or 22.0 statute miles per hour) were sufficient to cut completely through the struck ship, resulting in a probable sinking; refer to Figure D4-1. On the other hand, at 2 meters per second (3.8 knots or 4.4 statute miles per hour) only the four heavier ships would even penetrate the hull of the struck ship, and at or below 1 meters per second (1.9 knots or 2.2 statute miles per hour) the hull was not punctured for striking ships of any displacement.

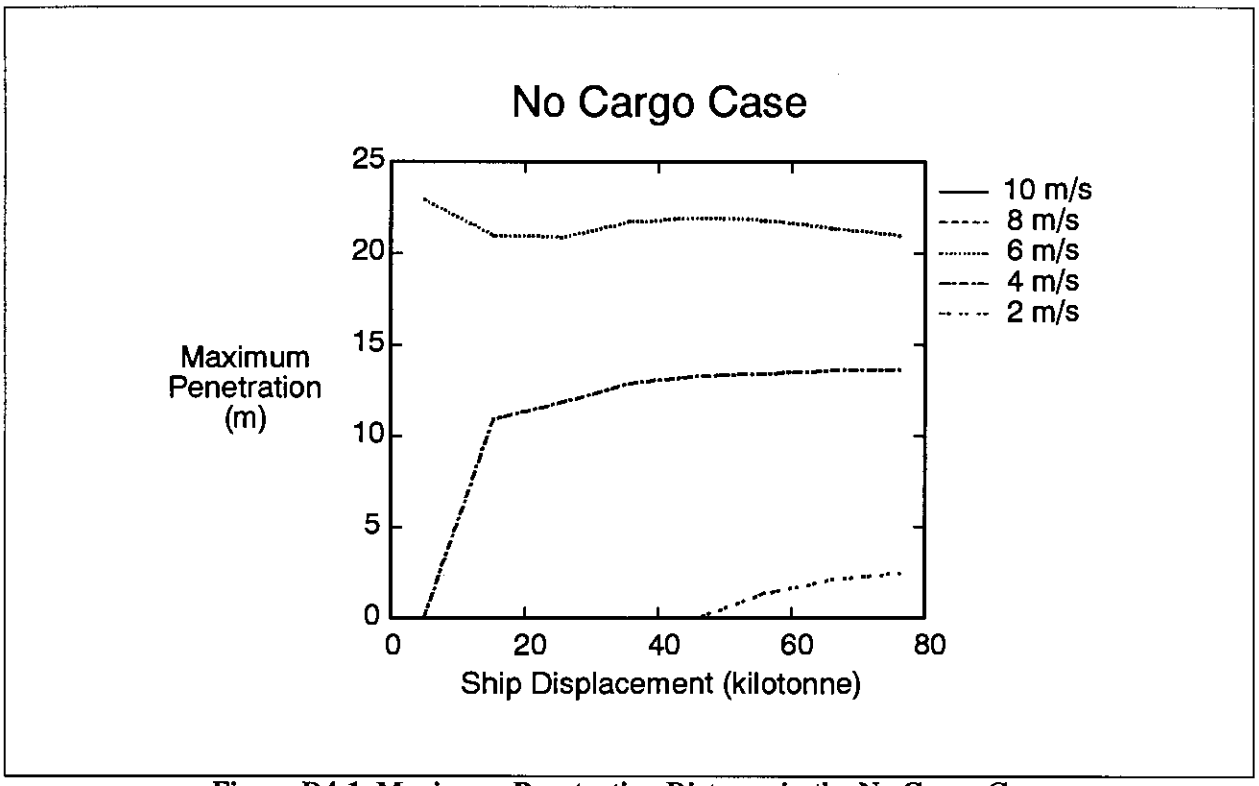

Figure D4-1 Maximum Penetration Distance in the No Cargo Case

Figure D4-2 shows the corresponding information for the light cargo case. Because of the packing fraction for this case, 0.6, the cargo effect does not begin until penetration has reached 15 m (49.2 ft). The figure shows the results as a function of the displacement of the striking ship, for normal impact speeds from 2 meters per second (3.8 knots or 4.4 statute miles per hour) to 10 meters per second (19.0 knots or 22.0 statute miles per hour). There were no cases where the struck ship would be completely cut through. At the two lower speeds, the cargo did not close up, hence was not a factor in absorbing the impact energy. There was no penetration at 1 meter per second (1.9 knots or 2.2 statute miles per hour) for any of the eight striking ship displacements considered.

The medium and heavy cargo results are shown in Figures D4-3 and D4-4, respectively. Figure D4-3 shows the cargo effect beginning at 5 m of penetration, and is important down to impact speeds of 4 meters per second (7.6 knots or 8.8 statute miles per hour). The cargo did not close up at smaller speeds, so was not a factor in determining the penetration depths. A similar result was obtained for heavy cargo; in both cases there was a strong influence by the cargo on the maximum penetration depth.

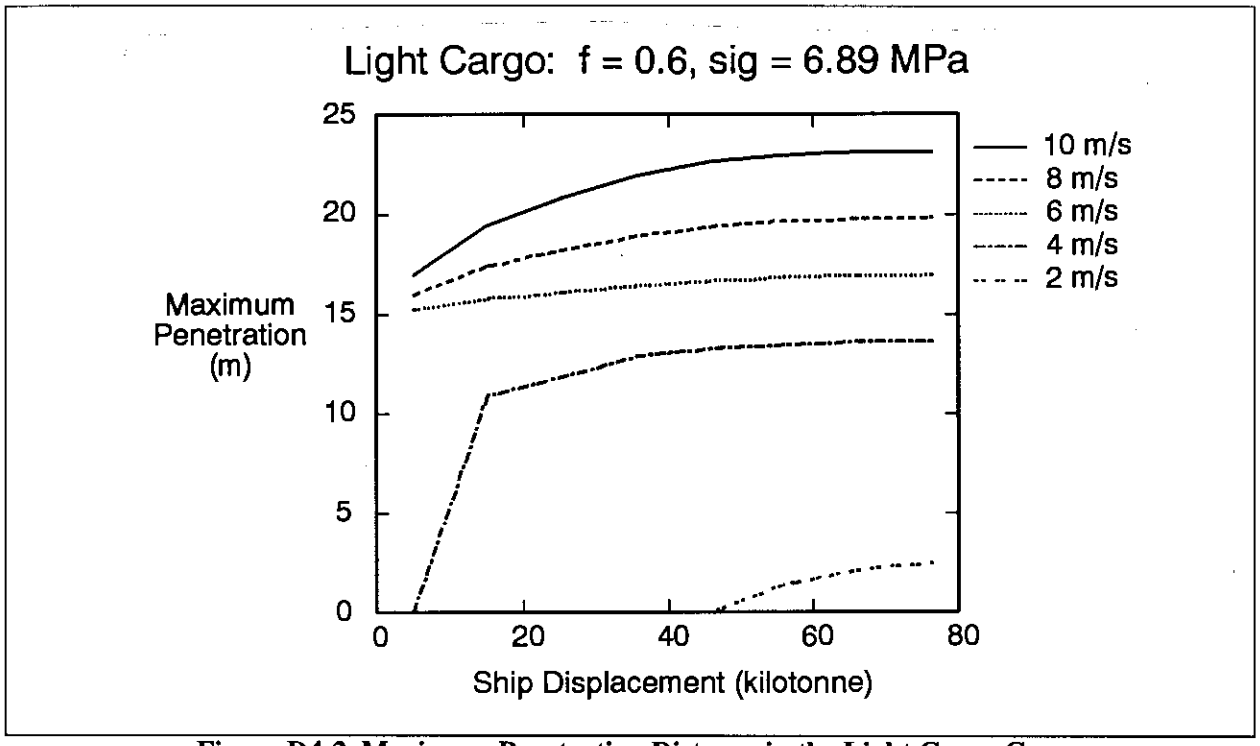

Figure D4-2 Maximum Penetration Distance in the Light Cargo Case

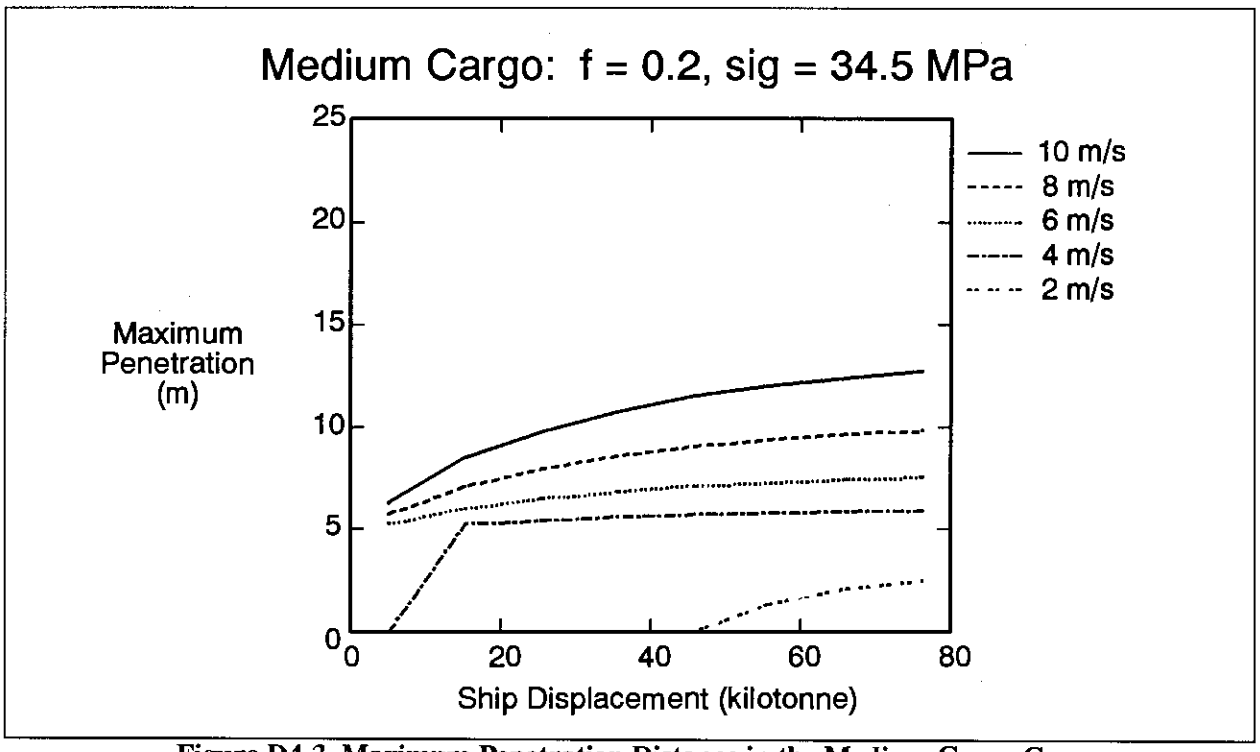

Figure D4-3 Maximum Penetration Distance in the Medium Cargo Case

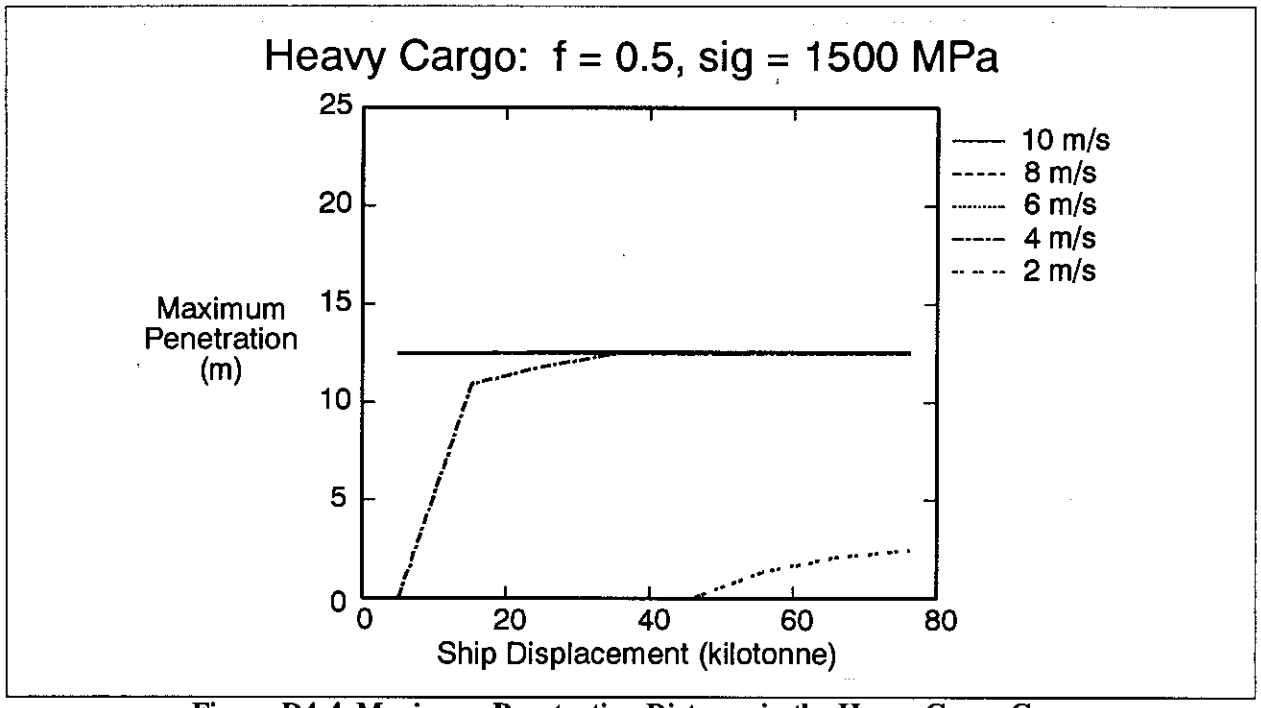

Figure D4-4 Maximum Penetration Distance in the Heavy Cargo Case

D4.2 Impact Forces During the Collision

Fuel elements experience impact forces if during a strong acceleration event they are driven against the inside of the cask or basket, or come into hard element to element contact. It is shown in Sanders et al., (Sanders, 1992) that accelerations of 100 g can be produced in the hypothetical accident conditions defined by NRC, which involve 9-m (29.5-ft) drops onto unyielding targets (NRC, 1990). They also showed there is a resulting cladding breach probability that for some power fuel types can be up to 0.0002 per rod in such events. We show here that the average acceleration experienced in ship collisions is very much smaller, usually below 1g, and conclude that inertial effects on the fuel are not significant for ship collisions.

The acceleration as a result of a ship collision is the time derivative of the transverse speed of the struck ship:

$$a = \frac{dv}{dt} = \frac{dv}{dx}\frac{dx}{dt} = v\frac{dv}{dx}$$

Performing the derivatives yields:

$$a = \frac{v(x)\,F(x)/(m+dm)}{\sqrt{V^2\cos^2(\theta) - \frac{2A\,W(x)}{m+dm}}}$$

where F(x) = dW(x)/dx. Notice that the acceleration peaks at the end of the collision, because the argument of the square root goes to zero there. The acceleration has a vertical asymptote at the maximum distance of penetration; the average acceleration, however, remains small.

The average acceleration during the collision is:

$$<a> = \frac{\int_o^d a(x)dx}{d} = \frac{V^2 \cos^2(\theta)}{2\,d\,A^2}$$

where d is the maximum penetration depth. This is an improper integral since a (x) has a singularity at 'd'. However, the integrand increases sufficiently slowly in the neighborhood of 'd', like $(x-d)^{-1/2}$, for the integral to converge.

When there is no other cargo in the hold with the spent fuel cask, the average acceleration is only a fraction of 1g (9.8 meters per second^2) in all cases, with the average acceleration always less than 5 meters per second^2. Similar results hold for the light and medium cargo cases. Even in the extreme case of heavy cargo, the average accelerations found were less than 2.5g. The highest acceleration, corresponding to a 75,000 metric tons (82,500 ton) ship striking with a normal speed of 10 meters per second (19.0 knots or 22.0 statute miles per hour), was about 0.2g (2 meters per second^2).

Because of these low average accelerations, generally on the order 1 percent relative to the accelerations expected in the NRC regulatory accident conditions, impact of fuel elements inside the cask is not expected to do any damage to the fuel as a result of collisions either in port or on the high seas. We conclude $P_{impact} = 0.0$.

D4.3 Crush Loads on the Fuel Package During the Collision

The spent fuel package of interest is the Pegase transportation cask, a cask of french design. It is a lead shielded cask, with a mass of 18.9 metric tons (20.8 ton), a diameter of 1.875 m (6.2 ft), and a height of 2.239 m (7.3 ft). It has a body composed of two stainless steel shells built around a lead shield. It is designed to carry fuel or other radioactive material in baskets of differing design which fit into the cylindrical cavity of the cask. A detailed analysis of the mechanical response of the Pegase transportation cask to crush forces is not available, however it is similar in construction to the lead shielded cask analyzed in the study of Fischer et al., (1987).

Fischer et al., developed a curve for the static force versus deflection for sidewise loading of a cask which was 4.9 m (16.1 ft) high, with a lead shield 0.133 m (0.4 ft) thick enclosed by an outer layer of stainless steel 0.0318 m 1.25 in) thick and an inner layer 0.0127 m (0.5 in) thick. Because a Pegase transportation cask is much shorter, but of similar construction, it will be at least as resistant to sidewise loading as Fisher's generic lead shielded cask. Fisher's results show that it requires a load of about 8.9 million newtons to produce a deflection of the cask body of 0.0254 m (1 in). A deflection of 0.0254 m (1 in) is judged to be a conservative deflection that could occur without damage to the fuel. Said another way, sidewise cask loading on a Pegase transportation cask in excess of 8.9 million newtons would probably result in some disruption of the fuel. Now the questions is, can crush forces on the cask as high as 8.9 million newtons be produced in a ship collision? To the extent that the homogeneous cargo models are applicable, the answer is "yes." The force applied by the cargo in these models, after closeup, is a constant equal to:

$$F_{cargo} = \sigma\, hd$$

where σ is the cargo crush strength, and 'hd' is the cross sectional area of the cask; for the Pegase transportation cask, 'hd' is 4.198 m^2 (45.2ft^2). Thus, the force in the light cargo case is 56.0 million newtons, and for the medium and heavy cargo cases it is many times larger. These values so far exceed the damage threshold at 8.9 million Newtons that major damage to the fuel and cask can be expected.

But if the cargo does not close up because the penetration is shallow or there is no other cargo in the hold, the cask does not see this force. Then, unless it is within the penetration region, it will not be significantly affected.

Inside the penetration region the cask can be crushed without the cargo going solid, or even if there is no other cargo in the struck hold. Cask tiedowns are designed, under U.S. regulatory practice, to withstand about 5 million newtons of transverse force (NRC, 1990). The difference between this value and the 8.9 million newtons required to produce a 0.025 m (1.0 in) deflection in the cask wall of the generic cask is not considered significant; moreover in ORI's opinion "the RAM [Radioactive Material] package could conceivably be restrained from sliding, even in an empty hold, after the fittings failed. A buckled deck for example could do this and in effect act as an infinitely strong fitting" (ORI, 1981a).

Thus there are two cases to consider for failure due to crush forces. In the first the penetration depth exceeds the cargo close-up distance, while in the second it exceeds the cask stowage location. We assume fuel damage and closure failure in both types of events.

Cask Failure Probability

This section evaluates the probability that a cask will fail when the ship carrying it is struck in a collision with another ship. Since there are two different scenarios, the total probability of cask failure is the sum of two terms, one of cargo going solid, the other for the ship over-running the cask location, or

$$P_{\text{crush}} = P_{\text{solid}} + P_{\text{contact}}.$$

P_{solid} and P_{contact} were evaluated by comparing the maximum penetration distance against the closeup distance and the stowage position, assumed to be on the centerline of the hull, for all combinations of striking ship displacement, speed, and angle given in Tables D4-3 to D4-5. Each individual case was counted as either resulting in cask failure (meaning the fuel is damaged and the cask seal is broken) or not, and the probability of the case was assigned according to the probability values in the referenced tables. The sum $P_{\text{solid}} + P_{\text{contact}}$ of the probabilities of all failure cases is P_{crush}.

The results are shown in Figure D4-5. The successive columns refer to the four models considered, for no cargo, and light, medium and heavy cargo. For other than the medium cargo model, the total crush probability is about 0.29, although the fraction due to the cargo going solid varies from 0 for the no cargo case to 1 for the heavy cargo case. The medium case, which as the smallest fraction of open hold space at 0.2, also has the highest failure rate, about 0.45. Of the four cases considered, this is the only case where the cargo goes solid well before the midline of the ship is reached, thus permitting a greater proportion of all the collisions to be significant from a cask damage point of view. Since this case shows the greatest probability, it is conservative to take $P_{\text{crush}} = 0.45$.

Alternate Case

Because the top speed in a harbor is controlled, the ORI distribution was adjusted to a top speed of 8.23 meters per second (15.6 knots or 18.1 statute miles per hour). This reduced the number of speed intervals to eight, and eliminated the three highest speed categories in Table D4-4. The total number of combinations of striking ship displacement, speed, and angle was therefore reduced from 968 to 704. Figure D4-6 shows the revised cask failure probabilities for the four cases. The highest failure probability is still from the medium cargo case, probably because this case has the earliest cargo closeup distance and fails most often from collisions which do not penetrate far into the target ship. The failure probability goes down more in the other cases because they involve penetrations going past the midline of the ship. Such events are sensitive to the high end of the speed distribution. The cask crush probability for this alternative is set equal to the largest result, $P_{\text{crush}} = 0.40$.

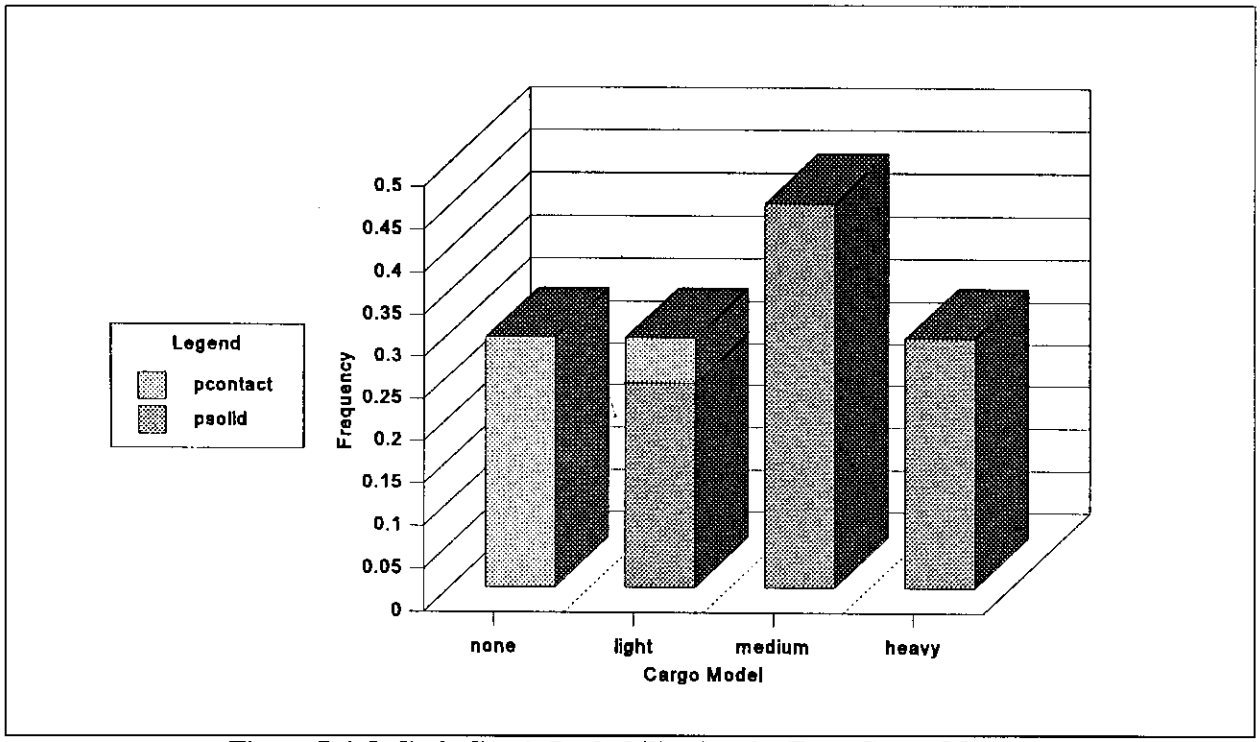

Figure D4-5 Cask Crush Probability for the Four Cargo Models

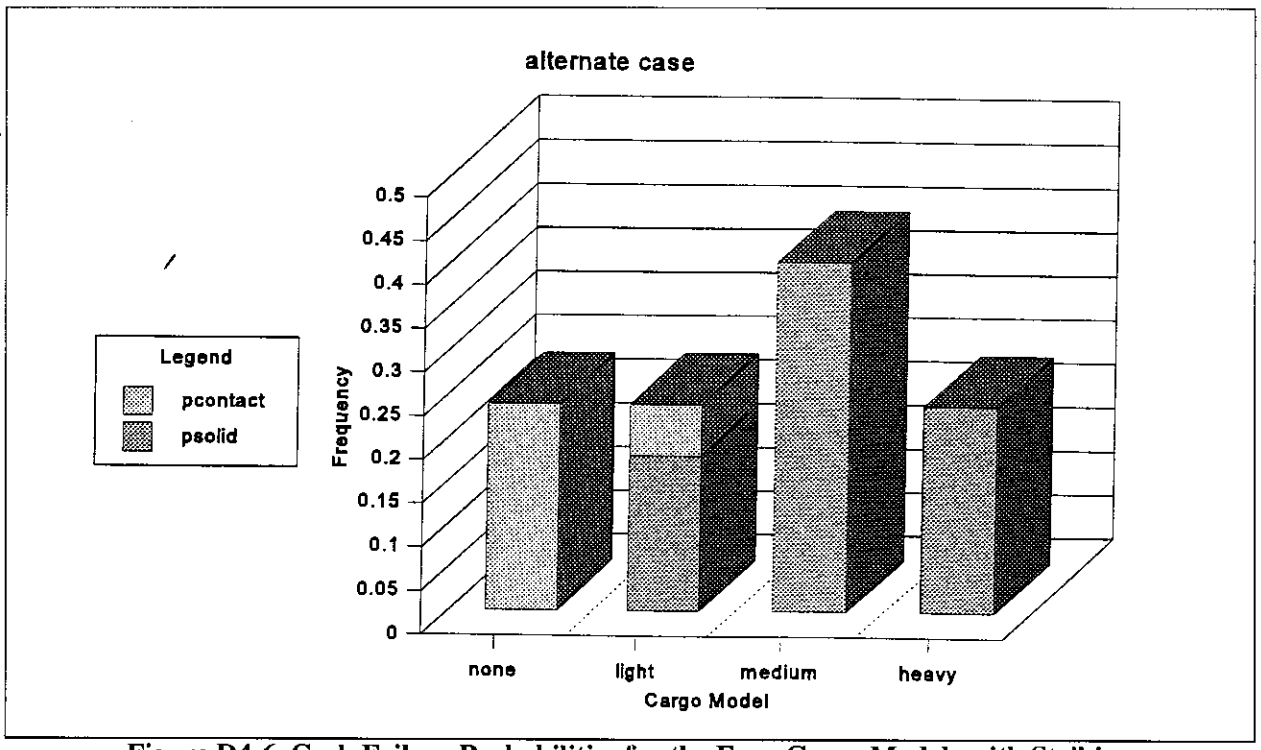

Figure D4-6 Cask Failure Probabilities for the Four Cargo Models with Striking Ship Speeds Truncated at 8.23 m/s

Attachment D5
High Temperature Effects on Research Reactor Fuel Release Fractions

D5.1 Introduction

Previous assessments of the accident risks associated with the transport of research reactor fuel did not specifically address certain high temperature (somewhat above 900°K or 1160°F) effects on the fuel. In this temperature range, aluminum based fuels (the aluminum-uranium alloys used in research reactor fuels) are susceptible to melting. Additionally, TRIGA fuel is spontaneously combustible in this same temperature range, if sufficient oxygen is available. The melting point for the uranium dioxide fuels that had been used as the basis for the development of estimates of the release fractions for earlier assessments is considerably higher than for the aluminum based fuels. These earlier assessments were the basis for the release fractions used in the base case analysis of this study. This attachment provides an assessment of the impact of these high temperature effects on the release fractions, and the probabilities of the accident severity categories, used in the base case study. This assessment forms the basis for the sensitivity study provided in Section 5.4.3.2 of Appendix D.

D5.2 Fission Product Release At High Temperatures

Table D-21 shows that accident severity category 4 accidents are caused by a ship collision that fails the seal of the spent fuel transport cask; that category 5 accidents add a severe engulfing fire to the conditions that characterize category 4 accidents; and that category 6 accidents assume an engulfing fire and a more severe cask failure (one medium sized hole or two or more small holes), one that allows the severe fire to induce substantial convective flow of air through the failed cask. Table D-21 also shows that the principal difference between severity category 5 and severity category 4 release fractions is a five order-of-magnitude increase (from 1.0×10^{-8} to 9.0×10^{-4}) in the release fraction for cesium; and that the principal difference between severity category 6 and severity category 5 release fractions is a 42-fold increase (from 1.0×10^{-6} to 4.2×10^{-5}) in the release fraction for ruthenium.

Much increased cesium volatility at the elevated temperatures to which the spent fuel is heated by the severe fire is the cause of the five order-of-magnitude increase in cesium release assumed for category 5 accidents. Conversion of elemental ruthenium to volatile ruthenium oxide (RuO_4) by oxygen, due to convective air flow through the failed cask, is the cause of the 40 fold increase of ruthenium release assumed for category 6 accidents.

The release fractions listed in Table D-21 and used in the base case analysis were constructed from release estimates developed (Wilmot, 1981; Wilmot et al., 1981) for power reactor fuel (uranium dioxide pellets clad in zircaloy). The fuels used in research reactors are not uranium dioxide pellets clad in zircaloy. TRIGA reactors use large pellets formed from a mixture of uranium, zirconium, and zirconium hydride (ZrH_2) that are clad in stainless steel. All of the other reactors considered in this assessment (BR-2 and RHF) use aluminum-clad metallic fuels where the metal is an alloy of aluminum and uranium (Al-U). At elevated temperatures (above 900°K or 1,160°F) these fuels melt and, if exposed to air, TRIGA fuel burns. Therefore, if a ship collision leads to a fire that heats these fuels to temperatures much above 900°K (1,160°F), fission product releases from these fuels will differ markedly from that predicted for uranium dioxide power reactor fuels. Therefore, the properties of metallic aluminum-uranium alloy fuels and of TRIGA fuel were reviewed to identify any significant differences between releases from these fuels and releases from power reactor fuel when these fuels are heated to elevated temperatures.

The BR-2, and RHF fuels considered by this study are fabricated as stacks of aluminum-uranium alloy plates or cylinders that are contained in aluminum-cladding. Release of fission products from aluminum-uranium alloy fuels has been reviewed (Ellison, 1993). Fission product release is minor below about 923°K (1,202°F), the melting point of the aluminum-uranium alloy from which these fuels are fabricated. Once the aluminum-uranium alloy has melted, fission products volatile at melt temperatures are rapidly released to the gas space above the molten alloy. Although molten aluminum can dissolve both the stainless steel spacers that support individual fuel bundles and the alloy fuel plates, melting of the aluminum-cladding that surrounds these alloy fuels does not significantly affect release because the melting temperature of the clad, 933°K (1,220°F), is slightly higher than the melting temperature of the alloy fuel.

The effects of air ingression on the release of fission products from commercial reactor fuel have been reviewed (Powers, 1994). That review indicates that ruthenium release fractions from uranium dioxide fuel will equal or exceed 4.2×10^{-5}, the release fraction for ruthenium used in the base case analysis for category 6 accidents, if the fuel is exposed to air for 15 to 30 minutes while heated to 700°K (800°F). The review also indicates that release increases rapidly as temperature rises or exposure times lengthen, and that for temperatures less than 1,200°K (1,700°F), ruthenium is released principally as ruthenium-oxide.

TRIGA fuel is a uranium-zirconium-hydrogen alloy that burns spontaneously in air at temperatures above 925°K or 1,205°F (Benedict, 1981). Because this combustion process is highly exothermic, if a severe fire heats a failed cask containing TRIGA spent nuclear fuel to temperatures above 925°K (1,205°F), air ingression due to convection or contraction of cask gases upon cask cooling would be expected to initiate spontaneous combustion of the fuel alloy, which should lead to substantially increased release from the fuel to the cask interior of krypton, cesium (most likely as cesium hydroxide, CsOH), and ruthenium [by conversion to volatile ruthenium oxide (RuO_4)].

Theoretical (NRC, 1988; SNL, 1989; GNS, 1993; Shaffer, 1994) and experimental (Babrauskas, 1986b; Nelsen, 1986; Gregory, 1987; Gregory, 1989; Schneider, 1989; Keltner, 1994) estimates of the thermal loads on casks produced by engulfing fires indicate that only engulfing fires with durations of an hour or more caused by the combustion of high-grade fuels (gasoline, jet fuel, diesel fuel) with an ample oxygen supply can raise the spent fuel contained in the casks to temperatures that approach 1,000°K (1,340°F). These studies also indicate that cask temperatures this high are not attained for fires of similar duration caused by poorer fuels (e.g., crude oil, wood). Thus, short duration fires involving low-grade fuels or mixtures of low and high-grade fuels are unlikely to raise cask temperatures high enough to significantly increase cesium vaporization or to cause substantial conversion of ruthenium to volatile ruthenium oxide. Fires involving high-grade fuels that are oxygen-starved because hold covers are closed or suppressed by the operation of fire fighting systems are also unlikely to result in elevated release fractions. Conversely, engulfing fires of about one hour duration that involve high-grade fuels could, for some accidents, be able to heat cask interiors to temperatures where (1) aluminum-uranium alloy fuels melt, (2) krypton, cesium, and ruthenium are easily vaporized, both from TRIGA fuel pellets and from melted aluminum-uranium alloy fuels, and (3) conversion of ruthenium to ruthenium oxide is substantial, if either fuel is exposed to air. The impact of these high temperature affects on the accident severity category 5 and 6 release fractions are discussed in the following paragraphs.

Accident Severity Category 5

When fuel temperatures remain below 900°K (1,160°F), that is, below the ignition point of TRIGA fuel in air and the melting point of aluminum-uranium alloy fuels, the release fractions from TRIGA fuel should be similar to that from uranium dioxide fuels. Also, the releases from aluminum-uranium alloy fuels should be very small, perhaps negligible, since diffusion in the metal plates from which the fuel is fabricated will be too slow to cause significant release to the cask, much less to the environment.

When research reactor fuels are heated significantly above 900°K (1,160°F) the release to the cask from TRIGA fuel pellets and from melted aluminum-uranium alloy fuels of krypton, volatile cesium, and ruthenium should be substantial (Cubicoitti, 1984; Cordfunke, 1990). Once released to the cask interior, transport of these fission products from the cask to the environment (past the failed cask seal) will only be efficient when the gases in the cask expand significantly due to heating of the cask to temperatures well above 900°K (1,160°F). For example, if melting of an aluminum-uranium alloy fuel at 923°K (1,202°F) causes essentially all of the krypton trapped in the fuel to be released to the cask interior, then further heating of cask gases to 1,023°K (1,382°F) by the fire will cause approximately 10 percent of the gases in the cask, including the krypton that escaped from the fuel to the cask interior, to be lost to the environment by expansion past the failed cask seal.

After the fire dies out, cooling of the hot cask will cause air to be drawn into the cask as the gases in the cask cool and contract. Thus, almost any hot fire of substantial duration will lead to substantial air ingression into a failed cask. Enhanced ruthenium release will then occur only if large amounts of fuel are exposed to the air, if this exposure occurs when the fuel is still hot enough to allow ruthenium to be oxidized to a volatile species, and if there is a transport process operating that causes the volatile ruthenium species to be released from the failed cask.

Because aluminum-uranium alloy fuels are molten at temperatures above 923°K (I,202°F), after air is drawn into the cask by cooling, if still molten, substantial exposure of fuel to air will occur, and therefore oxidation of ruthenium to ruthenium oxide will occur. However, after release to the cask interior, release of ruthenium to the environment can only occur by an inefficient transport mechanism, diffusion against the inflow of air since the cask is now cooling down. Thus, category 5 accident conditions, even those that reach unusually high temperatures, are not expected to significantly increase ruthenium release from aluminum-uranium alloy fuels, unless after dying down and drawing air into the cask, the fire flares up anew and again heats the cask to elevated temperatures whereupon gas expansion would transport some of the oxidized ruthenium vapors from the cask to the environment.

Because TRIGA fuel burns spontaneously and exothermically at temperatures above 900°K (1,160°F), if cask cooling draws air into a cask that contains TRIGA fuel while the fuel is still at such elevated temperatures, fuel burning will convert ruthenium to ruthenium oxide, and heating of the fuel and the cask gases by the highly exothermic oxidation of the hydride fuel will cause the oxidized ruthenium to vaporize, the cask gases to expand, and some of the vapors to be transported from the cask to the environment.

Accident Severity Category 6

During category 6 accidents, release from fuel to the cask interior of krypton, cesium, and ruthenium (after conversion to ruthenium oxide by exposure to air), occurs by the same processes that were just discussed for category 5 accidents. Gas convection through the failed cask is, by definition, substantial during category 6 accidents. Exposure of hot fuel to air causes substantial conversion of ruthenium to ruthenium oxide. Additionally, all vapors released from the fuel to the cask are transported from the cask to the environment by the convective flow of gases.

D5.3 Release Fractions for High-Temperature Events

The discussion presented in Section D5.2 indicates that, at elevated temperatures, release fractions for aluminum-uranium alloy and TRIGA fuels will differ substantially from those assumed in earlier studies of research reactor fuel transportation accidents for category 6 events and also for category 5 events that reach unusually high temperatures. To allow the consequences of such high-temperature events to be examined, the severity category strategy used in the base case analysis was modified by dividing both categories 5 and 6 into a low temperature and a high temperature category. Release fractions were then estimated for all of the categories in the modified strategy (categories 4, 5A and 5B, and 6A and 6B) and sensitivity calculations were performed to estimate the effects of the new release fractions on accident consequences.

Fire events that do not heat cask contents above 900°K (1,160°F) are placed in categories 5A and 6A. Fire events that heat cask contents above 900°K (1,160°F) are placed in categories 5B and 6B. Events that lead to seal failure are placed in category 4 and 5. Events that lead to cask failures (one medium hole, two or more small holes) that allow significant convective flow of gases through the failed cask are placed in category 6. Thus, transport of fission products released from fuel to the cask interior for category 5 events must be driven by expansion of cask gases due to heating of the cask by the fire, while for category 6 events, transport from the cask to the environments is efficiently driven by convective flow of gases through the cask. Table D5-1 summarizes the attributes of the modified severity categories.

Table D5-1 Category Attributes for the Modified Release Category Strategy

Category	*Cask Failure Mode*	*Transport from Cask*	*Temperature of Cask Contents*
5A	Seal Failure	Gas Expansion	T < 900°K
5B	Seal Failure	Gas Expansion	T > 900°K
6A	One medium hole, two small holes	Convection	T < 900°K
6B	One medium hole, two small holes	Convection	T > 900°K

Table D5-2 presents the release fractions developed for this modified strategy. Summarized in the footnotes of Table D5-2 are the basis for these release fractions. This table also compares the revised release fractions to the release fractions that were used in all of the base case calculations performed in this study. The sensitivity calculations that were performed using these new release fraction are described in Appendix D Section 5.4.3.2.

Table D5-2 Modified Release Fractions for Severity Categories 4, 5, and 6

Severity Category	*Study*	*Fuel*	*Chemical Element Group*			
			Krypton	*Cesium*	*Ruthenium*	*Particulate*
4	Base Case	Both	0.01	1.0x10-8	$1.0x10^{-8}$	$1.0x10^{-8}$
	Sensitivity Case	TRIGA	0.1	1.0x10-7	$1.0x10^{-7}$	$1.0x10^{-7}$
		Aluminum-uranium	$1.0x10^{-8}$	$1.0x10^{-8}$	$1.0x10^{-8}$	$1.0x10^{-8}$
5	Base Case	Both	0.1	$9.0x10^{-4}$	$1.0x10^{-6}$	$5.0x10^{-8}$
5A	Sensitivity Case	TRIGA	0.26	$1.0x10^{-3}$	$2.3x10^{-6}$	$1.3x10^{-6}$
		Aluminum-uranium	$1.3x10^{-7}$	$1.3x10^{-7}$	$1.3x10^{-7}$	$1.3x10^{-7}$
5B	Sensitivity Case	TRIGA	0.31	$1.1x10^{-2}$	$9.8x10^{-3}$	$3.3x10^{-4}$
		Aluminum-uranium	0.098	$9.8x10^{-3}$	$1.7x10^{-6}$	$3.0x10^{-7}$
6	Base Case	Both	0.11	$9.8x10^{-4}$	$4.2x10^{-5}$	$5.0x10^{-8}$
6A	Sensitivity Case	TRIGA	0.35	$1.6x10^{-3}$	$3.6x10^{-6}$	$2.0x10^{-6}$
		Aluminum-uranium	2.0x10-7	$2.0x10^{-7}$	$2.0x10^{-7}$	$2.0x10^{-7}$
6B	Sensitivity Case	TRIGA	1.0	0.3	0.3	0.01
		Aluminum-uranium	1.0	0.1	$1.6x10^{-5}$	$1.6x10^{-6}$

In order to develop release fraction values for the sensitivity study accident categories, several parameters need to be defined. These parameters are defined in Table D5-3.

Table D5-3 Definitions of Parameters used in the Sensitivity Study Accident Categories

F_{B1}	Fraction of fuel elements failed by the ships collision
F_{C1}	Release fraction for fission products from the fuel to the cask cavity due to the mechanical effects of the ship collision
F_{CE1}	Fraction of the fission products released to the cask cavity that escape from the cask in the absence of a fire
F_{FC2}	Fraction of fission products released from the fuel to cask cavity due to heating of the fuel from ambient temperature (T_a)to some elevated temperatures (T_f) less than 900°K
F_{B2}	Fraction of the fuel elements failed by burst rupture due to heating from T_a to T_f
F_{CE2}	1 - (T_a/T_f) where $T_a/T_f = V_a/V_f$ = the fraction of the gases in the cask at ambient temperature that remain in the cask after heating to Tf
F_{FC3}	Fraction of fission products released from the fuel to the cask cavity after the fuel has been heated to T_{FC3}(=temperature where aluminum-uranium fuel melts and TRIGA fuel burns if exposed to air
F_{B3}	The fraction of fuel elements failed by burst rupture due to heating from T_{FC3} to T_f
F_{CE3}	1 - (T_{FC3}/T_f) where $T_{FC3}/T_f = V_{TC3}/V_f$ = the fraction of the gases in the cask after heating to T_{FC3} that remain in the cask after further heating to T_f

Then, the release fraction (F_{R4}) for Category 4 events is given by

$$F_{R4} = F_{B1}F_{FC1}F_{CE1} \quad (1)$$

If the collision leads to a fire that heats the cask to elevated temperatures that do not exceed 900°K (1160°F) heating of the fuel may cause more fission products to be released from the fuel to the 900° cask cavity, and expansion of cask gases due to heating by the fire will cause a substantial fraction of the gas borne fission products to be transported from the cask interior through the failed cask seal to the environment. Thus, the release fraction (F_{R5A}) for Category 5A events is given by

$$F_{R5A} = F_{R4} + F_{B1}F_{FC1}(1 - F_{CE1})F_{CE2} + F_{B2}F_{FC2}F_{CE2} \quad (2)$$

If the collision has led to cask failures (a single medium hole or two smaller holes) that allow substantial convective flow through the cask, then all fission products released to the cask interior will be transported from the cask to the environment. Thus, the release fraction (F_{R6A}) for Category 6A events is given by

$$F_{R6A} = F_{R4} + F_{B1}F_{FC1}(1 - F_{CE1}) + F_{B2}F_{FC2} \quad (3)$$

as by definition $F_{CE2} = 1.0$ for Category 6 events.

The release fraction (F_{R58}) for fire events that heat the cask to temperatures above 900°K (1160°F), i.e. Category 5B events where Al-U alloy fuels melt and TRIGA fuel burns if exposed to oxygen is given by

$$F_{R5B} = F_{R5A} + F_{B3}F_{FC3}F_{CE3} \quad (4)$$

where

Again, if a Category 6 event has occurred, the release fraction (F_{R6B}) will be

$$F_{R6B} = F_{R6A} + F_{B3}F_{FC3} \quad (5)$$

since by definition $F_{CE3} = 1.0$ for Category 6 events.

The release fractions used in the base case assessment are the same as those (Wilmot 1981) developed for air-cooled casks for release of fission products from spent commercial UO_2 fuel for three processes: impact, burst, and oxidation. Base case Category 4 release fractions are the same as those developed by Wilmot for impact events involving air-cooled casks. Except for cesium, Category 5 release fractions are equal to the sum of Wilmot's release fractions for impact and burst, and Category 6 release fractions are equal to the sum of Wilmot's release fractions for impact, burst, and oxidation. For cesium, the base case uses release fractions that have been adjusted somewhat to reflect the effect of metallic fuel properties on cesium release. This information is used as the basis to derive several of the values for the parameters identified in Table D5-3.

For impact events, Wilmot uses $F_{B1} = 0.1$, $F_{FC1} = 0.2$ and $F_{CE1} = 0.5$ for krypton; and $F_{FC1} = 2x10^{-6}$ and $F_{CE1} = 0.05$ for cesium, ruthenium and particulates for release of fuel fines and thus the fission products trapped in the fines. For burst events, Wilmot assumes that $F_{B2} = 0.9$. Table D5-2 shows that the base case used values of 0.1, $9x10^{-4}$, $1x10^{-6}$, and $5x10^{-8}$, respectively, for the release fractions for krypton, cesuim, ruthenium, and particulates for Category 5 events. If Equation 2 is solved for F_{FC2} using the base case values for Category 5 events for F_{R5A} and Wilmot's values for F_{B1}, F_{B2}, F_{FC1}, and F_{CE1}, then the following values are obtained for F_{CE2}: 0.15 for krypton, $1.6x10^{-3}$ for cesium, $1.6x10^{-6}$ for ruthenium, and 0 for particulates.

The analysis presented in Attachment D4 of cask damage caused by impact and crush concludes that damage will not result from the impacts forces experienced by cask during ship collisions, and that if the cask is subjected to crush forces, they will always be large enough to fail all of the fuel elements contained in the cask. Therefore, $F_{B1} = F_{B2} = F_{B3} = 1.0$.

To facilitate comparison of the new release fractions developed here to the release fractions used in the base case, the release fractions for the cesium, ruthenium, and particulate chemical element groups for Category 4 events were forced to be the same as the value used in the base case. Although aluminum-uranium alloy fuels should have very little, if any, fuel fines associated with the metal plates from which the fuel bundles are fabricated, to achieve this equivalence, it was assumed that aluminum-uranium alloy fuels have amounts of fuel fines one-tenth of those assumed by Wilmot for uranium dioxide fuels. Thus, for aluminum-uranium alloy fuels, $F_{FC1} = 2x10^{-7}$ and therefore, because $F_{B1} = 1.0$, $F_{R4} = 2x10^{-8}$, which is the value that the base case used for the realease fraction for cesium, ruthenium, and particulate for Category 4 events.

Reasonable choices for F_{FC3} for aluminum-uranium alloy fuels, that is, for release to the cask cavity upon melting for the alloy fuel, are 1.0 for krypton, 0.1 for cesium, $1.6x10^{-5}$ for ruthenium, and $1.6x10^{-6}$ for particulate, where ruthenium release from metallic fuel upon melting has been assumed to be ten times the ruthenium release from commercial uranium dioxide fuel estimated for Category 5A events (the value of F_{FC2} for ruthenium release from uranium dioxide fuel for Category 5A events. Particulate release has been assumed to be about the same as ruthenium release from uranium dioxide fuel for Category 5A events and about ten times larger than particulate releases from aluminum-uranium alloy fuels for Category 4 events (the value of F_{FC1} for particulate release from aluminum-uranium alloy fuels for Category 4 events), as the melting of aluminum uranium alloy fuels due to heating of the cask by a fire is not likely to be violent.

Reasonable choices for F_{FC3} for TRIGA fuel are 1.0 for krypton; 0.3 for cesium; 0.3 for ruthenium, since burning of the fuel means that ruthenium will be converted to a volatile oxide by exposure to air; and 0.01 for particulate, on the assumption that the high exothermicity of the combusion process will cause one percent of the fuel mass to be aerosolized. For Category 5B, these values were decreased by a factor of 3, because air can only enter the cask due to cooling, which will not lead to fuel burning if the fuel cools

below 900°K, (1160°F). Even if burning does occur, efficient transport of fission products released by the burning from the cask to the environment can occur only by gas expansion caused by the heat released by fuel burning. Thus, the cask atmosphere must breath (pass through several cooling/burning cycles), if significant quantities of fission products are to be released by fuel burning, when there is no convective flow of air through the cask.

Table D5-4 lists the parameters used in Equations 1 through 5, and presents the values used for each parameter to calculate values for the release fractions F_{R4}, F_{R5A}, F_{R6A}, F_{R5B}, and F_{R6B}. For the four EA5 results for UO_2 fuel, the result calculated is the F_{FC2} value, not the F_{R5} value, which is an input and is set equal to the value used in the EA for the indicated element group.

Table D5-4 Parameters Used to Generate High Temperatures Fire Sensitivity Study Release Fractions

Accident Category	*Fuel*	*Element*	*Parameter*[1]									
			F_{B1}	F_{C1}	F_{CE1}	F_{B2}	F_{FC2}	T_a	T_{FC3}	T_f	F_{FC3}	F_R
Base Case 4		krypton	0.1	0.2	0.5							0.01
		all others	0.1	$2x10^{-6}$	0.05							$1x10^{-8}$
Base Case 5		krypton	0.1	0.2	0.5	0.9	0.15	300		800		0.1
		cesium	0.1	$2x10^{-6}$	0.05	0.9	$1.6x10^{-3}$	300		800		$9x10^{-4}$
		ruthenium	0.1	$2x10^{-6}$	0.05	0.9	$1.6x10^{-6}$	300		800		$1x10^{-6}$
		particulates	0.1	$2x10^{-6}$	0.05	0.9	0.0	300		800		$5x10^{-8}$
Sensitivity Study 4	Al-U	all	1.0	$2x10^{-7}$	0.05							$1x10^{-8}$
	TRIGA	krypton	1.0	0.2	0.5							0.1
	TRIGA	all others	1.0	$2x10^{-6}$	0.05							$1x10^{-7}$
Sensitivity Study 5A	TRIGA	krypton	1.0	0.2	0.05	1.0	0.15	300		800		0.26
	TRIGA	cesium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-3}$	300		800		0.001
	TRIGA	ruthenium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-6}$	300		800		$2.3x10^{-6}$
	TRIGA	particulates	1.0	$2x10^{-6}$	0.05	1.0	0.0	300		800		$1.3x10^{-6}$
	Al-U	all	1.0	$2x10^{-7}$	0.05	1.0	0.0	300		800		$1.3x10^{-7}$
Sensitivity Study 5B	TRIGA	krypton	1.0	0.2	0.5	1.0	0.15	300	923	1023	0.33	0.31
	TRIGA	cesium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-3}$	300	923	1023	0.1	0.011
	TRIGA	ruthenium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-6}$	300	923	1023	0.1	0.0098
	TRIGA	particulates	1.0	$2x10^{-6}$	0.05	1.0	0.0	300	923	1023	0.0033	$3.3x10^{-4}$
	Al-U	krypton	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	1.0	0.098
	Al-U	cesium	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	0.1	0.0098
	Al-U	ruthenium	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	$1.6x10^{-5}$	$1.7x10^{-6}$
	Al-U	particulates	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	1.6x10-6	$3.0x10^{-7}$
Sensitivity Study 6A	TRIGA	krypton	1.0	0.2	0.5	1.0	0.15	300		800		0.35
	TRIGA	cesium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-3}$	300		800		0.0016
	TRIGA	ruthenium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-6}$	300		800		$3.6x10^{-6}$
	TRIGA	particulates	1.0	$2x10^{-6}$	0.05	1.0	0.0	300		800		$2.0x10^{-6}$
	Al-U	all	1.0	$2x10^{-7}$	0.05	1.0	0.0	300		800		$2.0x10^{-7}$
Sensitivity Study 6B	TRIGA	krypton	1.0	0.2	0.5	1.0	0.15	300	923	1023	1.0	1.0
	TRIGA	cesium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-3}$	300	923	1023	0.3	0.3
	TRIGA	ruthenium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-6}$	300	923	1023	0.3	0.3
	TRIGA	particulates	1.0	$2x10^{-6}$	0.05	1.0	0.0	300	923	1023	0.01	0.01
	Al-U	krypton	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	1.0	1.0
	Al-U	cesium	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	0.1	0.1
	Al-U	ruthenium	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	$1.6x10^{-5}$	$1.6x10^{-5}$
	Al-U	particulates	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	$1.6x10^{-6}$	$1.8x10^{-6}$

Inspection of Table D5-2 allows the size of the new release fractions developed for aluminum-uranium alloy and TRIGa fuels to be conpared to the release fractions used in the base case calculations. Table D5-5 summarizes these comparisons.

Table D5-5 Relative Size of the Sensitivity Study Release Fractions Compared to the Base Case Release Fractions Used to Perform the Base Case Calculations

Fuel	*Severity Category*		*Size of New Sensitivity Study Release Fractions Compared to Base Case Release Fractions*
	Sensitivity Study	*Base Case*	
Aluminum-Uranium Alloy	4	4	About the same (krypton much smaller)
	5A	5	Smaller (cesium 10,000 times smaller)
	5B	5	Cesium 10 times larger
	6A	6	Cesium 5000 times smaller
	6B	6	Cesium 100 times larger
TRIGA	4	4	10 times larger
	5A	5	About the same
	5B	5	Cesium 10 times larger
	6A	6	About the same
	6B	6	Cesium 300 times larger

D5.4 Probability of High-Temperature Events

Data on the temperatures of real ship fires is nearly non-existent. Only one of the five severe fires identified by searching the Lloyd's of London data (Lloyd's, 1991) attained temperatures where steel beams buckled due to thermal stress. Carbon steels begin to soften at about 475°K (395°F) and have lost 90 percent of their strength at about 925°K (1,205°F). Thus, buckling of ship structures due to thermal stress might be expected to occur at about 700°K (800°F), the midpoint of this temperature range, which suggests that one severe fire in five attains temperatures at about 700°K (800°F) and also that P_{T900K} is less than 0.2. Due to the lack of ship board fire temperature data the an attempt has been made to estimate the likelihood of a fire that exceeds 900°K (1160°F).

A shipboard fire can heat the contents of a transportation cask to temperatures above 900°K (1,160°F) only if three conditions are met: (1) the fire must consume a high quality fuel such as gasoline or jet fuel, (2) enough fuel must be available to cause the fire to burn for an hour or more, and (3) the fire cannot be smothered by lack of air or the operation of fire suppression systems. Most severe ships fires involve the burning of the ship's own fuel (bunker or diesel fuel) or of crude oil, when the collision that leads to the fire involves an oil tanker. Thus, P_{T900K}, the chance that a ship fire can heat the contents of a transportation cask to temperatures above 900°K (1,160°F), can be estimated as follows:

$$PT>900K = P_{good\ fuel} \times P_{enough\ fuel} \times P_{enough\ oxygen}$$

Diesel fuel, bunker fuel, and crude oil all have peak flame temperatures that exceed 900°K or 1,160°F (Mudan, 1988), and most polymeric materials (e.g., plastics, wood) have peak flame temperatures of about 1,200°K or 1,700°F (Babrauskas, 1986a). Since fires in cargo holds should behave like enclosure fires, hold fires that burn wood could attain peak temperatures of about 1200°K (1,700°F), if post flashover conditions are attained (Babrauskas, 1986a). So, the fuels and solid materials that are likely to be involved in shipboard fires in cargo holds should be able to heat cask contents to temperatures significantly above

900°K (1,160°F), provided the fire burns long enough and isn't suppressed by lack of oxygen or the operation of fire suppression systems. Thus, $P_{good\ fuel}$, the chance that a long burning hold fire is supported by the burning of a good fuel, is not likely to be small and is here assumed to be 0.9.

The review of ship fires prepared by the French Bureau Veritas for the International Maritime Organization (IMO, 1992) contains data on ship fire durations. Most ship fires (70 to 80 percent) do not burn for an hour. However, most severe ship fires (95 percent) burn for more than an hour. Therefore, the chance that a severe fire involves enough fuel to burn for an hour or more, $P_{enough\ fuel}$, is assumed to be 0.95.

Figure D5-1 presents an event tree for oxygen availability during fires in cargo holds, and is used to estimate $P_{enough\ oxygen}$. The tree shows that most cargo hold fires will be partially starved for oxygen for two reasons, because hold covers will be closed when the fire starts, or will be deliberately closed after it starts in order to smother the fire; or because CO_2 fire suppression systems are installed in the hold and operate successfully. To quantify the event tree provided in Figure D5-1, it was necessary to derive the probability of these two events. The probability that a cargo hold is closed during a collision can be estimated using the following relationship.

$$P_{closed} = 1 - P_{open}$$

where

$$P_{open} = P_{all\ not\ closed} P_{worked} P_{location} \sum (N_i P_{deck}).$$

The derivation of each of the terms in this relationship is described in the following paragraphs.

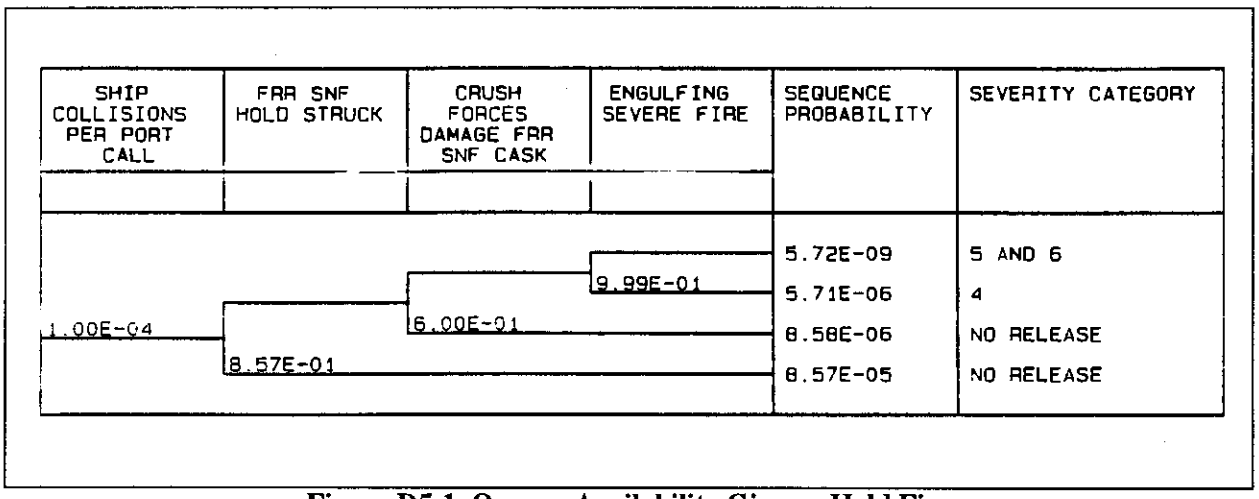

Figure D5-1 Oxygen Availability Given a Hold Fire

Cargo hold covers are normally closed except during loading of unloading of cargo. Thus, if a typical port call takes approximately three days (half a day entering the port and docking, two days anchored at the dock with two-thirds of that time, (two eight-hour shifts per day) spent loading and unloading cargo, and one-half day leaving the port) then all holds will be closed about half of the time while a ship is in port. Conversely, about half of the time at least one hold will be open. Thus $P_{all\ not\ closed}$=½ or 0.5.

When a break-bulk freighter like the seven-hold ship used in these analyses is being loaded or unloaded, usually three or four holds are being worked at any given time. Thus, when the ship is being loaded or unloaded, P_{worked}, the probability that a given hold is being worked is ½ or 0.5.

The break-bulk freighter used in these analyses has seven holds. Five of these holds contain three cargo decks, one contains four cargo decks, and one contains only two cargo decks. Thus, there are 21 possible deck locations for a spent fuel cask in this typical ship. Accordingly, $P_{location}$, the chance that a spent nuclear fuel cask has been loaded onto a given deck in one of the seven holds is 0.048.

All hold openings have covers, not just the opening in the main deck through which the hold is loaded and unloaded, but also the openings in the cargo decks within each hold. When a deck in a cargo hold is being loaded or unloaded, all openings above that deck must be open and the opening in the deck and all openings in lower decks are normally closed. Thus, while a hold is being worked, upper decks in that hold will be open to outside air more often than lower decks. For example, for a three-deck hold, while the hold is being worked, the upper deck will always be open to the outside air, the second deck will be open about two-thirds of the time, and the lowest deck will be open about one-third of the time. Thus, if N is the number of holds with two, three, or four decks, and P_{deck} is the probability that deck i in a hold is open to outside air while that hold is being worked, then P_{closed}, the chance that an engulfing fire is partially starved for oxygen because there is a cargo deck or main deck hold cover in place between the fire and the outside air will be:

$$P_{closed} = 1 - \{(0.5)(0.5)(0.048)([5(1 + \tfrac{2}{3} + \tfrac{1}{3})] + [1(1 + \tfrac{3}{4} + \tfrac{1}{2} + \tfrac{1}{4})] + [1(1 + \tfrac{1}{2})])\}$$

$$= 0.833$$

The ORI study (ORI, 1981a) found that over half (60 percent) of all cargo ships are equipped with fire detectors and CO_2 fire suppression systems. Because CO_2 fire suppression systems are not complicated, they should operate reliably on demand most of the time. To be conservative, failed operation during one of five fire events is assumed.

Using this data, the event tree in Figure D5-1 can be quantified to determine the probability of the event $P_{enough\ oxygen}$. Two branches of the oxygen availability tree lead to the outcome "enough air." The probabilities of these two branches sum to 0.087. Thus, 0.09 is a reasonable estimate for $P_{enough\ oxygen}$, the chance that a fire has adequate oxygen available to burn freely and generate maximum heat loads.

Combining the probability estimates for $P_{good\ fuel}$, $P_{enough\ fuel}$, and $P_{enough\ oxygen}$ allows $P_{T900\ K}$ to be estimated as follows:

$$P_{T900\ K} = P_{good\ fuel} \times P_{enough\ fuel} \times P_{enough\ oxygen}$$

$$= 0.9 \times 0.95 \times 0.09 = 0.077$$

Rounding to the next order of magnitude yields a conservative estimate of 0.1 for the chance that a severe engulfing fire with a duration of at least an hour will heat the contents of a transportation cask engulfed by the fire to temperatures significantly higher than 900°K (1,160°F).

D5.5 Probability of Convective Flow through the Failed Cask

Non-uniform heating of the cask during engulfing fires is expected to produce substantial flow of gases through the cask if two or more small holes or one medium hole have been produced in the cask by the ship collision. Because transportation casks bottoms and lid seats are welded to the cylindrical shell of the cask using full-penetration welds that are as strong or stronger than the parent material, when the cask shell is subjected to a severe stress (e.g, high impact or crush forces), the cask shell should yield before the welds fail. In fact, extra-regulatory 60 mph drop tests produced large plastic strains in the cylindrical shell of the test cask without failing its welds (Ludwigsen and Ammerman, 1995). Thus, during a ship collision, crush forces should collapse the cask walls inward without producing catastrophic failure of the lid, its seat, or the welds that attach the seat or the bottom of the cask to the cask walls. Therefore, an unusual configuration of cargo and/or deformed ship structures must be produced during the ship collision in order to subject the cask to forces that will produce failures substantially worse than failure of the lid seal. Either the lid seat must be bent significantly, or at least two penetrations must break, or the cask walls must be sheared or punctured. Although data for such failures is lacking, because casks normally do not fail by these mechanisms, the probability that a failure substantially worse than seal failure occurs is assumed to be no larger than 0.1.

D5.6 Severity Category Event Trees

Figures D5-2 and D5-3 present event trees that represent the sequence of events that lead to category 4, 5A, 5B, 6A, and 6B releases from transportation casks due to ship collisions. After rounding to the nearest integer, Figure D5-3 shows that these categories have the probabilities per port call provided in Table D5-4.

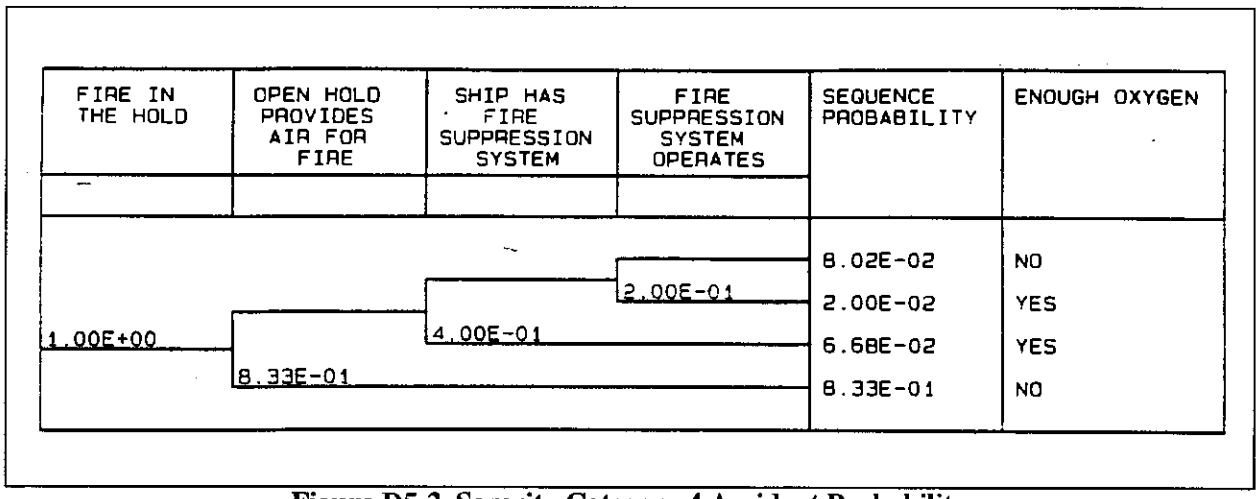

Figure D5-2 Severity Category 4 Accident Probability

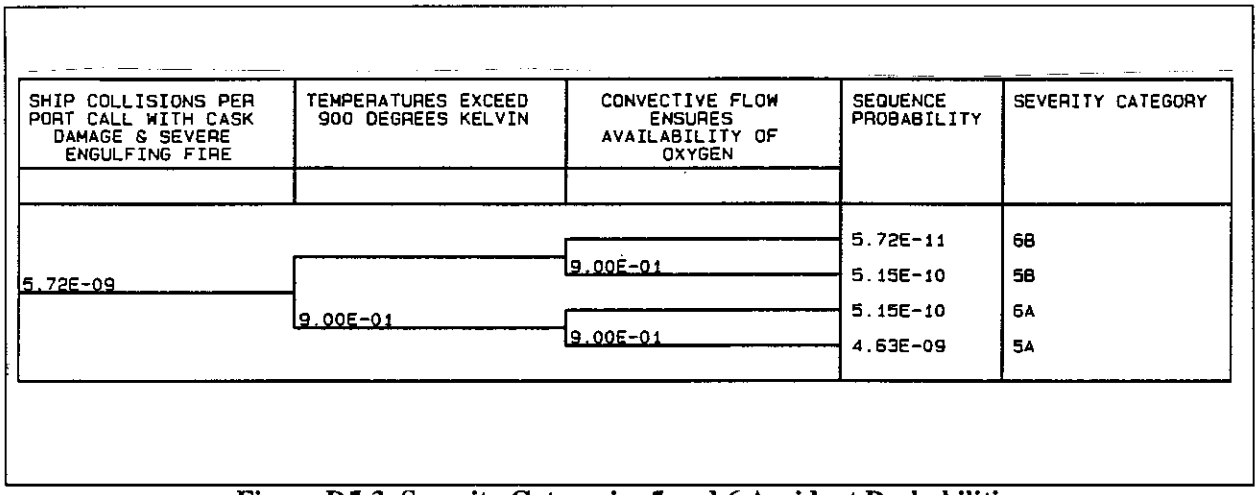

Figure D5-3 Severity Categories 5 and 6 Accident Probabilities

Table D5-4 Sensitivity Study Accident Severity Category Probabilities

Severity Category	*Probability Per Port Call*
4	$6x10^{-6}$
5A	$5x10^{-9}$
5B	$5x10^{-10}$
6A	$5x10^{-10}$
6B	$6x10^{-11}$

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FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix E
Evaluation of Human Health Effects of Overland Transportation

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Appendix E
Evaluation of Human Health Effects of Overland Transportation

E.1 Introduction

The overland transportation of any commodity involves a risk to both transportation crew members and members of the public. This risk results directly from transportation-related accidents and indirectly from the increased levels of pollution from vehicle emissions, regardless of the cargo. The transportation of certain materials, such as hazardous or radioactive waste, can pose an additional risk due to the unique nature of the material itself. In order to permit a complete appraisal of the environmental impacts of the proposed action and alternatives, the human health risks associated with the overland transportation of foreign research reactor spent nuclear fuel have been assessed.

This appendix provides an overview of the approach used to assess the human health risks that may result from the overland transportation of foreign research reactor spent nuclear fuel. The appendix includes discussion of the scope of the assessment, analytical methods used for the risk assessment (i.e., computer models), important assessment assumptions, determination of potential transportation routes, and presents the results of the assessment. In addition, to aid in the understanding and interpretation of the results, specific areas of uncertainty are described, with an emphasis on how the uncertainties may affect comparisons of the alternatives.

The approach used in this appendix is modeled after that used in the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Final Environmental Impact Statement (SNF&INEL Final EIS) (DOE, 1995). The SNF&INEL Final EIS did not perform as detailed an analysis on the specific actions taken for foreign research reactor spent nuclear fuel because of the breadth necessary to analyze the entire spent fuel management program. However, the fundamental assumptions used in this analysis are consistent with those used in the SNF&INEL Final EIS (DOE, 1995), and the same computer codes and generic release and accident data are used.

The risk assessment results are presented in this appendix in terms of "per-shipment" risk factors, as well as for the total risks associated with each alternative. Per-shipment risk factors provide an estimate of the risk from a single spent nuclear fuel shipment between a specific origin and destination. They are calculated for all possible origin and destination pairs for each spent nuclear fuel type. The total risks for a given alternative are found by multiplying the expected number of shipments by the appropriate per-shipment risk factors. This approach provides maximum flexibility for determining the risks for a large number of potential alternatives.

E.2 Scope of Assessment

The scope of the overland transportation human health risk assessment, including the alternatives and options, transportation activities, potential radiological and nonradiological impacts, and transportation modes considered, is described below. Additional details of the assessment are provided in the remaining sections of the appendix.

Proposed Action and Alternatives: The transportation risk assessment conducted for this EIS estimates the human health risks associated with the transportation of spent nuclear fuel for a number of management and implementation alternatives. The alternatives differ primarily in the number and location of possible ports of entry and Phase I management sites (storage sites that would be used until a repository was ready). The alternatives considered are described in detail in Chapter 2 of this EIS.

For transportation assessment purposes, each option is defined as an individual or pair of U.S. Department of Energy (DOE) sites used for initial management and an individual or pair of DOE sites used for final interim management. The transportation risk assessment determines risks by considering the total amount of spent nuclear fuel shipped over each representative route. The assessment takes into account differences in the physical and radiological properties of spent nuclear fuel types and characteristics of the potential routes to and between sites.

A large number of potentially applicable marine ports of entry and Canadian border crossings, including commercial and military ports on the Atlantic, Pacific, and Gulf of Mexico coasts are considered in this risk analysis. The port selection process is described in Appendix D. The Canadian border crossing points are representative points based on a qualitative judgment of previously used shipment routes (NRC, 1993). The alternatives in this EIS define the acceptance of the fuel, while the SNF&INEL Final EIS (DOE, 1995) alternative selected defines the DOE site or sites that would receive foreign research reactor spent nuclear fuel; and the options identified in this EIS define the various ways in which the foreign research reactor spent nuclear fuel could be handled to meet the SNF&INEL Final EIS selected alternative.

Transportation-Related Activities: The transportation risk assessment is limited to estimating the human health risks incurred during the overland transportation of spent nuclear fuel for each alternative. The risks to workers or to the public during spent nuclear fuel loading, unloading, and handling prior to or after shipment are not included in the overland transportation assessment, they are addressed in Appendices C and D. Similarly, the transportation risk assessment does not address possible impacts from increased transportation levels on local traffic flow, noise levels, or infrastructure.

Radiological Impacts: For each alternative, radiological risks (i.e., those risks that result from the radioactive nature of the spent nuclear fuel) are assessed for both incident-free (i.e., normal) and accident transportation conditions. The radiological risk associated with incident-free transportation conditions would result from the potential exposure of people to external radiation in the vicinity of a loaded shipment. The radiological risk from transportation accidents would come from the potential release and dispersal of radioactive material into the environment during an accident and the subsequent exposure of people through multiple exposure pathways (i.e., exposure to contaminated ground or air, or ingestion of contaminated food).

All radiological-related impacts are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent (10 CFR Part 20), which is the sum of the effective dose equivalent (EDE) from external radiation exposure and the 50-yr committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of person-rem for collective populations and rem for individuals. The impacts are further expressed as health risks in terms of latent cancer fatalities (LCF) and cancer incidence in exposed populations. The health risk conversion factors (expected health effects per dose absorbed) were derived from International Commission on Radiological Protection Publication 60 (ICRP, 1991).

Nonradiological Impacts: In addition to the radiological risks posed by overland transportation activities, vehicle-related risks are also assessed for nonradiological causes (i.e., related to the transport vehicles and not the radioactive cargo) for the same transportation routes. The nonradiological transportation risks are

independent of the radioactive nature of the cargo and would be incurred for similar shipments of any commodity. The nonradiological risks are assessed for both incident-free and accident conditions. Nonradiological risks during incident-free transportation conditions would be caused by potential exposure to increased vehicle exhaust emissions. The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities unrelated to the shipment cargo. State-specific transportation fatality rates are used in the assessment. Nonradiological risks are presented in terms of estimated fatalities.

Transportation Modes: All spent nuclear fuel shipments have been assumed to take place either by truck or rail transportation modes. Per-shipment risk factors are presented separately for truck and rail modes. For the alternatives, risks have been calculated separately for all truck and all rail options, although the actual transportation operation for a selected alternative may involve a combination of the two modes.

Barge transport has certain disadvantages. First, barge transport limits site and port selection for both the SNF&INEL Final EIS and this EIS to Savannah River Site (available both phases) and Hanford Site (available in Phase 2 only). These sites are only served by the ports of Savannah, GA and Portland, OR, respectively. Additionally, barge transportation would require additional intermodal transfers at the port and at the site. At the port, the cask would be removed from the ocean-going vessel and moved by truck to the barge terminal for loading onto a barge. When the barge arrives at the DOE site, the cask would have to be moved to a truck for transport across the site to the receiving basin. Other reasons for not using barge transportation include DOE's lack of extensive experience in shipping casks via barge, the lack of alternative routes, and low speeds. DOE, however, has performed a scoping analysis of barge transportation to assess its relative impacts.

Receptors: Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck and rail crew members involved in the actual overland transportation of spent nuclear fuel. The general public includes all persons who could be exposed to a shipment while it is moving or stopped en route. Potential risks are estimated for the collective populations of exposed people, as well as for the hypothetical maximally exposed individual (MEI). The collective population risk is a measure of the radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing various alternatives.

Cumulative Impacts: The cumulative impacts of the transportation of foreign research reactor spent nuclear fuel are calculated and presented as a relative proportion of those described in the SNF&INEL Final EIS (DOE, 1995). The collective dose to the general population and workers is the measure used to quantify cumulative transportation impacts.

E.3 Spent Nuclear Fuel Packaging and Representative Shipment Configurations

Regulations that govern the transportation of radioactive materials are designed to protect the public from the potential loss or dispersal of radioactive materials as well as from routine radiation doses during transit. The primary regulatory approach to ensure safety is through the specification of standards for the packaging of radioactive materials. Because packaging represents the primary barrier between the radioactive material being transported and radiation exposure to the public and the environment, packaging requirements are an important consideration for the transportation risk assessment. Regulatory packaging requirements are discussed briefly below and in Chapter 5. In addition, the representative packaging and shipment configurations assumed for this EIS are described.

E.3.1 Packaging Overview

Although several Federal and State organizations are involved in the regulation of radioactive waste transportation, primary regulatory responsibility resides with the U.S. Department of Transportation and the U.S. Nuclear Regulatory Commission (NRC). All transportation activities must take place in accordance with the applicable regulations of these agencies specified in 49 Code of Federal Regulations (CFR) Part 173 and 10 CFR Part 71.

Transportation packaging for radioactive materials must be designed, constructed, and maintained to ensure that the packages will contain and shield their contents during normal transport conditions. For more highly radioactive material, such as spent nuclear fuel, they must contain and shield their contents in the event of severe accident conditions. The type of packaging used is determined by the total radioactive hazard presented by the material within the packaging. The basic types of packaging required by the applicable regulations are designated as Type A, Type B, or "strong and tight".

"Strong and tight" packages are designed such that no radioactive material will leak or be released during transportation. They can only be used for low-specific-activity material. Type A packaging must withstand the conditions of incident-free transportation without the loss or dispersal of the radioactive contents. Incident-free transportation refers to all conditions of transportation except those that result from accidents or sabotage. Approval of Type A packaging is achieved by demonstrating that the packaging can withstand specified test conditions which are intended to simulate incident-free transportation conditions. Type A packaging, typically a 55-gallon (gal) drum or standard waste box, is commonly used to transport wastes having low activities of radioactive material.

The transportation of spent nuclear fuel requires the use of Type B packaging. In addition to meeting the standards for Type A packaging, Type B packaging must provide a high degree of assurance that even in severe accidents the integrity of the package will be maintained with essentially no loss of the radioactive contents or serious impairment of the shielding capability. Type B packaging must satisfy stringent testing criteria specified in 10 CFR Part 71. The testing criteria were developed to simulate severe hypothetical accident conditions, including impact, puncture, fire, and water immersion. The massive casks used to transport spent nuclear fuel represent the most widely recognized Type B packaging.

For risk assessment purposes, it is important to note that all packaging of a given type is designed to meet the same performance criteria. Therefore, two spent nuclear fuel casks of different designs would be expected to perform similarly during incident-free and accident transportation conditions. The specific cask selected, however, will determine the total number of shipments necessary to transport a given quantity of spent nuclear fuel.

External radiation allowed to escape from a package must be below specified limits that minimize the exposure of the handling personnel and general public. The foreign research reactor spent nuclear fuel shipments would be handled only by the shipper and the receiver, an arrangement referred to as an "exclusive-use" shipment. For these types of shipments, the external radiation dose rate during normal transportation conditions must be maintained below the following limits of 49 CFR Part 173:

- 10 mrem per hr at any point 2 meters (m) (6.6 ft) from the vertical planes projected by the outer lateral surfaces of the transport vehicle (referred to as the regulatory limit throughout this document), and
- 2 mrem per hr in any normally occupied position in the transport vehicle.

Although additional restrictions apply to package surface radiation levels, these restrictions are not important for the transportation radiological risk assessment.

The NRC recently issued revised regulations, 10 CFR Part 71, governing the transportation of radioactive materials. These regulations become effective on April 1, 1996 (NRC, 1995). The revised regulations conform with those of the International Atomic Energy Agency and current legislative requirements. The revised regulations affecting "Type B" casks require that a spent nuclear fuel transportation cask with activity greater than one million curies (Ci) be designed and 290 psi, or immersion in 200 m (656 ft) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask.

E.3.2 Packaging and Representative Shipment Configurations for Foreign Research Reactor Spent Nuclear Fuel

To conduct the overland transportation risk assessment, assumptions must be made concerning the types of packaging, transport vehicles, and shipment capacities that could be used for future spent nuclear fuel shipments. In all cases, it is assumed that spent nuclear fuel would be characterized, treated, packaged, and labeled in accordance with applicable regulations prior to shipment.

The transportation of all foreign research reactor spent nuclear fuel would take place in casks certified by foreign competent authorities and revalidated by Department of Transportation in accordance with 49 CFR 173. In addition, it is assumed that only exclusive-use vehicles would be used. Highway transportation is assumed to take place by legal weight heavy-haul combination (tractor-trailer) trucks. Rail transportation is assumed to take place by regular freight train service.

E.3.3 Description of Transportation Activities

The proposed action could involve transporting foreign research reactor spent nuclear fuel from the ports of entry (both marine ports and Canadian border crossings) to DOE sites, and could involve transporting foreign research reactor spent nuclear fuel between DOE sites. The interim management site or sites for the foreign research reactor spent nuclear fuel in the United States have been determined on the basis of the SNF&INEL Final EIS (DOE, 1995).

In this section, the assumptions and logic used to model the transportation requirements for the basic implementation of Management Alternative 1 of the proposed action are described. In general, the same assumptions are used to analyze the management and implementation alternatives. Therefore, the transportation requirements for management and implementation alternatives will be described in relation to the basic implementation.

Certain assumptions are required in order to simply and consistently describe the manner in which foreign research reactor spent nuclear fuel would be transported to the sites. The shipments were divided into east coast and west coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East and parts of South and Central America were designated as east coast shipments, and all others were designated as west coast shipments. Shipments from Canada were assumed to enter the United States from either an eastern or western point of entry, depending on the Canadian point of origin. Under these assumptions, the east coast would receive approximately 535 cask shipments and the west coast approximately 186 cask shipments. Approximately 116 shipments from Canada would arrive in the eastern United States.

Regarding foreign research reactor spent nuclear fuel transportation, the SNF&INEL Final EIS (DOE, 1995) analyzes the use of any one of five candidate sites and seven distinct combinations of sites. Eight of the alternatives involve sites that could not be ready to accept spent nuclear fuel at the onset of the foreign research reactor spent nuclear fuel program. Therefore, a two-phased approach is assumed using one or both of the sites that are ready to accept spent nuclear fuel (Savannah River Site and Idaho National Engineering Laboratory) as a near-term management location. Phase 1 is defined, for the purposes of analyzing transportation, as the period of time in which shipments of foreign research reactor spent nuclear fuel are transported to a near-term management site. For analytical purposes, Phase 1 is assumed to last from the beginning of 1996 to the beginning of 2006.

The amount of fuel that would arrive in Phase 1 versus Phase 2 cannot be precisely determined at this time. In order to proceed with the risk analysis, it is necessary to make assumptions based on the available information. The total number of casks that would be required to transport the 22,700 spent fuel elements is estimated to be 837, per Appendix B. The split between Phase 1 and Phase 2 depends on the rate at which casks are received and the time the Phase 2 site(s) is ready to receive fuel. For calculational purposes, the casks are assumed to arrive at a uniform rate, and the Phase 2 site(s) is assumed to be ready 10 years after the implementation of the policy.

The disposition of foreign research reactor spent nuclear fuel during Phase 1 is analyzed in this EIS. Logically, Phase 1 could entail any one of four options: A) splitting foreign research reactor spent nuclear fuel by fuel type [TRIGA (which stands for Training, Research, and Isotope reactors built by General Atomic) to Idaho National Engineering Laboratory and Aluminum-based to Savannah River Site], B) splitting the spent nuclear fuel geographically by port of entry, C) transporting all spent nuclear fuel to Idaho National Engineering Laboratory, or D) transporting all spent nuclear fuel to Savannah River Site. Not all Phase 1 strategies are consistent with all Phase 2 strategies.

Phase 2 begins when Oak Ridge Reservation, Hanford Site, or Nevada Test Site would be ready to receive fuel from ports and, when applicable, from a DOE site being used for near-term management. In all cases, Phase 2 is dependent on decisions based on the SNF&INEL Final EIS (DOE, 1995). During Phase 2, all foreign research reactor spent nuclear fuel arriving at ports of entry would be transported to the appropriate site. Additionally, intersite shipments from the near-term management site could also be arriving at the SNF&INEL Final EIS selected site(s).

The following is a description of the shipping program, organized by SNF&INEL Final EIS (DOE, 1995) alternatives:

No Action - DOE cannot accept foreign research reactor spent nuclear fuel under this alternative.

Decentralization - Foreign research reactor spent nuclear fuel arriving on the east coast would be transported to Savannah River Site, and foreign research reactor spent nuclear fuel arriving on the west coast would be transported to Idaho National Engineering Laboratory. Since both Idaho National Engineering Laboratory and Savannah River Site are capable of receiving fuel in late 1995, there is no need for a two-phase program or intersite shipments. The total number of shipments for this alternative would be approximately 837. Savannah River Site would receive 651 casks from the east, and Idaho National Engineering Laboratory would receive 186 casks from the west. The transportation under this alternative is illustrated in Figure E-1. No intersite shipment would be anticipated under this single-phased alternative.

Figure E-1 Decentralization: Spent Nuclear Fuel to Idaho National Engineering Laboratory and Savannah River Site

1992-1993 Planning Basis - The SNF&INEL Final EIS (DOE, 1995) provides no specific guidance for foreign research reactor spent nuclear fuel. The transportation analysis in the SNF&INEL Final EIS assumed that half the foreign research reactor spent nuclear fuel would be transported to Idaho National Engineering Laboratory and half to Savannah River Site. The disposition of foreign research reactor spent nuclear fuel could correspond to Decentralization (described above), Regionalization (described below), Centralization to Idaho National Engineering Laboratory or Savannah River Site (described below), or an arbitrary split as described in the SNF&INEL Final EIS (DOE, 1995).

Regionalization - There are two distinct subalternatives under Regionalization: Regionalization by Fuel Type, and Regionalization by Geography. These subalternatives are described below.

Regionalization Subalternative A - Under Regionalization by Fuel Type, the foreign research reactor spent nuclear fuel would be split by fuel type, regardless of the port of entry. The TRIGA fuel would be shipped to Idaho National Engineering Laboratory and the aluminum-based Material Test Reactor (MTR) fuel would be shipped to Savannah River Site. Savannah River Site would receive 675 casks of fuel: 544 from the east and 131 from the west. Idaho National Engineering Laboratory would receive 162 casks of fuel: 107 from the east and 55 from the west. The transportation under this alternative is illustrated in Figure E-2. No intersite shipment would be anticipated under this single-phased alternative.

Regionalization Subalternative B - Under Regionalization by Geography, foreign research reactor spent nuclear fuel would be distributed between an Eastern Regional Site (Oak Ridge Reservation or Savannah River Site) and a Western Regional Site (either Hanford Site, Idaho National Engineering Laboratory, or Nevada Test Site). The foreign research reactor spent nuclear fuel arriving at an eastern port would go to the Eastern Regional Site, and the foreign research reactor spent nuclear fuel arriving at a western port would go to the Western Regional Site. If the chosen sites were Savannah River Site and Idaho National

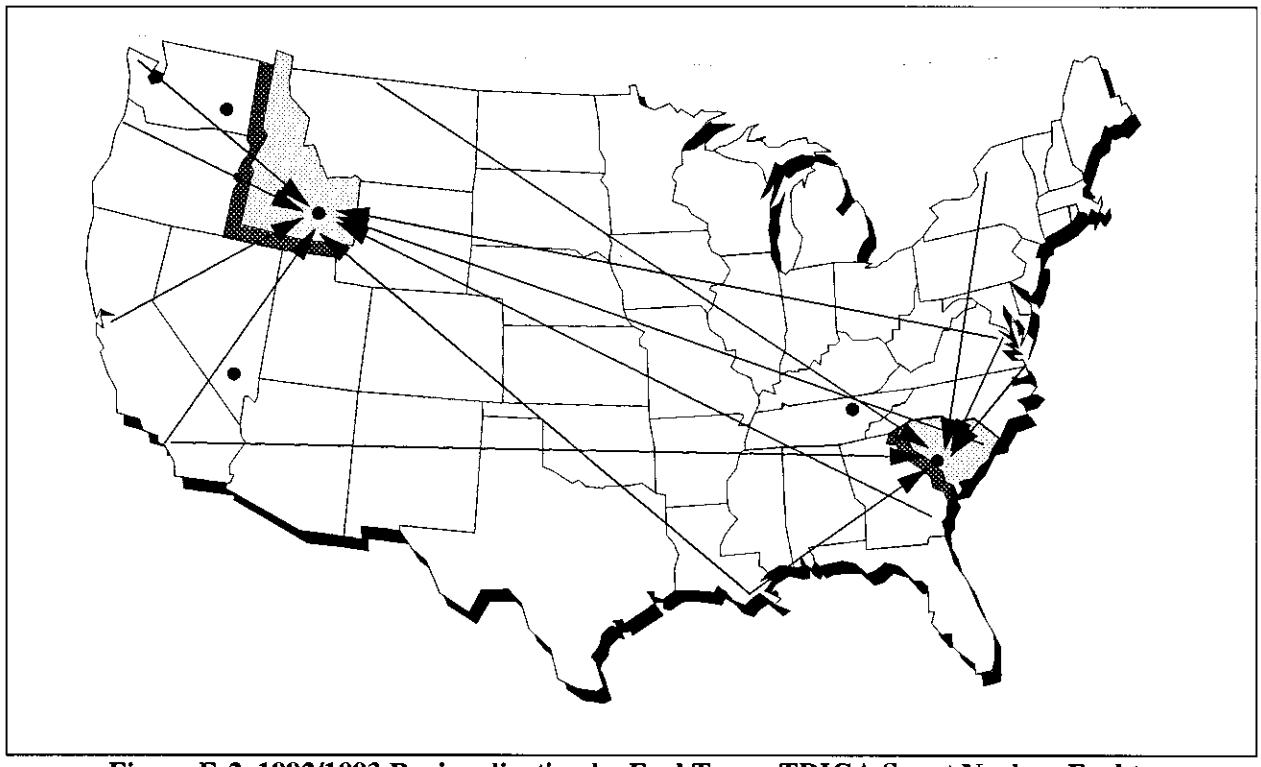

Figure E-2 1992/1993 Regionalization by Fuel Type: TRIGA Spent Nuclear Fuel to Idaho National Engineering Laboratory and MTR Spent Nuclear Fuel to Savannah River Site

Engineering Laboratory, the transportation would be the same as that described in the Decentralization Alternative and Figure E-2. No intersite shipment would be anticipated under this single-phased alternative.

A two-phased program would be required if a site other than Idaho National Engineering Laboratory or Savannah River Site were selected as a regional site under this programmatic alternative. The remaining possible site pairs for Regionalization are Idaho National Engineering Laboratory/Oak Ridge Reservation, Nevada Test Site/Savannah River Site, Nevada Test Site/Oak Ridge Reservation, Hanford Site/Savannah River Site, and Hanford Site/Oak Ridge Reservation. Splitting fuel by both geography and fuel type was considered as a logical Phase 1 approach for each site pair, but transporting all fuel to Savannah River Site or Idaho National Engineering Laboratory for near-term management was not considered for the following reasons:

- If Idaho National Engineering Laboratory were selected as the Western Regional Site, and Savannah River Site were not selected as the Eastern Regional Site, it would not be reasonable to ship all foreign research reactor spent nuclear fuel to Savannah River Site during Phase 1. Since Idaho National Engineering Laboratory is currently capable of receiving fuel and is much closer to west coast ports, it would be unreasonable to ship all fuel across the country to Savannah River Site only to move the fuel again. However, the option to ship all MTR fuel to Savannah River Site, for onsite logistical reasons, is a logical Phase 1 option even if Savannah River Site is not an ultimate interim management location. Thus, shipment of all fuel to SRS during Phase 1 was not considered reasonable if Idaho National Engineering Laboratory were to be chosen as the Western Regional Site.

- Conversely, if Savannah River Site were selected as the Eastern Regional Site, and Idaho National Engineering Laboratory were not selected as the Western Regional Site, it would not be reasonable to ship all foreign research reactor spent nuclear fuel to Idaho National Engineering Laboratory during Phase 1. Since Savannah River Site is currently capable of receiving fuel and is much closer to east coast ports, it would be unreasonable to ship all fuel across the country to Idaho National Engineering Laboratory only to move the fuel again. However, the option to ship all TRIGA fuel to Idaho National Engineering Laboratory, for onsite logistical reasons, is a logical Phase 1 option, even if Idaho National Engineering Laboratory is not an ultimate interim management location. Thus, shipment of all fuel to Idaho Engineering National Laboratory during Phase 1 was not considered reasonable if Savannah River Site were to be chosen as the Eastern Regional Site.

Figures E-3 through E-7 show the transportation schemes for site pairs Idaho National Engineering Laboratory/Oak Ridge Reservation, Nevada Test Site/Savannah River Site, Nevada Test Site/Oak Ridge Reservation, Hanford Site/Savannah River Site, and Hanford Site/Oak Ridge Reservation, respectively. The origins of the arrows representing shipments on the figures are selected for illustrative purposes, not to show specifically selected ports. Shipments would be expected to arrive at eastern, western, and Gulf Coast ports, and from eastern and western Canada. Because of their relative proximity to eastern sites, Gulf Coast ports are assigned the same transportation schemes as east coast ports. Note that there is no TRIGA fuel in Canada, so there is no planned route from Canada to Idaho National Engineering Laboratory for the Regionalization by Fuel Type alternative.

The number of shipments in the basic implementation for each site pair is described in Table E-1. The number of intersite shipments is based on the assumption that spent nuclear fuel arriving from foreign countries in small casks would be rearranged such that intersite shipments could be made in larger casks. The fuel assemblies would be cut to more efficient shapes, the fuel would be older and, thus, less radioactive and would produce less heat. For analysis purposes, it is assumed that the amount of foreign research reactor spent nuclear fuel originally shipped in four of the casks used for importing fuel could be shipped in one intersite truck cask. Rail shipment allows the use of even larger casks; and, thus, it is assumed that 10 cask loads of foreign research reactor spent nuclear fuel could be intersite shipped in 1 rail cask. Since the potential shipments would be scheduled to occur at least 10 years in the future, it is difficult to predict what casks would be used. Appendix B describes a variety of candidate casks. DOE would use fewer but larger shipments when shipping from site-to-site.

Centralization - Any one of the five DOE sites could be chosen by the SNF&INEL Final EIS (DOE, 1995) for receipt of all foreign research reactor spent nuclear fuel. If that site were Idaho National Engineering Laboratory or Savannah River Site, a two-phased approach would not be necessary. From the beginning of the program, all fuel could be accepted at either of these sites. Figures E-8 and E-9 describe the single-phase centralization options to Idaho National Engineering Laboratory and Savannah River Site, respectively.

However, a two-phase program would be required if the site selected were Hanford Site, Oak Ridge Reservation, or Nevada Test Site, none of which would be ready to receive foreign research reactor spent nuclear fuel at the beginning of the program. The option for execution of Phase 1 shipments is assumed to be independent from Phase 2 and the chosen SNF&INEL Final EIS (DOE, 1995) alternative. As with the Regionalization options, during Phase 1, DOE could choose to divide the fuel by either geography or fuel type between the two initially-capable DOE sites (Idaho National Engineering Laboratory and Savannah River Site). Alternatively, all fuel could be initially shipped to Idaho National Engineering Laboratory or Savannah River Site. Figures E-8 through E-12 show the transportation schemes for all five sites. The

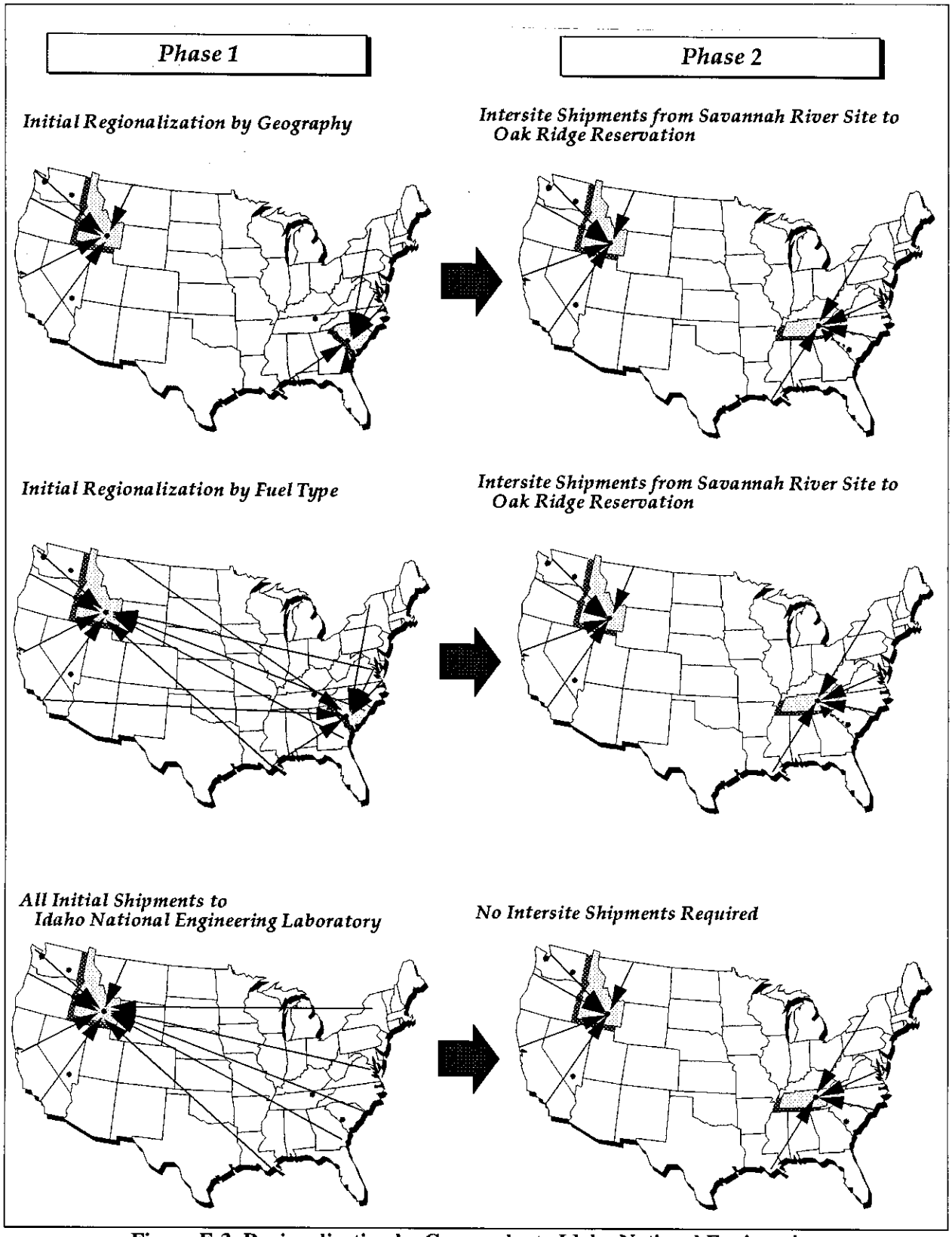

Figure E-3 Regionalization by Geography to Idaho National Engineering Laboratory and Oak Ridge Reservation

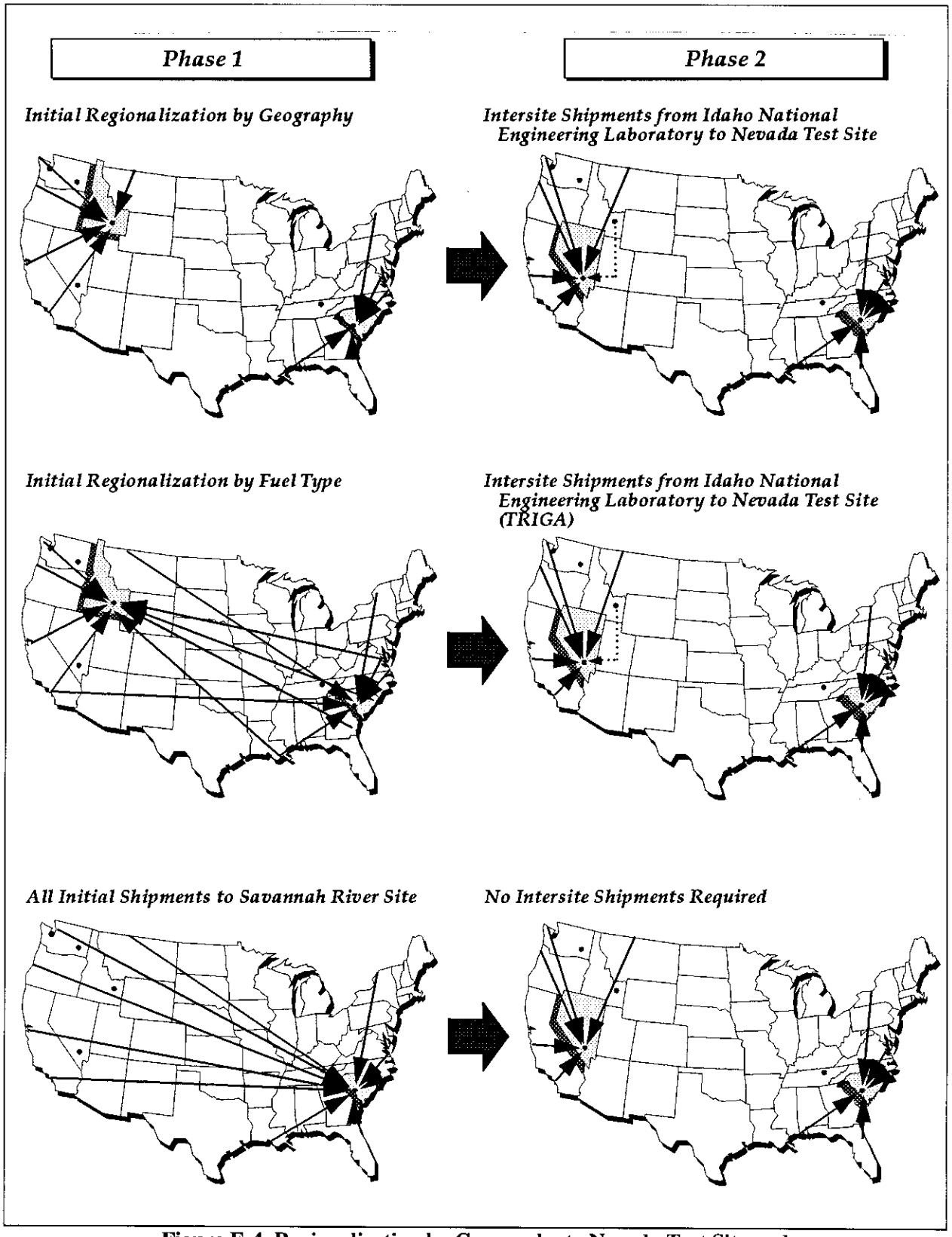

Figure E-4 Regionalization by Geography to Nevada Test Site and Savannah River Site

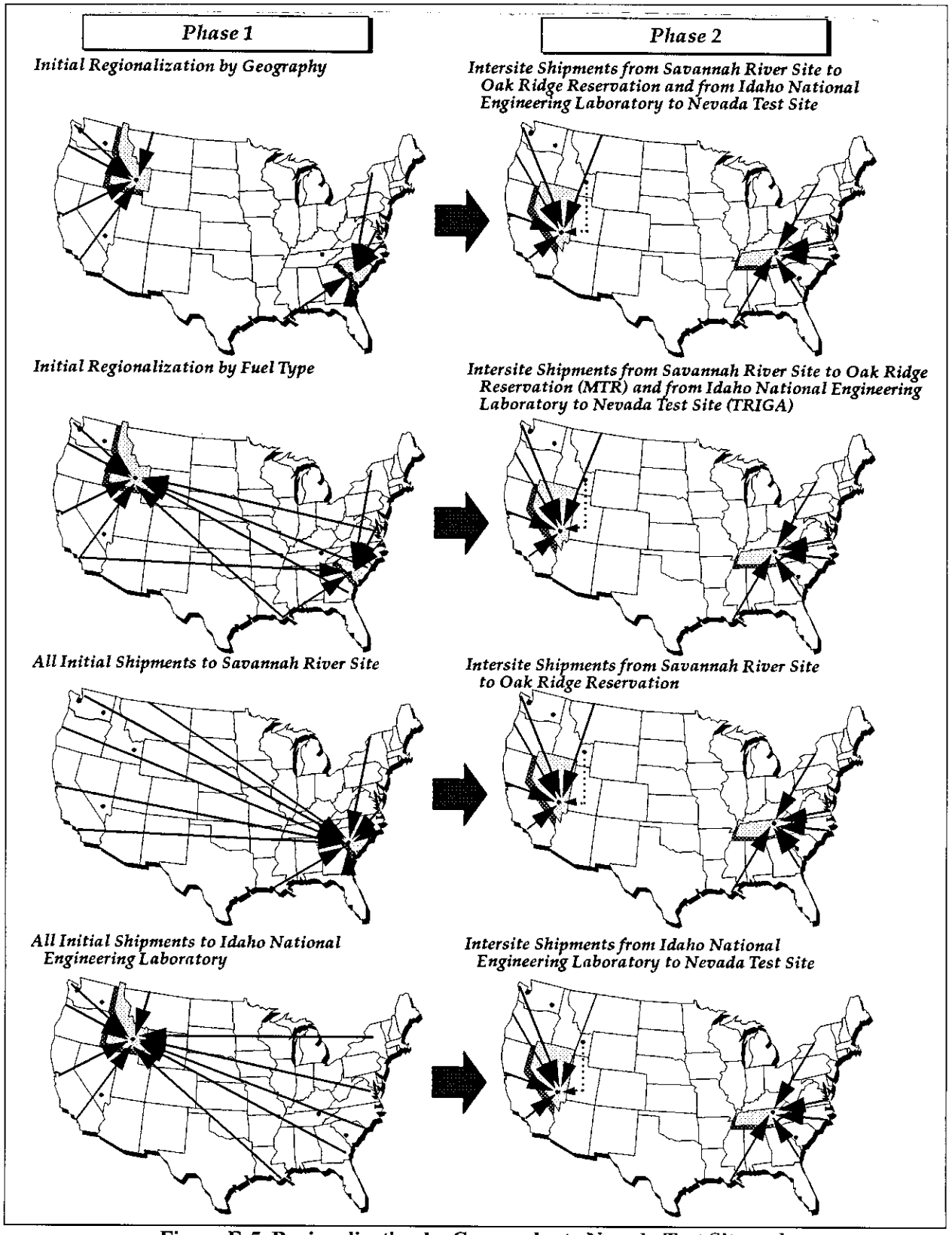

Figure E-5 Regionalization by Geography to Nevada Test Site and Oak Ridge Reservation

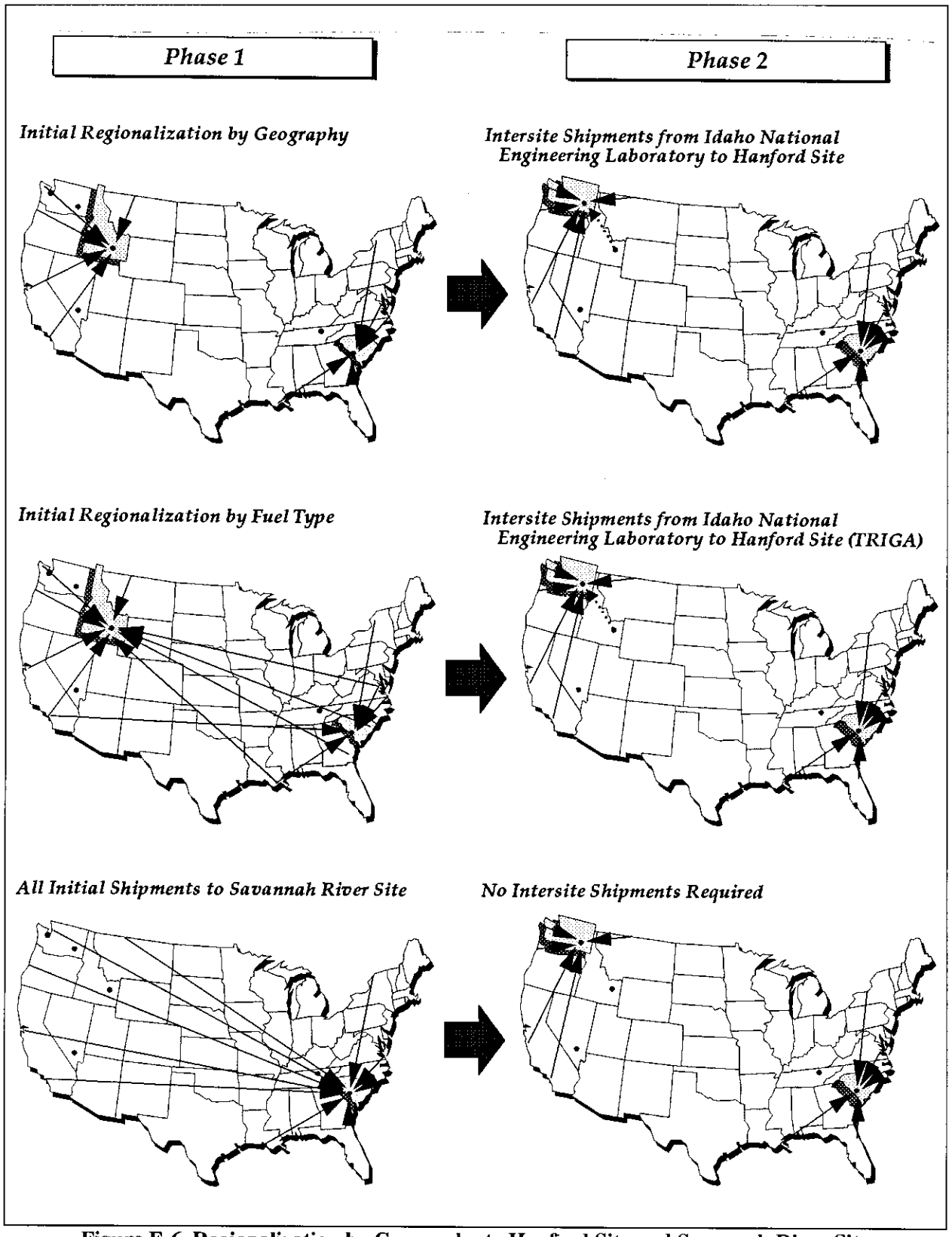

Figure E-6 Regionalization by Geography to Hanford Site and Savannah River Site

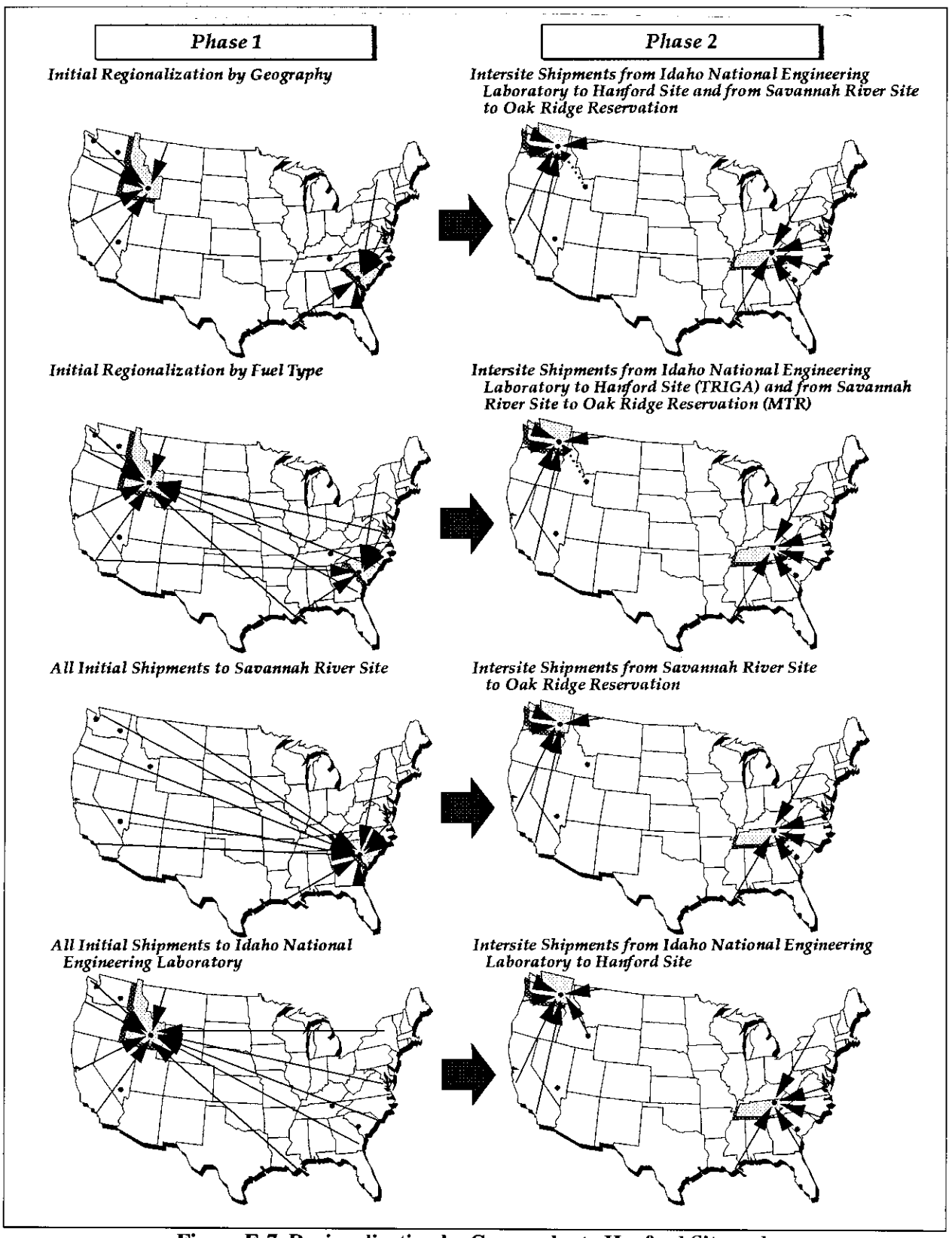

Figure E-7 Regionalization by Geography to Hanford Site and Oak Ridge Reservation

Table E-1 Shipment Summary for Regionalization Alternatives

Spent Nuclear Fuel Site Option	*Phase 1 Approach*	*Phase 1 Port-to-Site Shipments*	*Site-to-Site Shipments*[a]	*Phase 2 or Port-to-Final Site Shipments*	*Total Number of Shipments*
INEL/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51	East to ORR: 150 West to INEL: 43	963/888
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52	East to ORR: 150 West to INEL: 43	967/889
	All to INEL	644	None	East to ORR: 150 West to INEL: 43	837
NTS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to NTS: 36/15	East to SRS: 150 West to NTS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to NTS: 31/13	East to SRS: 150 West to NTS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to NTS: 43	837
NTS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to NTS: 36/15	East to ORR: 150 West to NTS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to NTS: 31/13	East to ORR: 150 West to NTS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to NTS: 43	998/902
	All to INEL	644	INEL to NTS: 161/65	East to ORR: 150 West to NTS: 43	998/902
HS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to HS: 36/15	East to SRS: 150 West to HS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to HS: 31/13	East to SRS: 150 West to HS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to HS: 43	837
HS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to HS: 36/15	East to ORR: 150 West to HS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to HS: 31/13	East to ORR: 150 West to HS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to HS: 43	998/902
	All to INEL	644	INEL to HS: 161/65	East to ORR: 150 West to HS: 43	998/902

[a] *Truck/Rail shipments, assuming that the truck casks used for intersite shipments are capable due to consolidation of carrying 4 times as much fuel, and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor.*

INEL = Idaho National Engineering Laboratory; ORR = Oak Ridge Reservation; SRS = Savannah River Site; NTS = Nevada Test Site; HS = Hanford Site

number of shipments for each site pair is shown in Table E-2. The number of intersite shipments is based on a 4-cask-to-1 conversion if trucks were used, and a 10-cask-to-1 conversion if trains were used, as explained in the previous section.

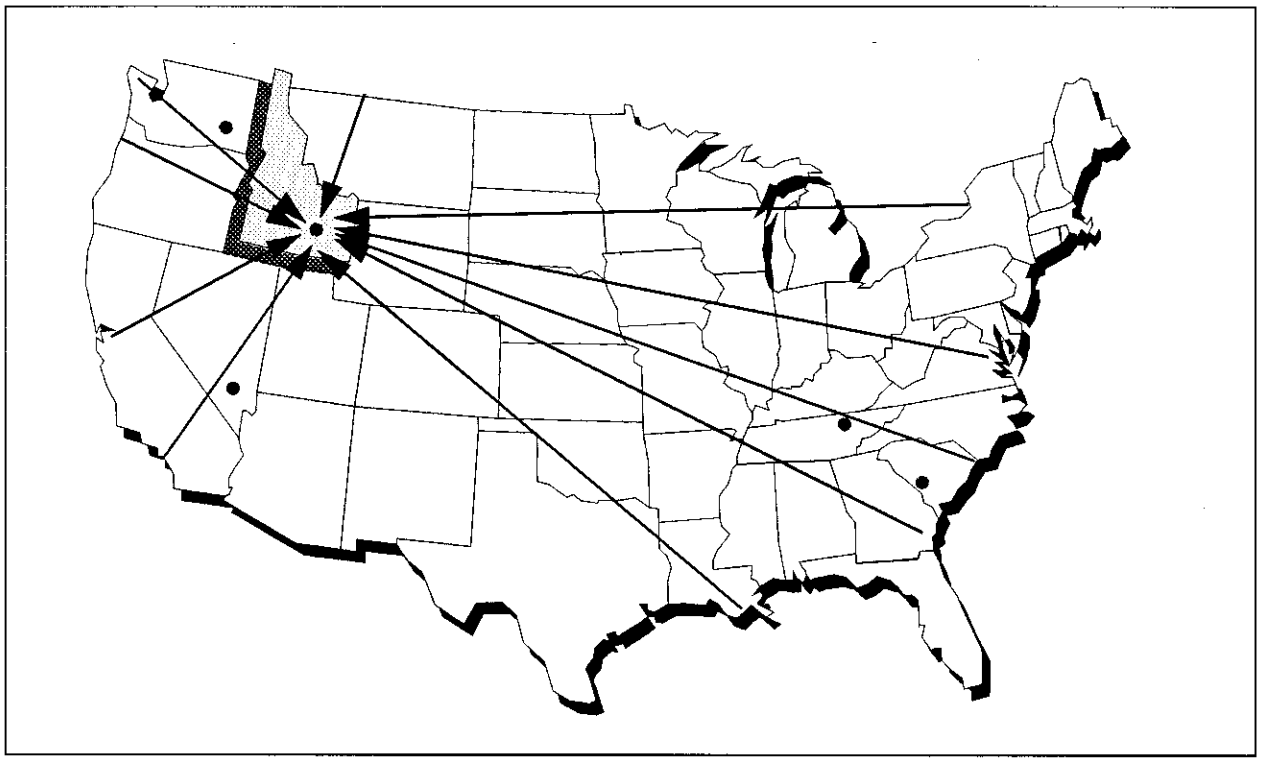

Figure E-8 Centralization to Idaho National Engineering Laboratory

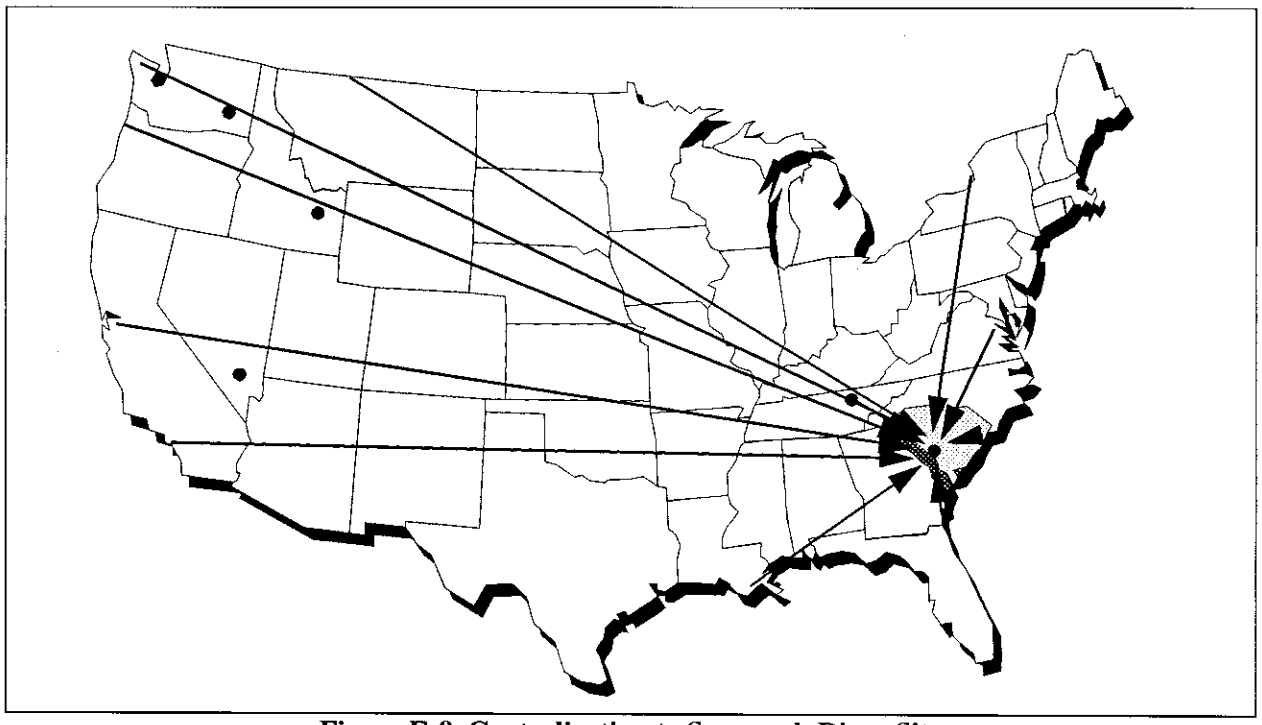

Figure E-9 Centralization to Savannah River Site

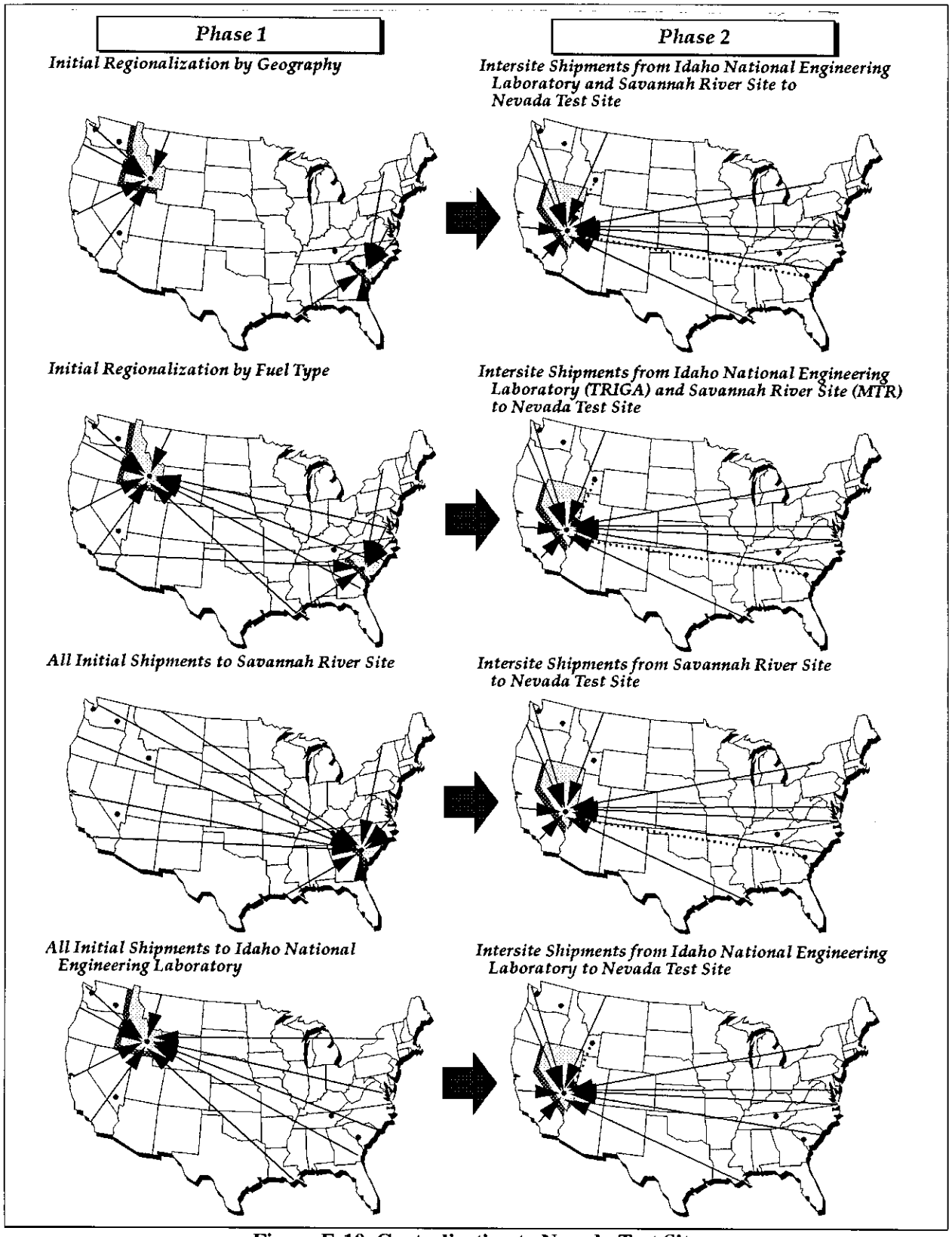

Figure E-10 Centralization to Nevada Test Site

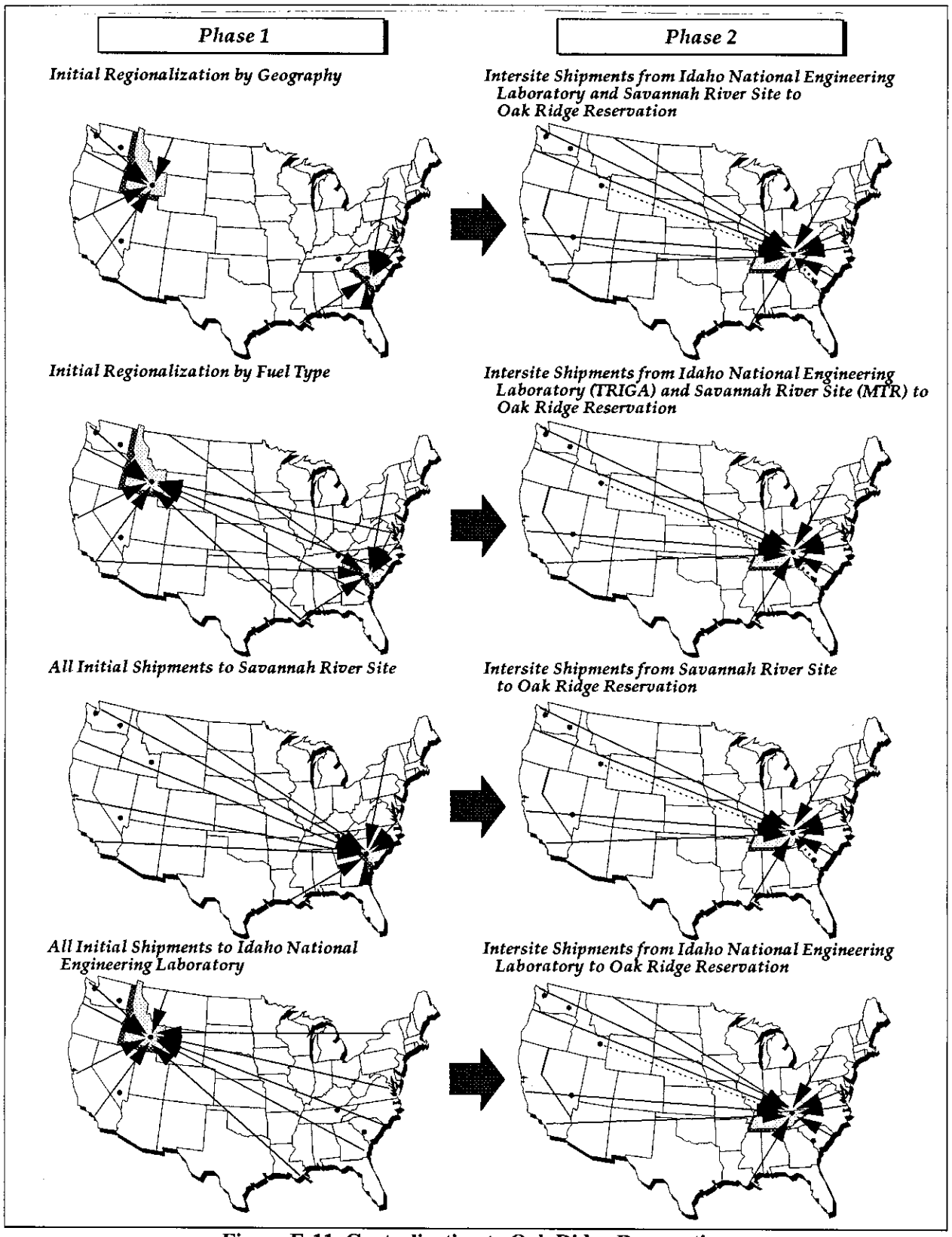

Figure E-11 Centralization to Oak Ridge Reservation

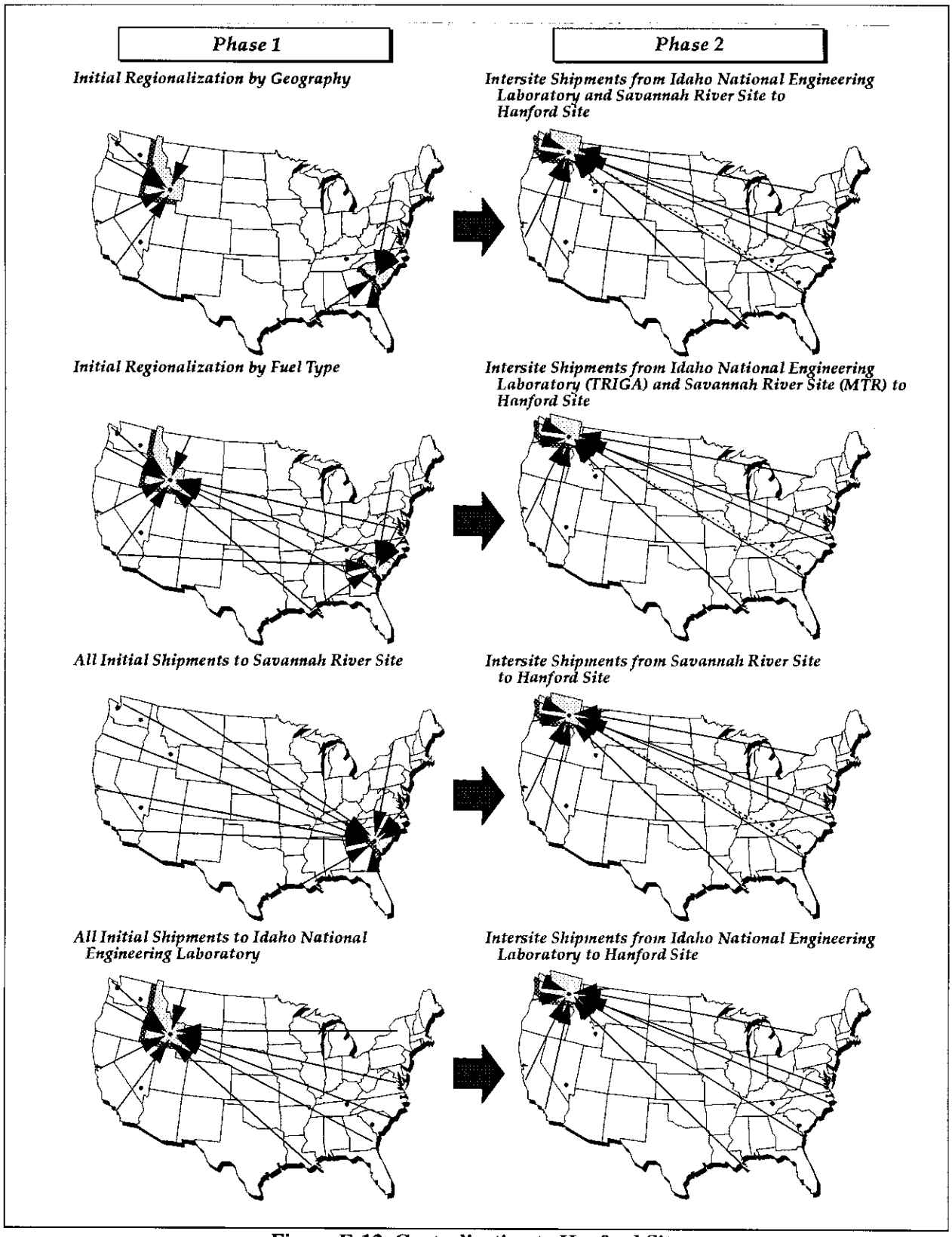

Figure E-12 Centralization to Hanford Site

Table E-2 Shipment Summary for Centralization Alternatives

Spent Nuclear Fuel Site Option	*Phase 1 Approach*	*Phase 1 Port-to-Site Shipments*	*Site-to-Site Shipments*[a]	*Phase 2 or Port-to-Final Site Shipments*	*Total Number of Shipments*
INEL	N/A - Single phase program			837	837
SRS	N/A - Single phase program			837	837
NTS	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
ORR	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	998/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
Hanford Site	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902

[a] *Truck/Rail shipments assuming that the truck casks used for intersite shipments are capable of carrying 4 times as much fuel and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor due to consolidation.*

INEL = Idaho National Engineering Laboratory; ORR = Oak Ridge Reservation; SRS = Savannah River Site; NTS = Nevada Test Site; HS = Hanford Site

E.4 Truck and Rail Routing Analysis

Both rail and highway shipping capabilities are available at all potential ports of entry, and each of the five DOE sites is or could be made capable of receiving foreign research reactor spent nuclear fuel transported by rail or highway. Therefore, shipment of spent nuclear fuel will be analyzed along representative highway and railway routes for all ports and SNF&INEL Final EIS (DOE, 1995) alternatives.

As discussed above, each alternative can be defined as a set of origin and destination pairs representing shipment linkages between ports of entry and interim management sites. The calculation of the overland transportation risk for an alternative depends in part on characteristics of the transportation routes between the origin and destination sites. Regulatory routing criteria and the methods used to determine representative truck and rail routes for the transportation risk assessment are described below. In addition, the route characteristics that are important for the risk assessment are summarized.

E.4.1 Routing Regulations

Department of Transportation's public highway routing regulations are prescribed in 49 CFR Part 397. The regulations' objectives are to reduce the impacts of transporting radioactive materials, to establish consistent and uniform requirements for route selection, and to identify the role of State and local governments in the routing of radioactive materials. The regulations attempt to reduce potential hazards by avoiding populous areas and by minimizing travel times. Further, they require that the carrier of radioactive materials ensure that the vehicle is operated on routes that minimize radiological risks, and that accident rates, transit times, population density and activity, time of day, and day of week are considered in determining risk.

A shipment of a "highway route controlled quantity" of radioactive material, such as spent nuclear fuel, is required by 49 CFR 397 Subpart D to use the interstate highway system except when moving from origin to interstate or from interstate to destination, when making necessary repair or rest stops, or when emergency conditions make continued use of the interstate unsafe or impossible. Carriers are required to use interstate circumferential or bypass routes, if available, to avoid populous areas. Other "preferred highways" may be designated by any State or Tribe to replace or supplement the interstate system (DOT, 1992). Under its authority to regulate interstate transportation safety, the Department of Transportation can prohibit State and local bans and restrictions as "undue restraint of interstate commerce." State or local bans will be pre-empted if inconsistent with 49 CFR 397.

Currently, there are no Department of Transportation railroad routing regulations specific to the transportation of radioactive materials. Routes are generally fixed by the location of rail lines, and urban areas cannot be readily bypassed.

E.4.2 Determination of Representative Transportation Routes

Representative overland truck and rail routes have been determined for all pairs of origin and destination sites considered by the alternatives. The routes were selected consistent with current routing practices and all applicable routing regulations and guidelines. However, because the routes were determined for risk assessment purposes, they do not necessarily represent the actual routes that would be used to transport foreign research reactor spent nuclear fuel in the future. Specific routes cannot be identified in advance because the route would not be finalized until it had been reviewed and approved by the NRC. The selection of the actual route would be responsive to environmental and other conditions that were in effect or could reasonably be predicted at the time of shipment. Such conditions could include adverse weather conditions, truck or road conditions, bridge closures, etc. (Massey, 1994).

For both truck and rail transportation modes, the route characteristics that are important to the radiological risk assessment include the total shipment distance between each origin and destination pair and the population distribution along the route. The specific route selected determines both the total potentially exposed population and the expected frequency of transportation-related accidents. Route characteristics are summarized in Tables E-3 and E-4 for the ports of entry and management sites considered in this assessment. The ports of Philadelphia, PA; Elizabeth, NJ; and Long Beach, CA are included in the list to show the effects on overland transportation of choosing high population ports. The routes from Canada are representative for risk analysis purposes, many other routes are available for use. They are not included in the risk analysis described later in this appendix. The exposed population includes all persons living within 800 m [0.5 mile (mi)] on each side of the route. The representative routes are shown in Attachment 1 to this appendix.

Table E-3 Summary of Route Distances for Truck and Rail Modes

Shipments to Hanford Site:				
		Percentage in Zone		
Route[a]	*Distance km (mi)*	*Rural*	*Suburban*	*Urban*
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	4579 (2858)	85.5	13.3	1.2
Charleston, SC (Wando Terminal)	4603 (2873)	85.3	13.4	1.3
Elizabeth, NJ	4527 (2812)	84.4	14.1	1.4
Galveston, TX	3746 (2327)	86.0	11.8	2.3
Jacksonville, FL	4708 (2924)	83.9	14.6	1.5
Newport News, VA	4682 (2908)	84.9	13.3	1.8
Norfolk, VA	4748 (2949)	84.4	13.8	1.7
Philadelphia, PA	4617 (2868)	82.9	15.6	1.5
Portsmouth, VA	4717 (2930)	84.5	13.6	1.9
Savannah, GA	4529 (2813)	84.7	13.8	1.5
MOTSU, NC	4617 (2868)	85.7	13.1	1.3
Wilmington, NC	4770 (2963)	85.3	13.5	1.1
Rail:				
Charleston, SC (NWS)	4925 (3059)	84.5	13.7	1.8
Charleston, SC (Wando Terminal)	4925 (3059)	84.5	13.7	1.8
Elizabeth, NJ	4846 (3010)	76.1	19.5	4.4
Galveston, TX	3851 (2392)	89.9	9.1	1.0
Jacksonville, FL	4941 (3069)	85.4	13.0	1.6
Newport News, VA	4972 (3088)	83.6	13.7	2.7
Norfolk, VA	5131 (3187)	83.8	13.6	2.7
Philadelphia, PA	4769 (2962)	77.1	18.6	4.3
Portsmouth, VA	5083 (3157)	84.0	13.4	2.6
Savannah, GA	4977 (3091)	85.3	13.2	1.4
MOTSU, NC	5157 (3203)	83.6	14.8	1.5
Wilmington, NC	5142 (3194)	83.7	14.7	1.5
From Western Ports				
Truck:				
Long Beach CA	1986 (1241)	80.5	14.3	5.2
NWS Concord, CA	1378 (856)	79.4	18.0	2.6
Portland, OR	407 (253)	81.5	15.3	3.3
Tacoma, WA	399 (248)	73.4	22.8	3.8
Rail:				
Long Beach, CA	2553 (1587)	85.6	8.9	5.5
NWS Concord, CA	1531 (951)	80.3	14.7	5.0
Portland, OR	385 (239)	82.1	13.4	4.5
Tacoma, WA	602 (374)	79.2	17.2	3.6
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	4456 (2768)	82.8	15.6	1.6
Idaho National Engineering Laboratory	964 (599)	91.3	7.6	1.1
Nevada Test Site	1816 (1128)	86.5	10.9	2.6
Oak Ridge Reservation	3967 (2464)	87.8	11.0	1.2
Savannah River	4390 (2727)	84.3	14.2	1.5
Sweetgrass, MT	1407 (874)	89.4	10.0	0.6
Rail:				
Alexandria Bay, NY	4634 (2878)	79.6	16.6	3.8
Idaho National Engineering Laboratory	1059 (658)	91.4	7.1	1.4

Shipments to Hanford Site:				
		Percentage in Zone		
Route[a]	***Distance km (mi)***	***Rural***	***Suburban***	***Urban***
Nevada Test Site	2096 (1302)	93.0	5.9	1.1
Oak Ridge Reservation	4188 (2601)	91.2	7.4	1.3
Savannah River	4754 (2953)	84.7	13.5	1.8
Sweetgrass, MT	976 (606)	91.7	6.8	1.6

Shipments to Idaho National Engineering Laboratory:				
		Percentage in Zone		
Route[a]	***Distance km (mi)***	***Rural***	***Suburban***	***Urban***
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	3910 (2441)	84.4	14.3	1.3
Charleston, SC (Wando Terminal)	3935 (2456)	84.2	14.4	1.4
Elizabeth, NJ	3858 (2396)	82.9	15.5	1.5
Galveston, TX	3077 (1911)	84.5	13.0	2.5
Jacksonville, FL	4031 (2504)	82.5	15.9	1.5
Newport News, VA	4012 (2492)	83.5	14.6	1.9
Norfolk, VA	4073 (2530)	83.1	15.1	1.8
Philadelphia, PA	3948 (2452)	81.2	17.2	1.6
Portsmouth, VA	4048 (2514)	83.1	14.8	2.1
Savannah, GA	3861 (2398)	83.3	15.1	1.6
MOTSU, NC	3875 (2407)	85.3	13.5	1.2
Wilmington, NC	4099 (2546)	84.1	14.8	1.2
Rail:				
Charleston, SC (NWS)	4046 (2513)	82.6	15.3	2.1
Charleston, SC (Wando Terminal)	4046 (2513)	82.6	15.3	2.1
Elizabeth, NJ	3967 (2464)	72.3	22.5	5.2
Galveston, TX	2972 (1846)	88.9	10.1	1.0
Jacksonville, FL	4062 (2523)	83.7	14.6	1.7
Newport News, VA	4093 (2542)	81.5	15.4	3.1
Norfolk, VA	4252 (2641)	81.8	15.2	3.0
Philadelphia, PA	3890 (2416)	73.4	21.5	5.1
Portsmouth, VA	4204 (2611)	82.1	14.9	3.0
Savannah, GA	4097 (2545)	83.6	14.8	1.6
MOTSU, NC	4278 (2657)	81.6	16.7	1.7
Wilmington, NC	4263 (2648)	81.8	16.5	1.7
From Western Ports				
Truck:				
Long Beach, CA	1575 (979)	77.7	14.2	8.1
NWS Concord, CA	1518 (943)	85.9	11.1	3.1
Portland, OR	1188 (738)	88.6	9.8	1.7
Tacoma, WA	1312 (815)	87.0	11.4	1.6
Rail:				
Long Beach, CA	1675 (1041)	81.5	10.5	8.0
NWS Concord, CA	1473 (915)	89.0	8.7	2.4
Portland, OR	1264 (785)	92.6	5.8	1.6
Tacoma, WA	1504 (934)	88.6	9.2	2.2
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	3787 (2352)	81.0	17.2	1.7
Hanford Site	964 (599)	91.3	7.6	1.1
Nevada Test Site	1146 (712)	82.8	13.7	3.5

Shipments to Idaho National Engineering Laboratory:				
		Percentage in Zone		
Route[a]	*Distance km (mi)*	*Rural*	*Suburban*	*Urban*
Oak Ridge Reservation	3297 (2048)	86.8	12.0	1.2
Savannah River	3721 (2311)	82.8	15.6	1.6
Sweetgrass, MT	874 (543)	94.8	4.8	0.4
Rail:				
Alexandria Bay, NY	3755 (2332)	76.4	19.1	4.5
Hanford Site	1059 (658)	91.4	7.1	1.4
Nevada Test Site	1217 (756)	92.8	5.9	1.3
Oak Ridge Reservation	3309 (2055)	90.7	7.8	1.5
Savannah River	3875 (2407)	82.8	15.2	2.0
Sweetgrass, MT	1982 (1231)	93.2	5.8	1.0

Shipments to Nevada Test Site:				
		Percentage in Zone		
Route[a]	*Distance km (mi)*	*Rural*	*Suburban*	*Urban*
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	3930 (2543)	84.5	14.1	1.4
Charleston, SC (Wando Terminal)	4098 (2558)	84.3	14.2	1.5
Elizabeth, NJ	4302 (2672)	80.5	17.2	2.3
Galveston ,TX	2998 (1862)	85.4	11.5	3.2
Jacksonville, FL	4197 (2607)	82.8	15.4	1.8
Newport News, VA	4178 (2595)	83.8	14.1	2.1
Norfolk, VA	4239 (2633)	83.4	14.6	2.0
Philadelphia, PA	4223 (2623)	80.4	17.4	2.2
Portsmouth, VA	4213 (2617)	83.4	14.3	2.3
Savannah, GA	4027 (2501)	83.6	14.6	1.8
MOTSU, NC	3956 (2457)	83.0	15.0	2.0
Wilmington, NC	4267 (2650)	84.3	14.3	1.4
Rail:				
Charleston, SC (NWS)	4741 (2945)	84.3	13.7	2.0
Charleston, SC (Wando Terminal)	4741 (2945)	84.3	13.7	2.0
Elizabeth, NJ	4661 (2895)	75.6	19.7	4.7
Galveston, TX	3148 (1955)	92.0	7.2	0.8
Jacksonville, FL	4758 (2955)	85.3	13.1	1.7
Newport News, VA	4787 (2973)	83.4	13.8	2.9
Norfolk, VA	4948 (3073)	83.6	13.6	2.8
Philadelphia, PA	4585 (2848)	76.6	18.8	4.6
Portsmouth, VA	4898 (3042)	83.8	13.4	2.8
Savannah, GA	4793 (2977)	85.2	13.2	1.5
MOTSU, NC	4973 (3089)	83.4	14.9	1.7
Wilmington, NC	4959 (3080)	83.5	14.8	1.7
From Western Ports				
Truck:				
Long Beach, CA	645 (401)	71.3	12.7	16.0
NWS Concord, CA	1146 (712)	81.8	11.3	6.9
Portland, OR	2045 (1270)	85.5	11.5	2.9
Tacoma, WA	2164 (1344)	84.7	12.6	2.7
Rail:				
Long Beach, CA	777 (483)	70.5	14.3	15.3
NWS Concord, CA	1369 (850)	77.8	16.7	5.6
Portland, OR	2301 (1429)	93.5	5.3	1.2

Shipments to Nevada Test Site:				
		Percentage in Zone		
Route[a]	***Distance km (mi)***	***Rural***	***Suburban***	***Urban***
Tacoma, WA	2542 (1579)	91.0	7.4	1.6
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	4217 (2619)	82.0	16.0	1.9
Hanford Site	1816 (1128)	86.5	10.9	2.6
Idaho National Engineering Laboratory	1146 (712)	82.8	13.7	3.5
Oak Ridge Reservation	3463 (2151)	86.9	11.5	1.6
Savannah River	3887 (2414)	83.1	15.1	1.8
Sweetgrass, MT	1900 (1180)	87.5	10.0	2.5
Rail:				
Alexandria Bay, NY	4448 (2763)	79.2	16.7	4.0
Hanford Site	2096 (1302)	93.0	5.9	1.1
Idaho National Engineering Laboratory	1217 (756)	92.8	5.9	1.3
Oak Ridge Reservation	4004 (2487)	91.4	7.2	1.5
Savannah River	4571 (2839)	84.5	13.5	1.9
Sweetgrass, MT	3019 (1875)	93.7	5.4	0.9

Shipments to Oak Ridge Reservation:				
		Percentage in Zone		
Route[a]	***Distance km (mi)***	***Rural***	***Suburban***	***Urban***
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	644 (402)	71.6	27.6	0.8
Charleston, SC (Wando Terminal)	668 (417)	70.9	27.8	1.3
Elizabeth, NJ	1188 (738)	62.2	35.7	2.1
Galveston, TX	1550 (963)	73.3	24.6	2.1
Jacksonville, FL	913 (567)	66.8	32.0	1.3
Newport News, VA	890 (553)	69.8	27.6	2.6
Norfolk, VA	886 (550)	68.4	30.2	1.3
Philadelphia, PA	1095 (680)	64.7	31.7	3.6
Portsmouth, VA	926 (575)	68.4	28.2	3.4
Savannah, GA	723 (449)	74.3	25.0	0.6
MOTSU, NC	799 (496)	72.4	26.7	0.9
Wilmington, NC	819 (509)	72.6	26.5	0.9
Rail:				
Charleston, SC (NWS)	935 (581)	65.2	33.3	1.5
Charleston, SC (Wando Terminal)	935 (581)	65.2	33.3	1.5
Elizabeth, NJ	1264 (785)	44.7	43.2	12.2
Galveston, TX	1695 (1053)	70.5	26.2	3.3
Jacksonville, FL	910 (565)	65.7	31.9	2.4
Newport News, VA	1230 (764)	59.2	38.7	2.0
Norfolk, VA	1109 (689)	62.2	36.3	1.6
Philadelphia, PA	1129 (701)	48.6	43.0	8.4
Portsmouth, VA	1061 (659)	62.3	36.4	1.3
Savannah, GA	945 (587)	66.2	32.1	1.7
MOTSU, NC	873 (542)	61.5	37.1	1.5
Wilmington, NC	857 (532)	61.7	36.8	1.5
From Western Ports				
Truck:				
Long Beach, CA	3614 (2246)	85.0	11.0	3.8
NWS Concord, CA	4117 (2557)	86.3	10.9	2.8

Shipments to Oak Ridge Reservation:				
		Percentage in Zone		
Route[a]	Distance km (mi)	Rural	Suburban	Urban
Portland, OR	4200 (2609)	87.0	11.5	1.5
Tacoma, WA	4279 (2658)	88.0	11.0	1.0
Rail:				
Long Beach, CA	4302 (2674)	86.5	9.7	3.9
NWS Concord, CA	4524 (2810)	87.5	10.4	2.2
Portland, OR	4551 (2827)	85.5	12.1	2.4
Tacoma, WA	4568 (2837)	83.7	13.3	3.0
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	1492 (927)	65.9	33.5	0.7
Hanford Site	3967 (2464)	87.8	11.0	1.2
Idaho National Engineering Laboratory	3297 (2048)	86.8	12.0	1.2
Nevada Test Site	3463 (2151)	86.9	11.5	1.6
Savannah River	610 (379)	59.1	38.5	2.4
Sweetgrass, MT	1900 (1180)	87.5	10.0	2.5
Rail:				
Alexandria Bay, NY	1565 (972)	57.5	35.7	6.8
Hanford Site	4188 (2601)	91.2	7.4	1.3
Idaho National Engineering Laboratory	3309 (2055)	90.7	7.8	1.5
Nevada Test Site	4004 (2487)	91.4	7.2	1.5
Savannah River	671 (417)	68.8	29.8	1.4
Sweetgrass, MT	3375 (2096)	83.7	13.7	2.6

Shipments to Savannah River Site:				
		Percentage in Zone		
Route[a]	Distance km (mi)	Rural	Suburban	Urban
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	301 (188)	72.9	26.2	0.9
Charleston, SC (Wando Terminal)	325 (203)	71.6	26.6	1.8
Elizabeth, NJ	1325 (823)	63.8	34.2	2.1
Galveston, TX	1610 (1000)	70.5	27.0	2.5
Jacksonville, FL	607 (377)	81.5	18.4	0.0
Newport News, VA	836 (519)	71.1	26.8	2.1
Norfolk, VA	802 (498)	72.8	26.2	1.0
Philadelphia, PA	1193 (741)	62.1	34.0	3.9
Portsmouth, VA	807 (501)	72.7	26.1	1.1
Savannah, GA	403 (250)	79.1	20.8	0.0
MOTSU, NC	403 (250)	82.5	17.2	0.3
Wilmington, NC	499 (310)	75.5	24.0	0.5
Rail:				
Charleston, SC (NWS)	225 (140)	83.9	13.6	2.5
Charleston, SC (Wando Terminal)	225 (140)	83.9	13.6	2.5
Elizabeth, NJ	1404 (872)	56.2	33.0	10.8
Galveston, TX	1890 (1174)	69.6	26.2	4.2
Jacksonville, FL	417 (259)	83.7	13.7	2.6
Newport News, VA	972 (604)	69.1	28.7	2.2
Norfolk, VA	852 (529)	74.3	24.1	1.6
Philadelphia, PA	1270 (789)	60.9	31.8	7.2
Portsmouth, VA	803 (499)	75.2	23.5	1.3
Savannah, GA	184 (114)	87.9	10.9	1.2

Shipments to Savannah River Site:				
		Percentage in Zone		
Route[a]	*Distance km (mi)*	*Rural*	*Suburban*	*Urban*
MOTSU, NC	615 (382)	77.9	20.5	1.6
Wilmington, NC	601 (373)	78.7	19.7	1.6
From Western Ports				
Truck:				
Long Beach, CA	3931 (2443)	78.8	18.0	3.3
NWS Concord, CA	4482 (2784)	79.4	17.2	3.3
Portland, OR	4635 (2879)	83.9	14.4	1.7
Tacoma, WA	4719 (2931)	84.8	13.9	1.3
Rail:				
Long Beach, CA	5212 (3239)	80.9	15.3	3.7
NWS Concord, CA	5123 (3182)	80.0	16.4	3.6
Portland, OR	5078 (3154)	82.0	15.4	2.6
Tacoma, WA	5096 (3165)	80.4	16.5	3.1
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	1629 (1012)	66.8	32.4	0.8
Hanford Site	4390 (2727)	84.3	14.2	1.5
Idaho National Engineering Laboratory	3721 (2311)	82.8	15.6	1.6
Nevada Test Site	3887 (2414)	83.1	15.1	1.8
Oak Ridge Reservation	610 (379)	59.1	38.5	2.4
Sweetgrass, MT	4147 (2576)	85.2	13.6	1.3
Rail:				
Alexandria Bay, NY	2062 (1281)	53.8	35.5	10.7
Hanford Site	4754 (2953)	84.7	13.5	1.8
Idaho National Engineering Laboratory	3875 (2407)	82.8	15.2	2.0
Nevada Test Site	4571 (2839)	84.5	13.5	1.9
Oak Ridge Reservation	671 (417)	68.8	29.8	1.4
Sweetgrass, MT	3903 (2424)	79.4	17.8	2.8

[a] *Route characteristics were generated using the routing models HIGHWAY (Johnson et al., 1993a) and INTERLINE (Johnson et al., 1993b) for truck and rail modes, respectively.*

Table E-4 Summary of the Population Distributions Along Routes for Truck and Rail Modes

Shipments to Hanford Site:				
		Average Persons/km^2		
Route[a]	*Number of Affected Persons*[b]	*Rural*	*Suburban*	*Urban*
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	550,000	7.0	342.5	2149.1
Charleston, SC (Wando Terminal)	569,000	7.0	346.1	2158.6
Elizabeth, NJ	585,000	7.8	318.3	2233.1
Galveston, TX	575,000	4.9	401.5	2139.5
Jacksonville, FL	643,000	7.1	338.6	2180.5
Newport News, VA	677,000	7.5	356.9	2254.3
Norfolk, VA	694,000	7.6	362.0	2219.3
Philadelphia, PA	622,000	7.4	317.4	2079.3
Portsmouth, VA	718,000	7.5	364.3	2243.7
Savannah, GA	602,000	6.8	344.2	2205.1
MOTSU, NC	548,000	7.6	332.1	2146.8

Shipments to Hanford Site:				
Route[a]	***Number of Affected Persons***[b]	***Average Persons/km²***		
		Rural	***Suburban***	***Urban***
Wilmington, NC	556,000	7.5	330.0	2149.9
Rail:				
Charleston, SC (NWS)	731,000	6.9	354.6	2296.8
Charleston, SC (Wando Terminal)	731,000	6.9	354.6	2296.8
Elizabeth, NJ	1,380,000	7.2	355.0	2506.4
Galveston, TX	347,000	4.8	374.8	2034.6
Jacksonville, FL	657,000	6.9	343.6	2272.5
Newport News, VA	936,000	7.5	329.1	2623.1
Norfolk, VA	960,000	7.6	338.8	2592.7
Philadelphia, PA	1,350,000	7.1	358.5	2567.1
Portsmouth, VA	934,000	7.6	334.2	2608.4
Savannah, GA	641,000	7.0	343.1	2244.5
MOTSU, NC	739,000	7.7	346.1	2288.1
Wilmington, NC	736,000	7.7	346.7	2288.1
From Western Ports				
Truck:				
Long Beach, CA	617,000	7.9	381.0	2693.6
NWS Concord, CA	263,000	9.3	335.1	2159.0
Portland, OR	85,700	6.3	413.3	2088.6
Tacoma, WA	98,600	7.7	321.9	2120.5
Rail:				
Long Beach, CA	783,000	3.6	471.4	2781.1
NWS Concord, CA	419,000	7.0	368.7	2363.7
Portland, OR	99,500	6.1	450.0	2294.4
Tacoma, WA	136,000	10.6	355.9	2161.1
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	612,000	7.7	300.4	2211.8
Idaho National Engineering Laboratory	82,800	5.5	363.0	2034.6
Nevada Test Site	305,000	4.1	447.3	2176.8
Oak Ridge Reservation	429,000	6.0	351.1	2207.3
Savannah River	599,000	6.7	354.7	2198.1
Sweetgrass, MT	106,000	4.5	314.4	2152.3
Rail:				
Alexandria Bay, NY	1,170,000	7.0	360.2	2584.5
Idaho National Engineering Laboratory	95,400	4.2	373.6	1935.8
Nevada Test Site	157,000	3.5	402.3	1980.5
Oak Ridge Reservation	410,000	6.7	375.7	2220.3
Savannah River	690,000	6.8	355.8	2267.6
Sweetgrass, MT	92,400	4.1	394.4	1979.9

Shipments to Idaho National Engineering Laboratory:				
Route[a]	***Number of Affected Persons***[b]	***Average Persons/km²***		
		Rural	***Suburban***	***Urban***
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	498,000	7.4	334.0	2157.4
Charleston, SC (Wando Terminal)	518,000	7.4	338.1	2167.2
Elizabeth, NJ	536,000	8.5	315.6	2257.0
Galveston, TX	526,000	5.1	405.8	2149.1

Shipments to Idaho National Engineering Laboratory:				
Route[a]	***Number of Affected Persons***[b]	***Average Persons/km²***		
		Rural	***Suburban***	***Urban***
Jacksonville, FL	576,000	7.6	332.2	2224.8
Newport News, VA	628,000	8.0	356.8	2274.7
Norfolk, VA	631,000	8.1	362.5	2220.6
Philadelphia, PA	573,000	7.9	314.9	2084.2
Portsmouth, VA	670,000	8.0	364.7	2261.3
Savannah, GA	553,000	7.3	343.2	2224.8
MOTSU, NC	463,000	8.1	327.9	2155.2
Wilmington, NC	507,000	8.1	328.1	2166.9
Rail:				
Charleston, SC (NWS)	671,000	7.6	348.5	2332.6
Charleston, SC (Wando Terminal)	671,000	7.6	348.5	2332.6
Elizabeth, NJ	1, 320,000	8.2	350.8	2528.3
Galveston, TX	286,000	5.1	365.7	2068.7
Jacksonville, FL	597,000	7.7	336.3	2312.5
Newport News, VA	875,000	8.4	321.1	2665.7
Norfolk, VA	900,000	8.5	331.7	2632.7
Philadelphia, PA	1,290,000	8.1	354.2	2592.0
Portsmouth, VA	874,000	8.6	326.6	2650.6
Savannah, GA	580,000	7.8	335.9	2284.9
MOTSU, NC	679,000	8.7	340.2	2328.4
Wilmington, NC	675,000	8.7	340.8	2328.4
From Western Ports				
Truck:				
Long Beach, CA	692,000	3.8	487.0	2641.1
NWS Concord, CA	271,000	3.5	411.6	2181.5
Portland, OR	143,000	5.6	395.0	2082.7
Tacoma, WA	157,000	6.1	336.5	2098.8
Rail:				
Long Beach, CA	722,000	3.5	484.6	2830.1
NWS Concord, CA	198,000	4.4	337.2	2293.0
Portland, OR	116,000	4.3	330.2	2222.6
Tacoma, WA	199,000	6.1	326.5	2291.5
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	564,000	8.3	296.8	2230.8
Hanford Site	82,800	5.5	363.0	2034.6
Nevada Test Site	256,000	3.9	470.3	2201.5
Oak Ridge Reservation	380,000	6.3	350.4	2237.4
Savannah River	551,000	7.2	354.4	2217.9
Sweetgrass, MT	38,900	4.3	348.1	2057.3
Rail:				
Alexandria Bay NY	1,110,000	7.9	355.3	2614.7
Hanford Site	95,400	4.2	373.6	1935.8
Nevada Test Site	96,100	3.3	384.6	2022.2
Oak Ridge Reservation	350,000	7.5	365.3	2270.2
Savannah River	630,000	7.6	349.6	2303.3
Sweetgrass, MT	134,000	4.2	338.5	2068.1

Shipments to Nevada Test Site:				
Route[a]	Number of Affected Persons[b]	Average Persons/km^2		
		Rural	Suburban	Urban
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	540,000	6.7	347.4	2179.0
Charleston, SC (Wando Terminal)	559,000	6.7	351.2	2186.6
Elizabeth, NJ	782,000	7.5	343.2	2300.4
Galveston, TX	595,000	4.1	463.8	2277.2
Jacksonville, FL	639,000	6.9	344.9	2265.9
Newport News, VA	691,000	7.3	370.6	2301.4
Norfolk, VA	694,000	7.4	375.5	2257.2
Philadelphia, PA	756,000	7.6	349.2	2199.9
Portsmouth, VA	732,000	7.3	378.0	2287.8
Savannah, GA	616,000	6.5	357.1	2265.9
MOTSU, NC	619,000	8.6	336.2	2218.3
Wilmington, NC	570,000	7.4	341.6	2229.9
Rail:				
Charleston, SC (NWS)	733,000	6.5	362.0	2314.6
Charleston, SC (Wando Terminal)	733,000	6.5	362.0	2314.6
Elizabeth, NJ	1,390,000	6.8	360.2	2511.5
Galveston, TX	231,000	4.3	374.6	2124.4
Jacksonville, FL	659,000	6.6	350.9	2294.0
Newport News, VA	938,000	7.1	335.5	2629.1
Norfolk, VA	963,000	7.3	345.5	2599.2
Philadelphia, PA	1,350,000	6.7	364.2	2571.7
Portsmouth, VA	936,000	7.3	340.8	2614.8
Savannah, GA	643,000	6.7	350.2	2268.7
MOTSU, NC	742,000	7.4	352.3	2308.6
Wilmington, NC	738,000	7.4	353.0	2308.6
From Western Ports				
Truck:				
Long Beach, CA	518,000	2.6	550.5	2817.2
NWS Concord, CA	437,000	3.8	559.2	2617.4
Portland, OR	375,000	4.3	452.0	2154.7
Tacoma, WA	379,000	4.7	409.3	2174.8
Rail:				
Long Beach, CA	628,000	3.4	522.1	2934.2
NWS Concord, CA	407,000	6.2	360.9	2313.2
Portland, OR	177,000	3.6	376.6	2183.0
Tacoma, WA	261,000	005	357.6	2251.6
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay NY	644,000	7.5	308.8	2262.5
Hanford Site	305,000	4.1	447.3	2176.8
Idaho National Engineering Laboratory	256,000	3.9	470.3	2201.5
Oak Ridge Reservation	443,000	5.5	370.8	2291.1
Savannah River	613,000	6.4	368.3	2261.8
Sweetgrass, MT	304,000	4.0	455.9	2167.2
Rail:				
Alexandria Bay NY	1,170,000	6.6	366.8	2589.6
Hanford Site	157,000	3.5	402.3	1980.5

Shipments to Nevada Test Site:				
Route[a]	***Number of Affected Persons***[b]	***Average Persons/km²***		
		Rural	***Suburban***	***Urban***
Idaho National Engineering Laboratory	96,100	3.3	384.6	2022.2
Oak Ridge Reservation	413,000	6.3	394.1	2252.1
Savannah River	692,000	6.4	363.7	2287.5
Sweetgrass, MT	196,000	3.7	370.8	2067.5

Shipments to Oak Ridge Reservation:				
Route[a]	***Number of Affected Persons***[b]	***Average Persons/km²***		
		Rural	***Suburban***	***Urban***
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	108,000	15.0	297.5	1842.7
Charleston, SC (Wando Terminal)	127,000	14.7	311.5	2027.0
Elizabeth, NJ	290,000	19.9	273.2	2343.1
Galveston, TX	337,000	13.5	330.3	2358.5
Jacksonville, FL	175,000	15.3	266.0	2322.6
Newport News, VA	209,000	18.5	286.8	2316.6
Norfolk, VA	174,000	17.6	292.1	2073.3
Philadelphia, PA	335,000	18.8	335.1	2215.7
Portsmouth, VA	251,000	18.2	309.4	2270.2
Savannah, GA	101,000	14.1	274.1	1764.7
MOTSU, NC	128,000	17.6	283.4	1854.5
Wilmington, NC	128,000	16.7	280.4	1764.7
Rail:				
Charleston, SC (NWS)	194,000	17.4	272.6	2202.7
Charleston, SC (Wando Terminal)	194,000	17.4	272.6	2202.7
Elizabeth, NJ	949,000	17.1	353.3	2694.7
Galveston, TX	471,000	13.4	360.1	2306.2
Jacksonville, FL	235,000	11.6	335.9	2233.3
Newport News, VA	305,000	16.8	277.3	2175.5
Norfolk, VA	241,000	17.3	270.6	2077.6
Philadelphia, PA	649,000	16.9	333.2	2653.3
Portsmouth, VA	215,000	17.8	259.8	2075.0
Savannah, GA	218,000	12.1	335.1	2090.0
MOTSU, NC	186,000	17.7	263.7	2038.3
Wilmington, NC	183,000	17.8	263.3	2038.3
From Western Ports				
Truck:				
Long Beach, CA	823,000	6.2	380.8	2607.0
NWS Concord, CA	742,000	6.0	401.1	2426.5
Portland, OR	519,000	6.0	367.2	2195.4
Tacoma, WA	431,000	6.2	343.7	2213.8
Rail:				
Long Beach, CA	995,000	6.6	398.5	2736.4
NWS Concord, CA	664,000	6.9	379.3	2317.2
Portland, OR	765,000	7.6	393.4	2272.1
Tacoma, WA	919,000	7.7	407.7	2330.2
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	257,000	19.9	258.2	1896.7
Hanford Site	429,000	6.0	351.0	2207.3

Shipments to Oak Ridge Reservation:				
Route[a]	Number of Affected Persons[b]	Average Persons/km^2		
		Rural	Suburban	Urban
Idaho National Engineering Laboratory	380,000	6.3	350.0	2237.4
Nevada Test Site	443,000	5.5	371.0	2291.1
Savannah River	175,000	17	318.0	2244.1
Sweetgrass, MT	346,000	6.3	336.3	2180.9
Rail:				
Alexandria Bay NY	752,000	18.2	378.0	2443.0
Hanford Site	416,000	6.7	376.0	2220.3
Idaho National Engineering Laboratory	350,000	7.5	365.0	2270.2
Nevada Test Site	413,000	6.3	394.0	2252.1
Savannah River	132,000	15.2	289.0	2164.4
Sweetgrass, MT	627,000	8.7	395.5	2256.5

Shipments to Savannah River Site:				
Route[a]	Number of Affected Persons[b]	Average Person/km^2		
		Rural	Suburban	Urban
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	46,200	16.3	275.0	1764.7
Charleston, SC (Wando Terminal)	65,700	15.6	306.1	2077.9
Elizabeth, NJ	316,000	17.6	277.6	2377.5
Galveston, TX	430,000	12.7	359.1	2254.1
Jacksonville, FL	46,900	13.2	211.4	1764.7
Newport News, VA	181,000	16.2	302.9	2281.5
Norfolk, VA	131,000	16.4	283.9	2007.9
Philadelphia, PA	397,000	16.5	348.5	2228.9
Portsmouth, VA	135,000	16.3	281.8	2033.1
Savannah, GA	37,300	13.6	233.4	1764.7
MOTSU, NC	34,200	15.0	213.0	1925.6
Wilmington, NC	64,700	17.7	256.7	1764.7
Rail:				
Charleston, SC (NWS)	41,200	6.8	328.6	2735.5
Charleston, SC (Wando Terminal)	41,200	6.8	328.6	2735.5
Elizabeth, NJ	903,000	14.2	353.0	2726.4
Galveston, TX	609,000	11.9	394.0	2330.3
Jacksonville, FL	72,200	10.6	290.3	2466.3
Newport News, VA	218,000	13.2	285.6	2444.8
Norfolk, VA	153,000	13.5	275.3	2469.8
Philadelphia, PA	603,000	14.0	328.8	2704.1
Portsmouth, VA	128,000	13.9	253.7	2615.8
Savannah, GA	21,300	9.6	309.0	2707.8
MOTSU, NC	99,000	12.7	260.9	2580.4
Wilmington, NC	95,500	12.8	259.6	2580.4
From Western Ports				
Truck:				
Long Beach, CA	714,000	7.5	369.4	2905.8
NWS Concord, CA	1,040,000	7.3	378.5	2381.7
Portland, OR	686,000	6.7	365.2	2188.9
Tacoma, WA	601,000	6.8	349.2	2202.0
Rail:				
Long Beach, CA	1,280,000	6.9	359.9	2653.0

Shipments to Savannah River Site:				
Route[a]	***Number of Affected Persons***[b]	***Average Person/km*** ^{2}		
		Rural	***Suburban***	***Urban***
NWS Concord, CA	1,210,000	7.1	381.6	2369.0
Portland, OR	950,000	7.5	369.0	2246.2
Tacoma, WA	1,100,000	7.5	381.0	2300.7
From DOE Sites/Canadian Border				
Truck:				
Alexandria, Bay NY	284,000	18.0	262.7	2072.4
Hanford Site	599,000	6.7	354.7	2198.1
Idaho National Engineering Laboratory	551,000	7.2	354.4	2217.9
Nevada Test Site	613,000	6.4	368.3	2261.8
Oak Ridge Reservation	175,000	17.0	317.7	2244.1
Sweetgrass, MT	513,000	7.1	344.0	2175.8
Rail:				
Alexandria Bay, NY	1,340,000	14.8	333.1	2756.8
Hanford Site	690,000	6.8	355.8	2267.6
Idaho National Engineering Laboratory	630,000	7.6	349.6	2303.3
Nevada Test Site	692,000	6.4	363.7	2287.5
Oak Ridge Reservation	132,000	15.2	289.2	2164.4
Sweetgrass, MT	812,000	8.4	367.3	2228.5

[a] *Route characteristics were generated using the routing models HIGHWAY (Johnson et al., 1993a) and INTERLINE (Johnson et al., 1993b) for truck and rail modes, respectively.*

[b] *The affected population includes all persons within 800 m (0.5 mi) of the route.*

The representative truck and rail routes were determined by using the routing models HIGHWAY (Johnson et al., 1993a) and INTERLINE (Johnson et al., 1993b), respectively. These models are described briefly below.

The HIGHWAY computer program is used for selecting highway routes for transporting radioactive materials within the United States by truck. The HIGHWAY data base is a computerized road atlas that currently describes approximately 386,400 kilometer (km) (240,000 mi) of roads. A complete description of the Interstate System and all United States highways is included in the database. In addition, most of the principal State highways and a number of local and community highways are also identified. The code is updated periodically to reflect current road conditions and has been benchmarked against reported mileages and observations of commercial truck firms.

Routes are calculated within the model by minimizing the total impedance between the origin and the destination. The impedance is basically defined as a function of distance and driving time along a particular highway segment. One of the special features of the HIGHWAY model is its ability to calculate routes that maximize the use of interstate highways. This feature allows the user to select routes for shipment of radioactive materials that conform to Department of Transportation regulations, specifically 49 CFR 397 Subpart D. The population densities along a route are derived from 1990 U.S. Bureau of the Census data. Rural, suburban, and urban areas are characterized according to the following breakdown: rural population densities range from 0 to 54 persons per km^2 (0 to 139 persons per mi^2); the suburban range is 55 to 1,284 persons per km^2 (140 to 3,326 persons per mi^2); and urban is taken as all population densities greater than 1,284 persons per km^2 (3,326 persons per mi^2).

The INTERLINE computer program is designed to simulate routing of the United States rail system. The INTERLINE database consists of 94 separate subnetworks and represents various competing rail companies in the United States. The database used by INTERLINE was originally based on Federal Railroad Administration data and reflected the United States railroad system in 1974. The data base has since been expanded and modified over the past 2 decades. The code is updated periodically to reflect current track conditions and has been benchmarked against reported mileages and observations of commercial rail firms.

The INTERLINE model uses a shortest-route algorithm that finds the minimum impedance path within an individual subnetwork. A separate routine is used to find paths along the subnetworks. The routes selected for this study used the standard assumptions in the INTERLINE model that simulate the selection process that railroads would use to direct shipments of spent nuclear fuel. The population densities along a route are derived from 1990 U.S. Bureau of the Census data. Rural, suburban, and urban areas are characterized according to the following breakdown: rural population densities range from 0 to 54 persons per km^2 (0 to 139 persons per mi^2); the suburban range is 55 to 1,284 persons per km^2 (140 to 3,326 persons per mi^2); and urban is taken as all population densities greater than 1,284 persons per km^2 (3,326 persons per mi^2).

E.5 Methods for Calculating Transportation Risks

The overland transportation risk assessment approach is summarized in Figure E-13. The first step in the ground transportation analysis was to determine the incident-free and accident risk factors, on a per-shipment basis, for transportation of the various types of spent nuclear fuel. Risk factors, as any risk estimate, are the product of the probability of exposure and the magnitude of the exposure. Accident risk factors were calculated for radiological and nonradiological traffic accidents. The probabilities, which are much lower than one, and the magnitudes of exposure were multiplied, yielding very low risk numbers. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the cask and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure is unity (one).

Radiological risk factors are expressed in units of rem. Later in the analysis, they will be multiplied by International Commission on Radiation Protection Publication 60 (ICRP, 1991) conversion factors and estimated numbers of shipments (see Section E.7.1) to give risk estimates in units of LCFs. The vehicle emission risk factors are calculated in latent mortalities, and the vehicle accident risk factors are calculated in mortalities. The nonradiological risk factors will be multiplied by the number of shipments.

For each alternative, risks were assessed for both incident-free transportation and accident conditions. For the incident-free assessment, risks were calculated for both collective populations of potentially exposed individuals and for MEIs. The accident assessment consists of two components: 1) a probabilistic accident risk assessment that considers the probabilities and consequences of a range of possible transportation accident environments, including low-probability accidents that have high consequences and high-probability accidents that have low consequences; and 2) an accident consequence assessment that considers only the consequences of the most severe transportation accidents postulated.

The RADTRAN 4 computer code (Neuhauser and Kanipe, 1993) is used for the incident-free and accident risk assessments to estimate the impacts to collective populations. RADTRAN 4 was developed by Sandia National Laboratories to calculate population risk associated with the transportation of radioactive materials by a variety of modes, including truck, rail, air, ship, and barge.

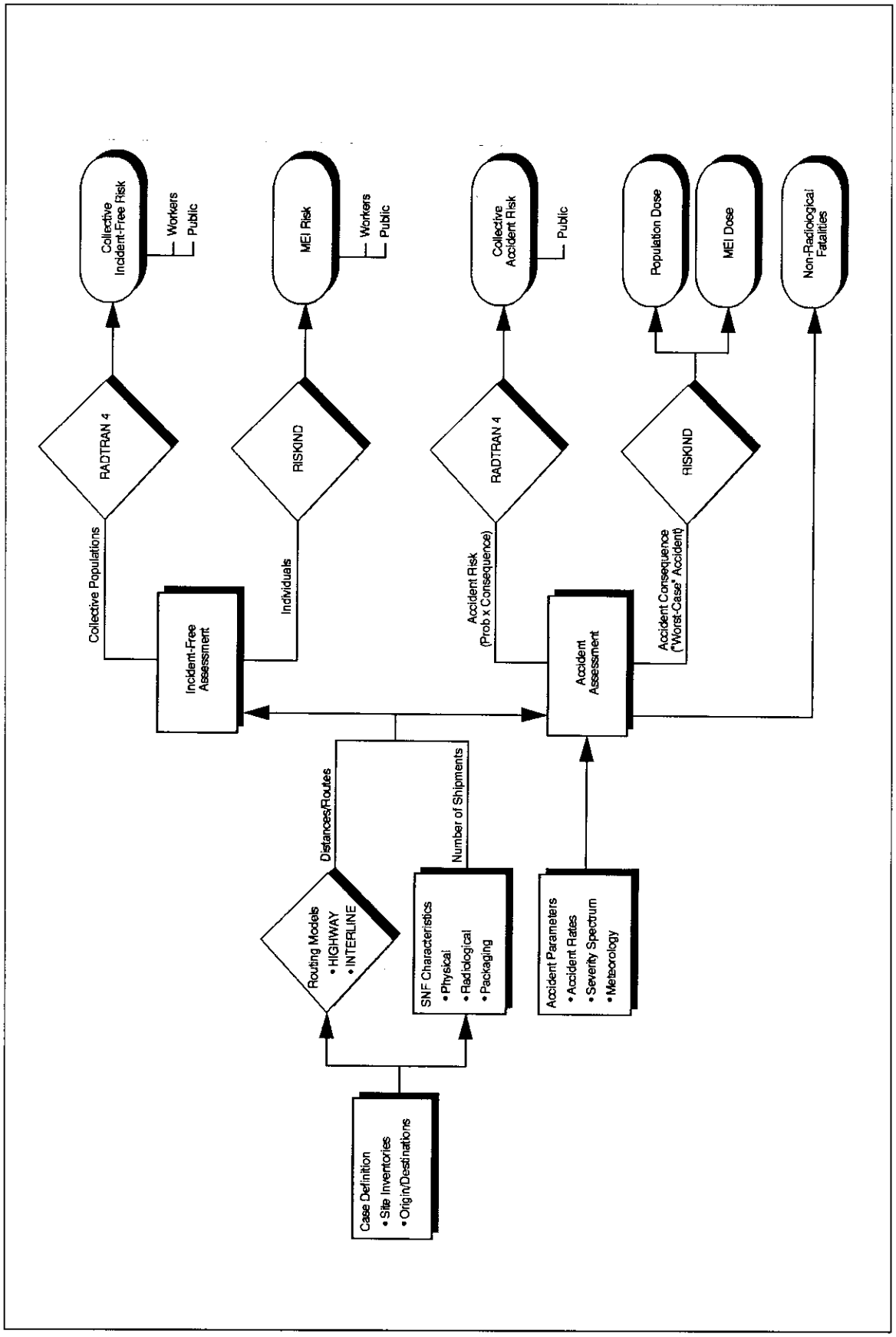

Figure E-13 Summary of the Assessment Approach for the Overland Transportation Risk Assessment

The RADTRAN 4 population risk calculations take into account both the consequences and probabilities of potential exposure events. The collective population risk is a measure of the total radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing the various alternatives.

The RISKIND computer code (Yuan et al., 1993) is used to estimate the incident-free doses to MEIs and for estimating impacts for the accident consequence assessment. The RISKIND computer code was developed for DOE's Office of Civilian Radioactive Waste Management to analyze the exposure of individuals during the incident-free transportation of spent nuclear fuel. In addition, the RISKIND code was designed to allow a detailed assessment of the consequences to individuals and population subgroups from severe spent nuclear fuel transportation accidents under various environmental settings.

The RISKIND calculations were conducted to supplement the collective risk results calculated with RADTRAN 4. Whereas the collective risk results provide a measure of the overall risks of each alternative, the RISKIND calculations are meant to address areas of specific concern to individuals and population subgroups. Essentially, the RISKIND analyses are meant to address "What if" questions, such as, "What if I live next to a site access road?" or "What if an accident happens near my town?"

E.5.1 Incident-Free Risk Assessment Methodology

Radiological dose during normal, incident-free transportation of spent nuclear fuel results from exposure to the external radiation field that surrounds the shipping containers. The dose is a function of the number of people exposed, their proximity to the containers, their length of time of exposure, and the intensity of the radiation field surrounding the containers.

Collective Population Risk: The consequences (dose) during incident-free conditions are expected to occur, therefore, the probability of incident-free consequences is taken to be unity (one) in the RADTRAN 4 code. The radiological risk associated with incident-free transportation conditions results from the potential exposure of people to external radiation in the vicinity of loaded shipments. The maximum allowable external dose rates for exclusive-use shipments were presented in Section E.3.

For incident-free transportation conditions, the RADTRAN 4 computer code considers all major groups of potentially exposed persons. The RADTRAN 4 risk calculations for incident-free highway and rail transportation include exposures of the following population groups:

- Persons along the route (off-link population): Collective doses are calculated for all persons living or working within 800 m (0.5 mi) on each side of a transportation route. The total number of persons within the 1.6 km (1 mi) corridor is calculated separately for each route considered in the assessment.

- Persons sharing the route (on-link population): Collective doses are calculated for persons in all vehicles sharing the transportation route. This group would include persons traveling in the same or opposite direction as the shipment, as well as persons in vehicles passing the shipment.

- Persons at stops: Collective doses are calculated for people who could be exposed while a shipment was stopped en route. For truck transportation, this would include refueling stops, food stops, and rest stops. For rail transportation, stops are assumed to occur for classification purposes.

- Crew Members: Collective doses are calculated for truck and rail transportation crew members.

The doses calculated for the first three population groups are added together to yield the collective dose to the general public. The dose calculated for the fourth group represents the collective dose to workers. The RADTRAN 4 incident-free dose models are not intended to be used for estimating specific risks to individuals.

The RADTRAN 4 incident-free dose calculations are based on expressing the dose rate as a function of distance from a point source (Neuhauser and Kanipe, 1993). Associated with the calculation of incident-free doses for each exposed population group are parameters such as the radiation field strength, source-receptor distance, exposure time, vehicle speed, stop time, traffic density, and route characteristics such as population density. The RADTRAN 4 code user's manual contains derivations of the equations and descriptions of these parameters (Neuhauser and Kanipe, 1993). The values for many of the most important parameters are presented in Section E.6.

The collective incident-free risks are calculated for each specific alternative as follows. Each alternative is first defined as a set of origin and destination pairs. Representative highway and rail routes are determined for each unique pair as described in Section E.4. For each pair, RADTRAN 4 is used to calculate the collective risks to workers and the public for a single shipment based on representative radiological and physical properties of the spent nuclear fuel. These estimates for a single shipment are referred to as per-shipment risk factors. The number of shipments transported across each linkage is then determined for both truck and rail modes. The collective risks for an alternative are calculated by multiplying the number of shipments by the appropriate per-shipment risk factor.

MEI Risk: In addition to the incident-free collective population risk assessment, the risk to MEIs has been estimated for a number of hypothetical exposure events using RISKIND. The receptors include transportation crew members, inspectors, and members of the public exposed during traffic delays, while working at a service station, or living near a port of entry or DOE site.

The dose to each MEI considered is calculated with RISKIND for a given distance, duration, and frequency of exposure specific to that receptor. The distances and durations of exposure are similar to those given in previous transportation assessments and are presented in Section E.6. The exposure scenarios are not meant to be exhaustive, but were selected to provide a realistic range of potential exposure situations.

The RISKIND external dose model considers direct external exposure and exposure from radiation scattered from the ground and air. RISKIND is used to calculate the dose as a function of distance (mrem per hr for stationary exposures and mrem per event for moving shipments) from a spent nuclear fuel shipment based on the dimensions of the shipment. The code models the shipment as a cylindrical volume source; and the calculated dose includes contributions from buildup, cloudshine, and groundshine. The dose rates calculated by using RISKIND have been compared with output from existing shielding codes. The RISKIND code has been found to produce realistic but conservative results. As a conservative measure, potential shielding between the cask and the receptor is not considered.

Nonradiological Risk (Vehicle Related): Vehicle-related health risks resulting from incident-free transport may be associated with the generation of air pollutants by transport vehicles during spent nuclear fuel shipment, and are independent of the radioactive nature of the shipment. The health end point assessed under incident-free transport conditions is the excess latent mortality due to inhalation of vehicle exhaust emissions. Risk factors for pollutant inhalation in terms of latent mortality have been generated (Rao et

al., 1982). These risks are $1x10^{-7}$ mortality per km ($1.6x10^{-7}$ per mi) and $1.3x10^{-7}$ mortality per km ($2.1x10^{-7}$ per mi) of truck and rail travel in urban areas, respectively. The risk factors are based on regression analyses of the effects of sulfur dioxide and particulate releases from diesel exhaust on mortality rates. Excess latent mortalities are assumed to be equivalent to LCF. Vehicle-related risks from incident-free transportation are calculated for each case by multiplying the total distance traveled in urban areas by the appropriate risk factor. Similar data are not available for rural and suburban areas.

Risks are summed over the entire route and over all shipments for each spent nuclear fuel case. This method has been used in several reports to calculate risks from incident-free transport. Lack of information for rural and suburban areas is an obvious data gap, although the risk factor would presumably be lower than for urban areas because of lower total emissions from all sources and lower population densities in rural and suburban areas.

E.5.2 Accident Assessment Methodology

The offsite spent nuclear fuel transportation accident analysis considers the impacts of accidents during the transportation of spent nuclear fuel by truck or rail. Under accident conditions, impacts to human health and the environment could result from the release and dispersal of radioactive material. Because of the rigorous design specifications for spent nuclear fuel shipping casks, the NRC has estimated that casks will withstand 99.4 percent of truck or rail accidents without sustaining damage sufficient to breach the cask (Fischer et al., 1987). The 0.6 percent of accidents that could potentially breach the cask are represented by a spectrum of accident severities and radioactive material release conditions. Accident analysis methodology has been developed by the NRC for calculating the probabilities and consequences from this spectrum of unlikely accidents, but it is not possible to predict where along the shipping route such accidents might occur. To provide an assessment of spent nuclear fuel transportation accident impacts, two types of analyses were performed. First, an accident risk assessment was performed that takes into account the probabilities and consequences of a spectrum of accident severities using methodology developed by the NRC (Fischer et al., 1987). The accident risk assessment used route-specific information for accident rates and population densities. For the spectrum of accidents considered in the analysis, accident consequences in terms of collective dose to the population within 80 km (50 mi) were multiplied by the accident probabilities to yield dose risk. Second, to represent the maximum reasonably foreseeable impacts to individuals and populations should an accident occur, radiological consequences were calculated for an accident of maximum credible severity in each population zone. An accident is considered credible if its probability of occurrence is greater than 1×10^{-7} per yr.

Accident Risk Assessment: The risk analysis of potential accidents differs from the incident-free analysis because accident occurrences are statistical in nature. The accident risk assessment is treated probabilistically in RADTRAN 4. Accident risk is defined as the product of the accident consequence (dose) and the probability of the accident occurring. In this respect, the RADTRAN 4 code estimates the collective accident risk to populations by considering a spectrum of transportation accidents. The accident spectrum is designed to encompass a range of possible accident environments, including low-probability accidents that have high consequences and high-probability accidents that have low consequences (i.e., "fender benders"). The collective accident risk results can be directly compared with the incident-free collective risk results because they incorporate the probabilities of accident occurrences.

The RADTRAN 4 calculation of collective accident risk employs models that quantify the range of potential accident severities and the responses of transport packages (i.e., casks) to accident environments. The accident severity spectrum is divided into a number of accident severity categories. Each severity category is assigned a conditional probability of occurrence; that is, the probability that an accident will be

of a particular severity if an accident occurs. The more severe the accident, the more remote the chance of such an accident. Release fractions, defined as the fraction of the material in a package that could be released in an accident, are assigned to each accident severity category based on the physical and chemical form of the spent nuclear fuel. The models take into account the transportation mode and the type of packaging being considered. The accident rates, definition of accident severity categories, and release fractions used in this analysis are discussed further in Section E.6.

For accidents involving the release of radioactive material, RADTRAN 4 assumes the material is dispersed in the environment according to standard Gaussian diffusion models. For the risk assessment, default atmospheric dispersion data were used representing an instantaneous ground-level release and a small diameter source cloud (Neuhauser and Kanipe, 1993). The calculation of collective population dose following the release and dispersal of radioactive material includes the following exposure pathways:

- external exposure to the passing radioactive cloud,
- external exposure to contaminated ground,
- internal exposure from inhalation of airborne contaminants, and
- internal exposure from the ingestion of contaminated food.

For the ingestion pathway, state-specific food transfer factors, which relate the amount of radioactive material ingested by people to the amount deposited on the ground, were derived in accordance with the methods described by NRC Guide 1.109 (NRC, 1977b). Radiation doses are calculated using standard dose conversion factors in DOE/EH-0070 (DOE, 1988a) and DOE/EH-0071 (DOE, 1988b).

The collective accident risk for each alternative is determined in a manner similar to that described for incident-free collective risks. Accident risks are first calculated for each unique origin and destination pair ("per-shipment" risk factors) and then summed over all pairs to estimate the total risk for the alternative. The accident risk assessment uses site- and spent nuclear fuel-type-specific radiological and physical characteristics, described further in Section E-6. In addition, the assessment uses route-specific population density information and accident rates derived for individual States.

Accident Consequence Assessment: The RISKIND code is used to provide a detailed assessment of the consequences of the most severe transportation accidents. Whereas the RADTRAN 4 accident risk assessment considers the entire range of accident severities and their related probabilities, the RISKIND accident consequence assessment assumes that an accident of the highest credible severity has occurred. The accident consequence assessment is intended to provide an estimate of the maximum potential impact posed by a severe transportation accident involving spent nuclear fuel.

The severe accidents considered in the consequence assessment are characterized by extreme mechanical and thermal forces. In all cases, these accidents result in a release of radioactive material to the environment. The accidents correspond to those within the highest accident severity category as described above. These accidents represent low-probability, high-consequence events. The probability of accidents of this magnitude occurring for each alternative depends on the total shipment distance. However, accidents of this severity are extremely rare in general.

RISKIND was used for the accident consequence assessment for two reasons. First, the code has the ability to model the complex atmospheric dispersion present in severe accident environments. The atmospheric dispersion is modeled as an instantaneous release using standard Gaussian puff methods. In addition, because severe accidents routinely involve fires, modeling of the potential radiological

consequences takes into account physical phenomena resulting from the fire, such as buoyant plume rise. Second, RISKIND can be used to estimate the dose to MEIs in the vicinity of an accident. The location of the MEI is determined by RISKIND based on the atmospheric conditions assumed at the time of the accident and thermal characteristics of the release.

The consequences of the most severe accidents are calculated for both local populations and MEIs. The population dose includes the population within 80 km (50 mi) of the accident site. The exposure pathways considered are similar to those discussed above for the accident risk assessment. Although post-accident remedial activities (e.g., immediate evacuation of the public or cleanup of dispersed radioactive material) would reduce the consequences of an accident, these activities were not given credit in the dose calculations.

Because it is impossible to predict the exact location of a severe transportation accident, separate accident consequences are calculated for accidents occurring in rural, suburban, and urban population density zones. Moreover, to address the effects of the atmospheric conditions existing at the time of an accident, two different atmospheric conditions are considered. The first case assumes neutral atmospheric conditions, and the second, stable conditions. Atmospheric conditions are discussed further in Section E.6.

Nonradiological Accident Risk Assessment: The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities that are not related to the shipment cargo. This risk represents fatalities from mechanical causes. State-specific transportation fatality rates are used in the assessment and are discussed in Section E.6. Nonradiological accident risks are calculated for each alternative by multiplying the total distance traveled in each State by the appropriate State fatality rate. In all cases, the nonradiological accident risks are calculated using round-trip shipment distances.

E.6 Input Parameters and Assumptions

The transportation risk assessment is designed to ensure—through uniform and judicious selection of models, data, and assumptions—that relative comparisons of risk among the various alternatives are meaningful. The major input parameters and assumptions used in the transportation risk assessment are discussed below.

Appendix B lists the casks that are being considered for intersite shipments. The sizes of casks identified vary considerably. Since it is not clear what size of cask would be used for intersite shipments, and since the shipments would not begin until 2005, hypothetical cask sizes are used in this assessment. Additionally, fuel that arrives at an interim site would be physically modified, depending on the dry or wet storage option chosen. Therefore, it is assumed that if spent nuclear fuel were shipped by truck, the number of shipments would be one-quarter of the number of shipments from ports. If the spent nuclear fuel were shipped by rail, the number of shipments would be one-tenth of the number of shipments from ports.

E.6.1 Spent Nuclear Fuel Inventory and Characterization Data

For the purposes of analysis, the foreign research reactor spent nuclear fuel has been characterized into five different spent nuclear fuel categories for shipments into ports and two for shipments between DOE sites. The detailed discussion of the fuel and casks is provided in Appendix B. The curie content of fully loaded shipments is summarized in Table E-5. The approach for calculating the number of shipments from the various countries is shown in Appendix B.

Table E-5 Curie Content of Fully Loaded Shipping Casks for Representative Fuel Types

Isotopes	Material Type					
	BR-2	RHF	TRIGA	NRU	HLW (1 yr)	Target
Tritium	$8.64x10^{+1}$	$3.70x10^{+1}$	$1.31x10^{+1}$	$9.48x10^{-1}$	~ 0	~ 0
Krypton 85	$2.47x10^{+3}$	$1.07x10^{+3}$	$3.63x10^{+2}$	$2.71x10^{+3}$	~ 0	~ 0
Strontium 89	$4.08x10^{+4}$	$1.76x10^{+4}$	$2.75x10^{+3}$	$9.72x10^{+3}$	$3.07x10^{+6}$	$1.95x10^{+2}$
Strontium 90	$2.08x10^{+4}$	$8.93x10^{+3}$	$3.16x10^{+3}$	$2.32x10^{+4}$	$1.74x10^{+6}$	$1.58x10^{+2}$
Yttrium 90	$2.08x10^{+4}$	$8.93x10^{+3}$	$3.16x10^{+3}$	$2.32x10^{+4}$	$1.74x10^{+6}$	$1.58x10^{+2}$
Yttrium 91	$7.30x10^{+4}$	$3.14x10^{+4}$	$4.56x10^{+3}$	$2.02x10^{+4}$	$5.48x10^{+6}$	$3.69x10^{+2}$
Zirconium 95	$1.07x10^{+5}$	$4.63x10^{+4}$	$6.48x10^{+3}$	$3.38x10^{+4}$	$8.08x10^{+6}$	$5.67x10^{+2}$
Niobium 95	$2.20x10^{+5}$	$9.49x10^{+4}$	$1.28x10^{+4}$	$7.34x10^{+4}$	$1.65x10^{+7}$	$1.21x10^{+3}$
Ruthenium 103	$8.90x10^{+3}$	$3.77x10^{+3}$	$8.44x10^{+2}$	$1.44x10^{+3}$	$7.16x10^{+5}$	$3.57x10^{+1}$
Rhodium 103m	$8.90x10^{+3}$	$3.77x10^{+3}$	$8.44x10^{+2}$	$1.44x10^{+3}$	$7.16x10^{+5}$	$3.57x10^{+1}$
Ruthenium 106	$2.15x10^{+4}$	$9.16x10^{+3}$	$2.54x10^{+3}$	$1.84x10^{+4}$	$1.88x10^{+6}$	$1.49x10^{+2}$
Rhodium 106m	$2.15x10^{+4}$	$9.16x10^{+3}$	$2.54x10^{+3}$	$1.84x10^{+4}$	$1.88x10^{+6}$	$1.49x10^{+2}$
Tin 123	$4.27x10^{+2}$	$1.84x10^{+2}$	$2.71x10^{+1}$	$2.40x10^{+2}$	$5.84x10^{+4}$	$2.70x10^{+0}$
Antimony 125	$8.90x10^{+2}$	$3.81x10^{+2}$	$1.19x10^{+2}$	$9.12x10^{+2}$	$7.57x10^{+4}$	$6.51x10^{+0}$
Tellurium 125m	$2.12x10^{+2}$	$9.06x10^{+1}$	$2.87x10^{+1}$	$2.21x10^{+2}$	$1.81x10^{+4}$	$1.56x10^{+0}$
Tellurium 127m	$8.87x10^{+2}$	$3.82x10^{+2}$	$5.57x10^{+1}$	$4.42x10^{+2}$	$6.97x10^{+4}$	$5.39x10^{+0}$
Tellurium 129m	$1.89x10^{+2}$	$7.98x10^{+1}$	$2.31x10^{+1}$	$2.30x10^{+1}$	$1.59x10^{+4}$	$6.73x10^{-1}$
Cesium 134	$1.64x10^{+4}$	$4.00x10^{+3}$	$1.16x10^{+3}$	$3.54x10^{+4}$	$1.41x10^{+6}$	$6.12x10^{-1}$
Cesium-137	$2.06x10^{+4}$	$8.87x10^{+3}$	$3.19x10^{+3}$	$2.30x10^{+4}$	$1.74x10^{+6}$	$1.56x10^{+2}$
Cerium 141	$5.74x10^{+3}$	$2.44x10^{+3}$	$6.97x10^{+2}$	$6.65x10^{+3}$	$5.59x10^{+5}$	$2.03x10^{+1}$
Cerium 144	$3.12x10^{+5}$	$1.35x10^{+5}$	$2.55x10^{+4}$	$2.54x10^{+5}$	$2.49x10^{+7}$	$2.18x10^{+3}$
Praseodymium 144	$3.12x10^{+5}$	$1.35x10^{+5}$	$2.55x10^{+4}$	$2.54x10^{+5}$	$2.49x10^{+7}$	$2.18x10^{+3}$
Promethium 147	$4.83x10^{+4}$	$2.46x10^{+4}$	$7.02x10^{+3}$	$2.98x10^{+4}$	$3.70x10^{+6}$	$5.14x10^{+2}$
Promethium 148m	$7.56x10^{+1}$	$2.92x10^{+1}$	$4.68x10^{+1}$	$1.40x10^{+0}$	$7.13x10^{+3}$	$2.43x10^{-2}$
Europium 154	$6.20x10^{+2}$	$1.63x10^{+2}$	$4.18x10^{+1}$	$1.35x10^{+3}$	$6.24x10^{+4}$	$7.90x10^{-2}$
Europium 155	$1.30x10^{+2}$	$4.56x10^{+1}$	$2.27x10^{+1}$	$2.45x10^{+2}$	$1.29x10^{+4}$	$3.35x10^{+0}$
Uranium 234	$9.14x10^{-4}$	$3.74x10^{-4}$	$1.81x10^{-4}$	$1.57x10^{-3}$	~ 0	$6.81x10^{-6}$
Uranium 235	$1.38x10^{-2}$	$1.09x10^{-2}$	$7.91x10^{-3}$	$6.06x10^{-3}$	~ 0	$3.98x10^{-3}$
Uranium 238	$3.41x10^{-4}$	$2.06x10^{-4}$	$6.51x10^{-3}$	$2.67x10^{-5}$	~ 0	$7.22x10^{-5}$
Plutonium 238	$6.42x10^{+1}$	$1.03x10^{+1}$	$3.03x10^{+0}$	$2.70x10^{+2}$	$8.48x10^{+3}$	$1.60x10^{-4}$
Plutonium 239	$1.84x10^{+0}$	$8.89x10^{-2}$	$5.50x10^{-1}$	$3.32x10^{-1}$	$4.05x10^{+2}$	$2.95x10^{-2}$
Plutonium 240	$1.20x10^{+0}$	$4.21x10^{-1}$	$2.09x10^{+0}$	$2.42x10^{-1}$	$3.26x10^{+2}$	$6.85x10^{-4}$
Plutonium 241	$2.84x10^{+2}$	$6.77x10^{+1}$	$2.13x10^{+2}$	$7.09x10^{+1}$	$7.84x10^{+4}$	$7.09x10^{-3}$
Americium 241	$3.96x10^{-1}$	$9.67x10^{-2}$	$4.07x10^{-1}$	$1.24x10^{-1}$	$9.84x10^{+1}$	$1.16x10^{-5}$
Americium 242m	$1.05x10^{-3}$	$1.55x10^{-4}$	$9.00x10^{-3}$	$6.00x10^{-4}$	$6.70x10^{-1}$	$2.13x10^{-10}$
Americium 243	$4.33x10^{-3}$	$3.76x10^{-3}$	$4.38x10^{-4}$	$3.51x10^{-3}$	$1.44x10^{+1}$	$1.47x10^{-10}$
Curium 244	$1.33x10^{+0}$	$9.26x10^{-3}$	$7.14x10^{-3}$	$2.70x10^{-1}$	$1.22x10^{+2}$	$1.63x10^{-10}$
Curium 242	$1.75x10^{+0}$	$1.27x10^{-1}$	$5.25x10^{+0}$	$1.03x10^{+0}$	$9.91x10^{+2}$	$6.86x10^{-8}$

E.6.2 Shipment External Dose Rates

The dose and corresponding risk to populations and MEIs during incident-free transportation conditions are directly proportional to the assumed shipment external dose rate. The Federal regulations for maximum allowable external dose rates for exclusive-use shipments were presented in Section E.3.

The actual shipment dose rate is a complex function of the composition and configuration of shielding and containment materials used in the cask, the geometry of the loaded shipments, and characteristics of the spent nuclear fuel itself. Based on actual measurements of the dose rate outside real shipping casks, a realistic dose rate of 1 mrem per hr at a distance of 2 m (6.6 ft) was estimated, as described in Appendix F.

However, since individual casks would be expected to exceed this average value, the analysis assumes that all casks would be at the regulatory limits of 10 mrem per hr at 2 m (6.6 ft). In practice, external dose rates would vary from spent nuclear fuel type to spent nuclear fuel type and from shipment to shipment.

E.6.3 Accident Involvement Rates

For the calculation of accident risks, vehicle accident and fatality rates are taken from data provided in other reports (Saricks and Kvitek, 1994). For each transport mode, accident rates are generically defined as the number of accident involvements (or fatalities) in a given year per unit of travel of that mode in that same year. Therefore, the rate is a fractional value, with accident-involvement count as the numerator of the fraction and vehicular activity (total travel distance) as its denominator. Accident rates are generally determined for a multi-year period. For assessment purposes, the total number of expected accidents or fatalities is calculated by multiplying the total shipment distance for a specific case by the appropriate accident or fatality rate.

For truck transportation, the rates presented are specifically for heavy combination trucks involved in interstate commerce (Saricks and Kvitek, 1994). Heavy combination trucks are rigs composed of a separable tractor unit containing the engine and one to three freight trailers connected to each other. Heavy combination trucks are typically used for radioactive waste shipments. The truck accident rates are computed for each State based on statistics compiled by the Department of Transportation Office of Motor Carriers for 1986–1988. Saricks and Kvitek present accident involvement and fatality counts; estimated kilometers of travel by State; and the corresponding average accident involvement, fatality, and injury rates for the 3 years investigated. Fatalities are deaths (including crew members) that are attributable to the accident or that occurred at any time within 30 days thereafter.

Rail accident rates are computed and presented similarly to truck accident rates; however, the unit of haulage is considered to be the railcar (Saricks and Kvitek, 1994). The State-specific rail accident involvement and fatality rates are based on statistics compiled by the Federal Railroad Administration for 1985–1988. Rail accident rates include both main line accidents and those occurring in railyards. It is important to note that the accident rates used in this assessment were computed using the universe of all interstate heavy combination truck shipments, independent of shipment cargo. The cited report points out that shippers and carriers of radioactive material generally have a higher-than-average awareness of transport risk and prepare cargoes and drivers for such shipments accordingly (Saricks and Kvitek, 1994). This preparation should have a twofold effect of reducing component/equipment failure and mitigating the human error contribution to accident causation. These effects were not given credit in the accident assessment.

E.6.4 Cask Accident Response Characteristics

E.6.4.1 Accident Severity Categories

A generic method to characterize the potential severity of transportation accidents was first described in an NRC report commonly referred to as NUREG-0170 (NRC, 1977a). The NRC method divided the spectrum of transportation accident severities into eight categories. Subsequently, other studies have divided the same accident spectrum into 6 categories (Wilmot, 1981) and 20 categories (Fischer et al., 1987). Results from the latter study, which utilizes 20 severity categories and is commonly referred to as the "modal study," are used in this analysis.

The modal study (Fischer et al., 1987) was the result of an initiative taken by the NRC to refine more precisely the analysis presented in NUREG-0170 (NRC, 1977a) for spent nuclear fuel shipping casks. Whereas the NUREG-0170 analysis was primarily performed using best engineering judgments and presumptions concerning cask response, the modal study relies on sophisticated structural and thermal engineering analysis and a probabilistic assessment of the conditions that could be experienced in severe transportation accidents. The modal study results are based on representative spent nuclear fuel casks that were assumed to have been designed, manufactured, operated, and maintained in accordance with national codes and standards. Design parameters of the representative casks were chosen to meet the minimum test criteria specified in 10 CFR Part 71. The study is believed to provide realistic, yet conservative, results for radiological releases under transport accident conditions.

In the modal study, potential accident damage to a cask is categorized according to the magnitude of the mechanical forces (impact) and thermal forces (fire) to which a cask may be subjected during an accident. Because all accidents can be described in these terms, severity is independent of the specific accident sequence. In other words, any sequence of events that results in an accident in which a cask is subjected to forces within a certain range of values is assigned to the accident severity category associated with that range. The accident severity scheme is designed to take into account all potential foreseeable transportation accidents, including accidents with low probability but high consequences and those with high probability but low consequences.

Each severity category actually represents a set of accidents defined by a combination of mechanical and thermal forces. A conditional probability of occurrence—that is, the probability that if an accident occurs, it is of a particular severity—is assigned to each category. The cask response regions and the fractional occurrences by accident severity category are shown in Figure E-14 for truck and rail accidents. Accidents in Region (1,1) are the least severe but most frequent, whereas accidents in Region (4,5) are very severe but very infrequent. To determine the expected frequency of an accident of a given severity, the conditional probability in the category is multiplied by the baseline accident rate. The entire spectrum of accident severities is considered in the accident risk assessment.

As discussed above, the accident consequence assessment only considers the potential impacts from the most severe transportation accidents. In terms of risk, the severity of an accident must be viewed in terms of potential radiological consequences, which are directly proportional to the fraction of the radioactive material within a cask that is released to the environment during the accident. In terms of the modal study accident characterization scheme (Figure E-15), the most severe transportation accidents correspond to those in Regions (4,1), (4,2), (4,3), (4,4), (4,5), (3,5), (2,5), and (1,5). Although these regions span the entire range of mechanical and thermal accident loads considered in the modal study, they are characterized by a single set of release fractions and are therefore considered together in the accident consequence assessment.

The conditional probability of the most severe accidents (i.e., the probability that an accident is of maximum severity, assuming that one has occurred) is found by summing the modal study conditional probabilities for the eight individual accident regions listed above. The resultant overall conditional probability is found to be 0.00000984 for truck transportation and 0.000124 for rail transportation. The stated probabilities encompass the entire spectrum of severe accidents, although over 97 percent of the severe truck accidents and nearly 100 percent of the severe rail accidents actually occur in Region (1,5), which is characterized by high thermal stresses and moderate mechanical stresses.

Structural Response (Maximum Strain on Inner Shell, %)					
S_3 (30)	R (4,1)	R (4,2)	R (4,3)	R (4,4)	R (4,5)
S_2 (2)	R (3,1)	R (3,2)	R (3,3)	R (3,4)	R (3,5)
S_1 (0.2)	R (2,1)	R (2,2)	R (2,3)	R (2,4)	R (2,5)
	R (1,1)	R (1,2)	R (1,3)	R (1,4)	R (1,5)
	T_1 (500)	T_2 (600)	T_3 (650)	T_4 (1050)	

Thermal Response (Lead Mid-Thickness Temperature, °F)

Figure E-14 Matrix of Cask Response Regions for Combined Mechanical and Thermal Loads

Legend:

(P_t) = Probability of occurrence assuming a truck accident occurs.
(P_r) = Probability of occurrence assuming a rail accident occurs.

Structural Response (maximum strain on inner shell, %)					
S_3 (30)	R(4,1) (P_t)1.532×10^{-7} (P_r)1.786×10^{-9}	R(4,2) 3.926×10^{-14} 3.290×10^{-13}	R(4,3) 1.495×10^{-14} 2.137×10^{-13}	R(4,4) 7.681×10^{-16} 1.644×10^{-13}	R(4,5) $<1 \times 10^{-16}$ 3.459×10^{-14}
S_2 (2)	R(3,1) (P_t)1.7984×10^{-3} (P_r)5.545×10^{-4}	R(3,2) 1.574×10^{-7} 1.021×10^{-7}	R(3,3) 2.034×10^{-7} 6.634×10^{-8}	R(3,4) 1.076×10^{-7} 5.162×10^{-8}	R(3,5) 4.873×10^{-8} 5.296×10^{-8}
S_1 (0.2)	R(2,1) (P_t)3.8192×10^{-3} (P_r)2.7204×10^{-3}	R(2,2) 2.330×10^{-7} 5.011×10^{-7}	R(2,3) 3.008×10^{-7} 3.255×10^{-7}	R(2,4) 1.592×10^{-7} 2.531×10^{-7}	R(2,5) 7.201×10^{-8} 1.075×10^{-8}
	R(1,1) (P_t)0.994316 (P_r)0.993962	R(1,2) 1.687×10^{-5} 1.2275×10^{-3}	R(1,3) 2.362×10^{-5} 7.9511×10^{-4}	R(1,4) 1.525×10^{-5} 6.140×10^{-4}	R(1,5) 9.570×10^{-6} 1.249×10^{-4}
	T_1 (500)	T_2 (600)	T_3 (650)	T_4 (1050)	

Thermal Response (lead midthickness temperature, °F)

Figure E-15 Fraction of Truck and Rail Accidents Expected within Each Severity Category, Assuming an Accident Occurs

E.6.4.2 Cask Release Fractions

Radiological consequences are calculated by assigning cask release fractions to each accident severity category. The release fraction is defined as the fraction of the radioactive material in a cask that could be released from the package in a given severity of accident. Release fractions take into account all mechanisms necessary to create a release of radioactive material from a damaged cask to the environment. Release fractions vary according to the spent nuclear fuel type and the physical and chemical characteristics of specific radionuclides within the spent nuclear fuel. For instance, most solid radionuclides are difficult to release in particulate form and are therefore relatively nondispersible. Conversely, gaseous radionuclides are relatively easy to release in the unlikely event that the cask and spent nuclear fuel elements are compromised in an accident.

Cask release fractions are given in Table E-6. Two sets of release fractions were used in the assessment depending on the spent nuclear fuel type, consistent with the SNF&INEL Final EIS (DOE, 1995). Release fractions developed for MTR spent nuclear fuel were used for aluminum-clad fuels including BR-2, RHF, and NRU spent nuclear fuel; Release fractions for TRIGA were used for the PRR-1 spent nuclear fuel.

Table E-6 Release Fractions Spent Nuclear Fuel

Cask Response Region	*Release Fractions*[a]				
	Inert Gas	*Iodine*	*Cesium*	*Ruthenium*	*Particulate*
TRIGA Fuel:					
R(1,1)	0.0	0.0	0.0	0.0	0.0
R(1,2), R(1,3)	0.0099	0.000075	0.000006	8.1×10^{-7}	6.0×10^{-8}
R(2,1), R(2,2), R(2,3)	0.03	0.00025	0.00002	0.0000027	2.0×10^{-7}
R(1,4), R(2,4), R(3,4)	0.39	0.0043	0.0002	0.000048	0.000002
R(3,1), R(3,2), R(3,3)	0.33	0.0025	0.0002	0.000027	0.000002
R(1,5), R(2,5), R(3,5), R(4,5), R(4,1), R(4,2), R(4,3), R(4,4)	0.63	0.043	0.002	0.00048	0.00002
Aluminum and Metallic Fuel:[b]					
R(1,1)	0.0	0.0	0.0	0.0	0.0
R(1,2), R(1,3)	0.0099	1.1×10^{-7}	3.0×10^{-8}	4.1×10^{-9}	3.0×10^{-10}
R(2,1), R(2,2), R(2,3)	0.033	3.5×10^{-7}	1.0×10^{-7}	1.4×10^{-8}	1.0×10^{-9}
R(1,4), R(2,4), R(3,4)	0.39	0.000006	0.000001	2.4×10^{-7}	1.0×10^{-8}
R(3,1), R(3,2), R(3,3)	0.33	0.0000035	0.000001	1.4×10^{-7}	1.0×10^{-8}
R(1,5), R(2,5), R(3,5), R(4,5), R(4,1), R(4,2), R(4,3), R(4,4)	0.63	0.00006	0.00001	0.0000024	1.0×10^{-7}

[a] *The fraction of the radioactive material released from a cask to the environment during an accident.*

[b] *These release fractions are applicable to all non-TRIGA, aluminum-clad fuel.*

For waste shipments of material other than spent nuclear fuel, the modal study results are not applicable. Therefore, more conservative release fractions from NUREG-0170 are used for vitrified high-level waste and target material. The NUREG-0170 recommendations for release fractions for Type B casks, regardless of content, are given below:

NUREG-0170 Severity Category	Release Fraction
1	0
2	0
3	0.01
4	0.1
5	1
6	1

Source: NRC, 1977a

The values indicate that in the most severe accidents, 100 percent of the material is released from the cask; a highly conservative assumption for most solid waste forms, and somewhat conservative for a powder or cake-like material. The accident assessment also utilizes the fraction of the release that is aerosolized and the fraction of the aerosol that is respirable. The values for high-level waste and target material (assumed to behave as a loose powdered material) were taken from the recommendations provided in RADTRAN 4. These values are shown below:

Physical Waste Form	Aerosolized Fraction	Respirable Fraction
Vitrified wastes	0.000001	0.05
Chunks (i.e., calcinated target material)	0.01	0.05
Loose powders (i.e., oxidized target material)	0.1	0.05

Source: Neuhauser and Kanipe, 1993

Therefore, the maximum total respirable release fraction for the most severe accidents is 5×10^{-8} for high-level waste shipments and 0.005 for shipments of target material. The values shown above have been used in the accident calculations for shipments of target material and vitrified material for the foreign research reactor spent nuclear fuel EIS.

E.6.5 Atmospheric Conditions

Radioactive material released to the atmosphere is transported by the wind. The amount of dispersion, or dilution, of the radioactive material concentrations in the air depends on the meteorological conditions at the time of the accident. Because it is impossible to predict the specific location of an overland transportation accident, generic atmospheric conditions were selected for the accident risk and consequence assessments.

For the accident risk assessment, neutral weather conditions (Pasquill Stability Class D with a wind speed of 4 m per sec or 9 mph) were assumed. Since neutral meteorological conditions are the most frequently occurring atmospheric stability condition in the United States, they are most likely to be present in the event of an accident involving a spent nuclear fuel shipment. On the basis of observations from National Weather Service surface meteorological stations at over 300 locations in the United States, on an annual average, neutral conditions (Pasquill Classes C and D) occur about half (50 percent) of the time, while stable (Pasquill Classes E and F) and unstable (Pasquill Class A and B) conditions occur about one-third (33 percent) and one-sixth (17 percent) of the time, respectively (Doty et al., 1976). The neutral category predominates in all seasons, but most frequently in the winter (nearly 60 percent of the observations).

For the accident consequence assessment, doses were assessed under both neutral (Pasquill Stability Class D with a wind speed of 4 m per sec or 9 mph) and stable (Pasquill Stability Class F with a wind speed of 1 m per sec or 2.4 mph) atmospheric conditions. The results calculated for neutral conditions represent the most likely consequences, and the results for stable conditions represent a "worst-case" weather situation.

E.6.6 Health Risk Conversion Factors

The health risk conversion factors used to estimate expected cancer fatalities were taken from International Commission on Radiation Protection Publication 60 (ICRP, 1991): 0.0005 and 0.0004 fatal cancer cases per person-rem for members of the public and workers, respectively. Cancer fatalities and incidence occur over the lifetimes of the exposed populations, and thus are called LCF.

E.6.7 Maximally Exposed Individual Exposure Scenarios

The risk to MEIs has been estimated for a number of hypothetical exposure scenarios during overland transportation using the RISKIND code. The receptors include crew members, departure inspectors, and members of the public exposed during traffic obstructions (traffic jam), while working at a service station, or by living near a port of entry or management site. The dose and risk to MEIs were calculated for given distances and durations of exposure. The distances and durations of exposure for each receptor are similar to those given in previous transportation assessments (DOE, 1987b; DOE, 1995), and are believed to be realistic but conservative. The exposure scenarios considered are the following:

- Crew Members: Truck and rail crew members are not assumed to be occupational radiation workers. Dose estimates are based on realistic locations and estimated travel time, and no credit is taken for shielding in addition to the cask.

- Inspectors (truck and rail): Inspectors are assumed to be either Federal or State vehicle inspectors, and are not assumed to be monitored by a dosimetry program. An average exposure distance of 3 m (10 ft) and an exposure time of 30 minutes (min) is assumed.

- Rail Yard Crew Member: A rail yard crew member is not assumed to be monitored by a dosimetry program. An average exposure distance of 10 m (33 ft) and an exposure time of 2 hr is assumed.

- Resident (truck and rail): A resident is assumed to live 30 m (100 ft) from a port or management site entrance route (truck or rail). Shipments are assumed to pass at a velocity of 24 km per hr (15 mph), and the resident is assumed to be exposed unshielded (i.e., no shielding in addition to the cask, such as that afforded by a structure.) Cumulative doses are assessed for each alternative based on the number of shipments entering or exiting the site and assuming the resident is present for 100 percent of the shipments.

- Person in Traffic Obstruction (truck and rail): A person is assumed to be stopped next to a spent nuclear fuel shipment (due to traffic, etc.). The person is assumed to be exposed (no credit is taken for radiation blocked by the individual's vehicle) at a distance of 1 m (3.3 ft) for a duration of 30 min.

- Person at a Truck Service Station: A person is assumed to be exposed at an average distance of 20 m (66 ft) for a duration of 2 hr. This receptor could be a worker at a truck stop, or a member of the public stopped at the same location.

- Resident Near a Rail Stop: A resident is assumed to live near a rail classification yard. The resident is assumed to be exposed unshielded at a distance of 200 m (660 ft) for a duration of 20 hr.

The largest uncertainty in predicting the dose to MEIs during transportation involves determining the frequency of exposure occurrences. This difficulty results from the uncertainties in future shipment schedules, route selection, and the inherent uncertainty in predicting the frequency of random or chance events. For instance, it is conceivable that an individual could be stopped in traffic next to a shipment of foreign research reactor spent nuclear fuel; however, it is difficult to predict how often the same individual would experience this event. Therefore, for the majority of receptors considered, doses are assessed on a per-event basis. To account for possible multiple exposures, ranges of realistic total doses are discussed qualitatively. One exception is the calculation of the dose to a hypothetical resident living near a port of entry or management site entrance route. For these residents, total doses are calculated based on the number of shipments entering or exiting each site for each of the alternatives.

E.6.8 General RADTRAN Input Parameters

In addition to the specific parameters discussed above, values for a number of general parameters must be specified within the RADTRAN code. These general parameters define basic shipment and traffic characteristics and are specific to the mode of transportation. The RADTRAN code user's manual (Neuhauser and Kanipe, 1993) contains derivations and descriptions of these parameters. The general RADTRAN input parameters used in the transportation risk assessment are summarized in Table E-7.

Table E-7 Summary of General RADTRAN Input Parameters

Parameter	*Truck*	*Rail*
Package type	Type B Cask	Type B Cask
Package dimension	3.2 m (10.6 ft)	3.2 m (10.6 ft)
Number of crewmen	2	5
Distance from source to crew	3 m (9.9 ft)	152 m (501.6 ft)
Velocity		
Rural	88 km/hr (55 mph)	64 km/hr (40 mph)
Suburban	40 km/hr (25 mph)	40 km/hr (25 mph)
Urban	24 km/hr (15 mph)	24 km/hr (15 mph)
Stop time per kilometer	0.011 hr/km (0.018 hr/mi)	0.033 hr/km (0.053 hr/mi)
Number of people exposed while stopped	50	Based on Suburban Population Density
Number of people per vehicle sharing route	2	3
Population densities (persons/km^2)	Route Specific (see Table E-4)	Route Specific (see Table E-4)
One-way traffic count (vehicles/hr)		
Rural	470	1
Suburban	780	5
Urban	2,800	5
Cask inventory (Ci)	(see Table E-5)	(see Table E-5)
Accident release fractions	(see Table E-6)	(see Table E-6)
Accident conditional probabilities	(see Figure E-15)	(see Figure E-15)

Source: Neuhauser and Kanipe, 1993.

E.7 Risk Assessment Results

In this section, the risk assessment results are presented for the ports of entry and management sites being considered. The collective population risk results are presented in Section E.7.1. First, the per-shipment risks results are presented in Section E.7.1.1. Then, in Section E.7.1.2, the results are analyzed, evaluated, and simplified so the different program alternatives and options can be evaluated in Section E.7.2.

The risks to MEIs during incident-free transportation conditions are provided in Section E.7.3. The accident consequence results calculated for the most severe transportation accidents are presented in Section E.7.4 for both collective populations and MEIs.

E.7.1 Collective Population Risk Results

E.7.1.1 Per-Shipment Risk Factors

Per-shipment risk factors have been calculated for the collective populations of exposed persons for shipments between all representative ports of entry and the five management sites. Results were calculated for both truck and rail modes, assuming that one cask would be shipped per truck or rail car. Additionally, the risk factors for the ports of Elizabeth, NJ; Philadelphia, PA; and Long Beach, CA are included to show the effect of using high population ports. Risk factors are included for some site-to-site routes, even though there are no shipments anticipated on these routes.

The radiological risks are presented in terms of dose per shipment for each unique route. The doses can be converted to health risks using the International Commission on Radiological Protection Publication 60 conversion factors described in Section E.6.6 (ICRP, 1991). The radiological dose per shipment factors for incident-free transportation conditions are presented in Table E-8 for crew members and members of the general public. The tabulated incident-free doses are based on the external dose rate which is conservatively assumed to be at the regulatory limit of 10 mrem per hr at 2 m. The radiological dose risk factors for accident transportation conditions are presented in Table E-9. The accident risk factors are referred to as "dose risk" because the values incorporate the spectrum of accident severity probabilities and the associated release fractions.

Table E-8 Incident-Free Dose per Shipment for All Spent Nuclear Fuel Types (Person-Rem/Shipment)[a]

Shipments to Hanford Site:						
Route(s)		*Crew*	*General Public*			
			Off-Link	*On-Link*	*Stops*	*Total*
From Eastern Ports						
Charleston, SC (NWS)	Truck	2.50×10^{-1}	9.26×10^{-3}	3.96×10^{-2}	5.92×10^{-1}	6.41×10^{-1}
	Rail	6.33×10^{-2}	2.91×10^{-2}	1.13×10^{-3}	1.70×10^{-2}	4.73×10^{-2}
Charleston, SC (Wando Terminal)	Truck	2.51×10^{-1}	9.61×10^{-3}	4.03×10^{-2}	5.95×10^{-1}	6.45×10^{-1}
	Rail	6.33×10^{-2}	2.91×10^{-2}	1.13×10^{-3}	1.70×10^{-2}	4.73×10^{-2}
Elizabeth, NJ	Truck	2.49×10^{-1}	9.96×10^{-3}	4.10×10^{-2}	5.82×10^{-1}	6.33×10^{-1}
	Rail	6.24×10^{-2}	6.03×10^{-2}	1.66×10^{-3}	1.68×10^{-2}	7.88×10^{-2}
Galveston, TX	Truck	2.05×10^{-1}	1.00×10^{-2}	3.77×10^{-2}	4.82×10^{-1}	5.30×10^{-1}
	Rail	5.20×10^{-2}	1.31×10^{-2}	6.59×10^{-4}	1.51×10^{-2}	2.88×10^{-2}
Jacksonville, FL	Truck	2.60×10^{-1}	1.09×10^{-2}	4.35×10^{-2}	6.05×10^{-1}	6.60×10^{-1}
	Rail	6.35×10^{-2}	2.58×10^{-2}	1.08×10^{-3}	1.65×10^{-2}	4.34×10^{-2}
Newport News, VA	Truck	2.57×10^{-1}	1.16×10^{-2}	4.44×10^{-2}	6.02×10^{-1}	6.58×10^{-1}
	Rail	6.38×10^{-2}	4.02×10^{-2}	1.25×10^{-3}	1.59×10^{-2}	5.73×10^{-2}

Shipments to Hanford Site:						
Route(s)		Crew	General Public			
			Off-Link	On-Link	Stops	Total
Norfolk, VA	Truck	$2.62x10^{-1}$	$1.19x10^{-2}$	$4.50x10^{-2}$	$6.10x10^{-1}$	$6.67x10^{-1}$
	Rail	$6.54x10^{-2}$	$4.10x10^{-2}$	$1.28x10^{-3}$	$1.67x10^{-2}$	$5.90x10^{-2}$
Philadelphia, PA	Truck	$2.58x10^{-1}$	$1.06x10^{-2}$	$4.31x10^{-2}$	$5.94x10^{-1}$	$6.47x10^{-1}$
	Rail	$6.16x10^{-2}$	$5.90x10^{-2}$	$1.58x10^{-3}$	$1.68x10^{-2}$	$7.74x10^{-2}$
Portsmouth, VA	Truck	$2.61x10^{-1}$	$1.24x10^{-2}$	$4.60x10^{-2}$	$6.07x10^{-1}$	$6.65x10^{-1}$
	Rail	$6.49x10^{-2}$	$3.99x10^{-2}$	$1.25x10^{-3}$	$1.64x10^{-2}$	$5.76x10^{-2}$
Savannah, GA	Truck	$2.48x10^{-1}$	$1.03x10^{-2}$	$4.13x10^{-2}$	$5.82x10^{-1}$	$6.34x10^{-1}$
	Rail	$6.38x10^{-2}$	$2.48x10^{-2}$	$1.08x10^{-3}$	$1.66x10^{-2}$	$4.24x10^{-2}$
MOTSU, NC	Truck	$2.50x10^{-1}$	$9.26x10^{-3}$	$4.00x10^{-2}$	$5.94x10^{-1}$	$6.43x10^{-1}$
	Rail	$6.57x10^{-2}$	$2.85x10^{-2}$	$1.20x10^{-3}$	$1.71x10^{-2}$	$4.68x10^{-2}$
Wilmington, NC	Truck	$2.59x10^{-1}$	$9.34x10^{-3}$	$4.07x10^{-2}$	$6.13x10^{-1}$	$6.63x10^{-1}$
	Rail	$6.56x10^{-2}$	$2.84x10^{-2}$	$1.19x10^{-3}$	$1.71x10^{-2}$	$4.67x10^{-2}$
From Western Ports						
NWS Concord, CA	Truck	$8.06x10^{-2}$	$4.55x10^{-3}$	$1.55x10^{-2}$	$1.77x10^{-1}$	$1.97x10^{-1}$
	Rail	$2.77x10^{-2}$	$1.88x10^{-2}$	$4.83x10^{-4}$	$8.77x10^{-3}$	$2.81x10^{-2}$
Long Beach, CA	Truck	$1.19x10^{-1}$	$1.13x10^{-2}$	$2.98x10^{-2}$	$2.57x10^{-1}$	$2.98x10^{-1}$
	Rail	$3.84x10^{-2}$	$3.71x10^{-2}$	$7.16x10^{-4}$	$1.46x10^{-2}$	$5.25x10^{-2}$
Portland, OR	Truck	$2.35x10^{-2}$	$1.50x10^{-3}$	$4.90x10^{-3}$	$5.24x10^{-2}$	$5.88x10^{-2}$
	Rail	$1.57x10^{-2}$	$4.36x10^{-3}$	$1.12x10^{-4}$	$7.04x10^{-3}$	$1.15x10^{-2}$
Tacoma, WA	Truck	$2.50x10^{-2}$	$1.73x10^{-3}$	$5.45x10^{-3}$	$5.14x10^{-2}$	$5.85x10^{-2}$
	Rail	$1.80x10^{-2}$	$5.69x10^{-3}$	$1.82x10^{-4}$	$6.12x10^{-3}$	$1.20x10^{-2}$
From DOE Sites/Canadian Border						
Alexandria Bay, NY	Truck	$2.49x10^{-1}$	$1.05x10^{-2}$	$4.23x10^{-2}$	$5.73x10^{-1}$	$6.26x10^{-1}$
	Rail	$6.02x10^{-2}$	$5.10x10^{-2}$	$1.40x10^{-3}$	$1.65x10^{-2}$	$6.89x10^{-2}$
Idaho National Engineering Laboratory	Truck	$4.92x10^{-1}$	$1.42x10^{-3}$	$7.63x10^{-3}$	$1.24x10^{-1}$	$1.33x10^{-1}$
	Rail	$2.28x10^{-2}$	$3.88x10^{-3}$	$1.76x10^{-4}$	$7.64x10^{-3}$	$1.17x10^{-2}$
Nevada Test Site	Truck	$9.91x10^{-2}$	$5.37x10^{-3}$	$1.92x10^{-2}$	$2.34x10^{-1}$	$2.58x10^{-1}$
	Rail	$3.36x10^{-1}$	$6.24x10^{-3}$	$3.08x10^{-4}$	$1.12x10^{-2}$	$1.77x10^{-2}$
Oak Ridge Reservation	Truck	$2.10x10^{-1}$	$7.29x10^{-3}$	$3.31x10^{-2}$	$5.10x10^{-1}$	$5.50x10^{-1}$
	Rail	$5.56x10^{-2}$	$1.64x10^{-2}$	$6.92x10^{-4}$	$1.60x10^{-1}$	$3.31x10^{-2}$
Savannah River	Truck	$2.42x10^{-1}$	$1.02x10^{-2}$	$4.01x10^{-2}$	$5.65x10^{-1}$	$6.15x10^{-1}$
	Rail	$6.15x10^{-2}$	$2.74x10^{-2}$	$1.08x10^{-3}$	$1.66x10^{-2}$	$4.51x10^{-2}$
Sweetgrass, MT	Truck	$7.27x10^{-1}$	$1.76x10^{-3}$	$1.03x10^{-2}$	$1.81x10^{-1}$	$1.93x10^{-1}$
	Rail	$2.19x10^{-2}$	$3.81x10^{-3}$	$1.62x10^{-4}$	$7.83x10^{-3}$	$1.18x10^{-2}$

Shipments to Idaho National Engineering Laboratory:						
Route(s)		Crew	General Public			
			Off-Link	On-Link	Stops	Total
From Eastern Ports						
Charleston, SC (NWS)	Truck	$2.16x10^{-1}$	$8.41x10^{-3}$	$3.48x10^{-2}$	$5.05x10^{-1}$	$5.49x10^{-1}$
	Rail	$5.41x10^{-2}$	$2.68x10^{-2}$	$1.01x10^{-3}$	$1.45x10^{-2}$	$4.24x10^{-2}$
Charleston, SC (Wando Terminal)	Truck	$2.18x10^{-1}$	$8.76x10^{-3}$	$3.56x10^{-2}$	$5.09x10^{-1}$	$5.53x10^{-1}$
	Rail	$5.41x10^{-2}$	$2.68x10^{-2}$	$1.01x10^{-3}$	$1.45x10^{-2}$	$4.24x10^{-2}$
Elizabeth, NJ	Truck	$2.15x10^{-1}$	$9.11x10^{-3}$	$3.59x10^{-2}$	$4.96x10^{-1}$	$5.41x10^{-1}$
	Rail	$5.32x10^{-2}$	$5.81x10^{-2}$	$1.53x10^{-3}$	$1.44x10^{-2}$	$7.40x10^{-2}$
Galveston, TX	Truck	$1.71x10^{-1}$	$9.18x10^{-3}$	$3.26x10^{-2}$	$3.96x10^{-1}$	$4.37x10^{-1}$
	Rail	$4.28x10^{-2}$	$1.08x10^{-2}$	$5.35x10^{-4}$	$1.24x10^{-2}$	$2.38x10^{-2}$
Jacksonville, FL	Truck	$2.26x10^{-1}$	$9.78x10^{-3}$	$3.77x10^{-2}$	$5.18x10^{-1}$	$5.66x10^{-1}$
	Rail	$5.42x10^{-2}$	$2.35x10^{-2}$	$9.54x10^{-4}$	$1.40x10^{-2}$	$3.85x10^{-2}$
Newport News, VA	Truck	$2.24x10^{-1}$	$1.08x10^{-2}$	$3.93x10^{-2}$	$5.16x10^{-1}$	$5.66x10^{-1}$
	Rail	$5.45x10^{-2}$	$3.79x10^{-2}$	$1.13x10^{-3}$	$1.35x10^{-2}$	$5.25x10^{-2}$

Shipments to Idaho National Engineering Laboratory:						
			General Public			
Route(s)		*Crew*	*Off-Link*	*On-Link*	*Stops*	*Total*
From Eastern Ports						
Norfolk, VA	Truck	2.28×10^{-1}	1.08×10^{-2}	3.93×10^{-2}	5.24×10^{-1}	5.74×10^{-1}
	Rail	5.62×10^{-2}	3.87×10^{-2}	1.16×10^{-3}	1.43×10^{-2}	5.42×10^{-2}
Philadelphia, PA	Truck	2.24×10^{-1}	9.71×10^{-3}	3.80×10^{-2}	5.08×10^{-1}	5.55×10^{-1}
	Rail	5.24×10^{-2}	5.67×10^{-2}	1.46×10^{-3}	1.44×10^{-2}	7.25×10^{-2}
Portsmouth, VA	Truck	2.27×10^{-1}	1.15×10^{-2}	4.09×10^{-2}	5.20×10^{-1}	5.73×10^{-1}
	Rail	5.57×10^{-2}	3.76×10^{-2}	1.13×10^{-3}	1.40×10^{-2}	5.27×10^{-2}
Savannah, GA	Truck	2.15×10^{-1}	9.41×10^{-3}	3.62×10^{-2}	4.96×10^{-1}	5.42×10^{-1}
	Rail	5.46×10^{-2}	2.25×10^{-2}	9.53×10^{-4}	1.41×10^{-2}	3.75×10^{-2}
MOTSU, NC	Truck	2.11×10^{-1}	7.80×10^{-3}	3.35×10^{-2}	4.98×10^{-1}	5.40×10^{-1}
	Rail	5.65×10^{-2}	2.62×10^{-2}	1.07×10^{-3}	1.47×10^{-2}	4.20×10^{-2}
Wilmington, NC	Truck	2.25×10^{-1}	8.50×10^{-3}	3.56×10^{-2}	5.27×10^{-1}	5.71×10^{-1}
	Rail	5.63×10^{-2}	2.61×10^{-2}	1.07×10^{-3}	1.47×10^{-2}	4.19×10^{-2}
From Western Ports						
NWS Concord, CA	Truck	8.40×10^{-2}	4.84×10^{-3}	1.72×10^{-2}	1.95×10^{-1}	2.17×10^{-1}
	Rail	2.71×10^{-2}	8.71×10^{-3}	2.96×10^{-4}	7.88×10^{-3}	1.69×10^{-2}
Long Beach, CA	Truck	9.93×10^{-2}	1.28×10^{-2}	3.06×10^{-2}	2.03×10^{-1}	2.46×10^{-1}
	Rail	2.92×10^{-2}	3.48×10^{-2}	5.92×10^{-4}	1.20×10^{-2}	4.74×10^{-2}
Portland, OR	Truck	6.27×10^{-2}	2.46×10^{-3}	1.06×10^{-2}	1.53×10^{-1}	1.66×10^{-1}
	Rail	2.49×10^{-2}	5.01×10^{-3}	2.01×10^{-4}	7.23×10^{-3}	1.24×10^{-2}
Tacoma, WA	Truck	7.04×10^{-2}	2.71×10^{-3}	1.19×10^{-2}	1.69×10^{-1}	1.83×10^{-1}
	Rail	2.74×10^{-2}	8.57×10^{-3}	3.04×10^{-4}	7.71×10^{-3}	1.66×10^{-2}
From DOE Sites/Canadian Border						
Alexandria Bay, NY	Truck	2.16×10^{-1}	9.64×10^{-3}	3.72×10^{-2}	4.87×10^{-1}	5.34×10^{-1}
	Rail	5.10×10^{-2}	4.87×10^{-2}	1.28×10^{-3}	1.41×10^{-2}	6.40×10^{-2}
Hanford Site	Truck	4.92×10^{-1}	1.42×10^{-2}	7.63×10^{-2}	1.24×10^{-1}	1.33×10^{-1}
	Rail	2.28×10^{-1}	3.88×10^{-2}	1.76×10^{-3}	7.64×10^{-2}	1.17×10^{-2}
Nevada Test Site	Truck	6.56×10^{-1}	4.52×10^{-2}	1.40×10^{-1}	1.47×10^{-1}	1.66×10^{-1}
	Rail	2.44×10^{-1}	3.95×10^{-2}	1.84×10^{-3}	8.29×10^{-2}	1.24×10^{-2}
Oak Ridge Reservation	Truck	1.76×10^{-1}	6.45×10^{-1}	2.80×10^{-1}	4.24×10^{-1}	4.58×10^{-1}
	Rail	4.63×10^{-1}	1.41×10^{-1}	5.69×10^{-3}	1.33×10^{-1}	2.80×10^{-2}
Savannah River	Truck	2.08×10^{-1}	9.34×10^{-2}	3.50×10^{-1}	4.79×10^{-1}	5.23×10^{-1}
	Rail	5.23×10^{-1}	2.51×10^{-1}	9.56×10^{-3}	1.41×10^{-1}	4.02×10^{-2}
Sweetgrass, MT	Truck	4.25×10^{-1}	6.45×10^{-3}	5.67×10^{-2}	1.12×10^{-1}	1.19×10^{-1}
	Rail	3.24×10^{-1}	5.37×10^{-2}	2.84×10^{-3}	9.14×10^{-2}	1.48×10^{-2}

Shipments to Nevada Test Site:						
			General Public			
Route(s)		*Crew*	*Off-Link*	*On-Link*	*Stops*	*Total*
From Eastern Ports						
Charleston, SC (NWS)	Truck	2.25×10^{-1}	9.15×10^{-3}	3.69×10^{-2}	5.27×10^{-1}	5.73×10^{-1}
	Rail	6.13×10^{-2}	2.95×10^{-2}	1.11×10^{-3}	1.69×10^{-2}	4.75×10^{-2}
Charleston, SC (Wando Terminal)	Truck	2.27×10^{-1}	9.50×10^{-3}	3.76×10^{-2}	5.30×10^{-1}	5.77×10^{-1}
	Rail	6.13×10^{-2}	2.95×10^{-2}	1.11×10^{-3}	1.69×10^{-2}	4.75×10^{-2}
Elizabeth, NJ	Truck	2.48×10^{-1}	1.35×10^{-2}	4.60×10^{-2}	5.53×10^{-1}	6.13×10^{-1}
	Rail	6.05×10^{-2}	6.07×10^{-3}	1.63×10^{-3}	1.66×10^{-2}	7.8×10^{-2}
Jacksonville, FL	Truck	2.35×10^{-1}	1.10×10^{-2}	4.07×10^{-2}	5.40×10^{-1}	5.91×10^{-1}
	Rail	6.15×10^{-2}	2.61×10^{-2}	1.05×10^{-3}	1.64×10^{-2}	4.36×10^{-2}
Newport News, VA	Truck	2.33×10^{-1}	1.20×10^{-2}	4.23×10^{-2}	5.37×10^{-1}	5.92×10^{-1}
	Rail	6.18×10^{-2}	4.05×10^{-2}	1.22×10^{-3}	1.57×10^{-2}	5.75×10^{-2}

Shipments to Nevada Test Site:						
			General Public			
Route(s)		*Crew*	*Off-Link*	*On-Link*	*Stops*	*Total*
Norfolk, VA	Truck	$2.73x10^{-1}$	$1.19x10^{-2}$	$4.23x10^{-2}$	$5.45x10^{-1}$	$5.99x10^{-1}$
	Rail	$6.35x10^{-2}$	$4.14x10^{-2}$	$1.2 x10^{-3}$	$1.66x10^{-2}$	$5.92x10^{-2}$
Philadelphia, PA	Truck	$2.44x10^{-1}$	$1.30x10^{-2}$	$4.49x10^{-2}$	$5.43x10^{-1}$	$6.01x10^{-1}$
	Rail	$5.97x10^{-2}$	$5.94x10^{-2}$	$1.56x10^{-3}$	$1.66x10^{-2}$	$7.75x10^{-2}$
Portsmouth, VA	Truck	$2.36x10^{-1}$	$1.27x10^{-2}$	$4.39x10^{-2}$	$5.42x10^{-1}$	$5.98x10^{-1}$
	Rail	$6.30x10^{-2}$	$4.3x10^{-2}$	$1.23x10^{-3}$	$1.63x10^{-2}$	$5.78x10^{-2}$
Savannah, GA	Truck	$2.24x10^{-1}$	$1.06x10^{-2}$	$3.92x10^{-2}$	$5.18x10^{-1}$	$5.67x10^{-1}$
	Rail	$6.19x10^{-2}$	$2.51x10^{-2}$	$1.05x10^{-3}$	$1.64x10^{-2}$	$4.26x10^{-2}$
MOTSU, NC	Truck	$2.22x10^{-1}$	$1.06x10^{-2}$	$3.95x10^{-2}$	$5.09x10^{-1}$	$5.59x10^{-1}$
	Rail	$6.38x10^{-2}$	$2.89x10^{-2}$	$1.17x10^{-3}$	$1.70x10^{-2}$	$4.70x10^{-2}$
Wilmington, NC	Truck	$2.35x10^{-1}$	$9.67x10^{-3}$	$3.86x10^{-2}$	$5.49x10^{-1}$	$5.97x10^{-1}$
	Rail	$6.36x10^{-2}$	$2.88x10^{-2}$	$1.16x10^{-3}$	$1.70x10^{-2}$	$4.69x10^{-2}$
From Western Ports						
NWS Concord, CA	Truck	$6.88x10^{-2}$	$8.04x10^{-3}$	$1.99x10^{-2}$	$1.47x10^{-1}$	$1.75x10^{-1}$
	Rail	$2.60x10^{-2}$	$1.83x10^{-2}$	$4.73x10^{-4}$	$8.17x10^{-3}$	$2.70x10^{-2}$
Long Beach, CA	Truck	$4.63x10^{-2}$	$9.79x10^{-3}$	$2.04x10^{-2}$	$8.30x10^{-2}$	$1.13x10^{-1}$
	Rail	$1.98x10^{-2}$	$3.09x10^{-2}$	$4.37x10^{-4}$	$9.63x10^{-3}$	$4.10x10^{-2}$
Portland, OR	Truck	$1.13x10^{-1}$	$6.62x10^{-3}$	$2.28x10^{-2}$	$2.63x10^{-1}$	$2.92x10^{-1}$
	Rail	$3.58x10^{-2}$	$7.38x10^{-3}$	$3.33x10^{-1}$	$1.10x10^{-2}$	$1.87x10^{-2}$
Tacoma, WA	Truck	$1.20x10^{-1}$	$6.66x10^{-3}$	$2.34x10^{-2}$	$2.78x10^{-1}$	$3.08x10^{-1}$
	Rail	$3.83x10^{-2}$	$1.09x10^{-2}$	$4.36x10^{-4}$	$1.11x10^{-2}$	$2.24x10^{-2}$
From DOE Sites/Canadian Border						
Alexandria Bay, NY	Truck	$2.39x10^{-1}$	$1.11x10^{-2}$	$4.21x10^{-2}$	$5.42x10^{-1}$	$5.95x10^{-1}$
	Rail	$5.83x10^{-2}$	$5.13x10^{-2}$	$1.37x10^{-3}$	$1.63x10^{-2}$	$6.90x10^{-2}$
Hanford Site	Truck	$9.91x10^{-2}$	$5.37x10^{-3}$	$1.92x10^{-2}$	$2.34x10^{-1}$	$2.58x10^{-1}$
	Rail	$3.36x10^{-2}$	$6.24x10^{-3}$	$3.08x10^{-1}$	$1.12x10^{-2}$	$1.77x10^{-2}$
Idaho National Engineering Laboratory	Truck	$6.56x10^{-2}$	$4.52x10^{-3}$	$1.40x10^{-2}$	$1.47x10^{-1}$	$1.66x10^{-1}$
	Rail	$2.44x10^{-2}$	$3.95x10^{-3}$	$1.84x10^{-1}$	$8.29x10^{-3}$	$1.24x10^{-2}$
Oak Ridge Reservation	Truck	$1.86x10^{-1}$	$7.6x10^{-3}$	$3.10x10^{-2}$	$4.45x10^{-1}$	$4.84x10^{-1}$
	Rail	$5.36x10^{-2}$	$1.68x10^{-2}$	$6.66x10^{-1}$	$1.63x10^{-2}$	$3.37x10^{-2}$
Savannah River	Truck	$2.17x10^{-1}$	$1.05x10^{-2}$	$3.80x10^{-2}$	$5.00x10^{-1}$	$5.48x10^{-1}$
	Rail	$5.96x10^{-2}$	$2.77x10^{-2}$	$1.05x10^{-3}$	$1.65x10^{-2}$	$4.53x10^{-2}$

Shipments to Oak Ridge Reservation:						
			General Public			
Route(s)		*Crew*	*Off-Link*	*On-Link*	*Stops*	*Total*
From Eastern Ports						
Charleston, SC (NWS)	Truck	$3.98x10^{-2}$	$1.73x10^{-3}$	$6.11x10^{-3}$	$8.32x10^{-2}$	$9.11x10^{-2}$
	Rail	$2.15x10^{-2}$	$6.88x10^{-3}$	$3.60x10^{-4}$	$5.33x10^{-3}$	$1.26x10^{-2}$
Charleston, SC (Wando Terminal)	Truck	$4.18x10^{-2}$	$2.09x10^{-3}$	$6.85x10^{-3}$	$8.63x10^{-2}$	$9.53x10^{-2}$
	Rail	$2.15x10^{-2}$	$6.88x10^{-3}$	$3.60x10^{-4}$	$5.33x10^{-3}$	$1.26x10^{-2}$
Elizabeth, NJ	Truck	$8.02x10^{-2}$	$4.85x10^{-3}$	$1.45x10^{-2}$	$1.53x10^{-1}$	$1.72x10^{-1}$
	Rail	$2.49x10^{-2}$	$4.30x10^{-2}$	$9.20x10^{-4}$	$7.73x10^{-3}$	$5.17x10^{-2}$
Galveston, TX	Truck	$9.53x10^{-2}$	$5.71x10^{-3}$	$1.73x10^{-2}$	$1.99x10^{-1}$	$2.22x10^{-1}$
	Rail	$2.94x10^{-2}$	$1.87x10^{-2}$	$6.28x10^{-4}$	$8.98x10^{-3}$	$2.83x10^{-2}$
Jacksonville, FL	Truck	$5.88x10^{-2}$	$2.88x10^{-3}$	$9.64x10^{-3}$	$1.17x10^{-1}$	$1.30x10^{-1}$
	Rail	$2.12x10^{-2}$	$8.80x10^{-3}$	$3.59x10^{-4}$	$6.51x10^{-3}$	$1.57x10^{-2}$
Newport News, VA	Truck	$5.68x10^{-2}$	$3.57x10^{-3}$	$1.09x10^{-2}$	$1.15x10^{-1}$	$1.29x10^{-1}$
	Rail	$2.45x10^{-2}$	$1.11x10^{-2}$	$5.45x10^{-4}$	$6.00x10^{-3}$	$1.77x10^{-2}$
Norfolk, VA	Truck	$5.63x10^{-2}$	$2.85x10^{-3}$	$9.27x10^{-3}$	$1.14x10^{-1}$	$1.26x10^{-1}$
	Rail	$2.33x10^{-2}$	$8.51x10^{-3}$	$4.56x10^{-4}$	$5.63x10^{-3}$	$1.46x10^{-2}$

Shipments to Oak Ridge Reservation:						
		General Public				
Route(s)		*Crew*	*Off-Link*	*On-Link*	*Stops*	*Total*
Philadelphia, PA	Truck	$7.36x10^{-2}$	$5.74x10^{-3}$	$1.56x10^{-2}$	$1.41x10^{-1}$	$1.62x10^{-1}$
	Rail	$2.35x10^{-2}$	$2.83x10^{-2}$	$7.16x10^{-4}$	$6.98x10^{-3}$	$3.60x10^{-2}$
Portsmouth, VA	Truck	$6.01x10^{-2}$	$4.32x10^{-3}$	$1.25x10^{-2}$	$1.19x10^{-1}$	$1.36x10^{-1}$
	Rail	$2.28x10^{-2}$	$7.44x10^{-3}$	$4.30x10^{-4}$	$5.31x10^{-3}$	$1.32x10^{-2}$
Savannah, GA	Truck	$4.33x10^{-2}$	$1.61x10^{-3}$	$6.44x10^{-3}$	$9.29x10^{-2}$	$1.01x10^{-1}$
	Rail	$2.16x10^{-2}$	$7.78x10^{-3}$	$3.58x10^{-4}$	$6.58x10^{-3}$	$1.47x10^{-2}$
MOTSU, NC	Truck	$4.88x10^{-2}$	$2.07x10^{-3}$	$7.52x10^{-3}$	$1.03x10^{-1}$	$1.12x10^{-1}$
	Rail	$2.08x10^{-2}$	$6.52x10^{-3}$	$3.63x10^{-4}$	$5.04x10^{-3}$	$1.19x10^{-2}$
Wilmington, NC	Truck	$5.00x10^{-2}$	$2.06x10^{-3}$	$7.70x10^{-3}$	$1.05x10^{-1}$	$1.15x10^{-1}$
	Rail	$2.06x10^{-2}$	$6.41x10^{-3}$	$3.55x10^{-4}$	$5.00x10^{-3}$	$1.18x10^{-2}$
From Western Ports						
NWS Concord, CA	Truck	$2.26x10^{-1}$	$1.32x10^{-2}$	$4.47x10^{-2}$	$5.29x10^{-1}$	$5.87x10^{-1}$
	Rail	$5.91x10^{-2}$	$2.76x10^{-2}$	$9.50x10^{-4}$	$1.71x10^{-2}$	$4.57x10^{-2}$
Long Beach, CA	Truck	$2.03x10^{-1}$	$1.49x10^{-2}$	$4.52x10^{-2}$	$4.65x10^{-1}$	$5.25x10^{-1}$
	Rail	$5.68x10^{-2}$	$4.55x10^{-2}$	$1.06x10^{-3}$	$1.73x10^{-2}$	$6.39x10^{-2}$
Portland, OR	Truck	$2.25x10^{-1}$	$8.88x10^{-3}$	$3.72x10^{-2}$	$5.40x10^{-1}$	$5.86x10^{-1}$
	Rail	$5.94x10^{-2}$	$3.16x10^{-2}$	$1.05x10^{-3}$	$1.78x10^{-2}$	$5.04x10^{-2}$
Tacoma, WA	Truck	$2.26x10^{-1}$	$7.26x10^{-3}$	$3.44x10^{-2}$	$5.50x10^{-1}$	$5.92x10^{-1}$
	Rail	$5.95x10^{-2}$	$3.86x10^{-2}$	$1.17x10^{-3}$	$1.85x10^{-2}$	$5.83x10^{-2}$
From DOE Sites/Canadian Border						
Alexandria Bay, NY	Truck	$9.64x10^{-2}$	$4.11x10^{-3}$	$1.46x10^{-2}$	$1.92x10^{-1}$	$2.11x10^{-1}$
	Rail	$2.80x10^{-2}$	$3.17x10^{-2}$	$8.38x10^{-4}$	$9.07x10^{-3}$	$4.16x10^{-2}$
Hanford Site	Truck	$2.10x10^{-1}$	$7.29x10^{-2}$	$3.31x10^{-1}$	$5.10x10^{-1}$	$5.50x10^{-1}$
	Rail	$5.56x10^{-1}$	$1.64x10^{-1}$	$6.92x10^{-3}$	$1.60x10^{-1}$	$3.31x10^{-2}$
Idaho National Engineering Laboratory	Truck	$1.76x10^{-1}$	$6.45x10^{-2}$	$2.80x10^{-1}$	$4.24x10^{-1}$	$4.58x10^{-1}$
	Rail	$4.63x10^{-1}$	$1.41x10^{-1}$	$5.69x10^{-3}$	$1.33x10^{-1}$	$2.80x10^{-2}$
Nevada Test Site	Truck	$1.86x10^{-1}$	$7.63x10^{-2}$	$3.10x10^{-1}$	$4.45x10^{-1}$	$4.84x10^{-1}$
	Rail	$5.36x10^{-1}$	$1.68x10^{-1}$	$6.66x10^{-3}$	$1.63x10^{-1}$	$3.37x10^{-2}$
Savannah River	Truck	$4.23x10^{-1}$	$2.93x10^{-2}$	$7.96x10^{-2}$	$7.84x10^{-1}$	$8.93x10^{-1}$
	Rail	$1.87x10^{-1}$	$4.69x10^{-2}$	$2.38x10^{-3}$	$5.11x10^{-2}$	$1.00x10^{-2}$
Sweetgrass, MT	Truck	$1.94x10^{-1}$	$5.82x10^{-2}$	$2.93x10^{-1}$	$4.79x10^{-1}$	$5.14x10^{-1}$
	Rail	$4.70x10^{-1}$	$2.57x10^{-1}$	$8.41x10^{-3}$	$1.46x10^{-1}$	$4.11x10^{-2}$

Shipments to Savannah River Site:						
			General Public			
Route(s)		*Crew*	*Off-Link*	*On-Link*	*Stops*	*Total*
From Eastern Ports						
Charleston, SC (NWS)	Truck	$1.84x10^{-2}$	$7.44x10^{-4}$	$2.83x10^{-3}$	$3.89x10^{-2}$	$4.25x10^{-2}$
	Rail	$1.40x10^{-2}$	$1.77x10^{-3}$	$5.53x10^{-6}$	$4.77x10^{-3}$	$6.59x10^{-3}$
Charleston, SC (Wando Terminal)	Truck	$2.04x10^{-2}$	$1.10x10^{-3}$	$3.57x10^{-3}$	$4.21x10^{-2}$	$4.67x10^{-2}$
	Rail	$1.40x10^{-2}$	$1.77x10^{-3}$	$5.53x10^{-6}$	$4.77x10^{-3}$	$6.59x10^{-3}$
Elizabeth, NJ	Truck	$8.83x10^{-2}$	$5.31x10^{-3}$	$1.60x10^{-2}$	$1.70x10^{-1}$	$1.92x10^{-1}$
	Rail	$2.64x10^{-2}$	$4.14x10^{-2}$	$8.55x10^{-4}$	$8.08x10^{-3}$	$5.04x10^{-2}$
Galveston, TX	Truck	$1.02x10^{-1}$	$6.82x10^{-3}$	$1.94x10^{-2}$	$2.07x10^{-1}$	$2.33x10^{-1}$
	Rail	$3.15x10^{-2}$	$2.48x10^{-2}$	$7.41x10^{-4}$	$1.04x10^{-2}$	$3.59x10^{-2}$
Jacksonville, FL	Truck	$3.37x10^{-2}$	$7.25x10^{-4}$	$4.42x10^{-3}$	$7.81x10^{-2}$	$8.32x10^{-2}$
	Rail	$1.60x10^{-2}$	$3.06x10^{-3}$	$1.04x10^{-4}$	$4.61x10^{-3}$	$7.77x10^{-3}$
Newport News, VA	Truck	$5.24x10^{-2}$	$3.06x10^{-3}$	$9.51x10^{-3}$	$1.07x10^{-1}$	$1.20x10^{-1}$
	Rail	$2.18x10^{-2}$	$8.37x10^{-3}$	$3.55x10^{-4}$	$5.66x10^{-3}$	$1.44x10^{-2}$
Norfolk, VA	Truck	$4.89x10^{-2}$	$2.13x10^{-3}$	$7.66x10^{-3}$	$1.03x10^{-1}$	$1.13x10^{-1}$
	Rail	$2.06x10^{-2}$	$5.78x10^{-3}$	$2.66x10^{-4}$	$5.22x10^{-3}$	$1.13x10^{-2}$

Shipments to Savannah River Site:						
			General Public			
Route(s)		*Crew*	*Off-Link*	*On-Link*	*Stops*	*Total*
Philadelphia, PA	Truck	$8.21x10^{-2}$	$6.81x10^{-3}$	$1.78x10^{-2}$	$1.53x10^{-1}$	$1.78x10^{-1}$
	Rail	$2.50x10^{-2}$	$2.67x10^{-2}$	$6.51x10^{-4}$	$7.21x10^{-3}$	$3.46x10^{-2}$
Portsmouth, VA	Truck	$4.93x10^{-2}$	$2.21x10^{-3}$	$7.90x10^{-3}$	$1.04x10^{-1}$	$1.14x10^{-1}$
	Rail	$2.01x10^{-2}$	$4.71x10^{-3}$	$2.39x10^{-4}$	$4.72x10^{-3}$	$9.67x10^{-3}$
Savannah, GA	Truck	$2.29x10^{-2}$	$5.78x10^{-4}$	$3.04x10^{-3}$	$5.18x10^{-2}$	$5.54x10^{-2}$
	Rail	$1.36x10^{-2}$	$8.37x10^{-4}$	$3.51x10^{-5}$	$4.39x10^{-3}$	$5.27x10^{-3}$
MOTSU, NC	Truck	$2.22x10^{-2}$	$5.41x10^{-4}$	$3.06x10^{-3}$	$5.17x10^{-2}$	$5.53x10^{-2}$
	Rail	$1.81x10^{-2}$	$3.82x10^{-3}$	$1.73x10^{-4}$	$4.51x10^{-3}$	$8.50x10^{-3}$
Wilmington, NC	Truck	$2.96x10^{-2}$	$1.03x10^{-3}$	$4.30x10^{-3}$	$6.42x10^{-2}$	$6.95x10^{-2}$
	Rail	$1.79x10^{-2}$	$3.71x10^{-3}$	$1.65x10^{-4}$	$4.46x10^{-3}$	$8.33x10^{-3}$
From Western Ports						
NWS Concord, CA	Truck	$2.64x10^{-1}$	$1.83x10^{-2}$	$5.52x10^{-2}$	$5.76x10^{-1}$	$6.50x10^{-1}$
	Rail	$6.54x10^{-2}$	$5.17x10^{-2}$	$1.52x10^{-3}$	$1.88x10^{-2}$	$7.20x10^{-2}$
Long Beach, CA	Truck	$2.33x10^{-1}$	$1.61x10^{-2}$	$4.83x10^{-2}$	$5.06x10^{-1}$	$5.70x10^{-1}$
	Rail	$6.63x10^{-2}$	$5.65x10^{-2}$	$1.51x10^{-3}$	$1.80x10^{-2}$	$7.60x10^{-2}$
Portland, OR	Truck	$2.57x10^{-1}$	$1.17x10^{-2}$	$4.42x10^{-2}$	$5.96x10^{-1}$	$6.52x10^{-1}$
	Rail	$6.49x10^{-2}$	$3.88x10^{-2}$	$1.34x10^{-3}$	$1.81x10^{-2}$	$5.82x10^{-2}$
Seattle, WA	Truck	$2.54x10^{-1}$	$9.50x10^{-3}$	$4.00x10^{-2}$	$6.00x10^{-1}$	$6.50x10^{-1}$
	Rail	$6.44x10^{-2}$	$4.13x10^{-2}$	$1.37x10^{-3}$	$1.82x10^{-2}$	$6.09x10^{-2}$
Tacoma, WA	Truck	$2.58x10^{-1}$	$1.02x10^{-2}$	$4.15x10^{-2}$	$6.07x10^{-1}$	$6.59x10^{-1}$
	Rail	$6.51x10^{-2}$	$4.59x10^{-2}$	$1.45x10^{-3}$	$1.87x10^{-2}$	$6.60x10^{-2}$
From DOE Sites/Canadian Border						
Alexandria Bay, NY	Truck	$1.04x10^{-1}$	$4.58x10^{-3}$	$1.61x10^{-2}$	$2.10x10^{-1}$	$2.30x10^{-1}$
	Rail	$3.33x10^{-2}$	$6.15x10^{-2}$	$1.30x10^{-3}$	$9.18x10^{-3}$	$7.20x10^{-2}$
Hanford Site	Truck	$2.42x10^{-1}$	$1.02x10^{-2}$	$4.01x10^{-2}$	$5.65x10^{-1}$	$6.15x10^{-1}$
	Rail	$6.15x10^{-2}$	$2.74x10^{-2}$	$1.08x10^{-3}$	$1.66x10^{-2}$	$4.51x10^{-2}$
Idaho National Engineering Laboratory	Truck	$2.08x10^{-1}$	$9.34x10^{-3}$	$3.50x10^{-2}$	$4.79x10^{-1}$	$5.23x10^{-1}$
	Rail	$5.23x10^{-2}$	$2.51x10^{-2}$	$9.56x10^{-4}$	$1.41x10^{-2}$	$4.02x10^{-2}$
Nevada Test Site	Truck	$2.17x10^{-1}$	$1.05x10^{-2}$	$3.80x10^{-2}$	$5.00x10^{-1}$	$5.48x10^{-1}$
	Rail	$5.96x10^{-2}$	$2.77x10^{-2}$	$1.05x10^{-3}$	$1.65x10^{-2}$	$4.53x10^{-2}$
Oak Ridge Reservation	Truck	$4.23x10^{-2}$	$2.93x10^{-3}$	$7.96x10^{-3}$	$7.84x10^{-2}$	$8.93x10^{-2}$
	Rail	$1.87x10^{-2}$	$4.69x10^{-3}$	$2.38x10^{-4}$	$5.11x10^{-3}$	$1.00x10^{-2}$
Sweetgrass, MT	Truck	$2.26x10^{-1}$	$8.66x10^{-3}$	$3.62x10^{-2}$	$5.33x10^{-1}$	$5.78x10^{-1}$
	Rail	$5.26x10^{-2}$	$3.30x10^{-2}$	$1.13x10^{-3}$	$1.49x10^{-2}$	$4.90x10^{-2}$

[a] *Incident free risk factors are based on dose rates of 10 mrem per hr at 2 m (6.6 ft) (the regulatory limit).*

MOTSU = Military Ocean Terminal at Sunny Point, NWS = Naval Weapons Station

Table E-9 Accident Dose Risk per Shipment for All Spent Nuclear Fuel Types (Person-Rem/shipment)

Shipments to Hanford Site:		
Source/Route	*Truck*	*Rail*
From Eastern Ports		
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	$1.51x10^{-4}$	$2.06x10^{-5}$
Charleston, SC (Wando Terminal)	$1.54x10^{-4}$	$2.06x10^{-5}$
Elizabeth, NJ	$1.30x10^{-4}$	$4.19x10^{-5}$
Galveston, TX	$9.53x10^{-5}$	$8.81x10^{-6}$
Jacksonville, FL	$1.71x10^{-4}$	$1.86x10^{-5}$

Shipments to Hanford Site:		
Source/Route	***Truck***	***Rail***
Newport News, VA	$1.44x10^{-4}$	$3.28x10^{-5}$
Norfolk, VA	$1.47x10^{-4}$	$3.33x10^{-5}$
Philadelphia, PA	$1.31x10^{-4}$	$3.75x10^{-5}$
Portsmouth, VA	$1.49x10^{-4}$	$3.28x10^{-5}$
Savannah, GA	$1.65x10^{-4}$	$1.86x10^{-5}$
MOTSU, NC	$1.45x10^{-4}$	$2.14x10^{-5}$
Wilmington, NC	$1.51x10^{-4}$	$2.13x10^{-5}$
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	$6.54x10^{-5}$	$8.86x10^{-6}$
Charleston, SC (Wando Terminal)	$6.67x10^{-5}$	$8.86x10^{-6}$
Elizabeth, NJ	$5.59x10^{-5}$	$1.80x10^{-5}$
Galveston, TX	$4.12x10^{-5}$	$3.80x10^{-6}$
Jacksonville, FL	$7.39x10^{-5}$	$8.00x10^{-6}$
Newport News, VA	$6.21x10^{-5}$	$1.41x10^{-5}$
Norfolk, VA	$6.35x10^{-5}$	$1.43x10^{-5}$
Philadelphia, PA	$5.67x10^{-5}$	$1.61x10^{-5}$
Portsmouth, VA	$6.44x10^{-5}$	$1.41x10^{-5}$
Savannah, GA	$7.11x10^{-5}$	$8.03x10^{-6}$
MOTSU, NC	$6.28x10^{-5}$	$9.20x10^{-6}$
Wilmington, NC	$6.51x10^{-5}$	$9.15x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	$2.09x10^{-4}$	$2.86x10^{-5}$
Charleston, SC (Wando Terminal)	$2.13x10^{-4}$	$2.86x10^{-5}$
Elizabeth, NJ	$1.79x10^{-4}$	$5.83x10^{-5}$
Galveston, TX	$1.32x10^{-4}$	$1.22x10^{-5}$
Jacksonville, FL	$2.36x10^{-4}$	$2.58x10^{-5}$
Newport News, VA	$1.99x10^{-4}$	$4.56x10^{-5}$
Norfolk, VA	$2.03x10^{-4}$	$4.64x10^{-5}$
Philadelphia, PA	$1.82x10^{-4}$	$5.22x10^{-5}$
Portsmouth, VA	$2.06x10^{-4}$	$4.57x10^{-5}$
Savannah, GA	$2.27x10^{-4}$	$2.59x10^{-5}$
MOTSU, NC	$2.01x10^{-4}$	$2.97x10^{-5}$
Wilmington, NC	$2.08x10^{-4}$	$2.95x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	$4.33x10^{-4}$	$9.80x10^{-5}$
Charleston, SC (Wando Terminal)	$4.40x10^{-4}$	$9.80x10^{-5}$
Elizabeth, NJ	$4.49x10^{-4}$	$2.11x10^{-4}$
Galveston, TX	$2.68x10^{-4}$	$3.71x10^{-5}$
Jacksonville, FL	$4.81x10^{-4}$	$9.21x10^{-5}$
Newport News, VA	$4.23x10^{-4}$	$1.83x10^{-4}$
Norfolk, VA	$4.31x10^{-4}$	$1.85x10^{-4}$
Philadelphia, PA	$4.63x10^{-4}$	$1.97x10^{-4}$
Portsmouth, VA	$4.36x10^{-4}$	$1.84x10^{-4}$
Savannah, GA	$4.64x10^{-4}$	$9.25x10^{-5}$
MOTSU, NC	$4.27x10^{-4}$	$1.01x10^{-4}$
Wilmington, NC	$4.32x10^{-4}$	$1.01x10^{-4}$
Calcined Target Material		
Charleston, SC (NWS)	$3.96x10^{-2}$	$7.21x10^{-3}$
Charleston, SC (Wando Terminal)	$3.98x10^{-2}$	$7.21x10^{-3}$
Elizabeth, NJ	$6.73x10^{-2}$	$1.66x10^{-2}$
Jacksonville, FL	$4.09x10^{-2}$	$7.17x10^{-3}$

Shipments to Hanford Site:		
Source/Route	***Truck***	***Rail***
Newport News, VA	$4.22x10^{-2}$	$1.62x10^{-2}$
Norfolk, VA	$4.27x10^{-2}$	$1.63x10^{-2}$
Philadelphia, PA	$7.31x10^{-2}$	$1.62x10^{-2}$
Portsmouth, VA	$4.25x10^{-2}$	$1.63x10^{-2}$
Savannah, GA	$4.01x10^{-2}$	$7.21x10^{-3}$
MOTSU, NC	$4.35x10^{-2}$	$7.41x10^{-3}$
Wilmington, NC	$4.07x10^{-2}$	$7.40x10^{-3}$
Galveston, TX	$2.25x10^{-2}$	$2.16x10^{-3}$
Oxidized Target Material		
Charleston, SC (NWS)	$9.91x10^{-2}$	$1.80x10^{-2}$
Charleston, SC (Wando Terminal)	$9.95x10^{-2}$	$1.80x10^{-2}$
Elizabeth, NJ	$1.68x10^{-1}$	$4.14x10^{-2}$
Jacksonville, FL	$1.02x10^{-1}$	$1.79x10^{-2}$
Newport News, VA	$1.05x10^{-1}$	$4.06x10^{-2}$
Norfolk, VA	$1.07x10^{-1}$	$4.09x10^{-2}$
Philadelphia, PA	$1.83x10^{-1}$	$4.06x10^{-2}$
Portsmouth, VA	$1.06x10^{-1}$	$4.08x10^{-2}$
Savannah, GA	$1.00x10^{-1}$	$1.80x10^{-2}$
MOTSU, NC	$1.09x10^{-1}$	$1.85x10^{-2}$
Wilmington, NC	$1.02x10^{-1}$	$1.85x10^{-2}$
Galveston, TX	$5.62x10^{-2}$	$5.40x10^{-3}$
From Western Ports		
BR-2 Belgium Spent Nuclear Fuel		
NWS, Concord CA	$4.46x10^{-5}$	$1.50x10^{-5}$
Long Beach, CA	$7.36x10^{-5}$	$1.59x10^{-5}$
Portland, OR	$1.15x10^{-5}$	$2.10x10^{-6}$
Tacoma, WA	$8.55x10^{-6}$	$2.08x10^{-6}$
RHF France Spent Nuclear Fuel		
NWS, Concord CA	$1.93x10^{-5}$	$6.47x10^{-6}$
Long Beach, CA	$3.18x10^{-5}$	$6.86x10^{-6}$
Portland, OR	$4.99x10^{-6}$	$9.08x10^{-7}$
Tacoma, WA	$3.69x10^{-6}$	$8.99x10^{-7}$
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	$6.15x10^{-5}$	$2.07x10^{-5}$
Long Beach, CA	$1.02x10^{-4}$	$2.20x10^{-5}$
Portland, OR	$1.59x10^{-5}$	$2.91x10^{-6}$
Tacoma, WA	$1.18x10^{-5}$	$2.88x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	$1.15x10^{-4}$	$5.15x10^{-5}$
Long Beach, CA	$1.90x10^{-4}$	$5.50x10^{-5}$
Portland, OR	$2.97x10^{-5}$	$7.50x10^{-6}$
Tacoma, WA	$2.46x10^{-5}$	$7.79x10^{-6}$
Calcined Target Material		
NWS Concord, CA	$4.92x10^{-3}$	$1.50x10^{-3}$
Long Beach, CA	$7.06x10^{-3}$	$1.63x10^{-3}$
Portland, OR	$1.28x10^{-3}$	$2.54x10^{-4}$
Tacoma, WA	$1.90x10^{-3}$	$3.11x10^{-4}$
Oxidized Target Material		
NWS Concord, CA	$1.23x10^{-2}$	$3.75x10^{-3}$
Long Beach, CA	$1.77x10^{-2}$	$4.08x10^{-3}$
Portland, OR	$3.21x10^{-3}$	$6.36x10^{-4}$

Shipments to Hanford Site:		
Source/Route	***Truck***	***Rail***
Tacoma, WA	4.75×10^{-3}	7.80×10^{-4}
From DOE Sites/Canadian Border		
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	1.52×10^{-4}	3.76×10^{-5}
Idaho National Engineering Laboratory	1.00×10^{-5}	3.08×10^{-6}
Nevada Test Site	5.07×10^{-5}	4.62×10^{-6}
Oak Ridge Reservation	1.13×10^{-4}	1.16×10^{-5}
Savannah River	1.61×10^{-4}	1.97×10^{-5}
Sweetgrass, MT	2.25×10^{-5}	1.56×10^{-6}
RHF France Spent Nuclear Fuel		
Alexandria Bay, NY	6.56×10^{-5}	1.62×10^{-5}
Idaho National Engineering Laboratory	4.33×10^{-6}	1.33×10^{-6}
Nevada Test Site	2.19×10^{-5}	1.99×10^{-6}
Oak Ridge Reservation	4.86×10^{-5}	4.97×10^{-6}
Savannah River	6.94×10^{-5}	8.47×10^{-6}
Sweetgrass, MT	9.72×10^{-6}	6.72×10^{-7}
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	2.10×10^{-4}	5.23×10^{-5}
Idaho National Engineering Laboratory	1.38×10^{-5}	4.27×10^{-6}
Nevada Test Site	6.99×10^{-5}	6.39×10^{-6}
Oak Ridge Reservation	1.56×10^{-4}	1.62×10^{-5}
Savannah River	2.22×10^{-4}	2.73×10^{-5}
Sweetgrass, MT	3.11×10^{-5}	2.16×10^{-6}
PRR-1 TRIGA Spent Nuclear Fuel		
Alexandria Bay, NY	4.99×10^{-4}	1.98×10^{-4}
Idaho National Engineering Laboratory	2.94×10^{-5}	1.21×10^{-5}
Nevada Test Site	1.29×10^{-4}	1.74×10^{-5}
Oak Ridge Reservation	3.35×10^{-4}	7.04×10^{-5}
Savannah River	4.53×10^{-4}	9.49×10^{-5}
Sweetgrass, MT	6.49×10^{-5}	6.18×10^{-6}
Calcined Target Material		
Alexandria Bay, NY	6.72×10^{-2}	1.63×10^{-2}
Idaho National Engineering Laboratory	2.69×10^{-3}	5.83×10^{-4}
Nevada Test Site	4.74×10^{-3}	7.41×10^{-4}
Oak Ridge Reservation	3.55×10^{-2}	6.78×10^{-3}
Savannah River	3.96×10^{-2}	7.11×10^{-3}
Sweetgrass, MT	6.01×10^{-3}	3.03×10^{-4}
Oxidized Target Material		
Alexandria Bay, NY	1.68×10^{-1}	4.07×10^{-2}
Idaho National Engineering Laboratory	6.74×10^{-3}	1.46×10^{-3}
Nevada Test Site	1.19×10^{-2}	1.85×10^{-3}
Oak Ridge Reservation	8.88×10^{-2}	1.70×10^{-2}
Savannah River	9.90×10^{-2}	1.78×10^{-2}
Sweetgrass, MT	1.50×10^{-2}	7.57×10^{-4}

Shipments to Idaho National Engineering Laboratory:		
Source/Route	***Truck***	***Rail***
From Eastern Ports		
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	1.45×10^{-4}	1.85×10^{-5}
Charleston, SC (Wando Terminal)	1.48×10^{-4}	1.85×10^{-5}
Elizabeth, NJ	1.23×10^{-4}	3.98×10^{-5}

Shipments to Idaho National Engineering Laboratory:		
Source/Route	***Truck***	***Rail***
Galveston, TX	$8.88x10^{-5}$	$6.71x10^{-6}$
Jacksonville, FL	$1.62x10^{-4}$	$1.65x10^{-5}$
Newport News, VA	$1.37x10^{-4}$	$3.07x10^{-5}$
Norfolk, VA	$1.39x10^{-4}$	$3.12x10^{-5}$
Philadelphia, PA	$1.25x10^{-4}$	$3.54x10^{-5}$
Portsmouth, VA	$1.43x10^{-4}$	$3.07x10^{-5}$
Savannah, GA	$1.58x10^{-4}$	$1.66x10^{-5}$
MOTSU, NC	$1.34x10^{-4}$	$1.93x10^{-5}$
Wilmington, NC	$1.44x10^{-4}$	$1.92x10^{-5}$
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	$6.26x10^{-5}$	$7.96x10^{-6}$
Charleston, SC (Wando Terminal)	$6.39x10^{-5}$	$7.96x10^{-6}$
Elizabeth, NJ	$5.31x10^{-5}$	$1.71x10^{-5}$
Galveston, TX	$3.84x10^{-5}$	$2.89x10^{-6}$
Jacksonville, FL	$7.02x10^{-5}$	$7.10x10^{-6}$
Newport News, VA	$5.93x10^{-5}$	$1.32x10^{-5}$
Norfolk, VA	$6.01x10^{-5}$	$1.34x10^{-5}$
Philadelphia, PA	$5.39x10^{-5}$	$1.52x10^{-5}$
Portsmouth, VA	$6.16x10^{-5}$	$1.32x10^{-5}$
Savannah, GA	$6.83x10^{-5}$	$7.12x10^{-6}$
MOTSU, NC	$5.78x10^{-5}$	$8.30x10^{-6}$
Wilmington, NC	$6.22x10^{-5}$	$8.25x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	$2.00x10^{-4}$	$2.57x10^{-5}$
Charleston, SC (Wando Terminal)	$2.04x10^{-4}$	$2.57x10^{-5}$
Elizabeth, NJ	$1.23x10^{-4}$	$5.54x10^{-5}$
Galveston, TX	$1.70x10^{-4}$	$9.29x10^{-6}$
Jacksonville, FL	$2.24x10^{-4}$	$2.29x10^{-5}$
Newport News, VA	$1.90x10^{-4}$	$4.27x10^{-5}$
Norfolk, VA	$1.92x10^{-4}$	$4.35x10^{-5}$
Portsmouth, VA	$1.97x10^{-4}$	$4.28x10^{-5}$
Savannah, GA	$2.18x10^{-4}$	$2.30x10^{-5}$
MOTSU, NC	$1.85x10^{-4}$	$2.68x10^{-5}$
Wilmington, NC	$1.99x10^{-4}$	$2.66x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	$4.14x10^{-4}$	$8.94x10^{-5}$
Charleston, SC (Wando Terminal)	$4.21x10^{-4}$	$8.94x10^{-5}$
Elizabeth, NJ	$4.30x10^{-4}$	$2.02x10^{-4}$
Galveston, TX	$4.49x10^{-4}$	$8.47x10^{-5}$
Jacksonville, FL	$4.57x10^{-4}$	$8.36x10^{-5}$
Newport News, VA	$4.04x10^{-4}$	$1.75x10^{-4}$
Norfolk, VA	$4.09x10^{-4}$	$1.77x10^{-4}$
Philadelphia, PA	$4.44x10^{-4}$	$1.88x10^{-4}$
Portsmouth, VA	$4.17x10^{-4}$	$1.75x10^{-4}$
Savannah, GA	$4.45x10^{-4}$	$8.39x10^{-5}$
MOTSU, NC	$3.97x10^{-4}$	$9.26x10^{-5}$
Wilmington, NC	$4.13x10^{-4}$	$9.22x10^{-5}$
Calcined Target Material		
Charleston, SC (NWS)	$3.79x10^{-2}$	$6.76x10^{-3}$
Charleston, SC (Wando Terminal)	$3.81x10^{-2}$	$6.76x10^{-3}$
Elizabeth, NJ	$6.57x10^{-2}$	$1.61x10^{-2}$

Shipments to Idaho National Engineering Laboratory:		
Source/Route	***Truck***	***Rail***
Jacksonville, FL	$3.91x10^{-2}$	$6.72x10^{-3}$
Newport News, VA	$4.05x10^{-2}$	$1.58x10^{-2}$
Norfolk, VA	$4.10x10^{-2}$	$1.59x10^{-2}$
Philadelphia, PA	$7.15x10^{-2}$	$1.58x10^{-2}$
Portsmouth, VA	$4.09x10^{-2}$	$1.59x10^{-2}$
Savannah, GA	$3.84x10^{-2}$	$6.76x10^{-3}$
MOTSU, NC	$4.18x10^{-2}$	$6.96x10^{-3}$
Wilmington, NC	$3.91x10^{-2}$	$6.95x10^{-3}$
Galveston, TX	$2.08x10^{-2}$	$1.71x10^{-3}$
Oxidized Target Material		
Charleston, SC (NWS)	$9.49x10^{-2}$	$1.69x10^{-2}$
Charleston, SC (Wando Terminal)	$9.53x10^{-2}$	$1.69x10^{-2}$
Elizabeth, NJ	$1.64x10^{-1}$	$4.03x10^{-2}$
Jacksonville, FL	$9.79x10^{-2}$	$1.68x10^{-2}$
Newport News, VA	$1.01x10^{-1}$	$3.95x10^{-2}$
Norfolk, VA	$1.03x10^{-1}$	$3.98x10^{-2}$
Philadelphia, PA	$1.79x10^{-1}$	$3.95x10^{-2}$
Portsmouth, VA	$1.02x10^{-1}$	$3.96x10^{-2}$
Savannah, GA	$9.62x10^{-2}$	$1.69x10^{-2}$
MOTSU, NC	$1.05x10^{-1}$	$1.74x10^{-2}$
Wilmington, NC	$9.78x10^{-2}$	$1.74x10^{-2}$
Galveston, TX	$5.21x10^{-2}$	$4.28x10^{-3}$
From Western Ports		
BR-2 Belgium Spent Nuclear Fuel		
NWS Concord, CA	$3.83x10^{-5}$	$4.00x10^{-6}$
Long Beach, CA	$7.40x10^{-5}$	$1.38x10^{-5}$
Portland, OR	$1.93x10^{-5}$	$4.94x10^{-6}$
Tacoma, WA	$1.66x10^{-5}$	$6.85x10^{-6}$
RHF France Spent Nuclear Fuel		
NWS Concord, CA	$1.66x10^{-5}$	$1.73x10^{-6}$
Long Beach, CA	$3.20x10^{-5}$	$5.96x10^{-6}$
Portland, OR	$8.35x10^{-6}$	$2.13x10^{-6}$
Tacoma, WA	$7.19x10^{-6}$	$2.95x10^{-6}$
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	$5.29x10^{-5}$	$5.53x10^{-6}$
Long Beach, CA	$1.02x10^{-4}$	$1.91x10^{-5}$
Portland, OR	$2.67x10^{-5}$	$6.84x10^{-6}$
Tacoma, WA	$2.30x10^{-5}$	$9.47x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	$9.68x10^{-5}$	$1.42x10^{-5}$
Long Beach, CA	$1.85x10^{-4}$	$4.64x10^{-5}$
Portland, OR	$5.19x10^{-5}$	$1.85x10^{-5}$
Tacoma, WA	$4.88x10^{-5}$	$2.50x10^{-5}$
Calcined Target Material		
NWS Concord, CA	$3.21x10^{-3}$	$4.70x10^{-4}$
Long Beach, CA	$4.75x10^{-3}$	$1.18x10^{-3}$
Portland, OR	$3.23x10^{-3}$	$7.61x10^{-4}$
Tacoma, WA	$4.35x10^{-3}$	$9.55x10^{-4}$
Oxidized Target Material		
NWS Concord, CA	$8.05x10^{-3}$	$1.18x10^{-3}$
Long Beach, CA	$1.19x10^{-2}$	$2.95x10^{-3}$

Shipments to Idaho National Engineering Laboratory:		
Source/Route	***Truck***	***Rail***
Portland, OR	$8.07x10^{-3}$	$1.91x10^{-3}$
Tacoma, WA	$1.09x10^{-2}$	$2.39x10^{-3}$
From DOE Sites/Canadian Border		
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	$1.45x10^{-4}$	$3.55x10^{-5}$
Hanford Site	$1.00x10^{-5}$	$3.08x10^{-6}$
Nevada Test Site	$4.41x10^{-5}$	$2.52x10^{-6}$
Oak Ridge Reservation	$1.06x10^{-4}$	$9.48x10^{-6}$
Savannah River	$1.54x10^{-4}$	$1.76x10^{-5}$
Sweetgrass, MT	$1.59x10^{-5}$	$3.84x10^{-6}$
RHF France Spent Nuclear Fuel		
Hanford Site	$4.33x10^{-6}$	$1.33x10^{-6}$
Nevada Test Site	$1.91x10^{-5}$	$1.09x10^{-6}$
Oak Ridge Reservation	$4.58x10^{-5}$	$4.07x10^{-6}$
Savannah River	$6.66x10^{-5}$	$7.57x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	$2.01x10^{-4}$	$4.94x10^{-5}$
Hanford Site	$1.38x10^{-5}$	$4.27x10^{-6}$
Nevada Test Site	$6.09x10^{-5}$	$3.48x10^{-6}$
Oak Ridge Reservation	$1.46x10^{-4}$	$1.33x10^{-5}$
Savannah River	$2.13x10^{-4}$	$2.44x10^{-5}$
Sweetgrass, MT	$2.20x10^{-5}$	$5.32x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
Hanford Site	$2.94x10^{-5}$	$1.21x10^{-5}$
Nevada Test Site	$1.09x10^{-4}$	$8.79x10^{-6}$
Oak Ridge Reservation	$3.16x10^{-4}$	$6.19x10^{-5}$
Savannah River	$4.34x10^{-4}$	$8.63x10^{-5}$
Calcined Target Material		
Alexandria Bay, NY	$6.55x10^{-2}$	$1.58x10^{-2}$
Hanford Site	$2.69x10^{-3}$	$5.83x10^{-4}$
Nevada Test Site	$3.10x10^{-3}$	$2.90x10^{-4}$
Oak Ridge Reservation	$3.38x10^{-2}$	$6.33x10^{-3}$
Savannah River	$3.79x10^{-2}$	$6.66x10^{-3}$
Sweetgrass, MT	$3.56x10^{-3}$	$8.19x10^{-4}$
Oxidized Target Material		
Alexandria Bay, NY	$1.64x10^{-1}$	$3.96x10^{-2}$
Hanford	$6.74x10^{-3}$	$1.46x10^{-3}$
Nevada Test Site	$7.76x10^{-3}$	$7.27x10^{-4}$
Oak Ridge Reservation	$8.47x10^{-2}$	$1.58x10^{-2}$
Savannah River	$9.49x10^{-2}$	$1.67x10^{-2}$
Sweetgrass, MT	$8.91x10^{-3}$	$2.05x10^{-3}$

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
From Eastern Ports		
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	$1.65x10^{-4}$	$1.99x10^{-5}$
Charleston, SC (Wando Terminal)	$1.68x10^{-4}$	$1.99x10^{-5}$
Elizabeth, NJ	$1.77x10^{-4}$	$4.12x10^{-5}$
Galveston, TX	$9.13x10^{-5}$	$5.16x10^{-6}$
Jacksonville, FL	$1.89x10^{-4}$	$1.79x10^{-5}$
Newport News, VA	$1.64x10^{-4}$	$3.21x10^{-5}$

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
Norfolk, VA	$1.66x10^{-4}$	$3.27x10^{-5}$
Philadelphia, PA	$1.75x10^{-4}$	$3.68x10^{-5}$
Portsmouth, VA	$1.69x10^{-4}$	$3.22x10^{-5}$
Savannah, GA	$1.85x10^{-4}$	$1.80x10^{-5}$
MOTSU, NC	$1.65x10^{-4}$	$2.07x10^{-5}$
Wilmington, NC	$1.71x10^{-4}$	$2.06x10^{-5}$
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	$7.12x10^{-5}$	$8.57x10^{-6}$
Charleston, SC (Wando Terminal)	$7.25x10^{-5}$	$8.57x10^{-6}$
Elizabeth, NJ	$7.66x10^{-5}$	$1.78x10^{-5}$
Galveston, TX	$3.95x10^{-5}$	$2.42x10^{-6}$
Jacksonville, FL	$8.17x10^{-5}$	$7.71x10^{-6}$
Newport News, VA	$7.08x10^{-5}$	$1.38x10^{-5}$
Norfolk, VA	$7.16x10^{-5}$	$1.40x10^{-5}$
Philadelphia, PA	$7.56x10^{-5}$	$1.58x10^{-5}$
Portsmouth, VA	$7.31x10^{-5}$	$1.38x10^{-5}$
Savannah, GA	$7.98x10^{-5}$	$7.74x10^{-6}$
MOTSU, NC	$7.14x10^{-5}$	$8.91x10^{-6}$
Wilmington, NC	$7.37x10^{-5}$	$8.86x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	$2.27x10^{-4}$	$2.76x10^{-5}$
Charleston, SC (Wando Terminal)	$2.32x10^{-4}$	$2.76x10^{-5}$
Elizabeth, NJ	$2.45x10^{-4}$	$5.73x10^{-5}$
Galveston, TX	$1.26x10^{-4}$	$7.77x10^{-6}$
Jacksonville, FL	$2.61x10^{-4}$	$2.49x10^{-5}$
Newport News, VA	$2.26x10^{-4}$	$4.47x10^{-5}$
Norfolk, VA	$2.29x10^{-4}$	$4.55x10^{-5}$
Philadelphia, PA	$2.42x10^{-4}$	$5.12x10^{-5}$
Portsmouth, VA	$2.34x10^{-4}$	$4.48x10^{-5}$
Savannah, GA	$2.55x10^{-4}$	$2.50x10^{-5}$
MOTSU, NC	$2.28x10^{-4}$	$2.87x10^{-5}$
Wilmington, NC	$2.36x10^{-4}$	$2.86x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	$4.62x10^{-4}$	$9.35x10^{-5}$
Charleston, SC (Wando Terminal)	$4.69x10^{-4}$	$9.35x10^{-5}$
Elizabeth, NJ	$5.19x10^{-4}$	$2.07x10^{-4}$
Galveston, TX	$2.37x10^{-4}$	$2.44x10^{-5}$
Jacksonville, FL	$5.21x10^{-4}$	$8.77x10^{-5}$
Newport News, VA	$4.68x10^{-4}$	$1.79x10^{-4}$
Norfolk, VA	$4.73x10^{-4}$	$1.81x10^{-4}$
Philadelphia, PA	$5.17x10^{-4}$	$1.92x10^{-4}$
Portsmouth, VA	$4.81x10^{-4}$	$1.79x10^{-4}$
Savannah, GA	$5.10x10^{-4}$	$8.80x10^{-5}$
MOTSU, NC	$4.19x10^{-4}$	$9.67x10^{-5}$
Wilmington, NC	$4.78x10^{-4}$	$9.63x10^{-5}$
20 yr old vitrified HLW		
Charleston, SC (NWS)	$1.40x10^{-3}$	$2.33x10^{-4}$
Charleston, SC (Wando Terminal)	$1.42x10^{-3}$	$2.33x10^{-4}$
Elizabeth, NJ	$1.70x10^{-3}$	$5.33x10^{-4}$
Jacksonville, FL	$1.45x10^{-3}$	$2.27x10^{-4}$
Newport News, VA	$1.45x10^{-3}$	$5.06x10^{-4}$

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
Norfolk, VA	$1.47x10^{-3}$	$5.10x10^{-4}$
Philadelphia, PA	$1.73x10^{-3}$	$5.14x10^{-4}$
Portsmouth, VA	$1.46x10^{-3}$	$5.07x10^{-4}$
Savannah, GA	$1.42x10^{-3}$	$2.28x10^{-4}$
MOTSU, NC	$7.07x10^{-4}$	$2.40x10^{-4}$
Wilmington, NC	$1.44x10^{-3}$	$2.40x10^{-4}$
Galveston, TX	$4.20x10^{-4}$	$4.70x10^{-5}$
Calcined Target Material		
Charleston, SC (NWS)	$3.90x10^{-2}$	$6.78x10^{-3}$
Charleston, SC (Wando Terminal)	$3.92x10^{-2}$	$6.78x10^{-3}$
Elizabeth, NJ	$4.86x10^{-2}$	$1.61x10^{-2}$
Jacksonville, FL	$4.03x10^{-2}$	$6.74x10^{-3}$
Newport News, VA	$4.16x10^{-2}$	$1.58x10^{-2}$
Norfolk, VA	$4.21x10^{-2}$	$1.59x10^{-2}$
Philadelphia, PA	$5.03x10^{-2}$	$1.58x10^{-2}$
Portsmouth, VA	$4.20x10^{-2}$	$1.59x10^{-2}$
Savannah, GA	$3.96x10^{-2}$	$6.78x10^{-3}$
MOTSU, NC	$1.79x10^{-2}$	$6.98x10^{-3}$
Wilmington, NC	$4.02x10^{-2}$	$6.97x10^{-3}$
Galveston, TX	$1.11x10^{-2}$	$1.43x10^{-3}$
Oxidized Target Material		
Charleston, SC (NWS)	$9.76x10^{-2}$	$1.70x10^{-2}$
Charleston, SC (Wando Terminal)	$9.81x10^{-2}$	$1.70x10^{-2}$
Elizabeth, NJ	$1.22x10^{-1}$	$4.04x10^{-2}$
Jacksonville, FL	$1.01x10^{-1}$	$1.69x10^{-2}$
Newport News, VA	$1.04x10^{-1}$	$3.95x10^{-2}$
Norfolk, VA	$1.05x10^{-1}$	$3.98x10^{-2}$
Philadelphia, PA	$1.26x10^{-1}$	$3.95x10^{-2}$
Portsmouth, VA	$1.05x10^{-1}$	$3.97x10^{-2}$
Savannah, GA	$9.90x10^{-2}$	$1.70x10^{-2}$
MOTSU, NC	$4.48x10^{-2}$	$1.75x10^{-2}$
Wilmington, NC	$1.01x10^{-1}$	$1.74x10^{-2}$
Galveston, TX	$2.78x10^{-2}$	$3.57x10^{-3}$
From Western Ports		
BR-2 Belgium Spent Nuclear Fuel		
NWS Concord, CA	$4.88x10^{-5}$	$8.09x10^{-6}$
Long Beach, CA	$5.28x10^{-5}$	$1.13x10^{-5}$
Portland, OR	$6.10x10^{-5}$	$6.48x10^{-6}$
Tacoma, WA	$5.73x10^{-5}$	$8.38x10^{-6}$
RHF France Spent Nuclear Fuel		
NWS Concord, CA	$2.11x10^{-5}$	$3.49x10^{-6}$
Long Beach, CA	$2.28x10^{-5}$	$4.89x10^{-6}$
Portland, OR	$2.64x10^{-5}$	$2.80x10^{-6}$
Tacoma, WA	$2.48x10^{-5}$	$3.62x10^{-6}$
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	$6.73x10^{-5}$	$1.12x10^{-5}$
Long Beach, CA	$7.28x10^{-5}$	$1.56x10^{-5}$
Portland, OR	$8.41x10^{-5}$	$8.96x10^{-6}$
Tacoma, WA	$7.91x10^{-5}$	$1.16x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	$1.26x10^{-4}$	$2.91x10^{-5}$

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
Long Beach, CA	$1.31x10^{-4}$	$3.77x10^{-5}$
Portland, OR	$1.53x10^{-4}$	$2.37x10^{-5}$
Tacoma, WA	$1.48x10^{-4}$	$3.02x10^{-5}$
Calcined Target Material		
NWS Concord, CA	$4.13x10^{-3}$	$9.39x10^{-4}$
Long Beach, CA	$2.42x10^{-3}$	$8.94x10^{-4}$
Portland, OR	$5.26x10^{-3}$	$9.19x10^{-4}$
Tacoma, WA	$6.39x10^{-3}$	$1.11x10^{-3}$
Oxidized Target Material		
NWS Concord, CA	$1.03x10^{-2}$	$2.35x10^{-3}$
Long Beach, CA	$6.05x10^{-3}$	$2.24x10^{-3}$
Portland, OR	$1.32x10^{-2}$	$2.30x10^{-3}$
Tacoma, WA	$1.60x10^{-2}$	$2.79x10^{-3}$
From DOE Sites/Canadian Border		
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	$1.79x10^{-4}$	$3.69x10^{-5}$
Hanford Site	$5.07x10^{-5}$	$4.62x10^{-6}$
Idaho National Engineering Laboratory	$4.41x10^{-5}$	$2.52x10^{-6}$
Oak Ridge Reservation	$1.33x10^{-4}$	$1.09x10^{-5}$
Savannah River	$1.81x10^{-4}$	$1.90x10^{-5}$
Sweetgrass, MT	$6.08x10^{-5}$	$5.38x10^{-6}$
RHF France Spent Nuclear Fuel		
Alexandria Bay, NY	$7.71x10^{-5}$	$1.59x10^{-5}$
Hanford Site	$2.19x10^{-5}$	$1.99x10^{-6}$
Idaho National Engineering Laboratory	$1.91x10^{-5}$	$1.09x10^{-6}$
Oak Ridge Reservation	$5.73x10^{-5}$	$4.68x10^{-6}$
Savannah River	$7.81x10^{-5}$	$8.18x10^{-6}$
Sweetgrass, MT	$2.63x10^{-5}$	$2.32x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	$2.47x10^{-4}$	$5.14x10^{-5}$
Hanford Site	$6.99x10^{-5}$	$6.39x10^{-6}$
Idaho National Engineering Laboratory	$6.09x10^{-5}$	$3.48x10^{-6}$
Oak Ridge Reservation	$1.83x10^{-4}$	$1.52x10^{-5}$
Savannah River	$2.49x10^{-4}$	$2.64x10^{-5}$
Sweetgrass, MT	$8.38x10^{-5}$	$7.45x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
Alexandria Bay, NY	$5.58x10^{-4}$	$1.93x10^{-4}$
Hanford Site	$1.29x10^{-4}$	$1.74x10^{-5}$
Idaho National Engineering Laboratory	$1.09x10^{-4}$	$8.79x10^{-6}$
Oak Ridge Reservation	$3.81x10^{-4}$	$6.60x10^{-5}$
Savannah River	$4.99x10^{-4}$	$9.04x10^{-5}$
Sweetgrass MT	$1.54x10^{-4}$	$2.09x10^{-5}$
Calcined Target Material		
Alexandria Bay, NY	$6.55x10^{-2}$	$1.58x10^{-2}$
Hanford	$4.74x10^{-3}$	$7.41x10^{-4}$
Idaho National Engineering Laboratory	$3.10x10^{-3}$	$2.90x10^{-4}$
Oak Ridge Reservation	$3.50x10^{-2}$	$6.35x10^{-3}$
Savannah River	$3.91x10^{-2}$	$6.68x10^{-3}$
Sweetgrass, MT	$6.29x10^{-3}$	$9.78x10^{-4}$
Oxidized Target Material		
Alexandria Bay, NY	$1.64x10^{-1}$	$3.96x10^{-2}$

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
Hanford	$1.19x10^{-2}$	$1.85x10^{-3}$
Idaho National Engineering Laboratory	$7.76x10^{-3}$	$7.27x10^{-4}$
Oak Ridge Reservation	$8.75x10^{-2}$	$1.59x10^{-2}$
Savannah River	$9.77x10^{-2}$	$1.67x10^{-2}$
Sweetgrass, MT	$1.58x10^{-2}$	$2.45x10^{-3}$

Shipments to Oak Ridge Reservation:		
Source/Route	***Truck***	***Rail***
	From Eastern Ports	
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	$3.28x10^{-5}$	$5.39x10^{-6}$
Charleston, SC (Wando Terminal)	$3.59x10^{-5}$	$5.39x10^{-6}$
Elizabeth, NJ	$5.86x10^{-5}$	$2.60x10^{-5}$
Galveston, TX	$6.99x10^{-5}$	$1.77x10^{-5}$
Jacksonville, FL	$4.98x10^{-5}$	$6.73x10^{-6}$
Newport News, VA	$4.20x10^{-5}$	$7.94x10^{-6}$
Norfolk, VA	$3.90x10^{-5}$	$6.68x10^{-6}$
Philadelphia, PA	$6.36x10^{-5}$	$1.76x10^{-5}$
Portsmouth, VA	$4.74x10^{-5}$	$6.18x10^{-6}$
Savannah, GA	$3.20x10^{-5}$	$6.79x10^{-6}$
MOTSU, NC	$3.60x10^{-5}$	$5.33x10^{-6}$
Wilmington, NC	$3.85x10^{-5}$	$5.22x10^{-6}$
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	$1.42x10^{-5}$	$2.33x10^{-6}$
Charleston, SC (Wando Terminal)	$1.55x10^{-5}$	$2.33x10^{-6}$
Elizabeth, NJ	$2.53x10^{-5}$	$1.13x10^{-5}$
Galveston, TX	$3.02x10^{-5}$	$7.64x10^{-6}$
Jacksonville, FL	$2.15x10^{-5}$	$2.91x10^{-6}$
Newport News, VA	$1.81x10^{-5}$	$3.43x10^{-6}$
Norfolk, VA	$1.69x10^{-5}$	$2.88x10^{-6}$
Philadelphia, PA	$2.75x10^{-5}$	$7.60x10^{-6}$
Portsmouth, VA	$2.05x10^{-5}$	$2.67x10^{-6}$
Savannah, GA	$1.38x10^{-5}$	$2.93x10^{-6}$
MOTSU, NC	$1.56x10^{-5}$	$2.30x10^{-6}$
Wilmington, NC	$1.66x10^{-5}$	$2.25x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	$4.53x10^{-5}$	$7.45x10^{-6}$
Charleston, SC (Wando Terminal)	$4.95x10^{-5}$	$7.45x10^{-6}$
Elizabeth, NJ	$8.08x10^{-5}$	$3.60x10^{-5}$
Galveston, TX	$9.63x10^{-5}$	$2.44x10^{-5}$
Jacksonville, FL	$6.86x10^{-5}$	$9.30x10^{-6}$
Newport News, VA	$5.79x10^{-5}$	$1.10x10^{-5}$
Norfolk, VA	$5.38x10^{-5}$	$9.23x10^{-6}$
Philadelphia, PA	$8.77x10^{-5}$	$2.43x10^{-5}$
Portsmouth, VA	$6.53x10^{-5}$	$8.54x10^{-6}$
Savannah, GA	$4.41x10^{-5}$	$9.38x10^{-6}$
MOTSU, NC	$4.97x10^{-5}$	$7.36x10^{-6}$
Wilmington, NC	$5.31x10^{-5}$	$7.21x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	$8.32x10^{-5}$	$1.94x10^{-5}$
Charleston, SC (Wando Terminal)	$9.06x10^{-5}$	$1.94x10^{-5}$
Elizabeth, NJ	$1.53x10^{-4}$	$8.64x10^{-5}$

Shipments to Oak Ridge Reservation:		
Source/Route	***Truck***	***Rail***
Galveston, TX	1.75×10^{-4}	6.15×10^{-5}
Jacksonville, FL	1.26×10^{-4}	2.40×10^{-5}
Newport News, VA	1.09×10^{-4}	2.84×10^{-5}
Norfolk, VA	1.01×10^{-4}	2.43×10^{-5}
Philadelphia, PA	1.66×10^{-4}	5.90×10^{-5}
Portsmouth, VA	1.22×10^{-4}	2.26×10^{-5}
Savannah, GA	8.13×10^{-5}	2.43×10^{-5}
MOTSU, NC	9.18×10^{-5}	1.94×10^{-5}
Wilmington, NC	9.85×10^{-5}	1.90×10^{-5}
Calcined Target Material		
Charleston, SC (NWS)	4.14×10^{-3}	7.70×10^{-4}
Charleston, SC (Wando Terminal)	4.32×10^{-3}	7.70×10^{-4}
Elizabeth, NJ	8.03×10^{-3}	2.18×10^{-3}
Jacksonville, FL	5.87×10^{-3}	9.16×10^{-4}
Newport News, VA	5.46×10^{-3}	1.10×10^{-3}
Norfolk, VA	5.50×10^{-3}	9.94×10^{-4}
Philadelphia, PA	8.57×10^{-3}	1.59×10^{-3}
Portsmouth, VA	5.79×10^{-3}	9.45×10^{-4}
Savannah, GA	4.19×10^{-3}	9.55×10^{-4}
MOTSU, NC	4.78×10^{-3}	8.03×10^{-4}
Wilmington, NC	5.32×10^{-3}	7.89×10^{-4}
Galveston, TX	7.44×10^{-3}	2.10×10^{-3}
Oxidized Target Material		
Charleston, SC (NWS)	1.04×10^{-2}	1.93×10^{-3}
Charleston, SC (Wando Terminal)	1.08×10^{-2}	1.93×10^{-3}
Elizabeth, NJ	2.01×10^{-2}	5.46×10^{-3}
Jacksonville, FL	1.47×10^{-2}	2.30×10^{-3}
Newport News, VA	1.37×10^{-2}	2.76×10^{-3}
Norfolk, VA	1.38×10^{-2}	2.49×10^{-3}
Philadelphia, PA	2.15×10^{-2}	3.97×10^{-3}
Portsmouth, VA	1.45×10^{-2}	2.37×10^{-3}
Savannah, GA	1.05×10^{-2}	2.39×10^{-3}
MOTSU, NC	1.20×10^{-2}	2.01×10^{-3}
Wilmington, NC	1.33×10^{-2}	1.98×10^{-3}
Galveston, TX	1.86×10^{-2}	5.25×10^{-3}
From Western Ports		
BR-2 Belgium Spent Nuclear Fuel		
NWS Concord, CA	1.46×10^{-4}	1.56×10^{-5}
Long Beach, CA	1.50×10^{-4}	2.08×10^{-5}
Portland, OR	1.33×10^{-4}	2.40×10^{-5}
Tacoma, WA	1.20×10^{-4}	2.55×10^{-5}
RHF France Spent Nuclear Fuel		
NWS Concord, CA	6.33×10^{-5}	6.72×10^{-6}
Long Beach, CA	6.50×10^{-5}	8.97×10^{-6}
Portland, OR	5.75×10^{-5}	1.03×10^{-5}
Tacoma, WA	5.16×10^{-5}	1.10×10^{-5}
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	2.02×10^{-4}	2.17×10^{-5}
Long Beach, CA	2.07×10^{-4}	2.89×10^{-5}
Portland, OR	1.84×10^{-4}	3.35×10^{-5}
Tacoma, WA	1.65×10^{-4}	3.56×10^{-5}

Shipments to Oak Ridge Reservation:		
Source/Route	***Truck***	***Rail***
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	$3.81x10^{-4}$	$7.68x10^{-5}$
Long Beach, CA	$3.86x10^{-4}$	$9.42x10^{-5}$
Portland, OR	$3.85x10^{-4}$	$1.25x10^{-4}$
Tacoma, WA	$3.78x10^{-4}$	$1.30x10^{-4}$
Calcined Target Material		
NWS Concord, CA	$1.77x10^{-2}$	$5.82x10^{-3}$
Long Beach, CA	$1.60x10^{-2}$	$6.15x10^{-3}$
Portland, OR	$3.65x10^{-2}$	$8.50x10^{-3}$
Tacoma, WA	$4.74x10^{-2}$	$8.59x10^{-3}$
Oxidized Target Material		
NWS Concord, CA	$4.43x10^{-2}$	$1.46x10^{-2}$
Long Beach, CA	$4.01x10^{-2}$	$1.54x10^{-2}$
Portland, OR	$9.13x10^{-2}$	$2.13x10^{-2}$
Tacoma, WA	$1.19x10^{-1}$	$2.15x10^{-2}$
From DOE Sites/Canadian Border		
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	$6.57x10^{-5}$	$1.83x10^{-5}$
Hanford Site	$1.13x10^{-4}$	$1.16x10^{-5}$
Idaho National Engineering Laboratory	$1.06x10^{-4}$	$9.48x10^{-6}$
Nevada Test Site	$1.33x10^{-4}$	$1.09x10^{-5}$
Savannah River	$5.10x10^{-5}$	$3.67x10^{-6}$
Sweetgrass, MT	$1.21x10^{-4}$	$2.12x10^{-5}$
RHF France Spent Nuclear Fuel		
Alexandria Bay, NY	$2.84x10^{-5}$	$7.91x10^{-6}$
Hanford Site	$4.86x10^{-5}$	$4.97x10^{-6}$
Idaho National Engineering Laboratory	$4.58x10^{-5}$	$4.07x10^{-6}$
Nevada Test Site	$5.73x10^{-5}$	$4.68x10^{-6}$
Savannah River	$2.21x10^{-5}$	$1.59x10^{-6}$
Sweetgrass, MT	$5.22x10^{-5}$	$9.11x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	$9.07x10^{-5}$	$2.53x10^{-5}$
Hanford Site	$1.56x10^{-4}$	$1.62x10^{-5}$
Idaho National Engineering Laboratory	$1.46x10^{-4}$	$1.33x10^{-5}$
Nevada Test Site	$1.83x10^{-4}$	$1.52x10^{-5}$
Savannah River	$7.03x10^{-5}$	$5.08x10^{-6}$
Sweetgrass, MT	$1.67x10^{-4}$	$2.95x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
Alexandria Bay, NY	$1.77x10^{-4}$	$6.60x10^{-5}$
Hanford Site	$3.35x10^{-4}$	$7.04x10^{-5}$
Idaho National Engineering Laboratory	$3.16x10^{-4}$	$6.19x10^{-5}$
Nevada Test Site	$3.81x10^{-4}$	$6.60x10^{-5}$
Savannah River	$1.27x10^{-4}$	$1.31x10^{-5}$
Sweetgrass, MT	$3.76x10^{-4}$	$1.14x10^{-4}$
Calcined Target Material		
Alexandria Bay, NY	$1.04x10^{-2}$	$2.20x10^{-3}$
Hanford Site	$3.55x10^{-2}$	$6.78x10^{-3}$
Idaho National Engineering Laboratory	$3.38x10^{-2}$	$6.33x10^{-3}$
Nevada Test Site	$3.50x10^{-2}$	$6.35x10^{-3}$
Savannah River	$5.03x10^{-3}$	$5.03x10^{-4}$
Sweetgrass, MT	$4.51x10^{-2}$	$8.02x10^{-3}$

Shipments to Oak Ridge Reservation:		
Source/Route	***Truck***	***Rail***
Oxidized Target Material		
Alexandria Bay, NY	2.61×10^{-2}	5.51×10^{-3}
Hanford Site	8.88×10^{-2}	1.70×10^{-2}
Idaho National Engineering Laboratory	8.47×10^{-2}	1.58×10^{-2}
Nevada Test Site	8.75×10^{-2}	1.59×10^{-2}
Savannah River	1.26×10^{-2}	1.26×10^{-3}
Sweetgrass, MT	1.13×10^{-1}	2.01×10^{-2}

Shipments to Savannah River Site:		
Source/Route	***Truck***	***Rail***
	From Eastern Ports	
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	8.42×10^{-6}	8.97×10^{-7}
Charleston, SC (Wando Terminal)	1.14×10^{-5}	8.97×10^{-7}
Elizabeth, NJ	5.89×10^{-5}	2.50×10^{-5}
Galveston, TX	8.20×10^{-5}	2.19×10^{-5}
Jacksonville, FL	9.46×10^{-6}	1.65×10^{-6}
Newport News, VA	3.16×10^{-5}	5.04×10^{-6}
Norfolk, VA	2.55×10^{-5}	3.77×10^{-6}
Philadelphia, PA	6.56×10^{-5}	1.65×10^{-5}
Portsmouth, VA	2.59×10^{-5}	3.27×10^{-6}
Savannah, GA	7.13×10^{-6}	5.84×10^{-7}
MOTSU, NC	6.37×10^{-6}	2.43×10^{-6}
Wilmington, NC	1.36×10^{-5}	2.32×10^{-6}
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	3.64×10^{-6}	3.87×10^{-7}
Charleston, SC (Wando Terminal)	4.95×10^{-6}	3.87×10^{-7}
Elizabeth, NJ	2.54×10^{-5}	1.08×10^{-5}
Galveston, TX	3.55×10^{-5}	9.46×10^{-6}
Jacksonville, FL	4.09×10^{-6}	7.13×10^{-7}
Newport News, VA	1.37×10^{-5}	2.17×10^{-6}
Norfolk, VA	1.10×10^{-5}	1.63×10^{-6}
Philadelphia, PA	2.83×10^{-5}	7.13×10^{-6}
Portsmouth, VA	1.12×10^{-5}	1.41×10^{-6}
Savannah GA	3.08×10^{-6}	2.52×10^{-7}
MOTSU NC	2.75×10^{-6}	1.05×10^{-6}
Wilmington NC	5.90×10^{-6}	1.00×10^{-6}
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	1.16×10^{-5}	1.24×10^{-6}
Charleston, SC (Wando Terminal)	1.58×10^{-5}	1.24×10^{-6}
Elizabeth, NJ	8.12×10^{-5}	3.45×10^{-5}
Galveston, TX	1.13×10^{-4}	3.02×10^{-5}
Jacksonville, FL	1.31×10^{-5}	2.28×10^{-6}
Newport News, VA	4.36×10^{-5}	6.96×10^{-6}
Norfolk, VA	3.52×10^{-5}	5.21×10^{-6}
Philadelphia, PA	9.04×10^{-5}	2.28×10^{-5}
Portsmouth, VA	3.57×10^{-5}	4.52×10^{-6}
Savannah, GA	9.83×10^{-6}	8.07×10^{-7}
MOTSU, NC	8.79×10^{-6}	3.36×10^{-6}
Wilmington, NC	1.88×10^{-5}	3.21×10^{-6}
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	2.14×10^{-5}	3.19×10^{-6}

Shipments to Savannah River Site:		
Source/Route	***Truck***	***Rail***
Charleston, SC (Wando Terminal)	$2.89x10^{-5}$	$3.19x10^{-6}$
Elizabeth, NJ	$1.54x10^{-4}$	$8.32x10^{-5}$
Galveston, TX	$2.05x10^{-4}$	$7.50x10^{-5}$
Jacksonville, FL	$2.56x10^{-5}$	$6.32x10^{-6}$
Newport News, VA	$8.21x10^{-5}$	$1.79x10^{-5}$
Norfolk, VA	$6.71x10^{-5}$	$1.37x10^{-5}$
Philadelphia, PA	$1.70x10^{-4}$	$5.58x10^{-5}$
Portsmouth, VA	$6.80x10^{-5}$	$1.20x10^{-5}$
Savannah, GA	$1.87x10^{-5}$	$2.22x10^{-6}$
MOTSU, NC	$1.73x10^{-5}$	$8.90x10^{-6}$
Wilmington, NC	$3.58x10^{-5}$	$8.52x10^{-6}$
HLW Vitrified		
Charleston, SC (NWS)	$1.27x10^{-4}$	$1.65x10^{-5}$
Charleston, SC (Wando Terminal)	$1.71x10^{-4}$	$1.65x10^{-5}$
Elizabeth, NJ	$1.19x10^{-3}$	$3.16x10^{-4}$
Galveston, TX	$1.21x10^{-3}$	$3.45x10^{-4}$
Jacksonville, FL	$2.93x10^{-4}$	$4.13x10^{-5}$
Newport News, VA	$6.64x10^{-4}$	$9.55x10^{-5}$
Norfolk, VA	$6.23x10^{-4}$	$7.95x10^{-5}$
Philadelphia, PA	$1.25x10^{-3}$	$2.28x10^{-4}$
Portsmouth, VA	$6.25x10^{-4}$	$7.24x10^{-5}$
Savannah, GA	$1.84x10^{-4}$	$1.44x10^{-5}$
MOTSU, NC	$1.98x10^{-4}$	$5.24x10^{-5}$
Wilmington, NC	$3.43x10^{-4}$	$5.04x10^{-5}$
Calcined Target Material		
Charleston, SC (NWS)	$1.15x10^{-3}$	$1.16x 10^{-4}$
Charleston, SC (Wando Terminal)	$1.33x10^{-3}$	$1.16x10^{-4}$
Elizabeth, NJ	$8.28x10^{-3}$	$2.17x10^{-3}$
Jacksonville, FL	$2.09x10^{-3}$	$2.96x10^{-4}$
Newport News, VA	$4.69x10^{-3}$	$6.69x10^{-4}$
Norfolk, VA	$4.41x10^{-3}$	$5.60x10^{-4}$
Philadelphia, PA	$8.79x10^{-3}$	$1.58x10^{-3}$
Portsmouth, VA	$4.42x10^{-3}$	$5.12x10^{-4}$
Savannah, GA	$1.30x10^{-3}$	$1.03x10^{-4}$
MOTSU, NC	$1.42x10^{-3}$	$3.71x10^{-4}$
Wilmington, NC	$2.43x10^{-3}$	$3.57x10^{-4}$
Galveston, TX	$8.43x10^{-3}$	$2.40x10^{-3}$
Oxidized Target Material		
Charleston, SC (NWS)	$2.88x10^{-3}$	$2.90x10^{-4}$
Charleston, SC (Wando Terminal)	$3.33x10^{-3}$	$2.90x10^{-4}$
Elizabeth, NJ	$2.07x10^{-2}$	$5.44x10^{-3}$
Jacksonville, FL	$5.23x10^{-3}$	$7.40x10^{-4}$
Newport News, VA	$1.17x10^{-2}$	$1.68x10^{-3}$
Norfolk, VA	$1.10x10^{-2}$	$1.40x10^{-3}$
Philadelphia, PA	$2.20x10^{-2}$	$3.95x10^{-3}$
Portsmouth, VA	$1.11x10^{-2}$	$1.28x10^{-3}$
Savannah, GA	$3.25x10^{-3}$	$2.57x10^{-4}$
MOTSU, NC	$3.54x10^{-3}$	$9.28x10^{-4}$
Wilmington, NC	$6.08x10^{-3}$	$8.93x10^{-4}$
Galveston, TX	$2.11x10^{-2}$	$6.00x10^{-3}$

Shipments to Savannah River Site:		
Source/Route	***Truck***	***Rail***
	From Western Ports	
BR-2 Belgium Spent Nuclear Fuel		
NWS Concord, CA	$1.52x10^{-4}$	$3.67x10^{-5}$
Long Beach, CA	$1.37x10^{-4}$	$3.15x10^{-5}$
Portland, OR	$1.80x10^{-4}$	$3.14x10^{-5}$
Tacoma, WA	$1.68x10^{-4}$	$3.29x10^{-5}$
RHF France Spent Nuclear Fuel		
NWS Concord, CA	$6.56x10^{-5}$	$1.58x10^{-5}$
Long Beach, CA	$5.94x10^{-5}$	$1.36x10^{-5}$
Portland, OR	$7.76x10^{-5}$	$1.35x10^{-5}$
Tacoma, WA	$7.24x10^{-5}$	$1.42x10^{-5}$
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	$2.09x10^{-4}$	$5.07x10^{-5}$
Long Beach, CA	$1.89x10^{-4}$	$4.37x10^{-5}$
Portland, OR	$2.48x10^{-4}$	$4.37x10^{-5}$
Tacoma, WA	$2.31x10^{-4}$	$4.57x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	$3.90x10^{-4}$	$1.31x10^{-4}$
Long Beach, CA	$3.51x10^{-4}$	$1.46x10^{-4}$
Portland, OR	$5.00x10^{-4}$	$1.55x10^{-4}$
Tacoma, WA	$4.96x10^{-4}$	$1.59x10^{-4}$
Calcined Target Material		
NWS Concord, CA	$1.81x10^{-2}$	$4.78x10^{-3}$
Long Beach, CA	$1.61x10^{-2}$	$1.01x10^{-2}$
Portland, OR	$4.05x10^{-2}$	$1.02x10^{-2}$
Tacoma, WA	$5.15x10^{-2}$	$1.03x10^{-2}$
Oxidized Target Material		
NWS Concord, CA	$4.54x10^{-2}$	$1.20x10^{-2}$
Long Beach, CA	$4.03x10^{-2}$	$2.53x10^{-2}$
Portland, OR	$1.01x10^{-1}$	$2.54x10^{-2}$
Tacoma, WA	$1.29x10^{-1}$	$2.57x10^{-2}$
	From DOE Sites/Canadian Border	
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	$6.60x10^{-5}$	$4.27x10^{-5}$
Hanford Site	$1.61x10^{-4}$	$1.97x10^{-5}$
Idaho National Engineering Laboratory	$1.54x10^{-4}$	$1.76x10^{-5}$
Nevada Test Site	$1.81x10^{-4}$	$1.90x10^{-5}$
Oak Ridge Reservation	$5.10x10^{-5}$	$3.67x10^{-6}$
Sweetgrass, MT	$1.67x10^{-4}$	$2.85x10^{-5}$
RHF France Spent Nuclear Fuel		
Alexandria Bay, NY	$2.85x10^{-5}$	$1.85x10^{-5}$
Hanford Site	$6.94x10^{-5}$	$8.47x10^{-6}$
Idaho National Engineering Laboratory	$6.66x10^{-5}$	$7.57x10^{-6}$
Nevada Test Site	$7.81x10^{-5}$	$8.18x10^{-6}$
Oak Ridge Reservation	$2.21x10^{-5}$	$1.59x10^{-6}$
Sweetgrass, MT	$7.23x10^{-5}$	$1.23x10^{-5}$
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	$9.11x10^{-5}$	$5.90x10^{-5}$
Hanford Site	$2.22x10^{-4}$	$2.73x10^{-5}$
Idaho National Engineering Laboratory	$2.13x10^{-4}$	$2.44x10^{-5}$
Nevada Test Site	$2.49x10^{-4}$	$2.64x10^{-5}$

Shipments to Savannah River Site:		
Source/Route	***Truck***	***Rail***
Oak Ridge Reservation	7.03×10^{-5}	5.08×10^{-6}
Sweetgrass, MT	2.31×10^{-4}	3.97×10^{-5}
PRR-1 TRIGA Spent Nuclear Fuel		
Alexandria Bay, NY	1.78×10^{-4}	1.43×10^{-4}
Hanford Site	4.53×10^{-4}	9.49×10^{-5}
Idaho National Engineering Laboratory	4.34×10^{-4}	8.63×10^{-5}
Nevada Test Site	4.99×10^{-4}	9.04×10^{-5}
Oak Ridge Reservation	1.27×10^{-4}	1.31×10^{-5}
Sweetgrass, MT	4.90×10^{-4}	1.44×10^{-4}
Calcined Target Material		
Alexandria Bay, NY	1.07×10^{-2}	3.61×10^{-3}
Hanford Site	3.96×10^{-2}	7.11×10^{-3}
Idaho National Engineering Laboratory	3.79×10^{-2}	6.66×10^{-3}
Nevada Test Site	3.91×10^{-2}	6.68×10^{-3}
Oak Ridge Reservation	5.03×10^{-3}	5.03×10^{-4}
Sweetgrass, MT	4.91×10^{-2}	9.69×10^{-3}
Oxidized Target Material		
Alexandria Bay, NY	2.68×10^{-2}	9.05×10^{-3}
Hanford Site	9.90×10^{-2}	1.78×10^{-2}
Idaho National Engineering Laboratory	9.49×10^{-2}	1.67×10^{-2}
Nevada Test Site	9.77×10^{-2}	1.67×10^{-2}
Oak Ridge Reservation	1.26×10^{-2}	1.26×10^{-3}
Sweetgrass, MT	1.23×10^{-1}	2.42×10^{-2}

The nonradiological risk factors are presented in terms of mortalities per shipment in Table E-10. Separate risk factors are provided for mortalities resulting from hydrocarbon emissions and transportation accidents (fatalities resulting from mechanical impact).

Table E-10 Vehicle-Related (Nonradiological) Risk Factors per Shipment to Spent Nuclear Fuel Types (Fatalities/Shipment)

Shipments to Hanford Site:			
Mode		***Emission***	***Accident***
		From Eastern Ports	
Truck			
	Charleston, SC (NWS)	1.11×10^{-5}	2.00×10^{-4}
	Charleston, SC (Wando Terminal)	1.18×10^{-5}	2.01×10^{-4}
	Elizabeth, NJ	1.31×10^{-5}	1.66×10^{-4}
	Galveston, TX	1.69×10^{-5}	1.50×10^{-4}
	Jacksonville, FL	1.44×10^{-5}	1.95×10^{-4}
	Newport News, VA	1.66×10^{-5}	1.80×10^{-4}
	Norfolk, VA	1.64×10^{-5}	1.83×10^{-4}
	Philadelphia, PA	1.42×10^{-5}	1.65×10^{-4}
	Portsmouth, VA	1.82×10^{-5}	1.82×10^{-4}
	Savannah, GA	1.36×10^{-5}	1.86×10^{-4}
	Sunny Point, NC	1.17×10^{-5}	1.82×10^{-4}
	Wilmington, NC	1.09×10^{-5}	2.10×10^{-4}
Rail			
	Charleston, SC (NWS)	2.35×10^{-5}	6.40×10^{-6}
	Charleston, SC (Wando Terminal)	2.35×10^{-5}	6.40×10^{-6}
	Elizabeth, NJ	5.58×10^{-5}	6.30×10^{-6}
	Galveston, TX	9.83×10^{-6}	5.00×10^{-6}

Shipments to Hanford Site:			
	Mode	***Emission***	***Accident***
	Jacksonville, FL	$2.01x10^{-5}$	$6.42x10^{-6}$
	Newport News, VA	$3.49x10^{-5}$	$6.46x10^{-6}$
	Norfolk, VA	$3.56x10^{-5}$	$6.67x10^{-6}$
	Philadelphia, PA	$5.38x10^{-5}$	$6.20x10^{-6}$
	Portsmouth, VA	$3.46x10^{-5}$	$6.60x10^{-6}$
	Savannah, GA	$1.86x10^{-5}$	$6.47x10^{-6}$
	Sunny Point, NC	$2.07x10^{-5}$	$6.70x10^{-6}$
	Wilmington, NC	$2.07x10^{-5}$	$6.68x10^{-6}$
		From Western Ports	
Truck			
	Concord, CA	$7.18x10^{-6}$	$4.81x10^{-5}$
	Long Beach, CA	$2.08x10^{-5}$	$7.74x10^{-5}$
	Portland, OR	$2.67x10^{-6}$	$1.04x10^{-5}$
	Tacoma, WA	$3.06x10^{-6}$	$1.03x10^{-5}$
Rail			
	Concord, CA	$1.99x10^{-5}$	$1.99x10^{-6}$
	Long Beach, CA	$3.67x10^{-5}$	$3.32x10^{-6}$
	Portland, OR	$4.48x10^{-6}$	$5.00x10^{-7}$
	Tacoma, WA	$5.61x10^{-6}$	$7.82x10^{-7}$
		From DOE Sites/Canadian Border	
Truck			
	Alexandria Bay, NY	$1.46x10^{-5}$	$1.49x10^{-4}$
	Idaho National Engineering Laboratory	$2.19x10^{-6}$	$3.07x10^{-5}$
	Nevada Test Site	$9.50x10^{-6}$	$6.38x10^{-5}$
	Oak Ridge Reservation	$9.50x10^{-6}$	$1.61x10^{-4}$
	Savannah River	$1.31x10^{-5}$	$1.81x10^{-4}$
	Sweetgrass, MT	$1.74x10^{-6}$	$4.15x10^{-5}$
Rail			
	Alexandria Bay, NY	$4.59x10^{-5}$	$6.02x10^{-6}$
	Idaho National Engineering Laboratory	$3.98x10^{-6}$	$1.38x10^{-6}$
	Nevada Test Site	$5.90x10^{-6}$	$2.72x10^{-6}$
	Oak Ridge Reservation	$1.45x10^{-5}$	$5.44x10^{-6}$
	Savannah River	$2.21x10^{-5}$	$6.18x10^{-6}$
	Sweetgrass MT	$3.98x10^{-6}$	$1.27x10^{-6}$

Shipments to Idaho National Engineering Laboratory:			
	Mode	***Emission***	***Accident***
		From Eastern Ports	
Truck			
	Charleston, SC (NWS)	$1.03x10^{-5}$	$1.80x10^{-4}$
	Charleston, SC (Wando Terminal)	$1.09x10^{-5}$	$1.81x10^{-4}$
	Elizabeth, NJ	$1.17x10^{-5}$	$1.46x10^{-4}$
	Galveston, TX	$1.54x10^{-5}$	$1.30x10^{-4}$
	Jacksonville, FL	$1.22x10^{-5}$	$1.75x10^{-4}$
	Newport News, VA	$1.52x10^{-5}$	$1.60x10^{-4}$
	Norfolk, VA	$1.43x10^{-5}$	$1.63x10^{-4}$
	Philadelphia, PA	$1.28x10^{-5}$	$1.45x10^{-4}$
	Portsmouth, VA	$1.67x10^{-5}$	$1.62x10^{-4}$
	Savannah, GA	$1.22x10^{-5}$	$1.66x10^{-4}$
	Sunny Point, NC	$9.53x10^{-6}$	$1.59x10^{-4}$
	Wilmington, NC	$9.50x10^{-6}$	$1.89x10^{-4}$

Shipments to Idaho National Engineering Laboratory:			
	Mode	***Emission***	***Accident***
Rail			
	Charleston, SC (NWS)	$2.16x10^{-5}$	$5.26x10^{-6}$
	Charleston, SC (Wando Terminal)	$2.16x10^{-5}$	$5.26x10^{-6}$
	Elizabeth, NJ	$5.39x10^{-5}$	$5.15x10^{-6}$
	Galveston, TX	$7.91x10^{-6}$	$3.86x10^{-6}$
	Jacksonville, FL	$1.82x10^{-5}$	$5.28x10^{-6}$
	Newport News, VA	$3.30x10^{-5}$	$5.32x10^{-6}$
	Norfolk, VA	$3.37x10^{-5}$	$5.53x10^{-6}$
	Philadelphia, PA	$5.19x10^{-5}$	$5.05x10^{-6}$
	Portsmouth, VA	$3.26x10^{-5}$	$5.46x10^{-6}$
	Savannah, GA	$1.67x10^{-5}$	$5.33x10^{-6}$
	Sunny Point, NC	$1.88x10^{-5}$	$5.56x10^{-6}$
	Wilmington, NC	$1.88x10^{-5}$	$5.54x10^{-6}$
		From Western Ports	
Truck			
	Concord, CA	$9.40x10^{-6}$	$5.52x10^{-5}$
	Long Beach, CA	$2.55x10^{-5}$	$6.21x10^{-5}$
	Portland, OR	$3.93x10^{-6}$	$3.62x10^{-5}$
	Sweetgrass, MT	$7.08x10^{-7}$	$2.89x10^{-5}$
	Tacoma, WA	$4.28x10^{-6}$	$3.97x10^{-5}$
Rail			
	Concord, CA	$9.08x10^{-6}$	$1.91x10^{-6}$
	Long Beach, CA	$3.48x10^{-5}$	$2.18x10^{-6}$
	Portland, OR	$5.36x10^{-6}$	$1.64x10^{-6}$
	Sweetgrass, MT	$5.06x10^{-6}$	$2.58x10^{-6}$
	Tacoma, WA	$8.62x10^{-6}$	$1.96x10^{-6}$
		From DOE Sites/Canadian Border	
Truck			
	Alexandria Bay, NY	$1.32x10^{-5}$	$1.29x10^{-4}$
	Hanford Site	$2.19x10^{-6}$	$3.07x10^{-5}$
	Nevada Test Site	$8.08x10^{-6}$	$4.36x10^{-5}$
	Oak Ridge Reservation	$8.08x10^{-6}$	$1.41x10^{-4}$
	Savannah River	$1.17x10^{-5}$	$1.61x10^{-4}$
Rail			
	Alexandria Bay, NY	$4.40x10^{-5}$	$4.88x10^{-6}$
	Hanford Site	$3.98x10^{-6}$	$1.38x10^{-6}$
	Nevada Test Site	$3.98x10^{-6}$	$1.58x10^{-6}$
	Oak Ridge Reservation	$1.26x10^{-5}$	$4.30x10^{-6}$
	Savannah River	$2.01x10^{-5}$	$5.04x10^{-6}$

Shipments to Nevada Test Site:			
	Mode	***Emission***	***Accident***
		From Eastern Ports	
Truck			
	Charleston, SC (NWS)	$1.16x10^{-5}$	$1.92x10^{-4}$
	Charleston, SC (Wando Terminal)	$1.23x10^{-5}$	$1.94x10^{-4}$
	Elizabeth, NJ	$1.98x10^{-5}$	$1.87x10^{-4}$
	Galveston, TX	$1.90x10^{-5}$	$1.32x10^{-4}$
	Jacksonville, FL	$1.49x10^{-5}$	$1.88x10^{-4}$
	Newport News, VA	$1.79x10^{-5}$	$1.74x10^{-4}$
	Norfolk, VA	$1.70x10^{-5}$	$1.76x10^{-4}$
	Philadelphia, PA	$1.90x10^{-5}$	$1.85x10^{-4}$

Shipments to Nevada Test Site:			
	Mode	***Emission***	***Accident***
	Portsmouth, VA	1.94×10^{-5}	1.75×10^{-4}
	Savannah, GA	1.49×10^{-5}	1.79×10^{-4}
	Sunny Point, NC	1.57×10^{-5}	1.90×10^{-4}
	Wilmington, NC	1.22×10^{-5}	2.03×10^{-4}
Rail			
	Charleston, SC (NWS)	2.42×10^{-5}	6.16×10^{-6}
	Charleston, SC (Wando Terminal)	2.42×10^{-5}	6.16×10^{-6}
	Elizabeth, NJ	5.64×10^{-5}	6.06×10^{-6}
	Galveston, TX	6.44×10^{-6}	4.09×10^{-6}
	Jacksonville, FL	2.08×10^{-5}	6.18×10^{-6}
	Newport News, VA	3.56×10^{-5}	6.22×10^{-6}
	Norfolk, VA	3.63×10^{-5}	6.43×10^{-6}
	Philadelphia, PA	5.45×10^{-5}	5.96×10^{-6}
	Portsmouth, VA	3.52×10^{-5}	6.37×10^{-6}
	Savannah, GA	1.92×10^{-5}	6.23×10^{-6}
	Sunny Point, NC	2.14×10^{-5}	6.46×10^{-6}
	Wilmington, NC	2.14×10^{-5}	6.44×10^{-6}
	From Western Ports		
Truck			
	Concord, CA	1.59×10^{-5}	5.34×10^{-5}
	Long Beach, CA	2.06×10^{-5}	2.83×10^{-5}
	Portland, OR	1.20×10^{-5}	6.95×10^{-5}
	Tacoma, WA	1.16×10^{-5}	7.28×10^{-5}
Rail			
	Concord, CA	1.97×10^{-5}	1.78×10^{-6}
	Long Beach, CA	3.08×10^{-5}	1.01×10^{-6}
	Portland, OR	7.28×10^{-6}	2.99×10^{-6}
	Tacoma, WA	1.05×10^{-5}	3.30×10^{-6}
	From DOE Sites/Canadian Border		
Truck			
	Alexandria Bay, NY	1.62×10^{-5}	1.58×10^{-4}
	Hanford Site	9.50×10^{-6}	6.38×10^{-5}
	Idaho National Engineering Laboratory	8.08×10^{-6}	4.36×10^{-5}
	Oak Ridge Reservation	1.08×10^{-5}	1.54×10^{-4}
	Savannah River	1.43×10^{-5}	1.74×10^{-4}
	Sweetgrass, MT	9.59×10^{-6}	6.77×10^{-5}
Rail			
	Alexandria Bay, NY	4.66×10^{-5}	5.78×10^{-6}
	Hanford Site	5.90×10^{-6}	2.72×10^{-6}
	Idaho National Engineering Laboratory	3.98×10^{-6}	1.58×10^{-6}
	Oak Ridge Reservation	1.52×10^{-5}	5.20×10^{-6}
	Savannah River	2.27×10^{-5}	5.94×10^{-6}
	Sweetgrass, MT	7.03×10^{-6}	3.92×10^{-6}

Shipments to Oak Ridge Reservation:			
	Mode	***Emission***	***Accident***
	From Eastern Ports		
Truck			
	Charleston, SC (NWS)	1.06×10^{-6}	3.81×10^{-5}
	Charleston, SC (Wando Terminal)	1.74×10^{-6}	3.94×10^{-5}
	Elizabeth, NJ	4.89×10^{-6}	5.42×10^{-5}
	Galveston, TX	6.60×10^{-6}	7.19×10^{-5}

Shipments to Oak Ridge Reservation:			
	Mode	***Emission***	***Accident***
	Jacksonville, FL	$2.29x10^{-6}$	$4.23x10^{-5}$
	Newport News, VA	$4.67x10^{-6}$	$3.66x10^{-5}$
	Norfolk, VA	$2.32x10^{-6}$	$3.64x10^{-5}$
	Philadelphia, PA	$7.95x10^{-6}$	$4.81x10^{-5}$
	Portsmouth, VA	$6.21x10^{-6}$	$3.79x10^{-5}$
	Savannah, GA	$9.33x10^{-7}$	$4.18x10^{-5}$
	Sunny Point, NC	$1.38x10^{-6}$	$4.66x10^{-5}$
	Wilmington, NC	$1.42x10^{-6}$	$4.94x10^{-5}$
Rail			
	Charleston, SC (NWS)	$3.60x10^{-6}$	$1.22x10^{-6}$
	Charleston, SC (Wando Terminal)	$3.60x10^{-6}$	$1.22x10^{-6}$
	Elizabeth, NJ	$4.00x10^{-5}$	$1.64x10^{-6}$
	Galveston, TX	$1.46x10^{-5}$	$2.20x10^{-6}$
	Jacksonville, FL	$5.61x10^{-6}$	$1.18x10^{-6}$
	Newport News, VA	$6.44x10^{-6}$	$1.60x10^{-6}$
	Norfolk, VA	$4.48x10^{-6}$	$1.44x10^{-6}$
	Philadelphia, PA	$2.46x10^{-5}$	$1.47x10^{-6}$
	Portsmouth, VA	$3.47x10^{-6}$	$1.38x10^{-6}$
	Savannah, GA	$4.06x10^{-6}$	$1.23x10^{-6}$
	Sunny Point, NC	$3.35x10^{-6}$	$1.13x10^{-6}$
	Wilmington, NC	$3.35x10^{-6}$	$1.11x10^{-6}$
	From Western Ports		
Truck			
	Concord, CA	$2.31x10^{-5}$	$1.95x10^{-4}$
	Long Beach, CA	$2.78x10^{-5}$	$1.70x10^{-4}$
	Portland, OR	$1.25x10^{-5}$	$1.68x10^{-4}$
	Tacoma, WA	$8.56x10^{-6}$	$1.38x10^{-4}$
Rail			
	Concord, CA	$2.53x10^{-5}$	$5.88x10^{-6}$
	Long Beach, CA	$4.33x10^{-5}$	$5.59x10^{-6}$
	Portland, OR	$2.87x10^{-5}$	$5.91x10^{-6}$
	Tacoma, WA	$3.56x10^{-5}$	$5.93x10^{-6}$
	From DOE Sites/Canadian Border		
Truck			
	Alexandria Bay, NY	$1.96x10^{-6}$	$6.31x10^{-5}$
	Hanford Site	$9.50x10^{-6}$	$1.61x10^{-4}$
	Idaho National Engineering Laboratory	$8.08x10^{-6}$	$1.41x10^{-4}$
	Nevada Test Site	$1.08x10^{-5}$	$1.54x10^{-4}$
	Savannah River	$2.96x10^{-6}$	$2.92x10^{-5}$
	Sweetgrass, MT	$6.98x10^{-6}$	$1.24x10^{-4}$
Rail			
	Alexandria Bay, NY	$2.78x10^{-5}$	$2.03x10^{-6}$
	Hanford Site	$1.45x10^{-5}$	$5.44x10^{-6}$
	Idaho National Engineering Laboratory	$1.26x10^{-5}$	$4.30x10^{-6}$
	Nevada Test Site	$1.52x10^{-5}$	$5.20x10^{-6}$
	Savannah River	$2.51x10^{-6}$	$8.72x 10^{-7}$
	Sweetgrass, MT	$2.31x10^{-5}$	$4.39x10^{-6}$

Shipments to Savannah River Site:			
	Mode	***Emission***	***Accident***
		From Eastern Ports	
Truck			
	Charleston, SC (NWS)	$5.15x10^{-7}$	$1.61x10^{-5}$
	Charleston, SC (Wando Terminal)	$1.19x10^{-6}$	$1.74x10^{-5}$
	Elizabeth, NJ	$5.47x10^{-6}$	$6.58x10^{-5}$
	Galveston, TX	$8.11x10^{-6}$	$7.15x10^{-5}$
	Jacksonville, FL	$3.22x10^{-8}$	$2.94x10^{-5}$
	Newport News, VA	$3.54x10^{-6}$	$4.57x10^{-5}$
	Norfolk, VA	$1.58x10^{-6}$	$4.45x10^{-5}$
	Philadelphia, PA	$9.37x10^{-6}$	$6.30x10^{-5}$
	Portsmouth, VA	$1.83x10^{-6}$	$4.47x10^{-5}$
	Savannah, GA	$3.22x10^{-8}$	$2.10x10^{-5}$
	Sunny Point, NC	$2.57x10^{-7}$	$2.17x10^{-5}$
	Wilmington, NC	$5.15x10^{-7}$	$2.87x10^{-5}$
Rail			
	Charleston, SC (NWS)	$1.46x10^{-6}$	$2.93x10^{-7}$
	Charleston, SC (Wando Terminal)	$1.46x10^{-6}$	$2.93x10^{-7}$
	Elizabeth, NJ	$3.92x10^{-5}$	$1.82x10^{-6}$
	Galveston, TX	$2.06x10^{-5}$	$2.46x10^{-6}$
	Jacksonville, FL	$2.80x10^{-6}$	$5.42x10^{-7}$
	Newport News, VA	$5.61x10^{-6}$	$1.26x10^{-6}$
	Norfolk, VA	$3.64x10^{-6}$	$1.11x10^{-6}$
	Philadelphia, PA	$2.39x10^{-5}$	$1.65x10^{-6}$
	Portsmouth, VA	$2.64x10^{-6}$	$1.04x10^{-6}$
	Savannah, GA	$5.86x10^{-7}$	$2.38x10^{-7}$
	Sunny Point, NC	$2.55x10^{-6}$	$7.99x10^{-7}$
	Wilmington, NC	$2.55x10^{-6}$	$7.80x10^{-7}$
		From Western Ports	
Truck			
	Concord, CA	$2.99x10^{-5}$	$1.96x10^{-4}$
	Long Beach, CA	$2.57x10^{-5}$	$1.68x10^{-4}$
	Portland, OR	$1.60x10^{-5}$	$1.88x10^{-4}$
	Tacoma, WA	$1.21x10^{-5}$	$1.59x10^{-4}$
Rail			
	Concord, CA	$4.80x10^{-5}$	$6.66x10^{-6}$
	Long Beach, CA	$5.07x10^{-5}$	$6.78x10^{-6}$
	Portland, OR	$3.44x10^{-5}$	$6.60x10^{-6}$
	Tacoma, WA	$4.13x10^{-5}$	$6.62x10^{-6}$
		From DOE Sites/Canadian Border	
Truck			
	Alexandria Bay, NY	$2.54x10^{-6}$	$7.47x10^{-5}$
	Hanford Site	$1.31x10^{-5}$	$1.81x10^{-4}$
	Idaho National Engineering Laboratory	$1.17x10^{-5}$	$1.61x10^{-4}$
	Nevada Test Site	$1.43x10^{-5}$	$1.74x10^{-4}$
	Oak Ridge Reservation	$2.96x10^{-6}$	$2.92x10^{-5}$
	Sweetgrass, MT	$1.05x10^{-5}$	$1.43x10^{-4}$
Rail			
	Alexandria Bay, NY	$5.76x10^{-5}$	$2.68x10^{-6}$
	Hanford Site	$2.21x10^{-5}$	$6.18x10^{-6}$
	Idaho National Engineering Laboratory	$2.01x10^{-5}$	$5.04x10^{-6}$
	Nevada Test Site	$2.27x10^{-5}$	$5.94x10^{-6}$

Shipments to Savannah River Site:		
Mode	*Emission*	*Accident*
Oak Ridge Reservation	2.51×10^{-6}	8.72×10^{-7}
Sweetgrass, MT	2.87×10^{-5}	5.07×10^{-6}

The total risks for any alternative or option can be calculated by multiplying the number of foreign research reactor spent nuclear fuel shipments by the per-shipment risk factors provided in Tables E-8 through E-10.

E.7.1.2 Characterization of Shipment Risks

The results of the per-shipment analysis are shown in Tables E-8 through E-10. From these tables, it is clear that the incident-free dose would be much higher than the accident dose for each of the fuel types. The accident doses are based on realistic, yet conservative fuel loadings. Since most of the public dose would be from incident-free exposure, it is not overly conservative to assume, for assessment purposes, that all spent nuclear fuel can be represented by the fuel type with the highest risk factors for the remainder of the transportation analysis.

E.7.2 Evaluation of the Basic Implementation

The following sections describe the evaluation of the basic implementation of the Management Alternative 1 of the proposed action. The evaluation of the management and implementation alternatives are described in Section E.8.

E.7.2.1 Shipments

Under all SNF&INEL Final EIS (DOE, 1995) alternatives, the shipment of foreign research reactor spent nuclear fuel would require the movement of 837 casks from points of entry (marine ports and Canadian border crossings) to DOE facilities. The basic assumption used in determining the number of shipments is that spent nuclear fuel from countries bordering the Atlantic Ocean and Mediterranean Sea was assumed to arrive on the east coast of the United States, and spent nuclear fuel from countries bordering the Indian and Pacific Oceans was assumed to arrive on the west coast. This is conservative from an overland transportation standpoint, because, as shown in Tables E-8 through E-10, shipment to the coast nearest the management site would reduce the risk factors for the overland shipment. Additionally, this assumption is considered to be realistic because the long shipping times required to ship from the Pacific Ocean to east coast ports and from the Atlantic Ocean to west coast ports, would be costly in terms of shipping, and would tie up the world's already short supply of casks. The foreign research reactor spent nuclear fuel could arrive at any port that meets the criteria identified in Appendix D, and would be likely to arrive at a variety of these ports. The basic shipment count, by point of origin is:

	East Coast		*West Coast*		*Totals*
	Aluminum	*TRIGA*	*Aluminum*	*TRIGA*	
Phase 1	419	82	101	42	644
Phase 2	125	25	30	13	193
Totals	544	107	131	55	837

Several of the SNF&INEL Final EIS (DOE, 1995) alternatives involve consolidation of all spent nuclear fuel to Idaho National Engineering Laboratory and/or Savannah River Site and, therefore, are single-phase programs that would require no additional shipments. However, many of the possible options require the use of Hanford Site, Nevada Test Site and/or Oak Ridge Reservation; and, thus, would require intersite shipments. The number of intersite shipments is calculated based on the assumption that the equivalent of

10 seagoing foreign research reactor casks would fit into a single rail cask that would travel between DOE sites. Similarly, it is assumed that the contents of four foreign research reactor casks would fit into a single truck cask for intersite shipment. This is based on the distribution of cask capacities described in Appendix B. As described in Appendix B, there is considerable uncertainty in what storage mode would be used at the Phase 1 site, and therefore in what form the fuel would be for intersite shipment. Additionally, it is not clear what casks would be licensed and available when the intersite shipments would begin (approximately 2006). Therefore, these assumptions, which are neither definitely conservative nor nonconservative, are considered to be reasonable and realistic.

The number of intersite shipments for SNF&INEL Final EIS (DOE, 1995) alternatives that would require two-phased approaches varies between 13 and 161. The variation is caused by the large number of unique combinations of Phase 1 and Phase 2 approaches depending on the specific management sites selected. Additionally, the variation is affected by the assumption that larger truck and rail casks would be used for intersite shipments. The actual numbers of shipments are shown in Tables E-1 and E-2.

E.7.2.2 Evaluation Using Risk Factors

Since the fuel would actually arrive at a variety of ports, average shipment risk factors were calculated for east coast ports to each DOE site, and an average shipment risk factor for west coast ports to each DOE site. This approach does not require that a specific port be selected for analysis purposes. It instead models the average affect the foreign research reactor spent nuclear fuel acceptance policy might actually have on the public. This approach is conservative since the dose rates and curie content of the fuel used for the analyses were selected to be conservative, but as realistic as possible, since it is impossible to predict the distribution of shipments among the capable ports.

The upper and lower bound risk estimates for the foreign research reactor spent nuclear fuel policy were also calculated. The upper bound assumes that DOE chooses the acceptable port with the highest per-shipment risk factors for all shipments, and the lower bound risk estimates assume that DOE chooses the acceptable port with the lowest per-shipment risk factors. In general, the highest risk factors result from the longest shipments, and the smallest risk factors from the shortest shipments.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.013 to 0.30 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crews.

The range of fatality estimates is caused by three factors: 1) the option of using truck or rail to transport spent nuclear fuel, 2) combinations of Phase 1 and Phase 2 sites that create varying shipment numbers and distances, and 3) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.006 to 0.071. The shipment by truck would yield higher crew exposures than the shipment by rail since the truck drivers would tend to sit closer to the cask than engineers. Doses to inspectors, security guards, and rail switchyard workers are also considered.

Truck and rail crew members are not radiation workers and, therefore, are not allowed to exceed a dosage of 100 mrem per yr. The regulatory limit for dose rate in occupied areas of the truck or train is 2 mrem per hr. Since a cross-country trip can take just over 50 hr of driving, if the radiation levels were at the

maximum allowed, a driver could exceed his or her annual limit. Therefore, DOE would implement administrative controls beyond those required by Federal regulations to ensure that vehicle operators would not exceed their annual dose limits.

The public would be exposed to a small amount of radiation emanating from the cask, and also to pollutants associated with the diesel exhaust. The estimated number of radiation-related LCFs for the general population ranged from 0.007 to 0.22, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.05. The fact that all these risk numbers are less than one means that the basic implementation would be unlikely to increase the total number of individuals that die of cancer in the United States (there are approximately 300,000 cancer deaths per yr in the United States) by a single fatality.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000004 to 0.00028 LCF from radiation and from 0.001 to 0.14 for traffic fatality, depending on the transportation mode and DOE sites selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation. These risks, especially in the case of radiological accident risks, are much lower than those for incident-free transportation. The risk estimates are probabilistic, which means that they take the probability of an accident's occurring and the consequences of these accidents into account. The risk estimates indicate that the likelihood of a death or an injury from a vehicle accident not involving radiation or radioactive release would be much higher than a death from a radiation-related accident. Both indicate an expectation of less than one fatality.

The impacts of overland transportation for all alternatives and options are shown in Tables E-11 through E-19. As shown in Tables E-1 and E-2, there are 35 distinct approaches to the basic implementation of Management Alternative 1 of the proposed action. These 35 approaches are all the Phase 1/Phase 2 combinations allowed by the SNF&INEL Final EIS (DOE, 1995). Each of these 35 approaches is evaluated for three different transportation mode assumptions: 1) all shipments are on trucks, 2) that shipments from ports to sites are on trucks and intersite shipments are on rail, and 3) that all shipments are on rail. The transportation mode assumptions of all by truck and all by rail are analyzed to bound the risks of any combination of transportation modes. The third mode assumption is provided as an example of a realistic approach. Each distinct approach and mode assumption is evaluated using the average, upper bound, and lower bound risk factors.

These tables are designed to provide risk estimate factors for all expected implementation alternatives. For example, if the SNF&INEL Final EIS alternative selected is Centralization to Nevada Test Site, "Centralization" should be in the first column of each table, and "Nevada Test Site" in the second column of each table. The Phase 1 approaches available are listed in the third column. The decision as to which of the possible Phase I approaches would be used will be part of the foreign research reactor spent nuclear fuel policy described in this EIS. The risk estimates for the foreign research reactor spent nuclear fuel EIS policy are given for "Geographic" distribution of spent nuclear fuel during Phase 1 (to Idaho National Engineering Laboratory and Savannah River Site), for "By Fuel" distribution of spent nuclear fuel during Phase 1 (TRIGA to Idaho National Engineering Laboratory and aluminum-based to Savannah River Site), for "All to Idaho National Engineering Laboratory" during Phase 1, and for "All to Savannah River Site" during Phase 1. The risks, expressed in LCF and traffic accident fatalities are provided. These risk estimates include Phase 1 port-to-site shipments (Savannah River Site and/or Idaho National Engineering Laboratory), intersite shipments to, in this case, Nevada Test Site, and Phase 2 port-to-Nevada Test Site

Table E-11 Tabulation of Overland Transportation Risks: Basic Implementation. All Shipments via Truck, Average Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.019	0.056	0.002	0.035	0.00002
1992/1993 Planning Basis	INEL/SRS		0.019	0.056	0.002	0.035	0.00002
Regionalization by Fuel Type	INEL/SRS		0.036	0.111	0.005	0.067	0.00004
Regionalization	INEL/SRS		0.019	0.056	0.002	0.035	0.00002
by	INEL/ORR	Geographic	0.022	0.063	0.003	0.040	0.00005
Geography		By Fuel	0.035	0.105	0.005	0.064	0.00007
		All to INEL	0.052	0.165	0.008	0.095	0.00006
	NTS/SRS	Geographic	0.020	0.060	0.003	0.038	0.00003
		By Fuel	0.033	0.102	0.005	0.062	0.00004
		All to SRS	0.030	0.091	0.005	0.056	0.00003
	NTS/ORR	Geographic	0.023	0.068	0.003	0.042	0.00006
		By Fuel	0.036	0.110	0.006	0.067	0.00008
		All to INEL	0.057	0.179	0.009	0.103	0.00009
		All to SRS	0.033	0.100	0.005	0.061	0.00007
	HS/SRS	Geographic	0.019	0.056	0.002	0.035	0.00002
		By Fuel	0.032	0.098	0.005	0.060	0.00004
		All to SRS	0.029	0.087	0.004	0.054	0.00003
	HS/ORR	Geographic	0.022	0.064	0.003	0.040	0.00005
		By Fuel	0.035	0.106	0.005	0.065	0.00007
		All to INEL	0.055	0.173	0.008	0.099	0.00007
		All to SRS	0.032	0.096	0.005	0.059	0.00007
Centralization	INEL		0.062	0.195	0.009	0.112	0.00007
	SRS		0.033	0.097	0.005	0.061	0.00003
	HS	Geographic	0.043	0.134	0.006	0.079	0.00015
		By Fuel	0.057	0.177	0.008	0.104	0.00017
		All to INEL	0.066	0.211	0.010	0.119	0.00008
		All to SRS	0.056	0.176	0.008	0.104	0.00019
	NTS	Geographic	0.042	0.127	0.007	0.080	0.00017
		By Fuel	0.055	0.170	0.009	0.105	0.00019
		All to INEL	0.067	0.212	0.011	0.123	0.00011
		All to SRS	0.054	0.167	0.009	0.104	0.00021
	ORR	Geographic	0.027	0.080	0.003	0.050	0.00008
		By Fuel	0.040	0.121	0.006	0.074	0.00009
		All to INEL	0.066	0.210	0.010	0.123	0.00016
		All to SRS	0.036	0.107	0.005	0.066	0.00008

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-12 Tabulation of Overland Transportation Risks: Basic Implementation, Shipments from Ports via Truck, Intersite Shipments via Rail, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.020	0.058	0.003	0.036	0.00002
		By Fuel	0.033	0.100	0.005	0.061	0.00004
		All to INEL	0.052	0.165	0.008	0.095	0.00006
	NTS/SRS	Geographic	0.019	0.057	0.003	0.036	0.00002
		By Fuel	0.033	0.100	0.005	0.061	0.00003
		All to SRS	0.030	0.091	0.005	0.056	0.00003
	NTS/ORR	Geographic	0.020	0.059	0.003	0.037	0.00002
		By Fuel	0.034	0.102	0.005	0.062	0.00004
		All to INEL	0.054	0.166	0.008	0.096	0.00006
		All to SRS	0.031	0.093	0.005	0.057	0.00004
	HS/SRS	Geographic	0.018	0.054	0.002	0.034	0.00002
		By Fuel	0.032	0.096	0.005	0.059	0.00003
		All to SRS	0.029	0.087	0.004	0.054	0.00003
	HS/ORR	Geographic	0.019	0.056	0.003	0.035	0.00002
		By Fuel	0.033	0.098	0.005	0.060	0.00004
		All to INEL	0.053	0.163	0.008	0.095	0.00006
		All to SRS	0.030	0.089	0.004	0.055	0.00004
Centralization	INEL						
	SRS						
	HS	Geographic	0.032	0.094	0.005	0.055	0.00005
		By Fuel	0.045	0.136	0.008	0.080	0.00007
		All to INEL	0.064	0.200	0.010	0.114	0.00007
		All to SRS	0.042	0.128	0.007	0.075	0.00007
	NTS	Geographic	0.031	0.093	0.006	0.057	0.00006
		By Fuel	0.044	0.135	0.008	0.081	0.00007
		All to INEL	0.063	0.199	0.010	0.116	0.00007
		All to SRS	0.042	0.126	0.008	0.076	0.00008
	ORR	Geographic	0.023	0.067	0.003	0.041	0.00003
		By Fuel	0.036	0.109	0.005	0.066	0.00005
		All to INEL	0.056	0.174	0.009	0.101	0.00008
		All to SRS	0.033	0.100	0.005	0.061	0.00004

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-13 Tabulation of Overland Transportation Risks: Basic Implementation. All Shipments via Rail, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.008	0.009	0.009	0.001	0.00001
1992/1993 Planning Basis	INEL/SRS		0.008	0.009	0.009	0.001	0.00001
Regionalization by Fuel Type	INEL/SRS		0.011	0.014	0.018	0.002	0.00001
Regionalization by Geography	INEL/SRS		0.008	0.009	0.009	0.001	0.00001
	INEL/ORR	Geographic	0.009	0.013	0.010	0.003	0.00001
		By Fuel	0.012	0.016	0.015	0.004	0.00001
		All to INEL	0.015	0.018	0.019	0.005	0.00001
	NTS/SRS	Geographic	0.009	0.014	0.011	0.004	0.00001
		By Fuel	0.012	0.018	0.016	0.005	0.00001
		All to SRS	0.011	0.018	0.016	0.005	0.00001
	NTS/ORR	Geographic	0.011	0.020	0.011	0.008	0.00004
		By Fuel	0.013	0.020	0.016	0.006	0.00002
		All to INEL	0.020	0.033	0.021	0.013	0.00005
		All to SRS	0.012	0.018	0.015	0.005	0.00002
	HS/SRS	Geographic	0.008	0.011	0.011	0.002	0.00001
		By Fuel	0.011	0.015	0.016	0.003	0.00001
		All to SRS	0.010	0.014	0.015	0.003	0.00001
	HS/ORR	Geographic	0.010	0.016	0.010	0.006	0.00004
		By Fuel	0.012	0.017	0.015	0.004	0.00002
		All to INEL	0.018	0.027	0.019	0.009	0.00002
		All to SRS	0.011	0.014	0.015	0.003	0.00001
Centralization	INEL		0.016	0.016	0.023	0.004	0.00002
	SRS		0.011	0.013	0.017	0.002	0.00001
	HS	Geographic	0.023	0.052	0.015	0.026	0.00012
		By Fuel	0.015	0.020	0.020	0.005	0.00004
		All to INEL	0.020	0.030	0.023	0.010	0.00003
		All to SRS	0.014	0.018	0.020	0.004	0.00004
	NTS	Geographic	0.022	0.049	0.016	0.027	0.00014
		By Fuel	0.016	0.024	0.021	0.007	0.00004
		All to INEL	0.022	0.036	0.024	0.014	0.00005
		All to SRS	0.015	0.021	0.020	0.006	0.00004
	ORR	Geographic	0.014	0.027	0.011	0.012	0.00005
		By Fuel	0.016	0.032	0.016	0.014	0.00004
		All to INEL	0.029	0.064	0.021	0.033	0.00012
		All to SRS	0.014	0.025	0.015	0.009	0.00002

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-14 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments via Truck, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radiological
Decentralization	INEL/SRS		0.013	0.040	0.001	0.024	0.000007
1992/1993 Planning Basis	INEL/SRS		0.013	0.040	0.001	0.024	0.000007
Regionalization by Fuel Type	INEL/SRS		0.030	0.093	0.003	0.052	0.000012
Regionalization by Geography	INEL/SRS		0.013	0.040	0.001	0.024	0.000007
	INEL/ORR	Geographic	0.017	0.049	0.002	0.030	0.000039
		By Fuel	0.029	0.089	0.003	0.052	0.000045
		All to INEL	0.044	0.138	0.006	0.078	0.000020
	NTS/SRS	Geographic	0.014	0.043	0.002	0.026	0.000015
		By Fuel	0.027	0.084	0.003	0.048	0.000018
		All to SRS	0.025	0.075	0.003	0.042	0.000011
	NTS/ORR	Geographic	0.018	0.052	0.002	0.032	0.000048
		By Fuel	0.030	0.092	0.004	0.054	0.000052
		All to INEL	0.048	0.151	0.008	0.086	0.000055
		All to SRS	0.028	0.085	0.003	0.049	0.000053
	HS/SRS	Geographic	0.013	0.040	0.001	0.024	0.000009
		By Fuel	0.026	0.081	0.003	0.046	0.000013
		All to SRS	0.024	0.072	0.002	0.040	0.000011
	HS/ORR	Geographic	0.017	0.049	0.002	0.030	0.000041
		By Fuel	0.029	0.089	0.003	0.052	0.000047
		All to INEL	0.046	0.146	0.006	0.082	0.000029
		All to SRS	0.028	0.082	0.003	0.047	0.000052
Centralization	INEL		0.051	0.163	0.007	0.091	0.000023
	SRS		0.028	0.085	0.003	0.047	0.000012
	HS	Geographic	0.036	0.114	0.004	0.065	0.000127
		By Fuel	0.050	0.156	0.006	0.088	0.000135
		All to INEL	0.056	0.179	0.008	0.098	0.000032
		All to SRS	0.050	0.157	0.006	0.088	0.000161
	NTS	Geographic	0.034	0.105	0.005	0.065	0.000145
		By Fuel	0.047	0.146	0.007	0.087	0.000152
		All to INEL	0.056	0.177	0.009	0.100	0.000059
		All to SRS	0.047	0.146	0.007	0.087	0.000176
	ORR	Geographic	0.022	0.066	0.002	0.039	0.000063
		By Fuel	0.034	0.105	0.004	0.060	0.000065
		All to INEL	0.058	0.184	0.007	0.105	0.000122
		All to SRS	0.031	0.094	0.003	0.053	0.000053

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-15 Tabulation of Overland Transportation Risks: Basic Implementation, Shipments from Ports via Truck, Intersite Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.015	0.043	0.001	0.026	0.000011
		By Fuel	0.028	0.084	0.003	0.048	0.000015
		All to INEL	0.044	0.138	0.006	0.078	0.000020
	NTS/SRS	Geographic	0.014	0.040	0.001	0.025	0.000008
		By Fuel	0.027	0.081	0.003	0.047	0.000012
		All to SRS	0.025	0.075	0.003	0.042	0.000011
	NTS/ORR	Geographic	0.015	0.043	0.002	0.027	0.000012
		By Fuel	0.028	0.084	0.003	0.049	0.000016
		All to INEL	0.044	0.138	0.007	0.079	0.000023
		All to SRS	0.026	0.078	0.003	0.045	0.000016
	HS/SRS	Geographic	0.013	0.038	0.001	0.023	0.000008
		By Fuel	0.026	0.079	0.003	0.045	0.000012
		All to SRS	0.024	0.072	0.002	0.040	0.000011
	HS/ORR	Geographic	0.014	0.041	0.001	0.025	0.000012
		By Fuel	0.027	0.082	0.003	0.047	0.000016
		All to INEL	0.043	0.136	0.006	0.077	0.000023
		All to SRS	0.025	0.075	0.002	0.043	0.000016
Centralization	INEL						
	SRS						
	HS	Geographic	0.025	0.074	0.004	0.042	0.000036
		By Fuel	0.038	0.115	0.005	0.063	0.000040
		All to INEL	0.053	0.168	0.008	0.093	0.000027
		All to SRS	0.036	0.109	0.005	0.059	0.000046
	NTS	Geographic	0.024	0.070	0.004	0.042	0.000035
		By Fuel	0.037	0.111	0.006	0.063	0.000040
		All to INEL	0.052	0.164	0.008	0.093	0.000026
		All to SRS	0.035	0.105	0.006	0.059	0.000045
	ORR	Geographic	0.018	0.052	0.002	0.031	0.000016
		By Fuel	0.031	0.093	0.003	0.052	0.000020
		All to INEL	0.048	0.148	0.007	0.083	0.000041
		All to SRS	0.029	0.087	0.003	0.048	0.000017

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-16 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.006	0.007	0.008	0.001	0.000004
1992/1993 Planning Basis	INEL/SRS		0.006	0.007	0.008	0.001	0.000004
Regionalization by Fuel Type	INEL/SRS		0.010	0.011	0.014	0.002	0.000005
Regionalization	INEL/SRS		0.006	0.007	0.008	0.001	0.000004
by	INEL/ORR	Geographic	0.008	0.010	0.008	0.002	0.000007
Geography		By Fuel	0.010	0.013	0.012	0.003	0.000008
		All to INEL	0.013	0.013	0.013	0.004	0.000004
	NTS/SRS	Geographic	0.007	0.010	0.008	0.003	0.000005
		By Fuel	0.010	0.013	0.013	0.004	0.000005
		All to SRS	0.009	0.014	0.012	0.004	0.000005
	NTS/ORR	Geographic	0.010	0.016	0.008	0.007	0.000036
		By Fuel	0.011	0.016	0.012	0.005	0.000015
		All to INEL	0.017	0.027	0.015	0.012	0.000039
		All to SRS	0.010	0.014	0.012	0.004	0.000009
	HS/SRS	Geographic	0.006	0.008	0.008	0.001	0.000005
		By Fuel	0.009	0.011	0.012	0.002	0.000005
		All to SRS	0.009	0.011	0.012	0.002	0.000005
	HS/ORR	Geographic	0.009	0.013	0.008	0.005	0.000036
		By Fuel	0.010	0.013	0.012	0.003	0.000009
		All to INEL	0.015	0.022	0.013	0.008	0.000013
		All to SRS	0.009	0.011	0.012	0.002	0.000008
Centralization	INEL		0.013	0.011	0.015	0.003	0.000004
	SRS		0.009	0.011	0.013	0.002	0.000005
	HS	Geographic	0.021	0.048	0.011	0.025	0.000119
		By Fuel	0.013	0.015	0.015	0.004	0.000031
		All to INEL	0.017	0.023	0.016	0.008	0.000013
		All to SRS	0.012	0.014	0.015	0.003	0.000036
	NTS	Geographic	0.020	0.044	0.012	0.025	0.000131
		By Fuel	0.013	0.018	0.016	0.006	0.000035
		All to INEL	0.019	0.028	0.017	0.012	0.000039
		All to SRS	0.013	0.016	0.016	0.005	0.000034
	ORR	Geographic	0.012	0.025	0.008	0.010	0.000041
		By Fuel	0.015	0.029	0.012	0.012	0.000028
		All to INEL	0.027	0.059	0.014	0.031	0.000106
		All to SRS	0.013	0.023	0.012	0.007	0.000009

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-17 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments via Truck, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Traffic	Radio-logical
Decentralization	INEL/SRS		0.033	0.096	0.006	0.057	0.00007
1992/1993 Planning Basis	INEL/SRS		0.033	0.096	0.006	0.057	0.00007
Regionalization by Fuel Type	INEL/SRS		0.048	0.143	0.010	0.088	0.00011
Regionalization by Geography	INEL/SRS		0.033	0.096	0.006	0.057	0.00007
	INEL/ORR	Geographic	0.035	0.101	0.007	0.061	0.00010
		By Fuel	0.046	0.137	0.009	0.085	0.00013
		All to INEL	0.057	0.179	0.011	0.110	0.00012
	NTS/SRS	Geographic	0.034	0.101	0.007	0.060	0.00008
		By Fuel	0.046	0.137	0.010	0.083	0.00011
		All to SRS	0.044	0.129	0.010	0.078	0.00010
	NTS/ORR	Geographic	0.036	0.105	0.007	0.063	0.00011
		By Fuel	0.048	0.142	0.010	0.087	0.00014
		All to INEL	0.062	0.194	0.012	0.118	0.00016
		All to SRS	0.046	0.136	0.010	0.083	0.00014
	HS/SRS	Geographic	0.034	0.098	0.006	0.058	0.00007
		By Fuel	0.045	0.134	0.009	0.082	0.00010
		All to SRS	0.043	0.127	0.009	0.077	0.00010
	HS/ORR	Geographic	0.035	0.103	0.007	0.061	0.00010
		By Fuel	0.047	0.139	0.009	0.085	0.00013
		All to INEL	0.060	0.189	0.011	0.115	0.00013
		All to SRS	0.045	0.133	0.009	0.081	0.00014
Centralization	INEL		0.065	0.205	0.012	0.126	0.00014
	SRS		0.046	0.137	0.010	0.083	0.00011
	HS	Geographic	0.055	0.169	0.010	0.100	0.00020
		By Fuel	0.067	0.206	0.012	0.124	0.00024
		All to INEL	0.070	0.222	0.013	0.134	0.00015
		All to SRS	0.068	0.209	0.013	0.125	0.00026
	NTS	Geographic	0.054	0.161	0.011	0.100	0.00023
		By Fuel	0.065	0.198	0.013	0.124	0.00026
		All to INEL	0.071	0.222	0.014	0.137	0.00018
		All to SRS	0.066	0.199	0.014	0.125	0.00028
	ORR	Geographic	0.040	0.117	0.007	0.072	0.00013
		By Fuel	0.051	0.152	0.010	0.095	0.00016
		All to INEL	0.071	0.224	0.013	0.139	0.00023
		All to SRS	0.048	0.142	0.010	0.088	0.00014

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-18 Tabulation of Overland Transportation Risks: Basic Implementation, Shipments from Ports via Truck, Intersite Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.033	0.095	0.006	0.057	0.00007
		By Fuel	0.044	0.131	0.009	0.081	0.00010
		All to INEL	0.057	0.179	0.011	0.110	0.00012
	NTS/SRS	Geographic	0.034	0.098	0.007	0.058	0.00007
		By Fuel	0.045	0.134	0.009	0.082	0.00010
		All to SRS	0.044	0.129	0.010	0.078	0.00010
	NTS/ORR	Geographic	0.034	0.097	0.007	0.058	0.00007
		By Fuel	0.045	0.134	0.009	0.082	0.00010
		All to INEL	0.058	0.181	0.011	0.111	0.00013
		All to SRS	0.044	0.129	0.010	0.078	0.00010
	HS/SRS	Geographic	0.033	0.096	0.006	0.057	0.00007
		By Fuel	0.044	0.132	0.009	0.081	0.00010
		All to SRS	0.043	0.127	0.009	0.077	0.00010
	HS/ORR	Geographic	0.033	0.095	0.006	0.057	0.00007
		By Fuel	0.044	0.131	0.009	0.081	0.00010
		All to INEL	0.058	0.179	0.011	0.110	0.00013
		All to SRS	0.043	0.126	0.009	0.077	0.00010
Centralization	INEL						
	SRS						
	HS	Geographic	0.044	0.129	0.009	0.076	0.00011
		By Fuel	0.055	0.165	0.012	0.100	0.00014
		All to INEL	0.068	0.212	0.013	0.129	0.00015
		All to SRS	0.054	0.161	0.012	0.096	0.00015
	NTS	Geographic	0.043	0.126	0.010	0.077	0.00012
		By Fuel	0.054	0.163	0.012	0.101	0.00015
		All to INEL	0.067	0.209	0.013	0.130	0.00015
		All to SRS	0.053	0.158	0.013	0.097	0.00015
	ORR	Geographic	0.036	0.103	0.007	0.063	0.00008
		By Fuel	0.047	0.140	0.010	0.087	0.00011
		All to INEL	0.061	0.188	0.012	0.116	0.00015
		All to SRS	0.046	0.135	0.010	0.083	0.00011

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-19 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.010	0.015	0.019	0.002	0.00003
1992/1993 Planning Basis	INEL/SRS		0.010	0.015	0.019	0.002	0.00003
Regionalization by Fuel Type	INEL/SRS		0.013	0.020	0.031	0.003	0.00004
Regionalization by Geography	INEL/SRS		0.010	0.015	0.019	0.002	0.00003
	INEL/ORR	Geographic	0.012	0.019	0.018	0.004	0.00003
		By Fuel	0.014	0.023	0.027	0.005	0.00004
		All to INEL	0.016	0.022	0.041	0.006	0.00005
	NTS/SRS	Geographic	0.012	0.022	0.020	0.005	0.00003
		By Fuel	0.014	0.025	0.028	0.006	0.00004
		All to SRS	0.014	0.026	0.025	0.006	0.00004
	NTS/ORR	Geographic	0.014	0.026	0.019	0.009	0.00006
		By Fuel	0.015	0.027	0.027	0.007	0.00005
		All to INEL	0.021	0.038	0.042	0.013	0.00008
		All to SRS	0.014	0.025	0.024	0.006	0.00004
	HS/SRS	Geographic	0.011	0.019	0.019	0.004	0.00003
		By Fuel	0.014	0.023	0.028	0.005	0.00004
		All to SRS	0.013	0.023	0.025	0.005	0.00004
	HS/ORR	Geographic	0.013	0.024	0.018	0.008	0.00006
		By Fuel	0.014	0.024	0.027	0.006	0.00004
		All to INEL	0.019	0.032	0.041	0.010	0.00006
		All to SRS	0.014	0.023	0.023	0.005	0.00004
Centralization	INEL		0.016	0.020	0.049	0.004	0.00005
	SRS		0.013	0.020	0.027	0.003	0.00004
	HS	Geographic	0.026	0.059	0.030	0.027	0.00015
		By Fuel	0.017	0.027	0.038	0.007	0.00007
		All to INEL	0.021	0.035	0.052	0.011	0.00006
		All to SRS	0.017	0.026	0.036	0.006	0.00007
	NTS	Geographic	0.025	0.056	0.027	0.028	0.00016
		By Fuel	0.018	0.030	0.035	0.008	0.00008
		All to INEL	0.023	0.040	0.049	0.014	0.00009
		All to SRS	0.017	0.028	0.032	0.007	0.00007
	ORR	Geographic	0.016	0.033	0.019	0.014	0.00007
		By Fuel	0.019	0.038	0.028	0.015	0.00007
		All to INEL	0.030	0.067	0.043	0.034	0.00015
		All to SRS	0.016	0.031	0.024	0.011	0.00005

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

shipments. Tables E-11 through E-13 present these risk estimates using average risk parameters. Tables E-14 through E-16 provide the lower bound risk estimates, and Tables E-17 through E-19 provide the upper bound risk estimates.

E.7.3 MEI Results for Routine Conditions

The risks to MEIs under incident-free transportation conditions have been estimated for the exposure scenarios described in Section E.6.7. The estimated dose to each of the receptors considered is presented in Table E-20 on a per-event basis (person-rem per event). Note that the potential exists for individual exposures if multiple exposure events occur. For instance, the dose to a person stuck in traffic next to a spent nuclear fuel shipment for 30 min is calculated to be 11 mrem. If the exposure duration was longer, the dose would rise proportionally. Therefore, it is conceivable that a person could receive a dose on the order of 30 to 50 mrem while stopped in traffic next to a shipment. In addition, a person working at a truck service station could receive a significant dose if trucks were to use the same stops repeatedly. If a truckstop worker was present for 100 shipment stops (at the distance and duration given above), the calculated dose is on the order of 30 mrem. Administrative controls could be instituted to control the location and duration of truck stops if multiple exposures were to happen routinely.

Table E-20 Estimated Doses (Rem/Event) to MEIs During Incident-free Transportation Conditions[a, b]

Receptor		*Dose to MEI*	
		Truck	*Rail*
Workers	Crew Member	0.1 rem/yr[c]	0.1 rem/yr[c]
	Inspector	0.0029 rem/event	0.0029 rem/event
	Rail Yard Crew Member	N/A	1.3×10^{-3} rem/event
Public	Resident	4.0×10^{-7} rem/event	4.0×10^{-7} rem/event
	Person in Traffic Obstruction	0.011 rem/event	0.011 rem/event
	Person at Service Station	0.00031 rem/event	N/A
	Resident Near Rail Stop	N/A	0.000013 rem/event

[a] *The exposure scenario assumptions are described in Section E.6.6.*

[b] *Doses are calculated assuming that the shipment external dose rate is equal to the regulatory limit of 10 mrem per hr at 2 m (6.6 ft) from the shipment.*

[c] *Dose to truck drivers could exceed the legal limit of rem per yr in the absence of administrative controls.*

The cumulative dose to a resident was calculated assuming all 837 shipments arrived at a single port or management site. The cumulative doses assume that the resident is present for every shipment and is unshielded at a distance of 30 m (66 ft) from the route. Therefore, the cumulative dose is only a function of the number of shipments passing a particular point and is independent of the actual site being considered. The maximum dose to this resident, if all the spent nuclear fuel were to be shipped to a single site, would be less than 0.1 mrem. The annual individual dose can be estimated by assuming that shipments would occur uniformly over a 15-year time period.

E.7.4 Accident Consequence Assessment - Maximum Severity Accident Results

The accident consequence assessment is intended to provide an estimate of the maximum potential impacts posed by the most severe potential transportation accidents involving a spent nuclear fuel shipment.

The accident consequence results are presented in Table E-21 for the maximum severity accidents as defined in the modal study. The population doses are for a uniform population density within an 80 km- (50 mi-) radius (Neuhuser and Kanipe, 1993). The location of the MEI is determined based on atmospheric conditions at the time of the accident and the buoyant characteristics of the released plume. The locations of maximum exposure would be 160 m (528 ft) and 400 m (1,320 ft) from the accident site for neutral and stable conditions, respectively. The dose to the MEI is independent of the location of the accident. In general, the dose to MEIs for the most severe accidents would be less than 10 mrem. No acute or early fatalities would be expected from radiological causes.

Table E-21 Potential Doses to Populations and MEIs for the Most Severe Transportation Accidents Involving Spent Nuclear Fuel[a,b]

	Neutral Conditions[c]				*Stable Conditions*[d]			
	Population[e]		*MEI*[f]		*Population*[e]		*MEI*[f]	
Mode and Accident Location	*Dose (person-rem)*	*Consequences (cancer fatalities)*	*Dose (rem)*	*Consequences (cancer fatality)*	*Dose (person-rem)*	*Consequences (cancer fatalities)*	*Dose (person-rem)*	*Consequences (cancer fatality)*
Truck								
Urban	14	0.007	0.0024	0.0000012	120	0.06	0.0079	0.000004
Suburban	2.7	0.0014	0.0024	0.0000012	21	0.01	0.0079	0.000004
Rural	0.15	0.000075	0.0024	0.0000012	1.2	0.0006	0.0079	0.000004
Rail								
Urban	14	0.007	0.0024	0.0000012	120	0.06	0.0079	0.000004
Suburban	2.7	0.0014	0.0024	0.0000012	21	0.01	0.0079	0.000004
Rural	0.15	0.000075	0.0024	0.0000012	1.2	0.0006	0.0079	0.000004

[a] *The most severe accidents correspond to the modal study accident severity category 6 (DOE, 1995).*

[b] *Buoyant plume rise resulting from fire for a severe accident was included in the exposure model.*

[c] *Neutral weather conditions result in moderate dispersion and dilution of the release plume. Neutral conditions were taken to be Pasquill stability Class D with a wind speed of 4 m per sec (9 mph). Neutral conditions occur approximately 50 percent of the time in the United States.*

[d] *Stable weather conditions result in minimal dispersion and dilution of the release plume and are thus unfavorable. Stable conditions were taken to be Pasquill stability Class F with a wind speed of 1 m per sec (2.2 mph). Stable conditions occur approximately one-third of the time in the United States.*

[e] *Populations extend at a uniform population density to a radius of 80 km (50 mi) from the accident site. Population exposure pathways include acute inhalation, acute cloudshine, groundshine, resuspended inhalation, resuspended cloudshine, and ingestion of food, including initially contaminated food (rural only). No decontamination or mitigative actions are taken.*

[f] *The MEI is assumed to be at the location of maximum exposure. The locations of maximum exposure would be 160 m (528 ft) and 400 m (1,320 ft) from the accident site under neutral and stable atmospheric conditions, respectively. Individual exposure pathways include acute inhalation, acute cloudshine, and groundshine during passage of the plume. No ingested dose is considered.*

The maximum foreseeable offsite transportation accident involves a shipment of spent nuclear fuel in a suburban population zone under neutral (average) weather conditions. The accident has a probability of occurrence of about 1 every 10,000,000 years and could result in 2.7 person-rem and no fatalities. The probability of an accident occurring is at least 10 times smaller in either an urban area or under stable atmospheric conditions, and the consequences are less than 10 times larger.

E.8 Impacts of Implementation Alternatives of the Spent Nuclear Fuel Acceptance Policy

E.8.1 Implementation Alternative - Implementing an Acceptance Policy of Alternative Amounts of Spent Nuclear Fuel - Accept only from Developing Nations

This implementation alternative was analyzed using the same set of assumptions as used in analyzing the basic implementation. The results are as follows:

Shipments

Under all SNF&INEL Final EIS (DOE, 1995) alternatives, the shipment of foreign research reactor spent nuclear fuel would require the movement of 168 casks from ports of entry to DOE facilities. The basic shipment count, by point of origin is:

	East Coast		*West Coast*		
	Aluminum	*TRIGA*	*Aluminum*	*TRIGA*	*Totals*
Phase 1	31	54	15	30	130
Phase 2	9	16	4	9	38
Totals	**40**	**70**	**19**	**39**	**168**

Calculated in the same manner as described in Section E.7.2.1, the number of intersite shipments for two-phased approaches to this alternative varies between 4 and 33. The variation is caused by the wide variety of phased approaches.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.002 to 0.06 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates is caused by three factors: 1) the option of using truck or rail to transport spent nuclear fuel, 2) combinations of Phase 1 and Phase 2 sites that created varying shipment numbers and distances, and 3) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.001 to 0.015. The estimated number of radiation-related LCFs for the general population ranged from 0.0006 to 0.045, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0002 to 0.01.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.0000001 to 0.00006 LCFs from radiation and from 0.0001 to 0.028 for traffic fatality, depending on the transportation mode and DOE sites selected. The reasons for the range of fatality estimates are the same as those described for incident-free transportation. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Tables E-22 through E-30. The analysis for this alternative implementation is analogous to the analysis performed for the Basic Implementation (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2.

The consequences of the most severe accident hypothesized are the same as described for the Basic Implementation since the material at risk is the same.

Table E-22 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Truck, Average Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.0034	0.0106	0.0005	0.0063	0.000003
1992/1993 Planning Basis	INEL/SRS		0.0034	0.0106	0.0005	0.0063	0.000003
Regionalization by Fuel Type	INEL/SRS		0.0098	0.0309	0.0016	0.0182	0.000011
Regionalization by Geography	INEL/SRS		0.0034	0.0106	0.0005	0.0063	0.000003
	INEL/ORR	Geographic	0.0040	0.0120	0.0006	0.0072	0.000009
		By Fuel	0.0087	0.0273	0.0014	0.0161	0.000012
		All to INEL	0.0096	0.0303	0.0015	0.0178	0.000010
	NTS/SRS	Geographic	0.0039	0.0120	0.0007	0.0071	0.000005
		By Fuel	0.0091	0.0285	0.0016	0.0167	0.000014
		All to SRS	0.0070	0.0215	0.0012	0.0127	0.000007
	NTS/ORR	Geographic	0.0044	0.0134	0.0008	0.0080	0.000011
		By Fuel	0.0094	0.0295	0.0017	0.0173	0.000017
		All to INEL	0.0106	0.0335	0.0018	0.0195	0.000018
		All to SRS	0.0077	0.0234	0.0014	0.0140	0.000016
	HS/SRS	Geographic	0.0035	0.0109	0.0005	0.0064	0.000003
		By Fuel	0.0086	0.0272	0.0013	0.0159	0.000010
		All to SRS	0.0067	0.0205	0.0011	0.0122	0.000007
	HS/ORR	Geographic	0.0040	0.0122	0.0006	0.0073	0.000009
		By Fuel	0.0090	0.0281	0.0014	0.0165	0.000013
		All to INEL	0.0101	0.0319	0.0015	0.0185	0.000012
		All to SRS	0.0074	0.0224	0.0012	0.0134	0.000016
Centralization	INEL		0.0112	0.0356	0.0017	0.0208	0.000012
	SRS		0.0078	0.0241	0.0013	0.0142	0.000008
	HS	Geographic	0.0078	0.0245	0.0011	0.0142	0.000025
		By Fuel	0.0119	0.0377	0.0018	0.0219	0.000023
		All to INEL	0.0120	0.0384	0.0018	0.0221	0.000014
		All to SRS	0.0120	0.0375	0.0019	0.0220	0.000039
	NTS	Geographic	0.0077	0.0237	0.0014	0.0146	0.000030
		By Fuel	0.0119	0.0376	0.0021	0.0224	0.000028
		All to INEL	0.0123	0.0390	0.0022	0.0229	0.000020
		All to SRS	0.0117	0.0361	0.0020	0.0221	0.000043
	ORR	Geographic	0.0056	0.0173	0.0008	0.0105	0.000017
		By Fuel	0.0110	0.0347	0.0017	0.0207	0.000026
		All to INEL	0.0127	0.0404	0.0019	0.0240	0.000032
		All to SRS	0.0083	0.0255	0.0014	0.0153	0.000017

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-23 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.0037	0.0111	0.0006	0.0065	0.000004
		By Fuel	0.0086	0.0268	0.0014	0.0157	0.000009
		All to INEL	0.0096	0.0303	0.0015	0.0178	0.000010
	NTS/SRS	Geographic	0.0036	0.0111	0.0006	0.0066	0.000003
		By Fuel	0.0086	0.0269	0.0015	0.0158	0.000009
		All to SRS	0.0070	0.0215	0.0012	0.0127	0.000007
	NTS/ORR	Geographic	0.0039	0.0115	0.0007	0.0068	0.000004
		By Fuel	0.0088	0.0273	0.0015	0.0160	0.000010
		All to INEL	0.0098	0.0308	0.0016	0.0181	0.000011
		All to SRS	0.0072	0.0220	0.0013	0.0130	0.000009
	HS/SRS	Geographic	0.0033	0.0101	0.0005	0.0060	0.000003
		By Fuel	0.0083	0.0258	0.0013	0.0152	0.000009
		All to SRS	0.0067	0.0205	0.0011	0.0122	0.000007
	HS/ORR	Geographic	0.0035	0.0105	0.0006	0.0063	0.000004
		By Fuel	0.0085	0.0263	0.0014	0.0155	0.000010
		All to INEL	0.0095	0.0298	0.0015	0.0175	0.000011
		All to SRS	0.0069	0.0210	0.0012	0.0125	0.000008
Centralization	INEL						
	SRS						
	HS	Geographic	0.0057	0.0171	0.0010	0.0099	0.000009
		By Fuel	0.0105	0.0328	0.0018	0.0191	0.000014
		All to INEL	0.0115	0.0363	0.0018	0.0211	0.000013
		All to SRS	0.0091	0.0277	0.0017	0.0161	0.000016
	NTS	Geographic	0.0057	0.0172	0.0012	0.0103	0.000010
		By Fuel	0.0106	0.0329	0.0019	0.0195	0.000014
		All to INEL	0.0115	0.0364	0.0019	0.0215	0.000013
		All to SRS	0.0091	0.0277	0.0019	0.0165	0.000016
	ORR	Geographic	0.0045	0.0137	0.0008	0.0082	0.000006
		By Fuel	0.0095	0.0295	0.0016	0.0174	0.000013
		All to INEL	0.0106	0.0331	0.0018	0.0194	0.000015
		All to SRS	0.0079	0.0241	0.0013	0.0143	0.000009

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-24 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Rail, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.0015	0.0011	0.0010	0.0002	0.000001
1992/1993 Planning Basis	INEL/SRS		0.0015	0.0011	0.0010	0.0002	0.000001
Regionalization by Fuel Type	INEL/SRS		0.0027	0.0027	0.0034	0.0006	0.000003
Regionalization by Geography	INEL/SRS		0.0015	0.0011	0.0010	0.0002	0.000001
	INEL/ORR	Geographic	0.0018	0.0023	0.0010	0.0008	0.000001
		By Fuel	0.0028	0.0035	0.0029	0.0011	0.000003
		All to INEL	0.0029	0.0036	0.0031	0.0011	0.000003
	NTS/SRS	Geographic	0.0019	0.0027	0.0011	0.0010	0.000001
		By Fuel	0.0029	0.0039	0.0030	0.0014	0.000003
		All to SRS	0.0025	0.0038	0.0025	0.0013	0.000002
	NTS/ORR	Geographic	0.0023	0.0037	0.0011	0.0017	0.000007
		By Fuel	0.0035	0.0057	0.0031	0.0023	0.000008
		All to INEL	0.0039	0.0068	0.0035	0.0028	0.000010
		All to SRS	0.0027	0.0039	0.0026	0.0013	0.000003
	HS/SRS	Geographic	0.0016	0.0017	0.0009	0.0005	0.000001
		By Fuel	0.0026	0.0029	0.0028	0.0008	0.000003
		All to SRS	0.0022	0.0028	0.0024	0.0007	0.000002
	HS/ORR	Geographic	0.0020	0.0027	0.0010	0.0011	0.000007
		By Fuel	0.0030	0.0044	0.0029	0.0014	0.000004
		All to INEL	0.0034	0.0052	0.0032	0.0019	0.000004
		All to SRS	0.0024	0.0029	0.0025	0.0007	0.000003
Centralization	INEL		0.0030	0.0028	0.0038	0.0007	0.000003
	SRS		0.0024	0.0026	0.0029	0.0005	0.000002
	HS	Geographic	0.0042	0.0089	0.0020	0.0046	0.000022
		By Fuel	0.0035	0.0048	0.0037	0.0016	0.000006
		All to INEL	0.0038	0.0056	0.0039	0.0020	0.000005
		All to SRS	0.0030	0.0036	0.0035	0.0009	0.000009
	NTS	Geographic	0.0042	0.0089	0.0020	0.0050	0.000024
		By Fuel	0.0039	0.0062	0.0039	0.0024	0.000010
		All to INEL	0.0043	0.0072	0.0041	0.0030	0.000010
		All to SRS	0.0033	0.0045	0.0035	0.0015	0.000009
	ORR	Geographic	0.0030	0.0059	0.0012	0.0030	0.000009
		By Fuel	0.0050	0.0109	0.0031	0.0056	0.000017
		All to INEL	0.0060	0.0137	0.0035	0.0074	0.000024
		All to SRS	0.0033	0.0061	0.0026	0.0026	0.000004

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-25 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipment via Truck, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radiological
Decentralization	INEL/SRS		0.0023	0.0072	0.0002	0.0039	0.0000004
1992/1993 Planning Basis	INEL/SRS		0.0023	0.0072	0.0002	0.0039	0.0000004
Regionalization by Fuel Type	INEL/SRS		0.0080	0.0256	0.0010	0.0142	0.0000022
Regionalization	INEL/SRS		0.0023	0.0072	0.0002	0.0039	0.0000004
by	INEL/ORR	Geographic	0.0029	0.0087	0.0003	0.0050	0.0000061
Geography		By Fuel	0.0071	0.0226	0.0009	0.0127	0.0000049
		All to INEL	0.0077	0.0245	0.0011	0.0141	0.0000020
	NTS/SRS	Geographic	0.0026	0.0082	0.0004	0.0046	0.0000031
		By Fuel	0.0073	0.0232	0.0011	0.0130	0.0000064
		All to SRS	0.0058	0.0181	0.0007	0.0096	0.0000018
	NTS/ORR	Geographic	0.0032	0.0098	0.0005	0.0058	0.0000088
		By Fuel	0.0077	0.0244	0.0012	0.0139	0.0000096
		All to INEL	0.0086	0.0273	0.0014	0.0157	0.0000093
		All to SRS	0.0066	0.0202	0.0008	0.0111	0.0000103
	HS/SRS	Geographic	0.0023	0.0073	0.0002	0.0039	0.0000010
		By Fuel	0.0069	0.0221	0.0009	0.0122	0.0000030
		All to SRS	0.0056	0.0173	0.0006	0.0091	0.0000017
	HS/ORR	Geographic	0.0029	0.0088	0.0003	0.0051	0.0000068
		By Fuel	0.0074	0.0233	0.0010	0.0130	0.0000062
		All to INEL	0.0081	0.0260	0.0011	0.0147	0.0000039
		All to SRS	0.0063	0.0194	0.0007	0.0105	0.0000102
Centralization	INEL		0.0090	0.0288	0.0013	0.0164	0.0000023
	SRS		0.0068	0.0212	0.0007	0.0110	0.0000021
	HS	Geographic	0.0063	0.0201	0.0008	0.0112	0.0000215
		By Fuel	0.0100	0.0319	0.0013	0.0177	0.0000143
		All to INEL	0.0098	0.0315	0.0014	0.0176	0.0000043
		All to SRS	0.0106	0.0336	0.0013	0.0184	0.0000320
	NTS	Geographic	0.0060	0.0188	0.0010	0.0113	0.0000255
		By Fuel	0.0098	0.0312	0.0016	0.0180	0.0000189
		All to INEL	0.0098	0.0316	0.0017	0.0181	0.0000096
		All to SRS	0.0101	0.0316	0.0015	0.0183	0.0000352
	ORR	Geographic	0.0045	0.0142	0.0005	0.0080	0.0000140
		By Fuel	0.0095	0.0301	0.0012	0.0170	0.0000185
		All to INEL	0.0108	0.0348	0.0014	0.0200	0.0000232
		All to SRS	0.0074	0.0229	0.0008	0.0122	0.0000105

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-26 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.0025	0.0078	0.0003	0.0044	0.0000011
		By Fuel	0.0070	0.0221	0.0009	0.0124	0.0000022
		All to INEL	0.0077	0.0245	0.0011	0.0141	0.0000020
	NTS/SRS	Geographic	0.0023	0.0072	0.0004	0.0041	0.0000007
		By Fuel	0.0068	0.0215	0.0010	0.0121	0.0000022
		All to SRS	0.0058	0.0181	0.0007	0.0096	0.0000018
	NTS/ORR	Geographic	0.0026	0.0079	0.0004	0.0046	0.0000014
		By Fuel	0.0071	0.0222	0.0010	0.0126	0.0000027
		All to INEL	0.0078	0.0247	0.0012	0.0143	0.0000027
		All to SRS	0.0061	0.0188	0.0008	0.0101	0.0000028
	HS/SRS	Geographic	0.0021	0.0065	0.0002	0.0035	0.0000006
		By Fuel	0.0066	0.0208	0.0009	0.0115	0.0000023
		All to SRS	0.0056	0.0173	0.0006	0.0091	0.0000017
	HS/ORR	Geographic	0.0024	0.0071	0.0003	0.0041	0.0000014
		By Fuel	0.0069	0.0214	0.0009	0.0121	0.0000027
		All to INEL	0.0076	0.0239	0.0011	0.0137	0.0000028
		All to SRS	0.0059	0.0180	0.0006	0.0096	0.0000026
Centralization	INEL						
	SRS						
	HS	Geographic	0.0042	0.0128	0.0007	0.0069	0.0000054
		By Fuel	0.0086	0.0270	0.0013	0.0149	0.0000051
		All to INEL	0.0092	0.0294	0.0013	0.0166	0.0000031
		All to SRS	0.0077	0.0237	0.0011	0.0125	0.0000083
	NTS	Geographic	0.0040	0.0123	0.0008	0.0071	0.0000052
		By Fuel	0.0084	0.0265	0.0014	0.0150	0.0000050
		All to INEL	0.0091	0.0289	0.0015	0.0167	0.0000030
		All to SRS	0.0076	0.0232	0.0013	0.0126	0.0000081
	ORR	Geographic	0.0035	0.0106	0.0004	0.0057	0.0000029
		By Fuel	0.0080	0.0249	0.0011	0.0137	0.0000053
		All to INEL	0.0088	0.0274	0.0013	0.0154	0.0000063
		All to SRS	0.0069	0.0214	0.0007	0.0112	0.0000030

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-27 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.0012	0.0006	0.0004	0.0001	0.0000001
1992/1993 Planning Basis	INEL/SRS		0.0012	0.0006	0.0004	0.0001	0.0000001
Regionalization by Fuel Type	INEL/SRS		0.0023	0.0017	0.0021	0.0005	0.0000003
Regionalization	INEL/SRS		0.0012	0.0006	0.0004	0.0001	0.0000001
by	INEL/ORR	Geographic	0.0015	0.0018	0.0004	0.0006	0.0000007
Geography		By Fuel	0.0024	0.0026	0.0017	0.0009	0.0000006
		All to INEL	0.0024	0.0025	0.0018	0.0009	0.0000002
	NTS/SRS	Geographic	0.0015	0.0017	0.0005	0.0008	0.0000004
		By Fuel	0.0024	0.0026	0.0018	0.0011	0.0000007
		All to SRS	0.0021	0.0027	0.0018	0.0010	0.0000005
	NTS/ORR	Geographic	0.0019	0.0028	0.0006	0.0015	0.0000060
		By Fuel	0.0029	0.0044	0.0020	0.0020	0.0000053
		All to INEL	0.0033	0.0053	0.0022	0.0026	0.0000075
		All to SRS	0.0023	0.0029	0.0019	0.0011	0.0000013
	HS/SRS	Geographic	0.0012	0.0010	0.0004	0.0002	0.0000004
		By Fuel	0.0021	0.0018	0.0017	0.0005	0.0000008
		All to SRS	0.0019	0.0020	0.0016	0.0005	0.0000003
	HS/ORR	Geographic	0.0017	0.0020	0.0005	0.0009	0.0000060
		By Fuel	0.0026	0.0033	0.0018	0.0012	0.0000018
		All to INEL	0.0029	0.0040	0.0019	0.0016	0.0000022
		All to SRS	0.0021	0.0021	0.0017	0.0005	0.0000012
Centralization	INEL		0.0025	0.0017	0.0021	0.0005	0.0000002
	SRS		0.0021	0.0020	0.0021	0.0004	0.0000004
	HS	Geographic	0.0037	0.0080	0.0011	0.0043	0.0000204
		By Fuel	0.0030	0.0036	0.0023	0.0013	0.0000039
		All to INEL	0.0032	0.0042	0.0022	0.0017	0.0000022
		All to SRS	0.0026	0.0026	0.0024	0.0007	0.0000066
	NTS	Geographic	0.0037	0.0077	0.0013	0.0047	0.0000224
		By Fuel	0.0033	0.0046	0.0025	0.0021	0.0000072
		All to INEL	0.0036	0.0054	0.0026	0.0026	0.0000075
		All to SRS	0.0027	0.0032	0.0025	0.0012	0.0000064
	ORR	Geographic	0.0028	0.0055	0.0006	0.0026	0.0000075
		By Fuel	0.0047	0.0101	0.0020	0.0052	0.0000142
		All to INEL	0.0056	0.0128	0.0021	0.0069	0.0000214
		All to SRS	0.0031	0.0056	0.0018	0.0022	0.0000015

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-28 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Truck, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.0064	0.0191	0.0014	0.0111	0.000014
1992/1993 Planning Basis	INEL/SRS		0.0064	0.0191	0.0014	0.0111	0.000014
Regionalization by Fuel Type	INEL/SRS		0.0113	0.0352	0.0024	0.0220	0.000027
Regionalization by Geography	INEL/SRS		0.0064	0.0191	0.0014	0.0111	0.000014
	INEL/ORR	Geographic	0.0067	0.0200	0.0015	0.0117	0.000019
		By Fuel	0.0104	0.0320	0.0022	0.0199	0.000027
		All to INEL	0.0107	0.0335	0.0021	0.0211	0.000024
	NTS/SRS	Geographic	0.0069	0.0207	0.0016	0.0118	0.000017
		By Fuel	0.0110	0.0340	0.0025	0.0207	0.000029
		All to SRS	0.0099	0.0296	0.0024	0.0176	0.000024
	NTS/ORR	Geographic	0.0072	0.0215	0.0016	0.0125	0.000022
		By Fuel	0.0111	0.0344	0.0025	0.0210	0.000032
		All to INEL	0.0117	0.0368	0.0025	0.0227	0.000032
		All to SRS	0.0104	0.0310	0.0025	0.0186	0.000032
	HS/SRS	Geographic	0.0067	0.0198	0.0014	0.0113	0.000015
		By Fuel	0.0106	0.0329	0.0022	0.0201	0.000025
		All to SRS	0.0097	0.0289	0.0023	0.0173	0.000023
	HS/ORR	Geographic	0.0070	0.0206	0.0015	0.0120	0.000020
		By Fuel	0.0108	0.0333	0.0022	0.0204	0.000028
		All to INEL	0.0113	0.0355	0.0022	0.0220	0.000026
		All to SRS	0.0102	0.0303	0.0024	0.0183	0.000031
Centralization	INEL		0.0120	0.0379	0.0024	0.0240	0.000028
	SRS		0.0106	0.0319	0.0026	0.0192	0.000026
	HS	Geographic	0.0104	0.0320	0.0020	0.0188	0.000038
		By Fuel	0.0134	0.0420	0.0026	0.0257	0.000040
		All to INEL	0.0130	0.0411	0.0025	0.0254	0.000030
		All to SRS	0.0145	0.0445	0.0030	0.0267	0.000057
	NTS	Geographic	0.0102	0.0310	0.0022	0.0189	0.000043
		By Fuel	0.0134	0.0417	0.0029	0.0260	0.000045
		All to INEL	0.0131	0.0415	0.0028	0.0260	0.000036
		All to SRS	0.0141	0.0428	0.0032	0.0267	0.000060
	ORR	Geographic	0.0083	0.0251	0.0017	0.0152	0.000029
		By Fuel	0.0126	0.0393	0.0026	0.0247	0.000042
		All to INEL	0.0137	0.0435	0.0026	0.0275	0.000047
		All to SRS	0.0109	0.0328	0.0026	0.0202	0.000033

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-29 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radiological
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.0064	0.0190	0.0014	0.0111	0.000014
		By Fuel	0.0102	0.0315	0.0022	0.0195	0.000024
		All to INEL	0.0107	0.0335	0.0021	0.0211	0.000024
	NTS/SRS	Geographic	0.0067	0.0197	0.0015	0.0113	0.000015
		By Fuel	0.0105	0.0323	0.0023	0.0198	0.000025
		All to SRS	0.0099	0.0296	0.0024	0.0176	0.000024
	NTS/ORR	Geographic	0.0067	0.0196	0.0015	0.0113	0.000015
		By Fuel	0.0105	0.0322	0.0023	0.0198	0.000025
		All to INEL	0.0110	0.0341	0.0023	0.0213	0.000025
		All to SRS	0.0099	0.0296	0.0024	0.0177	0.000024
	HS/SRS	Geographic	0.0065	0.0190	0.0014	0.0110	0.000015
		By Fuel	0.0103	0.0316	0.0022	0.0194	0.000025
		All to SRS	0.0097	0.0289	0.0023	0.0173	0.000023
	HS/ORR	Geographic	0.0065	0.0189	0.0014	0.0110	0.000015
		By Fuel	0.0103	0.0314	0.0022	0.0195	0.000025
		All to INEL	0.0108	0.0334	0.0022	0.0210	0.000025
		All to SRS	0.0097	0.0289	0.0023	0.0173	0.000024
Centralization	INEL						
	SRS						
	HS	Geographic	0.0083	0.0246	0.0019	0.0145	0.000022
		By Fuel	0.0120	0.0371	0.0026	0.0229	0.000030
		All to INEL	0.0124	0.0390	0.0024	0.0244	0.000029
		All to SRS	0.0116	0.0346	0.0029	0.0209	0.000033
	NTS	Geographic	0.0082	0.0245	0.0020	0.0146	0.000023
		By Fuel	0.0120	0.0370	0.0027	0.0231	0.000031
		All to INEL	0.0124	0.0388	0.0026	0.0246	0.000030
		All to SRS	0.0115	0.0345	0.0030	0.0210	0.000033
	ORR	Geographic	0.0073	0.0215	0.0017	0.0129	0.000018
		By Fuel	0.0111	0.0341	0.0025	0.0214	0.000029
		All to INEL	0.0116	0.0361	0.0025	0.0229	0.000030
		All to SRS	0.0104	0.0314	0.0025	0.0192	0.000026

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-30 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.0020	0.0025	0.0028	0.0004	0.000005
1992/1993 Planning Basis	INEL/SRS		0.0020	0.0025	0.0028	0.0004	0.000005
Regionalization by Fuel Type	INEL/SRS		0.0030	0.0036	0.0076	0.0007	0.000010
Regionalization by Geography	INEL/SRS		0.0020	0.0025	0.0028	0.0004	0.000005
	INEL/ORR	Geographic	0.0024	0.0037	0.0027	0.0011	0.000006
		By Fuel	0.0031	0.0046	0.0064	0.0013	0.000009
		All to INEL	0.0031	0.0044	0.0076	0.0013	0.000009
	NTS/SRS	Geographic	0.0026	0.0044	0.0029	0.0013	0.000006
		By Fuel	0.0034	0.0053	0.0067	0.0016	0.000010
		All to SRS	0.0032	0.0056	0.0046	0.0015	0.000009
	NTS/ORR	Geographic	0.0029	0.0053	0.0028	0.0019	0.000011
		By Fuel	0.0038	0.0069	0.0067	0.0025	0.000014
		All to INEL	0.0042	0.0078	0.0080	0.0030	0.000017
		All to SRS	0.0033	0.0056	0.0045	0.0015	0.000009
	HS/SRS	Geographic	0.0023	0.0037	0.0028	0.0010	0.000006
		By Fuel	0.0031	0.0046	0.0065	0.0012	0.000010
		All to SRS	0.0030	0.0049	0.0045	0.0012	0.000008
	HS/ORR	Geographic	0.0027	0.0046	0.0027	0.0016	0.000011
		By Fuel	0.0035	0.0059	0.0064	0.0019	0.000011
		All to INEL	0.0038	0.0065	0.0077	0.0023	0.000012
		All to SRS	0.0031	0.0048	0.0044	0.0012	0.000009
Centralization	INEL		0.0031	0.0035	0.0092	0.0007	0.000010
	SRS		0.0029	0.0041	0.0051	0.0007	0.000009
	HS	Geographic	0.0048	0.0107	0.0050	0.0051	0.000027
		By Fuel	0.0039	0.0063	0.0086	0.0020	0.000014
		All to INEL	0.0041	0.0069	0.0098	0.0024	0.000013
		All to SRS	0.0036	0.0055	0.0067	0.0014	0.000016
	NTS	Geographic	0.0048	0.0104	0.0043	0.0052	0.000029
		By Fuel	0.0043	0.0074	0.0080	0.0026	0.000018
		All to INEL	0.0045	0.0082	0.0092	0.0031	0.000019
		All to SRS	0.0038	0.0061	0.0060	0.0017	0.000016
	ORR	Geographic	0.0035	0.0072	0.0030	0.0036	0.000014
		By Fuel	0.0053	0.0118	0.0067	0.0061	0.000024
		All to INEL	0.0062	0.0144	0.0081	0.0078	0.000032
		All to SRS	0.0038	0.0074	0.0046	0.0031	0.000011

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

E.8.2 Implementation Alternative - Implementing an Acceptance Policy of Alternative Amounts of Spent Nuclear Fuel - Accept Only from Reactors that Use Highly-Enriched Uranium (HEU)

This alternative was not analyzed for policy reasons. See Chapter 4.

E.8.3 Implementation Alternative - Implementing an Acceptance Policy of Alternative Amounts of Spent Nuclear Fuel - Accept Target Material

Target material is currently stored overseas as a liquid. In order to allow shipment, it must be processed into a solid form by either calcination or oxidation. Calcination results in a solid, but easily crumbled material, and oxidation results in a powder. Oxidation removes the aluminum and, therefore, would lead to fewer shipments than calcination. Shipment counts in Appendix B indicate that just over five shipments would be arriving on the east coast. However, in order to be conservative, six full shipments are used for transportation risk analysis. Similarly, the amount of material that could arrive on the west coast is much less than one full cask. The analysis conservatively assumes one full cask.

Form	*Port of Entry*		
	East Coast	*West Coast*	*Eastern Canada*
Calcinate	14	1	125
Oxidized Powder	6	1	50

Analysis of the target material and potential casks indicates that the maximum dose rate from any cask would be 0.1 mrem per hr at 2 m (3.3 ft). This low radiation level is based on the low burn-up of target material. Because of the conservative release fractions assigned to the oxidized material (see Section E.6.4.2), the results are emphasized below. The risks tabulated in this section would be added to those associated with the basic implementation of Management Alternative 1 if both aspects of the policy were to be performed.

Impacts of Incident-Free Ground Transport

The incident-free transportation of oxidized target material was estimated to result in total latent fatalities that ranged from 0.0002 to 0.003 over the entire duration of the program. The calcinated target material results are 2.5 times higher. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew. This represents an increase to the risk associated with the basic implementation.

The range of fatality estimates is caused by three factors: 1) the option of using truck or rail to transport spent nuclear fuel 2) combinations of Phase 1 and Phase 2 sites that created varying shipment numbers and distances, and 3) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.00007 to 0.00074. The estimated number of radiation-related LCFs for the general population ranged from 0.00015 to 0.0023, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.00396.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire policy are estimated to range from 0.00023 to 0.0054 LCF from radiation and from 0.0001 to 0.013 for traffic fatality, depending on the transportation mode and DOE sites selected. The risks would be four times lower if calcinated material is transported. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Tables E-31 through E-39. The analysis for this implementation alternative is analogous to the analysis performed for the Basic Implementation (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2. The total policy risk with this implementation alternative is the sum of the values in the above referenced tables and those in Section E.7.2 describing the Basic Implementation.

Table E-40 gives the consequences for the most severe accident hypothesized if that accident were to occur at various locations. The maximum accident risks would be four times lower for calcinated material. The accident probabilities are described in Section 6 of this appendix.

E.8.4 Implementation Alternative - Implementing an Acceptance Policy for Varying Durations - Five-Year Spent Nuclear Fuel Acceptance

Under all SNF&INEL Final EIS (DOE, 1995) alternatives, the shipment of foreign research reactor spent nuclear fuel would require the movement of casks from ports of entry to DOE facilities. The basic shipment count, by point of origin is:

	East Coast		*West Coast*		Totals
	Aluminum	*TRIGA*	*Aluminum*	*TRIGA*	
Phase 1	419	101	105	53	678

Calculated in the same manner as described for the basic implementation of Management Alternative 1, the number of intersite shipments for the two-phased approaches to this strategy varies between 8 and 184. The variation is caused by the wide variety of phased approaches.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.01 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates is caused by two factors: 1) the option of using truck or rail to transport spent nuclear fuel, and 2) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.005 to 0.064. The estimated number of radiation-related LCFs for the general population ranged from 0.005 to 0.20, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.041.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000003 to 0.00026 LCFs from radiation and from 0.001 to 0.13 for traffic fatality, depending on the transportation mode and DOE sites selected. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Tables E-41 through E-49. The analysis for this implementation alternative is analogous to the analysis performed for the basic implementation of Management Alternative 1 (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2.

Table E-31 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Truck, Average Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.00022	0.00061	0.00013	0.0040	0.00070
1992/1993 Planning Basis	INEL/SRS		0.00022	0.00061	0.00013	0.0040	0.00070
Regionalization by Fuel Type	INEL/SRS		0.00023	0.00064	0.00016	0.0041	0.00074
Regionalization by Geography	INEL/SRS		0.00022	0.00061	0.00013	0.0040	0.00070
	INEL/ORR	Geographic	0.00029	0.00079	0.00026	0.0051	0.00071
		By Fuel	0.00030	0.00082	0.00028	0.0053	0.00075
		All to INEL	0.00042	0.00129	0.00060	0.0066	0.00352
	NTS/SRS	Geographic	0.00022	0.00062	0.00015	0.0040	0.00070
		By Fuel	0.00023	0.00064	0.00016	0.0041	0.00074
		All to SRS	0.00023	0.00064	0.00016	0.0041	0.00074
	NTS/ORR	Geographic	0.00029	0.00080	0.00027	0.0052	0.00071
		By Fuel	0.00030	0.00082	0.00028	0.0053	0.00075
		All to INEL	0.00054	0.00166	0.00095	0.0085	0.00353
		All to SRS	0.00030	0.00082	0.00028	0.0053	0.00075
	HS/SRS	Geographic	0.00022	0.00062	0.00015	0.0040	0.00070
		By Fuel	0.00023	0.00064	0.00016	0.0041	0.00074
		All to SRS	0.00023	0.00064	0.00016	0.0041	0.00074
	HS/ORR	Geographic	0.00029	0.00080	0.00027	0.0051	0.00071
		By Fuel	0.00030	0.00082	0.00028	0.0053	0.00075
		All to INEL	0.00051	0.00159	0.00069	0.0079	0.00352
		All to SRS	0.00030	0.00082	0.00028	0.0053	0.00075
Centralization	INEL		0.00049	0.00152	0.00074	0.0075	0.00439
	SRS		0.00023	0.00055	0.00016	0.0041	0.00074
	HS	Geographic	0.00072	0.00221	0.00087	0.0128	0.00163
		By Fuel	0.00073	0.00225	0.00089	0.0131	0.00167
		All to INEL	0.00059	0.00186	0.00086	0.0091	0.00442
		All to SRS	0.00073	0.00225	0.00089	0.0131	0.00167
	NTS	Geographic	0.00067	0.00199	0.00094	0.0127	0.00160
		By Fuel	0.00068	0.00203	0.00096	0.0129	0.00165
		All to INEL	0.00061	0.00191	0.00114	0.0098	0.00439
		All to SRS	0.00068	0.00203	0.00096	0.0129	0.00165
	ORR	Geographic	0.00030	0.00082	0.00027	0.0053	0.00071
		By Fuel	0.00030	0.00082	0.00028	0.0053	0.00075
		All to INEL	0.00073	0.00230	0.00095	0.0128	0.00355
		All to SRS	0.00030	0.00082	0.00028	0.0053	0.00075

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-32 Tabulation of Overland Transportation Risks: Accept Target Material Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Average Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase 1 Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.00025	0.00062	0.00025	0.0039	0.00070
		By Fuel	0.00026	0.00065	0.00026	0.0040	0.00074
		All to INEL	0.00042	0.00129	0.00060	0.0066	0.00352
	NTS/SRS	Geographic	0.00022	0.00061	0.00015	0.0040	0.00070
		By Fuel	0.00023	0.00064	0.00016	0.0041	0.00074
		All to SRS	0.00023	0.00064	0.00016	0.0041	0.00074
	NTS/ORR	Geographic	0.00025	0.00062	0.00025	0.0039	0.00070
		By Fuel	0.00026	0.00065	0.00026	0.0040	0.00074
		All to INEL	0.00047	0.00132	0.00077	0.0066	0.00352
		All to SRS	0.00026	0.00065	0.00026	0.0040	0.00074
	HS/SRS	Geographic	0.00022	0.00061	0.00015	0.0040	0.00070
		By Fuel	0.00023	0.00064	0.00016	0.0041	0.00074
		All to SRS	0.00023	0.00064	0.00016	0.0041	0.00074
	HS/ORR	Geographic	0.00025	0.00062	0.00025	0.0039	0.00070
		By Fuel	0.00026	0.00065	0.00026	0.0040	0.00074
		All to INEL	0.00046	0.00132	0.00077	0.0066	0.00352
		All to SRS	0.00026	0.00065	0.00026	0.0040	0.00074
Centralization	INEL						
	SRS						
	HS	Geographic	0.00041	0.00097	0.00125	0.0053	0.00161
		By Fuel	0.00041	0.00100	0.00129	0.0054	0.00165
		All to INEL	0.00054	0.00159	0.00094	0.0078	0.00442
		All to SRS	0.00041	0.00100	0.00129	0.0054	0.00165
	NTS	Geographic	0.00040	0.00094	0.00130	0.0054	0.00158
		By Fuel	0.00041	0.00097	0.00133	0.0055	0.00162
		All to INEL	0.00054	0.00157	0.00096	0.0079	0.00439
		All to SRS	0.00041	0.00097	0.00133	0.0055	0.00162
	ORR	Geographic	0.00025	0.00063	0.00026	0.0039	0.00070
		By Fuel	0.00026	0.00065	0.00026	0.0040	0.00074
		All to INEL	0.00050	0.00136	0.00115	0.0068	0.00353
		All to SRS	0.00026	0.00065	0.00026	0.0040	0.00074

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-33 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Rail, Average Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.00007	0.00018	0.00228	0.0001	0.00023
1992/1993 Planning Basis	INEL/SRS		0.00007	0.00018	0.00228	0.0001	0.00023
Regionalization by Fuel Type	INEL/SRS		0.00007	0.00019	0.00295	0.0001	0.00024
Regionalization	INEL/SRS		0.00007	0.00018	0.00228	0.0001	0.00023
by	INEL/ORR	Geographic	0.00010	0.00019	0.00267	0.0002	0.00021
Geography		By Fuel	0.00010	0.00019	0.00270	0.0002	0.00022
		All to INEL	0.00010	0.00016	0.00217	0.0002	0.00085
	NTS/SRS	Geographic	0.00007	0.00018	0.00292	0.0001	0.00023
		By Fuel	0.00007	0.00019	0.00295	0.0001	0.00024
		All to SRS	0.00007	0.00019	0.00295	0.0001	0.00024
	NTS/ORR	Geographic	0.00014	0.00036	0.00269	0.0014	0.00022
		By Fuel	0.00010	0.00019	0.00270	0.0002	0.00022
		All to INEL	0.00022	0.00052	0.00253	0.0022	0.00086
		All to SRS	0.00010	0.00019	0.00270	0.0002	0.00022
	HS/SRS	Geographic	0.00007	0.00018	0.00292	0.0001	0.00023
		By Fuel	0.00007	0.00019	0.00295	0.0001	0.00024
		All to SRS	0.00007	0.00019	0.00295	0.0001	0.00024
	HS/ORR	Geographic	0.00014	0.00036	0.00269	0.0014	0.00022
		By Fuel	0.00010	0.00019	0.00270	0.0002	0.00022
		All to INEL	0.00019	0.00045	0.00227	0.0016	0.00085
		All to SRS	0.00010	0.00019	0.00270	0.0002	0.00022
Centralization	INEL		0.00012	0.00016	0.00239	0.0003	0.00106
	SRS		0.00007	0.00015	0.00295	0.0001	0.00024
	HS	Geographic	0.00050	0.00151	0.00337	0.0080	0.00047
		By Fuel	0.00020	0.00029	0.00381	0.0005	0.00046
		All to INEL	0.00021	0.00047	0.00251	0.0016	0.00107
		All to SRS	0.00020	0.00029	0.00381	0.0005	0.00046
	NTS	Geographic	0.00046	0.00131	0.00343	0.0077	0.00047
		By Fuel	0.00019	0.00028	0.00384	0.0004	0.00046
		All to INEL	0.00024	0.00054	0.00278	0.0022	0.00107
		All to SRS	0.00019	0.00028	0.00384	0.0004	0.00046
	ORR	Geographic	0.00014	0.00036	0.00270	0.0014	0.00022
		By Fuel	0.00010	0.00019	0.00270	0.0002	0.00022
		All to INEL	0.00041	0.00117	0.00253	0.0064	0.00087
		All to SRS	0.00010	0.00019	0.00270	0.0002	0.00022

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-34 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Truck, Lower Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.00021	0.00060	0.00013	0.0039	0.00068
1992/1993 Planning Basis	INEL/SRS		0.00021	0.00060	0.00013	0.0039	0.00068
Regionalization by Fuel Type	INEL/SRS		0.00022	0.00062	0.00014	0.0040	0.00070
Regionalization by Geography	INEL/SRS		0.00021	0.00060	0.00013	0.0039	0.00068
	INEL/ORR	Geographic	0.00028	0.00078	0.00025	0.0050	0.00095
		By Fuel	0.00029	0.00081	0.00026	0.0052	0.00098
		All to INEL	0.00041	0.00126	0.00058	0.0064	0.00341
	NTS/SRS	Geographic	0.00022	0.00060	0.00014	0.0039	0.00069
		By Fuel	0.00022	0.00062	0.00014	0.0040	0.00070
		All to SRS	0.00022	0.00062	0.00014	0.0040	0.00070
	NTS/ORR	Geographic	0.00029	0.00079	0.00026	0.0050	0.00096
		By Fuel	0.00029	0.00081	0.00026	0.0052	0.00098
		All to INEL	0.00053	0.00163	0.00093	0.0083	0.00358
		All to SRS	0.00029	0.00081	0.00026	0.0052	0.00098
	HS/SRS	Geographic	0.00022	0.00060	0.00013	0.0039	0.00069
		By Fuel	0.00022	0.00062	0.00014	0.0040	0.00070
		All to SRS	0.00022	0.00062	0.00014	0.0040	0.00070
	HS/ORR	Geographic	0.00029	0.00079	0.00025	0.0050	0.00096
		By Fuel	0.00029	0.00081	0.00026	0.0052	0.00098
		All to INEL	0.00050	0.00156	0.00067	0.0077	0.00356
		All to SRS	0.00029	0.00081	0.00026	0.0052	0.00098
Centralization	INEL		0.00048	0.00147	0.00072	0.0073	0.00426
	SRS		0.00022	0.00062	0.00014	0.0040	0.00070
	HS	Geographic	0.00071	0.00219	0.00085	0.0127	0.00369
		By Fuel	0.00073	0.00224	0.00087	0.0130	0.00375
		All to INEL	0.00058	0.00183	0.00084	0.0089	0.00443
		All to SRS	0.00073	0.00224	0.00087	0.0130	0.00375
	NTS	Geographic	0.00066	0.00197	0.00093	0.0125	0.00362
		By Fuel	0.00068	0.00201	0.00094	0.0128	0.00369
		All to INEL	0.00060	0.00188	0.00111	0.0095	0.00442
		All to SRS	0.00068	0.00201	0.00094	0.0128	0.00369
	ORR	Geographic	0.00029	0.00080	0.00026	0.0051	0.00100
		By Fuel	0.00029	0.00081	0.00026	0.0052	0.00098
		All to INEL	0.00072	0.00227	0.00093	0.0126	0.00528
		All to SRS	0.00029	0.00081	0.00026	0.0052	0.00098

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-35 Tabulation of Overland Transportation Risks: Accept Target Material Only, Shipments from Ports via Truck, Intersite via Rail, Lower Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radiological
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.00024	0.00061	0.00023	0.0038	0.00071
		By Fuel	0.00025	0.00063	0.00024	0.0039	0.00073
		All to INEL	0.00041	0.00126	0.00058	0.0064	0.00341
	NTS/SRS	Geographic	0.00022	0.00060	0.00014	0.0039	0.00068
		By Fuel	0.00022	0.00062	0.00014	0.0040	0.00070
		All to SRS	0.00022	0.00062	0.00014	0.0040	0.00070
	NTS/ORR	Geographic	0.00025	0.00061	0.00024	0.0038	0.00071
		By Fuel	0.00025	0.00063	0.00024	0.0039	0.00073
		All to INEL	0.00046	0.00129	0.00075	0.0065	0.00343
		All to SRS	0.00025	0.00063	0.00024	0.0039	0.00073
	HS/SRS	Geographic	0.00022	0.00060	0.00014	0.0039	0.00068
		By Fuel	0.00022	0.00062	0.00014	0.0040	0.00070
		All to SRS	0.00022	0.00062	0.00014	0.0040	0.00070
	HS/ORR	Geographic	0.00025	0.00061	0.00024	0.0038	0.00071
		By Fuel	0.00025	0.00063	0.00024	0.0039	0.00073
		All to INEL	0.00045	0.00129	0.00075	0.0064	0.00344
		All to SRS	0.00025	0.00063	0.00024	0.0039	0.00073
Centralization	INEL						
	SRS						
	HS	Geographic	0.00040	0.00096	0.00124	0.0052	0.00194
		By Fuel	0.00041	0.00098	0.00127	0.0053	0.00197
		All to INEL	0.00053	0.00156	0.00091	0.0076	0.00432
		All to SRS	0.00041	0.00098	0.00127	0.0053	0.00197
	NTS	Geographic	0.00039	0.00092	0.00129	0.0052	0.00188
		By Fuel	0.00040	0.00095	0.00131	0.0054	0.00190
		All to INEL	0.00053	0.00154	0.00093	0.0077	0.00426
		All to SRS	0.00040	0.00095	0.00131	0.0054	0.00190
	ORR	Geographic	0.00025	0.00061	0.00025	0.0038	0.00072
		By Fuel	0.00025	0.00063	0.00024	0.0039	0.00073
		All to INEL	0.00049	0.00132	0.00113	0.0066	0.00376
		All to SRS	0.00025	0.00063	0.00024	0.0039	0.00073

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-36 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Rail, Lower Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.00007	0.00018	0.00289	0.0001	0.00023
1992/1993 Planning Basis	INEL/SRS		0.00007	0.00018	0.00289	0.0001	0.00023
Regionalization by Fuel Type	INEL/SRS		0.00007	0.00018	0.00292	0.0001	0.00023
Regionalization by Geography	INEL/SRS		0.00007	0.00018	0.00289	0.0001	0.00023
	INEL/ORR	Geographic	0.00010	0.00019	0.00264	0.0002	0.00023
		By Fuel	0.00010	0.00019	0.00267	0.0002	0.00024
		All to INEL	0.00010	0.00015	0.00210	0.0002	0.00080
	NTS/SRS	Geographic	0.00007	0.00018	0.00289	0.0001	0.00023
		By Fuel	0.00007	0.00018	0.00292	0.0001	0.00023
		All to SRS	0.00007	0.00018	0.00292	0.0001	0.00023
	NTS/ORR	Geographic	0.00014	0.00036	0.00267	0.0014	0.00048
		By Fuel	0.00010	0.00019	0.00267	0.0002	0.00024
		All to INEL	0.00022	0.00052	0.00245	0.0022	0.00097
		All to SRS	0.00010	0.00019	0.00267	0.0002	0.00024
	HS/SRS	Geographic	0.00007	0.00018	0.00289	0.0001	0.00023
		By Fuel	0.00007	0.00018	0.00292	0.0001	0.00023
		All to SRS	0.00007	0.00018	0.00292	0.0001	0.00023
	HS/ORR	Geographic	0.00014	0.00036	0.00267	0.0014	0.00048
		By Fuel	0.00010	0.00019	0.00267	0.0002	0.00024
		All to INEL	0.00019	0.00045	0.00219	0.0016	0.00095
		All to SRS	0.00010	0.00019	0.00267	0.0002	0.00024
Centralization	INEL		0.00011	0.00017	0.00231	0.0003	0.00100
	SRS		0.00007	0.00018	0.00292	0.0001	0.00023
	HS	Geographic	0.00050	0.00150	0.00333	0.0080	0.00255
		By Fuel	0.00020	0.00028	0.00377	0.0005	0.00082
		All to INEL	0.00020	0.00046	0.00243	0.0016	0.00116
		All to SRS	0.00020	0.00028	0.00377	0.0005	0.00082
	NTS	Geographic	0.00046	0.00131	0.00339	0.0077	0.00251
		By Fuel	0.00019	0.00027	0.00380	0.0004	0.00079
		All to INEL	0.00023	0.00054	0.00269	0.0022	0.00117
		All to SRS	0.00019	0.00027	0.00380	0.0004	0.00079
	ORR	Geographic	0.00014	0.00036	0.00267	0.0014	0.00049
		By Fuel	0.00010	0.00019	0.00267	0.0002	0.00024
		All to INEL	0.00041	0.00116	0.00245	0.0064	0.00266
		All to SRS	0.00010	0.00019	0.00267	0.0002	0.00024

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-37 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Truck, Upper Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF &	SNF Site	Phase I	Radiological		Nonradiological		Radio-
INEL EIS Alternative	Option	Approach	Crew	Public	Emis.	Traffic	logical
Decentralization	INEL/SRS		0.00024	0.00066	0.00019	0.0042	0.00074
1992/1993 Planning Basis	INEL/SRS		0.00024	0.00066	0.00019	0.0042	0.00074
Regionalization by Fuel Type	INEL/SRS		0.00024	0.00068	0.00021	0.0044	0.00080
Regionalization	INEL/SRS		0.00024	0.00066	0.00019	0.0042	0.00074
by	INEL/ORR	Geographic	0.00030	0.00084	0.00030	0.0053	0.00100
Geography		By Fuel	0.00031	0.00086	0.00033	0.0055	0.00107
		All to INEL	0.00043	0.00131	0.00062	0.0067	0.00355
	NTS/SRS	Geographic	0.00024	0.00066	0.00019	0.0043	0.00074
		By Fuel	0.00024	0.00068	0.00021	0.0044	0.00080
		All to SRS	0.00024	0.00068	0.00021	0.0044	0.00080
	NTS/ORR	Geographic	0.00031	0.00084	0.00031	0.0054	0.00101
		By Fuel	0.00031	0.00086	0.00033	0.0055	0.00107
		All to INEL	0.00054	0.00167	0.00098	0.0086	0.00372
		All to SRS	0.00031	0.00086	0.00033	0.0055	0.00107
	HS/SRS	Geographic	0.00024	0.00066	0.00019	0.0042	0.00074
		By Fuel	0.00024	0.00068	0.00021	0.0044	0.00080
		All to SRS	0.00024	0.00068	0.00021	0.0044	0.00080
	HS/ORR	Geographic	0.00031	0.00084	0.00031	0.0054	0.00101
		By Fuel	0.00031	0.00086	0.00033	0.0055	0.00107
		All to INEL	0.00051	0.00160	0.00072	0.0081	0.00370
		All to SRS	0.00031	0.00086	0.00033	0.0055	0.00107
Centralization	INEL		0.00049	0.00152	0.00077	0.0076	0.00442
	SRS		0.00024	0.00068	0.00021	0.0044	0.00080
	HS	Geographic	0.00073	0.00224	0.00091	0.0131	0.00376
		By Fuel	0.00074	0.00229	0.00094	0.0134	0.00387
		All to INEL	0.00059	0.00187	0.00088	0.0093	0.00459
		All to SRS	0.00074	0.00229	0.00094	0.0134	0.00387
	NTS	Geographic	0.00068	0.00203	0.00098	0.0129	0.00371
		By Fuel	0.00070	0.00207	0.00101	0.0131	0.00381
		All to INEL	0.00062	0.00192	0.00116	0.0099	0.00459
		All to SRS	0.00070	0.00207	0.00101	0.0131	0.00381
	ORR	Geographic	0.00031	0.00086	0.00031	0.0055	0.00105
		By Fuel	0.00031	0.00086	0.00033	0.0055	0.00107
		All to INEL	0.00074	0.00231	0.00098	0.0129	0.00541
		All to SRS	0.00031	0.00086	0.00033	0.0055	0.00107

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-38 Tabulation of Overland Transportation Risks: Accept Target Material Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Upper Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by	INEL/ORR	Geographic	0.00026	0.00067	0.00028	0.0041	0.00076
Geography		By Fuel	0.00027	0.00069	0.00031	0.0043	0.00082
		All to INEL	0.00043	0.00131	0.00062	0.0067	0.00355
	NTS/SRS	Geographic	0.00024	0.00066	0.00019	0.0042	0.00074
		By Fuel	0.00024	0.00068	0.00021	0.0044	0.00080
		All to SRS	0.00024	0.00068	0.00021	0.0044	0.00080
	NTS/ORR	Geographic	0.00027	0.00067	0.00029	0.0041	0.00076
		By Fuel	0.00027	0.00069	0.00031	0.0043	0.00082
		All to INEL	0.00047	0.00133	0.00080	0.0068	0.00357
		All to SRS	0.00027	0.00069	0.00031	0.0043	0.00082
	HS/SRS	Geographic	0.00024	0.00066	0.00019	0.0042	0.00074
		By Fuel	0.00024	0.00068	0.00021	0.0044	0.00080
		All to SRS	0.00024	0.00068	0.00021	0.0044	0.00080
	HS/ORR	Geographic	0.00027	0.00067	0.00029	0.0041	0.00076
		By Fuel	0.00027	0.00069	0.00031	0.0043	0.00082
		All to INEL	0.00047	0.00133	0.00080	0.0068	0.00358
		All to SRS	0.00027	0.00069	0.00031	0.0043	0.00082
Centralization	INEL						
	SRS						
	HS	Geographic	0.00042	0.00101	0.00129	0.0055	0.00201
		By Fuel	0.00043	0.00104	0.00133	0.0057	0.00208
		All to INEL	0.00055	0.00160	0.00096	0.0080	0.00448
		All to SRS	0.00043	0.00104	0.00133	0.0057	0.00208
	NTS	Geographic	0.00041	0.00098	0.00134	0.0056	0.00196
		By Fuel	0.00042	0.00100	0.00138	0.0058	0.00203
		All to INEL	0.00054	0.00158	0.00098	0.0081	0.00444
		All to SRS	0.00042	0.00100	0.00138	0.0058	0.00203
	ORR	Geographic	0.00027	0.00067	0.00030	0.0041	0.00077
		By Fuel	0.00027	0.00069	0.00031	0.0043	0.00082
		All to INEL	0.00051	0.00137	0.00118	0.0069	0.00390
		All to SRS	0.00027	0.00069	0.00031	0.0043	0.00082

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-39 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Rail, Upper Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.00008	0.00019	0.00301	0.0002	0.00025
1992/1993 Planning Basis	INEL/SRS		0.00008	0.00019	0.00301	0.0002	0.00025
Regionalization by Fuel Type	INEL/SRS		0.00008	0.00019	0.00305	0.0002	0.00026
Regionalization by Geography	INEL/SRS		0.00008	0.00019	0.00301	0.0002	0.00025
	INEL/ORR	Geographic	0.00010	0.00019	0.00276	0.0002	0.00025
		By Fuel	0.00011	0.00020	0.00280	0.0002	0.00026
		All to INEL	0.00010	0.00016	0.00242	0.0002	0.00089
	NTS/SRS	Geographic	0.00008	0.00019	0.00302	0.0002	0.00025
		By Fuel	0.00008	0.00019	0.00305	0.0002	0.00026
		All to SRS	0.00008	0.00019	0.00305	0.0002	0.00026
	NTS/ORR	Geographic	0.00015	0.00037	0.00278	0.0014	0.00050
		By Fuel	0.00011	0.00020	0.00280	0.0002	0.00026
		All to INEL	0.00022	0.00053	0.00278	0.0022	0.00106
		All to SRS	0.00011	0.00020	0.00280	0.0002	0.00026
	HS/SRS	Geographic	0.00008	0.00019	0.00302	0.0002	0.00025
		By Fuel	0.00008	0.00019	0.00305	0.0002	0.00026
		All to SRS	0.00008	0.00019	0.00305	0.0002	0.00026
	HS/ORR	Geographic	0.00015	0.00037	0.00278	0.0014	0.00050
		By Fuel	0.00011	0.00020	0.00280	0.0002	0.00026
		All to INEL	0.00019	0.00045	0.00252	0.0016	0.00104
		All to SRS	0.00011	0.00020	0.00280	0.0002	0.00026
Centralization	INEL		0.00012	0.00018	0.00268	0.0003	0.00111
	SRS		0.00008	0.00019	0.00305	0.0002	0.00026
	HS	Geographic	0.00051	0.00151	0.00352	0.0080	0.00258
		By Fuel	0.00020	0.00029	0.00396	0.0005	0.00086
		All to INEL	0.00021	0.00047	0.00282	0.0016	0.00127
		All to SRS	0.00020	0.00029	0.00396	0.0005	0.00086
	NTS	Geographic	0.00046	0.00132	0.00354	0.0077	0.00255
		By Fuel	0.00019	0.00028	0.00396	0.0005	0.00082
		All to INEL	0.00024	0.00055	0.00306	0.0022	0.00128
		All to SRS	0.00019	0.00028	0.00396	0.0005	0.00082
	ORR	Geographic	0.00015	0.00037	0.00279	0.0014	0.00050
		By Fuel	0.00011	0.00020	0.00280	0.0002	0.00026
		All to INEL	0.00041	0.00117	0.00278	0.0064	0.00275
		All to SRS	0.00011	0.00020	0.00280	0.0002	0.00026

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-40 Potential Consequences for the Most Severe Accidents Involving Shipments of Target Material[a,b]

Mode and Accident Location	Neutral Conditions[c]				Stable Conditions[d]			
	Population[e]		MEI[f]		Population[e]		MEI[f]	
	Dose (person-rem)	Consequences (LCF)	Dose (rem)	Consequences (LCF)	Dose (person-rem)	Consequences (LCF)	Dose (person-rem)	Consequences (LCF)
Truck:								
Urban	206	0.1	0.15	0.000074	1650	0.83	0.50	0.00025
Suburban	38.3	0.019	0.15	0.000074	307	0.15	0.50	0.00025
Rural	0.70	0.00035	0.15	0.000074	5.5	0.0028	0.50	0.00025
Rail:								
Urban	206	0.1	0.15	0.000074	1650	0.83	0.50	0.00025
Suburban	38.3	0.19	0.15	0.000074	307	0.15	0.50	0.00025
Rural	0.70	0.00035	0.15	0.000074	5.5	0.0028	0.50	0.00025

[a] *The most severe accidents correspond to modal study accident severity category 6 (DOE,1994b).*

[b] *Buoyant plume rise resulting from fire for a severe accident was included in the exposure model.*

[c] *Neutral weather conditions result in moderate dispersion and dilution of the release plume. Neutral conditions were taken to be Pasquill stability Class D with a wind speed of 4 m per sec (9 mph). Neutral conditions occur approximately 50 percent of the time in the United States.*

[d] *Stable weather conditions result in minimal dispersion and dilution of the release plume and are thus unfavorable. Stable conditions were taken to be Pasquill stability Class F with a wind speed of 1 m per sec (2.2 mph). Stable conditions occur approximately one-third of the time in the United States.*

[e] *Populations extend at a uniform population density to a radius of 80 km (50 mi) from the accident site. Population exposure pathways include acute inhalation, acute cloudshine, groundshine, resuspended inhalation, resuspended cloudshine, and ingestion of food, including initially contaminated food (rural only). No decontamination or mitigative actions are taken.*

[f] *The MEI is assumed to be at the location of maximum exposure. The locations of maximum exposure would be 160 m (528 ft) and 400 m (1,320 ft) from the accident site under neutral and stable atmospheric conditions, respectively. Individual exposure pathways include acute inhalation, acute cloudshine, and groundshine during passage of the plume. No ingested dose is considered.*

E.8.5 Implementation Alternative - Implementing an Acceptance Policy for Varying Durations - Indefinite HEU Acceptance

Since most LEU would come back within 10 years and spent nuclear fuel produced from the indefinite operation of HEU reactors is difficult to predict, it is reasonable to assume that the analysis for the basic implementation applies closely.

E.8.6 Implementation Alternative - Implementing an Acceptance Policy with Varying Financial Approaches

None of the financial approaches would have a significant effect on overland transportation. The effects calculated for the basic implementation adequately model this strategy.

Table E-41 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Truck, Average Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.015	0.045	0.002	0.028	0.000013
1992/1993 Planning Basis	INEL/SRS		0.015	0.045	0.002	0.028	0.000013
Regionalization by Fuel Type	INEL/SRS		0.030	0.093	0.004	0.056	0.000032
Regionalization	INEL/SRS		0.015	0.045	0.002	0.028	0.000013
by	INEL/ORR	Geographic	0.017	0.051	0.002	0.032	0.000046
Geography		By Fuel	0.032	0.099	0.005	0.060	0.000066
		All to INEL	0.050	0.157	0.008	0.090	0.000055
	NTS/SRS	Geographic	0.016	0.049	0.002	0.030	0.000022
		By Fuel	0.031	0.096	0.005	0.058	0.000041
		All to SRS	0.027	0.082	0.004	0.050	0.000029
	NTS/ORR	Geographic	0.018	0.055	0.003	0.034	0.000055
		By Fuel	0.033	0.102	0.005	0.062	0.000074
		All to INEL	0.054	0.171	0.009	0.097	0.000092
		All to SRS	0.030	0.090	0.005	0.055	0.000072
	HS/SRS	Geographic	0.016	0.048	0.002	0.030	0.000016
		By Fuel	0.031	0.095	0.005	0.057	0.000035
		All to SRS	0.027	0.082	0.004	0.050	0.000029
	HS/ORR	Geographic	0.018	0.054	0.002	0.033	0.000049
		By Fuel	0.033	0.101	0.005	0.061	0.000068
		All to INEL	0.053	0.168	0.008	0.095	0.000065
		All to SRS	0.030	0.090	0.005	0.055	0.000072
Centralization	INEL		0.050	0.157	0.008	0.090	0.000055
	SRS		0.027	0.082	0.004	0.050	0.000029
	HS	Geographic	0.029	0.088	0.004	0.053	0.000133
		By Fuel	0.044	0.136	0.006	0.081	0.000153
		All to INEL	0.053	0.168	0.008	0.095	0.000065
		All to SRS	0.044	0.134	0.006	0.081	0.000183
	NTS	Geographic	0.028	0.083	0.004	0.053	0.000152
		By Fuel	0.043	0.130	0.007	0.081	0.000172
		All to INEL	0.054	0.171	0.009	0.097	0.000092
		All to SRS	0.042	0.127	0.006	0.080	0.000199
	ORR	Geographic	0.020	0.060	0.003	0.038	0.000071
		By Fuel	0.035	0.108	0.005	0.065	0.000090
		All to INEL	0.061	0.195	0.009	0.114	0.000162
		All to SRS	0.030	0.090	0.005	0.055	0.000072

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-42 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Average Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.016	0.046	0.002	0.028	0.000017
		By Fuel	0.031	0.093	0.005	0.056	0.000036
		All to INEL	0.050	0.157	0.008	0.090	0.000055
	NTS/SRS	Geographic	0.015	0.046	0.002	0.028	0.000014
		By Fuel	0.030	0.093	0.004	0.056	0.000033
		All to SRS	0.027	0.082	0.004	0.050	0.000029
	NTS/ORR	Geographic	0.016	0.046	0.002	0.028	0.000017
		By Fuel	0.031	0.093	0.005	0.056	0.000037
		All to INEL	0.050	0.157	0.008	0.090	0.000058
		All to SRS	0.028	0.083	0.004	0.050	0.000033
	HS/SRS	Geographic	0.015	0.046	0.002	0.028	0.000014
		By Fuel	0.030	0.093	0.004	0.056	0.000033
		All to SRS	0.027	0.082	0.004	0.050	0.000029
	HS/ORR	Geographic	0.016	0.046	0.002	0.028	0.000018
		By Fuel	0.031	0.093	0.005	0.056	0.000037
		All to INEL	0.050	0.157	0.008	0.090	0.000059
		All to SRS	0.028	0.083	0.004	0.050	0.000033
Centralization	INEL						
	SRS						
	HS	Geographic	0.017	0.047	0.003	0.029	0.000039
		By Fuel	0.032	0.094	0.006	0.057	0.000058
		All to INEL	0.050	0.157	0.008	0.090	0.000059
		All to SRS	0.029	0.084	0.006	0.050	0.000061
	NTS	Geographic	0.017	0.047	0.003	0.029	0.000037
		By Fuel	0.032	0.094	0.006	0.056	0.000057
		All to INEL	0.050	0.157	0.008	0.090	0.000058
		All to SRS	0.029	0.084	0.006	0.050	0.000060
	ORR	Geographic	0.016	0.046	0.002	0.028	0.000022
		By Fuel	0.031	0.093	0.005	0.056	0.000041
		All to INEL	0.051	0.158	0.008	0.090	0.000076
		All to SRS	0.028	0.083	0.004	0.050	0.000033

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-43 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Rail, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.006	0.007	0.009	0.001	0.000005
1992/1993 Planning Basis	INEL/SRS		0.006	0.007	0.009	0.001	0.000005
Regionalization by Fuel Type	INEL/SRS		0.009	0.011	0.015	0.002	0.000010
Regionalization	INEL/SRS		0.006	0.007	0.009	0.001	0.000005
by	INEL/ORR	Geographic	0.007	0.007	0.009	0.001	0.000008
Geography		By Fuel	0.010	0.012	0.015	0.002	0.000013
		All to INEL	0.013	0.013	0.018	0.003	0.000013
	NTS/SRS	Geographic	0.006	0.007	0.009	0.001	0.000006
		By Fuel	0.009	0.011	0.015	0.002	0.000011
		All to SRS	0.009	0.011	0.014	0.002	0.000010
	NTS/ORR	Geographic	0.009	0.013	0.009	0.005	0.000039
		By Fuel	0.011	0.015	0.015	0.004	0.000022
		All to INEL	0.017	0.027	0.020	0.010	0.000050
		All to SRS	0.009	0.011	0.014	0.002	0.000014
	HS/SRS	Geographic	0.006	0.007	0.009	0.001	0.000006
		By Fuel	0.009	0.011	0.015	0.002	0.000011
		All to SRS	0.009	0.011	0.014	0.002	0.000010
	HS/ORR	Geographic	0.009	0.013	0.009	0.005	0.000039
		By Fuel	0.010	0.014	0.015	0.003	0.000016
		All to INEL	0.016	0.025	0.019	0.008	0.000023
		All to SRS	0.009	0.011	0.014	0.002	0.000014
Centralization	INEL		0.013	0.013	0.018	0.003	0.000013
	SRS		0.009	0.011	0.014	0.002	0.000010
	HS	Geographic	0.019	0.047	0.010	0.024	0.000124
		By Fuel	0.011	0.015	0.016	0.003	0.000037
		All to INEL	0.016	0.025	0.019	0.008	0.000023
		All to SRS	0.010	0.013	0.015	0.002	0.000042
	NTS	Geographic	0.018	0.041	0.010	0.024	0.000136
		By Fuel	0.012	0.016	0.016	0.004	0.000042
		All to INEL	0.017	0.027	0.020	0.010	0.000050
		All to SRS	0.010	0.012	0.015	0.002	0.000040
	ORR	Geographic	0.009	0.013	0.009	0.005	0.000043
		By Fuel	0.012	0.020	0.015	0.007	0.000038
		All to INEL	0.025	0.052	0.020	0.027	0.000120
		All to SRS	0.009	0.011	0.014	0.002	0.000014

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-44 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Truck, Lower Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.011	0.033	0.001	0.019	0.000005
1992/1993 Planning Basis	INEL/SRS		0.011	0.033	0.001	0.019	0.000005
Regionalization by Fuel Type	INEL/SRS		0.025	0.078	0.003	0.044	0.000010
Regionalization by Geography	INEL/SRS		0.011	0.033	0.001	0.019	0.000005
	INEL/ORR	Geographic	0.013	0.039	0.001	0.023	0.000038
		By Fuel	0.027	0.084	0.003	0.048	0.000043
		All to INEL	0.041	0.131	0.006	0.073	0.000018
	NTS/SRS	Geographic	0.012	0.036	0.001	0.021	0.000014
		By Fuel	0.026	0.081	0.003	0.046	0.000019
		All to SRS	0.023	0.071	0.002	0.039	0.000010
	NTS/ORR	Geographic	0.014	0.042	0.002	0.025	0.000047
		By Fuel	0.029	0.087	0.003	0.049	0.000052
		All to INEL	0.046	0.145	0.007	0.081	0.000055
		All to SRS	0.026	0.079	0.003	0.044	0.000053
	HS/SRS	Geographic	0.012	0.035	0.001	0.021	0.000008
		By Fuel	0.026	0.081	0.003	0.045	0.000013
		All to SRS	0.023	0.071	0.002	0.039	0.000010
	HS/ORR	Geographic	0.014	0.041	0.001	0.025	0.000041
		By Fuel	0.028	0.087	0.003	0.049	0.000046
		All to INEL	0.045	0.143	0.006	0.078	0.000028
		All to SRS	0.026	0.079	0.003	0.044	0.000053
Centralization	INEL		0.041	0.131	0.006	0.073	0.000018
	SRS		0.023	0.071	0.002	0.039	0.000010
	HS	Geographic	0.024	0.075	0.003	0.044	0.000125
		By Fuel	0.039	0.121	0.004	0.069	0.000131
		All to INEL	0.045	0.143	0.006	0.078	0.000028
		All to SRS	0.040	0.123	0.004	0.070	0.000164
	NTS	Geographic	0.023	0.070	0.003	0.044	0.000144
		By Fuel	0.038	0.116	0.005	0.068	0.000149
		All to INEL	0.046	0.145	0.007	0.081	0.000055
		All to SRS	0.038	0.115	0.005	0.068	0.000180
	ORR	Geographic	0.016	0.048	0.002	0.029	0.000064
		By Fuel	0.030	0.093	0.003	0.053	0.000068
		All to INEL	0.053	0.170	0.007	0.097	0.000125
		All to SRS	0.026	0.079	0.003	0.044	0.000053

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-45 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments from Ports via Truck, Intersite Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.011	0.033	0.001	0.020	0.000009
		By Fuel	0.026	0.078	0.003	0.044	0.000014
		All to INEL	0.041	0.131	0.006	0.073	0.000018
	NTS/SRS	Geographic	0.011	0.033	0.001	0.020	0.000006
		By Fuel	0.025	0.078	0.003	0.044	0.000011
		All to SRS	0.023	0.071	0.002	0.039	0.000010
	NTS/ORR	Geographic	0.011	0.033	0.001	0.020	0.000009
		By Fuel	0.026	0.078	0.003	0.044	0.000014
		All to INEL	0.042	0.132	0.006	0.073	0.000021
		All to SRS	0.024	0.071	0.002	0.039	0.000014
	HS/SRS	Geographic	0.011	0.033	0.001	0.020	0.000006
		By Fuel	0.025	0.078	0.003	0.044	0.000011
		All to SRS	0.023	0.071	0.002	0.039	0.000010
	HS/ORR	Geographic	0.011	0.033	0.001	0.020	0.000010
		By Fuel	0.026	0.078	0.003	0.044	0.000015
		All to INEL	0.042	0.132	0.006	0.073	0.000022
		All to SRS	0.024	0.071	0.002	0.039	0.000014
Centralization	INEL						
	SRS						
	HS	Geographic	0.012	0.034	0.002	0.020	0.000031
		By Fuel	0.027	0.079	0.004	0.044	0.000036
		All to INEL	0.042	0.132	0.006	0.073	0.000022
		All to SRS	0.025	0.073	0.004	0.039	0.000042
	NTS	Geographic	0.012	0.034	0.002	0.020	0.000030
		By Fuel	0.027	0.079	0.004	0.044	0.000035
		All to INEL	0.042	0.132	0.006	0.073	0.000021
		All to SRS	0.025	0.072	0.004	0.039	0.000041
	ORR	Geographic	0.012	0.033	0.001	0.020	0.000014
		By Fuel	0.026	0.079	0.003	0.044	0.000019
		All to INEL	0.042	0.132	0.007	0.074	0.000039
		All to SRS	0.024	0.071	0.002	0.039	0.000014

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-46 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.005	0.005	0.006	0.001	0.000003
1992/1993 Planning Basis	INEL/SRS		0.005	0.005	0.006	0.001	0.000003
Regionalization by Fuel Type	INEL/SRS		0.008	0.009	0.011	0.001	0.000004
Regionalization by Geography	INEL/SRS		0.005	0.005	0.006	0.001	0.000003
	INEL/ORR	Geographic	0.006	0.006	0.007	0.001	0.000006
		By Fuel	0.008	0.009	0.011	0.002	0.000007
		All to INEL	0.011	0.009	0.012	0.002	0.000003
	NTS/SRS	Geographic	0.005	0.006	0.006	0.001	0.000004
		By Fuel	0.008	0.009	0.011	0.002	0.000004
		All to SRS	0.008	0.009	0.011	0.001	0.000004
	NTS/ORR	Geographic	0.007	0.011	0.007	0.004	0.000037
		By Fuel	0.009	0.012	0.012	0.003	0.000016
		All to INEL	0.015	0.023	0.013	0.010	0.000040
		All to SRS	0.008	0.009	0.011	0.001	0.000008
	HS/SRS	Geographic	0.005	0.006	0.006	0.001	0.000004
		By Fuel	0.008	0.009	0.011	0.002	0.000005
		All to SRS	0.008	0.009	0.011	0.001	0.000004
	HS/ORR	Geographic	0.007	0.011	0.007	0.004	0.000037
		By Fuel	0.009	0.012	0.011	0.003	0.000009
		All to INEL	0.014	0.020	0.012	0.008	0.000013
		All to SRS	0.008	0.009	0.011	0.001	0.000008
Centralization	INEL		0.011	0.009	0.012	0.002	0.000003
	SRS		0.008	0.009	0.011	0.001	0.000004
	HS	Geographic	0.018	0.045	0.008	0.024	0.000122
		By Fuel	0.010	0.013	0.012	0.003	0.000031
		All to INEL	0.014	0.020	0.012	0.008	0.000013
		All to SRS	0.009	0.011	0.012	0.002	0.000036
	NTS	Geographic	0.017	0.040	0.008	0.023	0.000133
		By Fuel	0.010	0.013	0.013	0.004	0.000036
		All to INEL	0.015	0.023	0.013	0.010	0.000040
		All to SRS	0.009	0.010	0.013	0.002	0.000035
	ORR	Geographic	0.008	0.011	0.007	0.004	0.000041
		By Fuel	0.011	0.018	0.012	0.007	0.000032
		All to INEL	0.023	0.048	0.013	0.026	0.000110
		All to SRS	0.008	0.009	0.011	0.001	0.000008

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-47 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Truck, Upper Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.027	0.078	0.005	0.046	0.00006
1992/1993 Planning Basis	INEL/SRS		0.027	0.078	0.005	0.046	0.00006
Regionalization by Fuel Type	INEL/SRS		0.039	0.118	0.008	0.073	0.00009
Regionalization by Geography	INEL/SRS		0.027	0.078	0.005	0.046	0.00006
	INEL/ORR	Geographic	0.029	0.083	0.006	0.050	0.00009
		By Fuel	0.041	0.124	0.009	0.077	0.00012
		All to INEL	0.052	0.165	0.010	0.101	0.00011
	NTS/SRS	Geographic	0.028	0.081	0.006	0.048	0.00006
		By Fuel	0.040	0.121	0.009	0.075	0.00010
		All to SRS	0.038	0.113	0.008	0.068	0.00009
	NTS/ORR	Geographic	0.030	0.087	0.006	0.052	0.00010
		By Fuel	0.042	0.127	0.009	0.078	0.00013
		All to INEL	0.057	0.179	0.011	0.109	0.00015
		All to SRS	0.041	0.120	0.009	0.073	0.00013
	HS/SRS	Geographic	0.027	0.080	0.005	0.047	0.00006
		By Fuel	0.040	0.121	0.008	0.074	0.00009
		All to SRS	0.038	0.113	0.008	0.068	0.00009
	HS/ORR	Geographic	0.030	0.086	0.006	0.051	0.00009
		By Fuel	0.042	0.126	0.009	0.078	0.00013
		All to INEL	0.056	0.176	0.010	0.107	0.00012
		All to SRS	0.041	0.120	0.009	0.073	0.00013
Centralization	INEL		0.052	0.165	0.010	0.101	0.00011
	SRS		0.038	0.113	0.008	0.068	0.00009
	HS	Geographic	0.040	0.120	0.007	0.071	0.00018
		By Fuel	0.053	0.161	0.010	0.098	0.00021
		All to INEL	0.056	0.176	0.010	0.107	0.00012
		All to SRS	0.054	0.165	0.011	0.099	0.00024
	NTS	Geographic	0.039	0.115	0.007	0.071	0.00019
		By Fuel	0.052	0.156	0.010	0.097	0.00023
		All to INEL	0.057	0.179	0.011	0.109	0.00015
		All to SRS	0.053	0.157	0.011	0.098	0.00026
	ORR	Geographic	0.032	0.093	0.006	0.056	0.00011
		By Fuel	0.044	0.133	0.009	0.082	0.00015
		All to INEL	0.064	0.203	0.011	0.125	0.00022
		All to SRS	0.041	0.120	0.009	0.073	0.00013

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-48 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments from Ports via Truck, Intersite Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.040	0.118	0.008	0.073	0.00009
		All to INEL	0.052	0.165	0.010	0.101	0.00011
	NTS/SRS	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.039	0.118	0.008	0.073	0.00009
		All to SRS	0.038	0.113	0.008	0.068	0.00009
	NTS/ORR	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.040	0.118	0.008	0.073	0.00009
		All to INEL	0.053	0.165	0.010	0.102	0.00012
		All to SRS	0.038	0.113	0.009	0.069	0.00009
	HS/SRS	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.039	0.118	0.008	0.073	0.00009
		All to SRS	0.038	0.113	0.008	0.068	0.00009
	HS/ORR	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.040	0.118	0.008	0.073	0.00009
		All to INEL	0.053	0.165	0.010	0.102	0.00012
		All to SRS	0.038	0.113	0.009	0.069	0.00009
Centralization	INEL						
	SRS						
	HS	Geographic	0.028	0.079	0.006	0.047	0.00008
		By Fuel	0.041	0.119	0.009	0.073	0.00012
		All to INEL	0.053	0.165	0.010	0.102	0.00012
		All to SRS	0.040	0.114	0.010	0.069	0.00012
	NTS	Geographic	0.028	0.079	0.006	0.047	0.00008
		By Fuel	0.041	0.119	0.009	0.073	0.00011
		All to INEL	0.053	0.165	0.010	0.102	0.00012
		All to SRS	0.040	0.114	0.010	0.069	0.00012
	ORR	Geographic	0.027	0.078	0.006	0.046	0.00006
		By Fuel	0.040	0.118	0.009	0.073	0.00010
		All to INEL	0.054	0.165	0.011	0.102	0.00014
		All to SRS	0.038	0.113	0.009	0.069	0.00009

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-49 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.008	0.012	0.016	0.002	0.00002
1992/1993 Planning Basis	INEL/SRS		0.008	0.012	0.016	0.002	0.00002
Regionalization by Fuel Type	INEL/SRS		0.011	0.016	0.026	0.002	0.00003
Regionalization by Geography	INEL/SRS		0.008	0.012	0.016	0.002	0.00002
	INEL/ORR	Geographic	0.009	0.013	0.016	0.002	0.00002
		By Fuel	0.011	0.016	0.026	0.002	0.00004
		All to INEL	0.013	0.016	0.039	0.003	0.00004
	NTS/SRS	Geographic	0.009	0.012	0.016	0.002	0.00002
		By Fuel	0.011	0.016	0.026	0.002	0.00003
		All to SRS	0.011	0.017	0.022	0.002	0.00003
	NTS/ORR	Geographic	0.011	0.018	0.016	0.005	0.00005
		By Fuel	0.012	0.020	0.026	0.004	0.00004
		All to INEL	0.018	0.030	0.041	0.011	0.00008
		All to SRS	0.011	0.017	0.022	0.002	0.00004
	HS/SRS	Geographic	0.008	0.012	0.016	0.002	0.00002
		By Fuel	0.011	0.016	0.026	0.002	0.00003
		All to SRS	0.011	0.017	0.022	0.002	0.00003
	HS/ORR	Geographic	0.011	0.018	0.016	0.005	0.00005
		By Fuel	0.012	0.019	0.026	0.004	0.00004
		All to INEL	0.017	0.027	0.040	0.008	0.00005
		All to SRS	0.011	0.017	0.022	0.002	0.00004
Centralization	INEL		0.013	0.016	0.039	0.003	0.00004
	SRS		0.011	0.017	0.022	0.002	0.00003
	HS	Geographic	0.021	0.052	0.017	0.025	0.00014
		By Fuel	0.013	0.020	0.027	0.004	0.00006
		All to INEL	0.017	0.027	0.040	0.008	0.00005
		All to SRS	0.012	0.018	0.023	0.003	0.00006
	NTS	Geographic	0.020	0.046	0.017	0.024	0.00015
		By Fuel	0.013	0.020	0.027	0.004	0.00007
		All to INEL	0.018	0.030	0.041	0.011	0.00008
		All to SRS	0.012	0.018	0.023	0.003	0.00006
	ORR	Geographic	0.011	0.018	0.016	0.005	0.00006
		By Fuel	0.014	0.025	0.026	0.008	0.00006
		All to INEL	0.025	0.055	0.041	0.027	0.00015
		All to SRS	0.011	0.017	0.022	0.002	0.00004

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

E.8.7 Implementation Alternative - Implementing an Acceptance Policy by Taking Title at Varying Locations

The agency that has title to the spent nuclear fuel has no significant effect on overland transportation. The effects calculated for the basic implementation apply here.

E.8.8 Implementation Alternative - Implementing an Acceptance Policy and Storing Underwater

The use of underwater storage would have only minor effects on the location to which foreign research reactor spent nuclear fuel were delivered on the DOE sites. However, since there is some degree of uncertainty in the exact delivery location on all the DOE sites and intrasite transportation would be less likely, the effects calculated for the basic implementation apply here.

E.8.9 Implementation Alternative - Implementing an Acceptance Policy and Near-Term Chemical Separation in the United States

The performance of conventional or alternative chemical separation is only considered feasible at the Idaho National Engineering Laboratory and Savannah River Site sites. The requirements for overland transportation are not affected by the activities at the sites. Therefore, the impacts calculated in Section E.7 for the options to transport fuel to Idaho National Engineering Laboratory and/or Savannah River Site under the Regionalization by Fuel Type or Centralization alternatives would apply to this section. They are shown in Table E-50.

Table E-50 Tabulation of Overland Transportation Risks: Chemical Separation in the United States

Alternative/Option			*Incident-free*				*Accidental*
			Radiological		*Nonradiological*		
Implementation	*Mode*	*Risk Factors*	*Crew*	*Public*	*Emis.*	*Traffic*	*Radiological*
Regionalization by Fuel Type	Truck	Upper	0.048	0.143	0.010	0.088	0.000109
		Nominal	0.036	0.112	0.005	0.067	0.000039
		Lower	0.030	0.093	0.003	0.052	0.000012
	Rail	Upper	0.013	0.020	0.031	0.003	0.000040
		Nominal	0.011	0.014	0.018	0.002	0.000012
		Lower	0.010	0.011	0.014	0.002	0.000005
Centralization to Idaho National Engineering Laboratory	Truck	Upper	0.065	0.205	0.012	0.126	0.000143
		Nominal	0.062	0.195	0.009	0.112	0.000069
		Lower	0.051	0.163	0.007	0.091	0.000023
	Rail	Upper	0.016	0.020	0.049	0.004	0.000053
		Nominal	0.016	0.016	0.023	0.004	0.000016
		Lower	0.013	0.011	0.015	0.003	0.000004
Centralization to Savannah River Site	Truck	Upper	0.046	0.137	0.010	0.083	0.000107
		Nominal	0.033	0.097	0.005	0.062	0.000035
		Lower	0.028	0.085	0.003	0.047	0.000012
	Rail	Upper	0.013	0.020	0.027	0.003	0.000038
		Nominal	0.011	0.013	0.017	0.002	0.000012
		Lower	0.009	0.011	0.013	0.002	0.000005

All risks are expressed in latent cancer fatalities during the foreign research reactor spent nuclear fuel policy, except for the Accidental - Traffic column, which is the number of fatalities during the policy.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.020 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates is caused by two factors: 1) the option of using truck or rail to transport spent nuclear fuel, and, 2) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.009 to 0.065. The estimated number of radiation-related LCFs for the general population ranged from 0.011 to 0.21, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.003 to 0.05.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000004 to 0.00014 LCFs from radiation and from 0.002 to 0.13 for traffic fatality, depending on the transportation mode and DOE sites selected. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Table E-50. The analysis for this implementation alternative is analogous to the analysis performed for the basic implementation of Management Alternative 1 (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2.

The consequences of the most severe accident hypothesized are the same as described for the Basic Implementation since the material at risk is the same.

E.8.10 Developmental Processing Capabilities

The overland transportation impacts would be based on the site selected for processing, and would be determined after that site is selected.

E.8.11 Management Alternative - Adopt a Strategy of Managing Foreign Research Reactor Spent Nuclear Fuel Overseas: Store Overseas

There would be no overland transportation impacts in the United States if this alternative were implemented.

E.8.12 Policy Alternative - Adopt a Strategy of Managing Foreign Research Reactor Spent Nuclear Fuel Overseas: Process Overseas and Ship Vitrified High-Level Waste to the United States

The total amount of foreign research reactor spent nuclear fuel could be reduced into 16 vitrified waste logs, which could be carried in 8 casks. The contents of each cask is described isotopically in Table E-3. The curie content is based on the total number of curies expected to be returned to the United States under the basic implementation of Management Alternative 1. Realistically, the logs might have to be allowed to decay at the vitrification facility until the dose rate was below the regulatory-limit. Therefore, all incident-free calculations assume the dose rate is 10 mrem per hr at 2 m (6.6 ft).

This alternative is assumed to be independent of the SNF&INEL Final EIS (DOE, 1995) results. The only site considered for interim storage of vitrified high-level waste is the Savannah River Site. The only overseas facilities are in Europe, so all shipments are assumed to arrive on the east coast.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.0002 to 0.004 over the entire duration of the program. These results are the sum of the estimated number of radiation-related LCFs to the public and the crews.

The range of fatality estimates is caused by the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.00014 to 0.001. The estimated number of radiation-related LCFs for the general population ranged from 0.00009 to 0.003, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.0005.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 1.9×10^{-7} to 5.9×10^{-6} LCF from radiation and from 0.00003 to 0.002 for traffic fatality, depending on the transportation mode and DOE sites selected. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Tables E-51 and E-52.

Table E-51 Tabulation of Ground Transportation Risks: Vitrified High-Level Waste Acceptance Only

Alternative/Option			*Incident-free*				*Accidental*
			Radiological		*Nonradiological*		
Implementation	*Mode*	*Risk Factors*	*Crew*	*Public*	*Emis.*	*Traffic*	*Radiological*
Ship directly to repository	Truck	Upper	0.00076	0.00240	0.00016	0.00162	5.9×10^{-6}
		Nominal	0.00072	0.00227	0.00013	0.00143	5.0×10^{-6}
		Lower	0.00053	0.00172	0.00010	0.00106	1.7×10^{-6}
	Rail	Upper	0.00020	0.00024	0.00052	0.00005	2.0×10^{-6}
		Nominal	0.00019	0.00019	0.00025	0.00005	1.2×10^{-6}
		Lower	0.00014	0.00009	0.00015	0.00003	1.9×10^{-7}
Ship to Savannah River Site, then to repository	Truck	Upper	0.00102	0.00302	0.00018	0.00196	1.0×10^{-5}
		Nominal	0.00083	0.00249	0.00013	0.00168	7.6×10^{-6}
		Lower	0.00076	0.00227	0.00011	0.00153	6.3×10^{-6}
	Rail	Upper	0.00029	0.00030	0.00035	0.00007	2.3×10^{-6}
		Nominal	0.00025	0.00021	0.00022	0.00006	1.3×10^{-6}
		Lower	0.00023	0.00018	0.00019	0.00005	9.7×10^{-7}

All risks are expressed in latent cancer fatalities during the foreign research reactor spent nuclear fuel policy, except for the Accidental - Traffic column, which is the number of fatalities during the policy.

E.8.13 Management Alternative - The Hybrid Alternative

The hybrid alternative is based on the SNF&INEL Final EIS (DOE, 1995) Regionalization by Fuel Type. The origin of shipment count is described in detail in Chapter 2 and Appendix B. The shipment count is:

	East Coast		*West Coast*		*Totals*
	Aluminum	*TRIGA*	*Aluminum*	*TRIGA*	
Phase 1	212	82	101	42	437
Phase 2	63	25	30	13	131
Total	**275**	**107**	**131**	**55**	**568**

Table E-52 Potential Consequences for the Most Severe Accidents Involving Shipments of Foreign Research Reactor High-Level Waste[a,b]

	Neutral Conditions[c]				*Stable Conditions*[d]			
	Population[e]		*MEI*[f]		*Population*[e]		*MEI*[f]	
Mode and Accident Location	*Dose (person-rem)*	*Consequences (LCF)*	*Dose (rem)*	*Consequences (LCF)*	*Dose (person-rem)*	*Consequences (LCF)*	*Dose (rem)*	*Consequences (LCF)*
Truck								
Urban	121	0.06	0.09	0.000044	970	0.48	0.29	0.00015
Suburban	22.5	0.01	0.09	0.000044	180	0.09	0.29	0.00015
Rural	0.4	0.0002	0.09	0.000044	3.2	0.002	0.29	0.00015
Rail								
Urban	121	0.06	0.09	0.000044	907	0.48	0.29	0.00015
Suburban	22.5	0.01	0.09	0.000044	180	0.09	0.29	0.00015
Rural	0.4	0.0002	0.09	0.000044	3.2	0.002	0.29	0.00015

[a] *The most severe accidents correspond to the highest NUREG-0170 accident severity category (category VIII) (NRC, 1977a). It was assumed that 0.000001 of the radioactive material would be released from its packaging and 5 percent of the aerosolized release would be respirable following an accident.*

[b] *Buoyant plume rise resulting from fire for a severe accident was included in the exposure model.*

[c] *Neutral weather conditions result in moderate dispersion and dilution of the release plume. Neutral conditions were taken to be Pasquill stability Class D with a wind speed of 4 m per sec (9 mph). Neutral conditions occur approximately 50 percent of the time in the United States.*

[d] *Stable weather conditions result in minimal dispersion and dilution of the release plume and are thus unfavorable. Stable conditions were taken to be Pasquill stability Class F with a wind speed of 1 m per sec (2.2 mph). Stable conditions occur approximately one-third of the time in the United States.*

[e] *Populations extend at a uniform density to a radius of 80 km (50 mi) from the accident site. Population exposure pathways include acute inhalation; acute cloudshine; groundshine; resuspended inhalation; resuspended cloudshine; and ingestion of food, including initially contaminated food (rural only). No decontamination or mitigative actions are taken.*

[f] *The MEI is assumed to be at the location of maximum exposure. The locations of maximum exposure would be 160 m (528 ft) and 400 m (1,320 ft) from the accident site under neutral and stable atmospheric conditions, respectively. Individual exposure pathways include acute inhalation, acute cloudshine, and groundshine during passage of the plume. No ingested dose is considered.*

No intersite shipment is necessary for this alternative. The risk estimates are summarized in Table E-53.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.009 to 0.15 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates is caused by two factors: 1) the option of using truck or rail to transport spent nuclear fuel, and, 2) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.008 to 0.037. The estimated number of radiation-related LCFs for the general population ranged from 0.01 to 0.11, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.003 to 0.025.

Table E-53 Tabulation of Overland Transportation Risks: Management Alternative 3 (Hybrid Alternative)

Alternative/Option			*Incident-free*				*Accidental*
			Radiological		*Nonradiological*		
Implementation	*Mode*	*Risk Factors*	*Crew*	*Public*	*Emis.*	*Traffic*	*Radiological*
Regionalization by Fuel Type	Truck	Upper	0.037	0.112	0.008	0.069	0.000081
		Nominal	0.033	0.098	0.005	0.058	0.000035
		Lower	0.028	0.087	0.003	0.048	0.000012
	Rail	Upper	0.010	0.015	0.025	0.002	0.000030
		Nominal	0.009	0.012	0.016	0.002	0.000011
		Lower	0.008	0.010	0.013	0.002	0.000005

All risks are expressed in latent cancer fatalities during the foreign research reactor spent nuclear fuel except for the Accidental - Traffic column, which is the number of fatalities during the policy

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 4.5×10^{-6} to 0.000081 LCFs from radiation and from 0.0017 to 0.069 for traffic fatality, depending on the transportation mode and DOE sites selected. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Table E-53. The analysis for this implementation alternative is analogous to the analysis performed for the basic implementation of Management Alternative 1 (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2.

E.8.14 Transportation Implementation Example - Ship All Foreign Research Reactor Spent Nuclear Fuel to a Single Port, Regionalization-By-Fuel-Type

All the implementation alternatives analyzed in Section E.8 have been analyzed under the assumption that all foreign research reactor spent nuclear fuel would be delivered to ports on the coast nearest to the foreign research reactor (Section E.3.3). This assumption is a reasonable approximation and simplification to a complex set of possible implementation approaches. The following section, however, presents the results of the analysis associated with overland transportation risk of transporting the foreign research reactor spent nuclear fuel from a single commercial or military port.

DOE could bring all spent nuclear fuel through any identified military or commercial port. This would result in 721 shipments to that single port over the 13-year duration of the policy. Canadian fuel would be shipped overland as previously analyzed. The impacts can be directly compared with the impacts of the basic implementation of Management Alternative 1 previously analyzed under the assumption that the foreign research reactor spent nuclear fuel would arrive at the coast nearest the foreign research reactor.

Impacts of Incident-Free Ground Transportation

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.017 to 0.272 over the entire duration of the program. These fatalities are the sums of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates are caused by two factors: 1) the option of using truck or rail to transport spent nuclear fuel, and 2) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.008 to 0.069. The estimated number of radiation-related LCFs for the general population ranged from 0.009 to 0.213, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.002 to 0.035.

Impacts of Accidents During Ground Transportation

The cumulative transportation accident risks over the entire program are estimated to range from 0.00001 to 0.00015 LCFs from radiation and from 0.001 to 0.127 for traffic fatality, depending on the transportation mode and management site(s) selected. The reasons for the range of fatality estimates are the same as those described for incident-free transportation. Both show an expectation of less than one fatality.

The consequences of the most severe accident hypothesized are the same as described for the basic implementation of Management Alternative 1 since the material at risk is the same.

Conclusion

The overland transportation risk associated with bringing all foreign research reactor spent nuclear fuel to a single port is generally within the bounds of the previous analysis that assumed that spent nuclear fuel would arrive at the coast nearest to the foreign source. In the specific case of Regionalization-By-Fuel-Type, the overland transportation risks are reduced by shipping the aluminum-based spent nuclear fuel to an east coast port. The estimated impacts of overland transportation are driven by DOE's selection of a port and the transportation mode. The increased cost and risk associated with shipping the spent nuclear fuel from Asia and Australia to the U.S. east coast are analyzed in Appendices C and F. Table E-54 gives the risk estimates associated with implementing the entire policy from each of the selected ports. The risk estimates are tabulated in a form that can be compared with the other policy and implementation alternatives analyzed in this appendix.

E.8.15 Transportation Implementation Example - Transportation by Barge

As an alternative to truck or rail transport of foreign research reactor spent nuclear fuel, barge transport from Savannah, GA to the Savannah River Site and from Portland, OR to the Hanford Site was evaluated. This section summarizes the impacts.

The analysis of the impacts of barge transportation closely parallels the analysis of truck and rail transportation described in preceding sections. Routing data was generated using the INTERLINE code for the barge routes and the HIGHWAY code for the short trucking segment. A conservative dose rate of 10 mrem per hr at 2 m (6.6 feet) from the vehicle, which is the regulatory limit, was used for calculational purposes. The RADTRAN 4 code was used to calculate the incident-free doses to the public and barge crew. The analysis of port worker consequences on breakbulk vessels was used to estimate the dose to handlers.

The barge analysis used the same radionuclide inventories used in previous sections. The RADTRAN 4 code was used to calculate the impacts of hypothetical accidental releases to the air. A specific waterborne analysis, was performed for barge accidents. Two very conservative assumptions were used in estimating the quantity of material released following an accident:

- Release fractions that determine the source term for dispersion into the water are the same as those used for similar airborne release scenarios, and

Table E-54 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments to Any Single Port, Regionalization by Fuel Type

Port	Mode	Routine				Accidental
		Radiological		Non-Radiological		
		Crew	Public	Emission	Traffic	Radiological
Charleston, SC (NWS)	Truck	0.023	0.070	0.002	0.047	0.00004
	Rail	0.008	0.009	0.011	0.001	0.00001
Charleston, SC (Wando Terminal)	Truck	0.024	0.071	0.003	0.048	0.00005
	Rail	0.008	0.009	0.011	0.001	0.00001
Galveston, TX	Truck	0.039	0.114	0.007	0.070	0.00008
	Rail	0.011	0.016	0.031	0.002	0.00003
Newport News, VA	Truck	0.031	0.093	0.005	0.060	0.00006
	Rail	0.010	0.012	0.015	0.002	0.00002
Norfolk, VA	Truck	0.031	0.091	0.003	0.060	0.00006
	Rail	0.010	0.012	0.014	0.002	0.00002
Portsmouth, VA	Truck	0.031	0.092	0.004	0.060	0.00006
	Rail	0.010	0.011	0.013	0.002	0.00002
Jacksonville, FL	Truck	0.027	0.082	0.002	0.053	0.00005
	Rail	0.009	0.009	0.011	0.001	0.00001
MOTSU, NC	Truck	0.023	0.073	0.002	0.047	0.00004
	Rail	0.009	0.010	0.011	0.002	0.00001
NWS-Concord, CA	Truck	0.069	0.213	0.019	0.127	0.00012
	Rail	0.018	0.026	0.035	0.004	0.00004
Portland, OR	Truck	0.066	0.209	0.010	0.120	0.00015
	Rail	0.018	0.021	0.027	0.004	0.00005
Savannah, GA	Truck	0.024	0.073	0.002	0.047	0.00005
	Rail	0.008	0.009	0.010	0.001	0.00001
Tacoma, WA	Truck	0.067	0.212	0.008	0.104	0.00015
	Rail	0.018	0.024	0.031	0.004	0.00005
Wilmington, NC	Truck	0.026	0.079	0.002	0.055	0.00005
	Rail	0.009	0.010	0.011	0.002	0.00001

- All of the source term resulting from an accident event is dispersed into the waterway, uniformly, over a one month period (i.e., it takes one month to recover the cask).

Barge accident statistics (Hutchinson, 1986) were used to estimate the probabilities of the accident severity classes defined in the Modal Study (Fischer et al., 1987). Barge transportation fatality statistics from Saricks and Kvitek, 1994 were used to estimate accident fatality rates. The following exposure pathways were assessed using the methodology developed by the NRC in Regulatory Guide 1.109 (NRC, 1977b):

- drinking water
- ingestion of fish
- ingestion of irrigated foods
- ingestion of meat and milk from exposed cattle
- shoreline deposits (external exposure)
- swimming (external exposure).

Collective doses were calculated for average densities for rural, suburban and urban populations, using route-specific river parameters. Additionally, MEI doses were calculated for each accident scenario in a manner analogous to that in preceding sections.

Unlike previous sections, where impacts were reported in terms of implementation of the foreign research reactor spent nuclear fuel policy, impacts are reported on a per shipment basis. As shown in Figures E-1 through E-12, the policy can be carried out in many ways, depending on the outcome of the SNF&INEL EIS (DOE, 1995) and its Record of Decision. The SNF&INEL EIS alternative that could be implemented using only barge transportation is Centralization to the Savannah River Site or the Hanford Site. All others would require various mixtures of barge transportation and overland transportation via truck or rail. Therefore, barge transportation impacts are reported on a per shipment basis and compared on that basis to shipments via truck or rail for the same origin/destination pair.

The results of the barge transportation analysis, along with comparable results from the analysis of truck and rail transportation are summarized in Table E-55.

Table E-55 Tabulation of Inland Transportation Risk Factors: Basic Implementation, Shipments via Barge to Hanford and Savannah River Sites

Alternative/ Option		*Incident Free*				*Accidental*		
		Radiological			*Nonradiological*		*Radiological*	
Port - Site	*Mode*	*Crew*	*Handlers*	*Public*	*Emission*[a]	*Traffic*	*Air-borne*	*Water-borne*
Savannah, GA to Savannah River Site	Truck	8.96×10^{-6}	N/A	0.0000277	3.22×10^{-8}	0.000021	9.35×10^{-9}	N/A
	Rail	5.44×10^{-6}	N/A	2.64×10^{-6}	5.86×10^{-7}	2.38×10^{-7}	1.11×10^{-9}	N/A
	Barge	7.64×10^{-8}	9.60×10^{-7}	1.94×10^{-6}	2.39×10^{-7}	3.42×10^{-6}	5.90×10^{-10}	2.93×10^{-8}
Portland, OR to Hanford Site	Truck	9.40×10^{-6}	N/A	0.0000294	2.67×10^{-6}	0.0000104	1.49×10^{-8}	N/A
	Rail	6.28×10^{-6}	N/A	5.75×10^{-6}	4.48×10^{-6}	5.00×10^{-7}	3.75×10^{-9}	N/A
	Barge[b]	6.40×10^{-7}	1.92×10^{-6}	4.26×10^{-6}	2.44×10^{-6}	6.69×10^{-6}	1.65×10^{-8}	1.87×10^{-8}

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

[a] *Assumes the same emission rate as rail transportation, and two-way travel.*

[b] *Includes truck shipment from Richland, WA to Hanford Site*

E.8.15.1 Evaluation of Barge Transportation from Portland, OR to the Hanford Site

Transportation Routes

Barge transportation from the port of Portland, OR, up the Columbia River to the town of Richland, WA, followed by truck shipment to the Hanford Site was analyzed. It was assumed that the port facilities at Portland could be used to load the casks to a barge without having to transport it into areas accessed by the public. The barge could have sailed up the river to either Richland, Pasco or Kennewick, WA. The difference in the risk parameters would be less than 5 percent. Richland was chosen for analysis because it is the largest of the three cities.

Incident-Free Transportation

The incident-free transportation of spent nuclear fuel was estimated to result in 1.17×10^{-5} total latent fatalities per shipment. These fatalities are the sum of the estimated number of radiation-related and emission-related latent fatalities for the crew, handlers, and public.

The estimated number of radiation-related LCFs for the barge and truck crews is 6.40×10^{-7}. The number of radiation-related LCFs for handlers during handling activities (other than the initial off-load from the seagoing ship and the on-site handling) is 1.92×10^{-6} per shipment. The number of radiation-related LCFs for the general population is 4.26×10^{-6} per shipment. The number of nonradiological fatalities from vehicle emissions is 4.88×10^{-6} per shipment.

The MEI risk would be the same as that in the basic implementation of Management Alternative 1, which is 0.00052 LCF for the duration of the program. This estimate is based on the conservative assumption that one individual is involved in enough driving, handling and/or inspection to reach the regulatory limit of 100 mrem per year every year for the 13-year duration of the program.

Transportation Accidents

The barge transportation accident risks from radiation exposure are estimated to be 3.63×10^{-8} LCF per shipment. These fatalities are the sum of the estimated number of radiation-related fatalities from atmospheric and waterborne releases. The estimated number of radiation-related LCFs from atmospheric releases is 1.65×10^{-8} per shipment, and 1.87×10^{-8} per shipment for waterborne releases. The barge transportation accident risks from other accidents than radiation are estimated to be 6.6×10^{-6} fatalities per shipment.

The consequences of the maximum foreseeable offsite transportation accident are 0.0295 LCF. The likelihood of this accident is approximately 1×10^{-7}.

E.8.15.2 Evaluation of Barge Transportation from Savannah, GA to the Savannah River Site

Transportation Routes

Barge transportation from the port of Savannah, GA, up the Savannah River to the Savannah River Site was analyzed. The Savannah River Site has a barge receiving facility that could be used to off-load the casks. Handling at that facility and the onsite movement to the Receiving Basin for Offsite Fuels would not result in a significant change in calculated onsite risks (Appendix D).

Incident-Free Transportation

The incident-free transportation of spent nuclear fuel was estimated to result in 3.45×10^{-6} total latent fatalities per shipment. These fatalities are the sum of the estimated number of radiation-related and emission-related latent fatalities for the crew, handlers, and public.

The estimated number of radiation-related LCFs for the barge and truck crews is 7.64×10^{-8}. The number of radiation-related LCFs for handlers during handling activities (other than the initial off-load from the seagoing ship and the on-site handling) is 9.60×10^{-7} per shipment. The number of radiation-related LCFs for the general population is 1.94×10^{-6} per shipment. The number of nonradiological fatalities from vehicle emissions is 4.97×10^{-7} per shipment.

The MEI risk would be the same as that in the basic implementation of Management Alternative 1, which is 0.00052 LCF for the duration of the program. This estimate is based on the conservative assumption that one individual is involved in enough driving, handling and/or inspection to reach the regulatory limit of 100 mrem per year every year for the 13-year duration of the program.

Transportation Accidents

The barge transportation accident risks are estimated to be 2.12×10^{-8} LCF per shipment. These fatalities are the sum of the estimated number of radiation-related fatalities from atmospheric and waterborne releases. The estimated number of radiation-related LCFs from atmospheric releases is 5.90×10^{-10} per shipment, and 2.93×10^{-8} per shipment for waterborne releases. The barge transportation accident risks from other than radiation are estimated to be 3.42×10^{-6} fatalities per shipment.

The consequences of the maximum foreseeable offsite transportation accident are 0.0259 LCF. The likelihood of this accident is approximately 1×10^{-7}.

E.8.15.3 Conclusions on Barge Transportation

Table E-55 provides a comparison of barge shipment parameters to truck and rail shipment parameters between the same two points. For incident-free transportation, the risk to the public and onboard crew is lower than for truck or rail shipment. However, the risk increase associated with additional handling of casks negates this risk reduction. The net incident-free risk for barge transportation is essentially identical to that for rail transportation. The radiological accident risk associated with barge transportation is larger than that of truck or rail because of the consequences of a hypothetical accident in which a damaged cask is dropped into a river. As evident from Table E-55, fatality rates for barge transportation accidents are higher than traffic fatality rates for rail shipment and lower than those for highway shipment. In total, the difference between the risks of shipping by truck, rail or barge is very low.

When traveling along a river, a barge can be observed from a long distance, and due to its slow speed, can be boarded while underway. Although not considered in detail, these characteristics would increase the vulnerability to terrorist attack.

E.9 Historical Account of Spent Nuclear Fuel Shipments and Cumulative Impacts of Transportation

E.9.1 Spent Nuclear Fuel Shipment History

The SNF&INEL Final EIS (DOE, 1995) contains a survey of transportation incidents from 1949 to 1993. For 1949 through 1970, there were 14 incidents involving irradiated fuel elements. No packages approximating a Type B shipping cask were breached as a result of these incidents. Between 1971 and 1993, there were seven transportation accidents involving spent nuclear fuel. Three involved rail shipments, and four of these accidents involved truck shipments. None of these accidents resulted in damage to the structural integrity of a cask or release of contents.

The number of spent nuclear fuel shipments and amount of spent nuclear fuel shipped throughout the entire history of spent nuclear fuel shipment cannot be precisely determined from available information. The NRC keeps accurate records of more recent (since 1979) shipments of spent nuclear fuel.

Tables E-56 and E-57 describe the spent nuclear fuel shipments in the United States that have occurred since 1979. The data for the tables comes from NUREG-0725 (NRC, 1993). These tables show detailed spent nuclear fuel shipment information, including mode of shipment (highway or rail) and shipment trends over time. Data for shipment miles are taken primarily from a road atlas and have been rounded to the nearest hundred miles for each year. Data on quantity of spent fuel shipped were provided by shippers,

and have been rounded to the nearest hundred kg (220 lb) (when more than 100 kg (220 lb) were shipped). These tables do not include DOE shipments (including Naval) of spent nuclear fuel, since these shipments are not regulated by the NRC.

Table E-56 Domestic and International Spent Nuclear Fuel Shipments: 1979-1992

Year	Domestic		International		Transient
	Highway	Railway	Export	Import	
1979	2	11	0	14	0
1980	73	5	2	55	0
1981	30	2	3	48	0
1982	80	0	1	43	0
1983	92	0	2	23	0
1984	209	3	2	34	0
1985	114	18	0	21	0
1986	88	15	0	17	0
1987	85	15	3	19	0
1988	10	7	0	15	0
1989	11	6	1	4	0
1990	0	8	2	0	3
1991	7	10	4	0	1
1992	17	6	0	0	0
Totals	**818**	**106**	**20**	**293**	4

Source: NRC, 1993

Table E-56 shows the pattern of highway and rail shipments throughout the period 1979 to 1992. The number of shipments generally rose in the early 1980s and then declined steadily through 1992. Import shipments have generally declined since 1980, with no shipments since 1989.

Table E-57 shows that most (91.4 percent) of approximately 1,200 spent nuclear fuel shipments during the 1979 to 1992 period were completed over highways. The highway shipments accounted for a larger percentage of the mileage (94.8 percent), meaning that the longer distance shipments tended to use the highways rather than rail. However, rail shipments moved 70 percent (by weight) of the fuel. This indicates that rail has been chosen for the larger shipments over shorter distances. A review of spent nuclear fuel shipments indicates that rail transportation was often used for shipments of commercial spent nuclear fuel, and research reactors almost exclusively used trucks (NRC, 1993).

E.9.2 Cumulative Impacts of Transportation

The SNF&INEL Final EIS (DOE, 1995) analyzed the cumulative impacts of transportation, taking into account impacts from: 1) historical shipments of spent nuclear fuel to Hanford Site, Savannah River Site, Idaho National Engineering Laboratory, Oak Ridge Reservation, and Nevada Test Site; 2) the programmatic alternatives; 3) other reasonably foreseeable actions that include transportation of radioactive material; and 4) general radioactive materials transportation that is not related to a particular action.

The total worker and general population collective doses are summarized in Table E-58. Total collective worker doses from all types of shipments (historical, the alternatives, reasonably foreseeable actions, and general transportation) were estimated to be 320,000 person-rem (130 LCFs) for the period of time 1943 through 2035 (93 yr). Total general population collective doses were also estimated to be 320,000 person-rem (160 LCFs). The majority of the collective dose for workers and the general population was due to the general transportation of radioactive material. Examples of these activities are

Table E-57 Summary Data for 1979-1992 Spent Nuclear Fuel Shipment Information

Year	Number of Shipments		Kilograms Spent Fuel Shipped (thousands)[a]		Shipment Kilometers (thousands)[b]	
	Highway	Railway	Highway	Railway	Highway	Railway
1979	16	11	0.1	30.2	12.9	3.7
1980	130	5	10.0	13.6	186.6	1.6
1981	81	2	7.9	6.0	62.0	0.6
1982	124	0	7.1	0.0	171.9	0.0
1983	117	0	36.6	0.0	134.6	0.0
1984	245	3	84.5	23.8	291.9	2.6
1985	135	18	74.0	119.4	114.1	14.0
1986	105	15	40.4	97.5	77.0	14.0
1987	107	15	82.3	101.4	67.3	13.5
1988	25	7	12.8	41.8	18.4	6.9
1989	16	6	0.1	30.8	26.9	2.7
1990	2	8	0.03	65.5	2.4	1.6
1991	11	10	0.1	98.4	15.5	2.4
1992	17	6	0.1	61.3	15.7	0.8
Totals	**1,131**	**106**	**356.0**	**689.7**	**1197.2**	**64.4**

Source: NRC, 1993

[a] *To convert kilogram values to pounds, multiply values given by 2.2.*

[b] *To convert kilometer values to miles, multiply by 0.62.*

Table E-58 Cumulative Transportation-Related Radiological Collective Doses and LCFs (1943 to 2035)

Category	Collective Occupational Dose (person-rem)	Collective General Population Dose (person-rem)
Historical	200	110
Spent Nuclear Fuel Shipments for SNF&INEL Final EIS Alternatives 1-5		
Truck	1.5 to 1,000	0.34 to 2,400
Rail	1.5 to 150	0.34 to 190
Reasonably Foreseeable Actions		
Truck	11,000	50,000
Rail	820	1,700
General Transportation (1943 to 2035)	310,000	270,000
Total Collective Dose	320,000	320,000
Total LCFs	130	160

Source: DOE, 1995

shipments of radiopharmaceuticals to nuclear medicine laboratories and shipments of commercial low-level radioactive waste to commercial disposal facilities. The total number of LCFs over the time period 1943 through 2035 was estimated to be 290. Over this same period of time (93 yr), approximately 28,000,000 people would die from cancer, based on 300,000 LCFs per yr (NRC, 1977a). It should be noted that the estimated number of transportation-related LCFs would be indistinguishable from other LCFs, and the transportation-related LCFs are 0.0010 percent of the total number of LCFs.

The transportation of foreign research reactor spent nuclear fuel, under any of the proposed options or alternatives in this EIS, is included in the calculated totals under the spent nuclear fuel shipments for SNF&INEL Final EIS Alternatives 1-5 (DOE, 1995). Proposed transportation of domestic and foreign spent nuclear fuel accounts for less than one percent of the total LCFs, attributable to the transportation of radioactive material, and foreign research reactor spent nuclear fuel accounts for less than one-quarter of that one percent.

E.10 Uncertainty and Conservatism in Estimated Impacts

The sequence of analyses performed to generate the estimates of radiological risk for the transportation of spent nuclear fuel includes: 1) determination of the inventory and characteristics, 2) estimation of shipment requirements, 3) determination of route characteristics, 4) calculation of radiation doses to exposed individuals (including estimation of environmental transport and uptake of radionuclides), and 5) estimation of health effects. Uncertainties are associated with each of these steps. Uncertainties exist in the way that the physical systems being analyzed are represented by the computational models, in the data required to exercise the models (due to measurement errors, sampling errors, natural variability, or unknowns simply caused by the future nature of the actions being analyzed), and in the calculations themselves (for example, approximate algorithms used by the computers).

In principle, one can estimate the uncertainty associated with each input or computational source and predict the resultant uncertainty in each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final, or absolute, result; however, conducting such a full-scale quantitative uncertainty analysis is often impractical and sometimes impossible, especially for actions to be initiated at an unspecified time in the future. Instead, the risk analysis is designed to ensure, through uniform and judicious selection of scenarios, models, and input parameters, that relative comparisons of risk among the various alternatives are meaningful. In the transportation risk assessment, this design is accomplished by uniformly applying common input parameters and assumptions to each alternative. Therefore, although considerable uncertainty is inherent in the absolute magnitude of the transportation risk for each alternative, much less uncertainty is associated with the relative differences among the alternatives in a given measure of risk.

In the following sections, areas of uncertainty are discussed for the assessment steps enumerated above. Special emphasis is placed on identifying whether the uncertainties affect relative or absolute measures of risk. The degree of reality conservatism of the assumption is addressed. Where practical, the parameters that most significantly affect the risk assessment results are identified.

E.10.1 Uncertainties in Spent Nuclear Fuel Inventory and Characterization

The spent nuclear fuel inventories (i.e., number of shipments) and the physical and radiological characteristics are important input parameters to the transportation risk assessment. The potential amount of transportation for any alternative is determined primarily by the projected spent nuclear fuel inventory and assumptions concerning shipment capacities. The physical and radiological characteristics are important in determining the amount of material released during accidents and the subsequent doses to exposed individuals through multiple environmental exposure pathways.

The development of projected spent nuclear fuel inventory and characterization data used to support the EIS is described in Appendix B. Uncertainties in the spent nuclear fuel inventory and characterization will be reflected to some degree in the transportation risk results. If the spent nuclear fuel inventory (number of elements) is overestimated (or underestimated), the resulting transportation risk estimates also will be overestimated (or underestimated) by roughly the same factor. However, the same spent nuclear fuel

inventory estimates are used to analyze the transportation impacts of each of the EIS alternatives. Therefore, for comparative purposes, the observed differences in transportation risks among alternatives are believed to represent unbiased, reasonably accurate estimates from current information in terms of relative risk comparisons.

The spent nuclear fuel type selected for the accident risk calculations was chosen to maximize the potential accident risk results. All accidents were analyzed for fuel that is less than 1 year old. However, much of the fuel has already been out of the foreign research reactors for more than 1 year and may not be brought back for several years. For calculations of MEIs, the cask loaded with the maximum possible amount of radioactive material should have been and was considered. However, the risk values were calculated under the assumption that all casks were loaded to this maximum value. Depending on the implementation of the program, very few, if any, of the casks would be carrying fuel as new as that used in the accident analysis. Selection of another spent nuclear fuel type, or consideration of all spent nuclear fuel types in detail, would result in accident risks less than those reported in the assessment of alternatives in this appendix.

E.10.2 Uncertainties in Casks, Shipment Capacities and Number of Shipments

The amount of transportation required for each alternative is based in part on assumptions concerning the packaging characteristics and shipment capacities for truck and rail modes. Representative shipment capacities have been defined for assessment purposes based on probable future shipment capacities. In reality, the actual shipment capacities may differ from the predicted capacities, so that the projected number of shipments, and consequently the total transportation risk, would change. However, although the predicted transportation risks would increase or decrease accordingly, the relative differences in risks among alternatives would remain about the same. It is in fact likely that DOE would deploy a large capacity truck or rail cask for large intersite shipping campaigns.

For the purposes of analysis, Phase 1 was assumed to last exactly 10 years and Phase 2 was assumed to last exactly 3 years. Realistically, the Phase 2 site may be ready somewhat sooner or later. Additionally, the fractions of the fuel arriving during each phase may not be precisely proportional to the duration of the phase. However, the risk changes are small when compared with the conservatism introduced in the radiological calculations.

The number of shipments to and from various points comes from a complex series of models of how the policy may be implemented. They are not intended to define how the policy would be implemented. Instead, they describe somewhat generally how the policy would be implemented. The risk factors for all conceivable routes between DOE sites and ports of entry are given to show that a slight deviation from the shipment pattern modeled could have a negligible affect on risk. For example, if the policy were being implemented with fuel arriving in the eastern United States going to Savannah River Site and fuel arriving in the western United States going to Idaho National Engineering Laboratory, the risk impact of transporting a few casks from eastern ports to Idaho National Engineering Laboratory (presumably for onsite logistical reasons) would have a small impact on program risk.

E.10.3 Uncertainties in Route Determination

Representative routes have been determined between all origin and destination sites considered in the EIS. The routes have been determined consistent with current guidelines, regulations, and practices, but may not be the actual routes that would be used in the future. In reality, the actual routes could differ from the representative ones in terms of distances and total population along the routes. Moreover, in that the assessment considers spent nuclear fuel could be transported over an extended period of time starting at

some time in the future, the highway and rail infrastructures and the demographics along routes could change. These effects have not been accounted for in the transportation assessment, however, it is not anticipated that these changes would significantly affect relative comparisons of risk among the alternatives considered in the EIS.

E.10.4 Uncertainties in the Calculation of Radiation Doses

The models used to calculate radiation doses from transportation activities introduce a further uncertainty in the risk assessment process. It is generally difficult to estimate the accuracy or absolute uncertainty of the risk assessment results. The accuracy of the calculated results is closely related to the limitations of the computational models and to the uncertainties in each of the input parameters that the model requires. The single greatest limitation facing users of RADTRAN, or any computer code of this type, is the scarcity of data for certain input parameters.

Uncertainties associated with the computational models are minimized by using state-of-the-art computer codes that have undergone extensive review. Because there are numerous uncertainties that are recognized but difficult to quantify, assumptions are made at each step of the risk assessment process that are intended to produce conservative results (i.e., overestimate the calculated dose and radiological risk). Because parameters and assumptions are applied equally to all alternatives, this model bias is not expected to affect the meaningfulness of relative comparisons of risk; however, the results may not represent risks in an absolute sense.

In order to understand the most important uncertainties and conservatism in the transportation risk assessment, the results for all cases were examined to identify the largest contributors to the collective population risk. The results of this examination are discussed briefly below.

For truck shipments, the largest contributors to the collective population dose were found to be, in decreasing order of importance: 1) incident-free dose to members of the public at stops, 2) incident-free dose to transportation crew members, 3) incident-free dose to members of the public sharing the route (on-link dose), 4) incident-free dose to members of the public residing along the route (off-link dose), and 5) accident dose risk to members of the public. Approximately 80 percent of the estimated public dose was incurred at stops, 15 percent by the on-link population, and 5 percent by the off-link population. In general, the accident contribution to the total risk was negligible compared with the incident-free risk.

For rail shipments, the largest contributors to the collective population dose were found to be, in decreasing order of importance: 1) incident-free dose to transportation crew members, 2) incident-free dose to members of the public residing along the route (off-link dose), 3) incident-free dose to members of the public at stops, 4) incident-free dose to members of the public sharing the route (on-link dose), and 5) accident dose risk to members of the public. Approximately 70 percent of the estimated public dose was incurred by the off-link population, 25 percent by the population at stops, and 5 percent by the on-link population. As with truck shipments, the accident contribution to the total risk in general was negligible compared with the incident-free risk, even when the spent nuclear fuel type was selected to maximize the accident risk results.

As shown above, incident-free transportation risks are the dominant component of the total transportation risk for both truck and rail modes. The most important parameter in calculating incident-free doses is the shipment external dose rate (incident-free doses are directly proportional to the shipment external dose rate). For this assessment, it was assumed that all shipments would have an external dose rate at the regulatory limit of 10 mrem per hr at 2 m. In practice, the external dose rates would vary from shipment to shipment. Although it is conceivably possible to load a cask with enough fresh foreign research reactor

spent nuclear fuel to obtain a dose rate equal to the regulatory limit, experience has shown this to be unlikely. In fact, the observed average dose rate described in Appendix B is approximately ten times lower than the regulatory limit. During the shipments of foreign research reactor to MOTSU and ultimately to Savannah River Site, the State of North Carolina detected less than 1 mrem on contact with the cask and no radiation above background at 2 m (Massey, 1994). Therefore, the incident-free risks are conservative, and would be ten times lower if calculated with the observed average dose.

Finally, the single largest contributor to the collective population doses calculated with RADTRAN was found to be the dose to members of the public at truck stops. Currently, RADTRAN uses a simple point-source approximation for truck-stop exposures and assumes that the total stop time for a shipment is proportional to the shipment distance. The parameters used in the stop model were based on a survey of a very limited number of radioactive material shipments that examined a variety of shipment types in different areas of the country (Wilmot, 1981). It was assumed that stops occur as a function of distance, with a stop rate of 0.011 h per km (0.018 h per mile). It was further assumed that at each stop, an average of 50 people are exposed at a distance of 20 m (66 ft). In RADTRAN, the population dose is directly proportional to the external shipment dose rate and the number of people exposed, and inversely proportional to the square of the distance. The stop rate assumed results in an hour of stop time per 100 km (62 miles) of travel.

Based upon the qualitative discussion with shippers of spent nuclear fuel, the parameter values used in the assessment appear to be conservative. However, data do not exist to qualitatively assess the degree of conservatism in the stop-dose model. As a practical matter, it is conceivable that DOE could take steps to control the location, frequency, and duration of truck stops if necessary. However, based on the regulatory requirements for continuous escort of the material (10 CFR 73) and the requirement for two drivers, it is clear that the trucks would be on the move essentially one-hundred percent of the time until arrival at the destination. Therefore, the calculated impacts are extremely conservative. By using these conservative parameters, the calculations in this EIS are consistent with the RADTRAN default values and the SNF&INEL Final EIS (DOE, 1995).

Shielding of exposed populations is not considered. For all incident-free exposure scenarios, no credit has been taken for shielding of exposed individuals. In reality, shielding would be afforded by trucks and cars sharing the transport routes, natural topography, and the houses and buildings in which people reside. Incident-free exposures to external radiation could be reduced significantly depending on the type of shielding present. For residential houses, shielding factors (i.e., the ratio of shielded to unshielded exposure rates) have been estimated to range from 0.02 to 0.7, with a recommended value of 0.33. If shielding were to be considered for the maximally exposed resident living near a transport route, the calculated doses and risks would be reduced by approximately 70 percent. Similar levels of shielding may be provided to individuals exposed in vehicles. However, consideration of shielding does not significantly affect the overall incident-free risks to the general population.

Post-accident mitigative actions are not considered for dispersal accidents. For severe accidents involving the release and dispersal of radioactive materials in the environment, no post-accident mitigative actions, such as interdiction of crops or evacuation of the accident vicinity, have been considered in this risk assessment. In reality, mitigative actions would take place following an accident in accordance with U.S. Environmental Protection Agency radiation protection guides for nuclear incidents (EPA, 1991). The effects of mitigative actions on population accident doses are highly dependent upon the spent nuclear fuel type involved and the severity, location, and timing of the accident. For this risk assessment, ingestion doses are only calculated for accidents occurring in rural areas (the calculated ingestion dose, however, assumes all food grown on contaminated ground is consumed and is not limited to the rural population).

Examination of the severe accident consequence assessment results has shown that ingestion of contaminated foodstuffs contributes on the order of 50 percent of the total population dose for rural accidents. Interdiction of foodstuffs would act to reduce, but not eliminate, this contribution.

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Attachment E1
Representative Routes for Overland Transportation

The purpose of this Attachment is to show the representative routes that are used in the risk analysis of overland transportation of foreign research reactor spent nuclear fuel. Identification of these representative routes is necessary to carry out the risk analysis described in Appendix E. The characteristics of these representative routes are used to estimate the driving time, which is needed to estimate the risk to the crew operating the truck, train, or barge and to estimate population along the route, which is exposed to small amounts of radiation emanating from the cask and could conceivably be exposed to an accident.

Representative routes between all potential ports and Canadian border crossings and all 5 DOE sites, as well as between the DOE sites, were identified. Because of the large number of routes, complexity of routes and the difficulty associated with manually tabulating population data, DOE has developed computer codes to select routes and tabulate populations. The HIGHWAY (Johnson, 1993a) code was used for the road route analysis and the INTERLINE (Johnson, 1993b) code was used for the rail and barge route analysis. The codes are described in Section E.4.

Representative routes are used in the analysis because specific routes cannot be identified in advance. A transportation route is not finalized until it has been reviewed and approved by the NRC (Massey, 1994). Therefore, the routes identified in this section are representative of the routes that may be used to ship spent nuclear fuel, but the actual routes shown may or may not be used if the policy were to be implemented. The codes used to select the representative routes automatically select preferred routes and minimize the transport time, as required by regulation. However, it is required that route selection also considers information such as the time of day and the day of week during which transportation will occur, current conditions such as adverse weather conditions (e.g., flooding, snow), track or road conditions, bridge closures, and seasonal traffic. Additionally, numerous U.S. interstates serve the Canadian border and could be used to transport spent nuclear fuel. HIGHWAY and INTERLINE cannot take these environmental, and other, conditions into account.

Specific requirements for the selection of road routes are found in 49 CFR 397, Subpart D. This regulation requires selection of "preferred routes", which are defined as Interstate System highways for which an alternative route is not designated by a State routing agency and/or State-designated routes selected by a State routing agency. 49 CFR 397 also gives specific directions for the selection of a route between a point of pickup or delivery (i.e., a port facility) and a preferred route. In some cases, the HIGHWAY code does not explicitly cover this short distance. These stretches of road were evaluated, and it was determined that they were very short compared to the overland routes.

This attachment provides maps and listings of the representative routes used in the transportation analysis. Information on population can be found in Tables E-3 and E-4. For each port of entry, a road map followed by a listings of the routes, and a rail map, followed by listings of the routes are provided.

The route listings are from the HIGHWAY and INTERLINE computer code output files. The columns of the HIGHWAY output, from left to right indicate the following: 1) The roadway on which the truck would be traveling, 2) codified information describing the road (# - toll bridge, $ - toll road) and other numbers used to identify the stretch of road, 3) the city nearest the node, 4) the position of the node relative to the city, 5) the intersecting roads that define the node, and 6) the state in which the node is located.

The INTERLINE output, from left to right, shows the following: 1) the railroad that owns a section of the track or, in the case of barge transportation, the letters BRG, 2) the INTERLINE code number for the node, 3) the city nearest the node, 4) the state in which the node is located, and 5) the distance traveled since the last node. Transfers between rail carriers are noted by dashed lines.

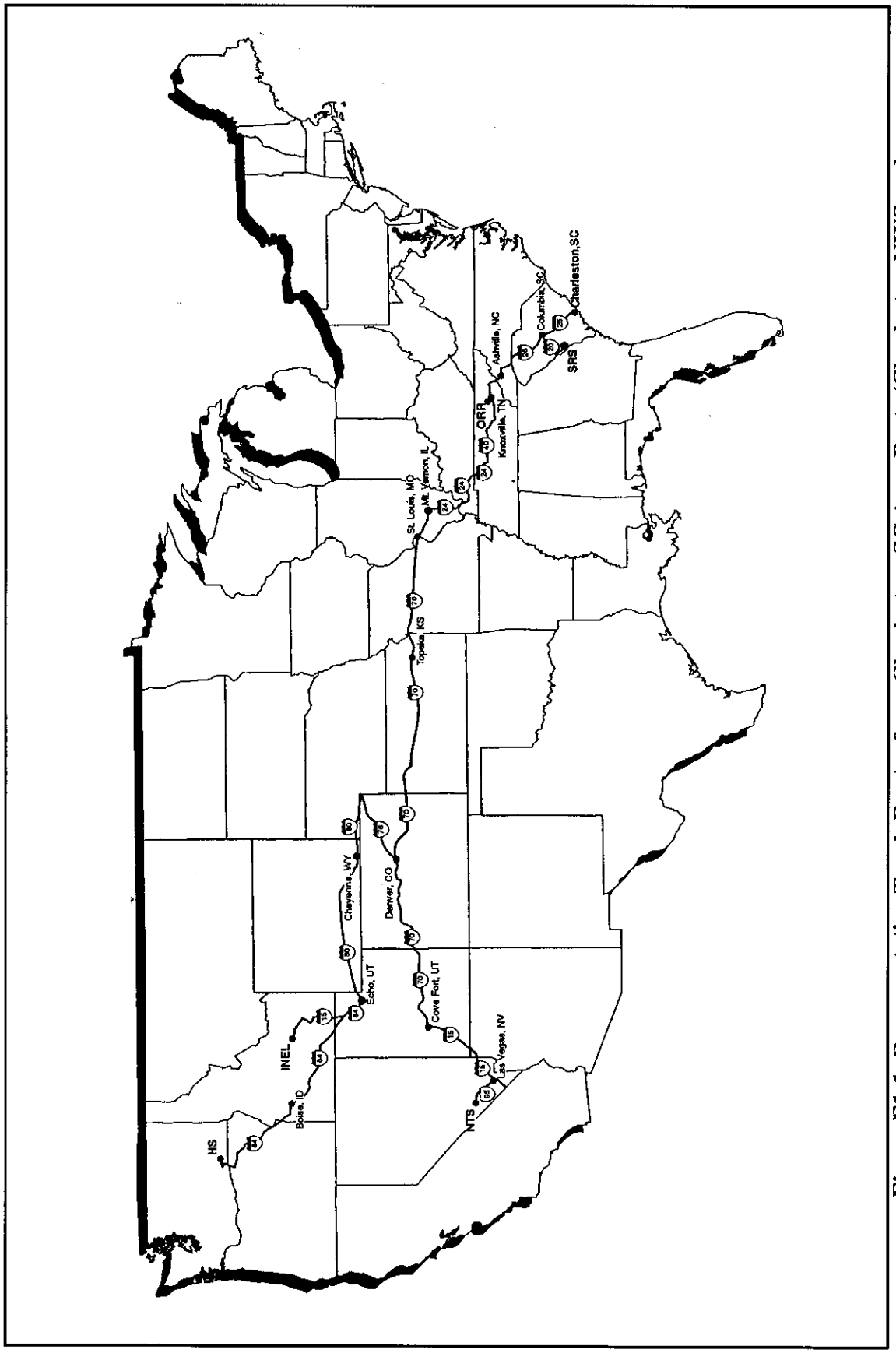

Figure E1-1 Representative Truck Routes from Charleston, SC Area Ports (Charleston-NWS and Charleston-Wando) to Department of Energy Management Sites

From: NWS CHARLESTON, SC
To : SRP BARCD 1

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		COLUMBIA	NW	I20	I26	SC
I20		BELVEDERE	N	I20	X5	SC
U25		NORTH AUGUSTA	SE	U25	S125	SC
S125		CLEARWATER	W	U1	U27B	SC
U278		BEECH ISLAND		U27B	S125	SC
S125		SRP BARCD 1		S125	LC	SC

From: NWS CHARLESTON, SC
To : ID NATL ENG LAB, ID

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB		U20	LOCL	ID

From: NWS CHARLESTON, SC
To : HANFORD, WA

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	X5	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: NWS CHARLESTON, SC
To : K-25, TN

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		KINGSTON	E	I40	X356	TN
S58		K-25				TN

From: NWS CHARLESTON, SC
To : MERCURY, NV

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWAROSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		LAS VEGAS	W	U95	U95B	NV
U95BU		LAS VEGAS	NW	U95	U95B	NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: CHARLESTON (WANDO TERMINAL), SC
To : SRP BARCD 1, SC

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		COLUMBIA	NW	I20	I26	SC
I20		BELVEOERE	N	I20	X5	SC
U25		NORTH AUGUSTA	SE	U25	S125	SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLANO		U278	S125	SC
S125		SRP 8ARCD 1		S125	LC	SC

From: CHARLESTON (WANDO TERMINAL), SC
To : ID NATL ENG LAB, ID

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB		U2D	LOCL	ID

From: CHARLESTON (WANDO TERMINAL), SC
To : HANFORD, WA

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOO	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EOWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I27D	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I7D	KS
I470$	TKST$	TOPEKA	S	I335	I47D	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	X5	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: CHARLESTON (WANDO TERMINAL), SC
To : K-25, TN

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		KINGSTON	E	I40	X356	TN
S58		K-25				TN

From: CHARLESTON (WANDO TERMINAL), SC
To : MERCURY, NV

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		LAS VEGAS	W	U95	U95B	NV
U95BU		LAS VEGAS	NW	U95	U95B	NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

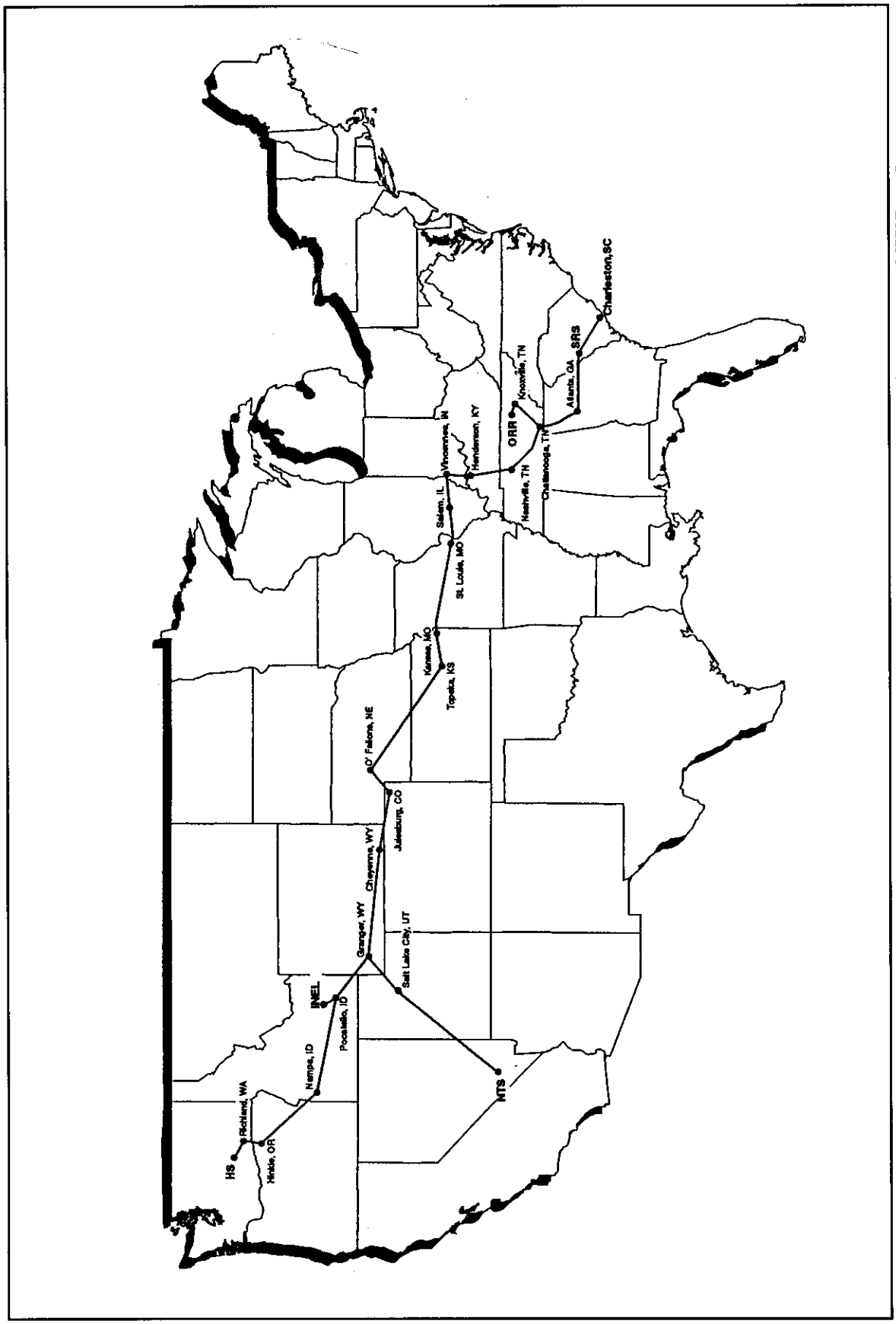

Figure E1-2 Representative Rail Routes from Charleston, SC Area Ports (Charleston-NWS and Charleston-Wando) to Department of Energy Management Sites

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: SRP, SC

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7739-FAIRFAX	SC	94.
CSXT	7732-ROBBINS	SC	123.
CSXT	7717-DUNBARTON / WELL	SC	132.
USG	7717-DUNBARTON / WELL	SC	132.
USG	15359-SRP	SC	140.

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7739-FAIRFAX	SC	94.
CSXT	7732-ROBBINS	SC	123.
CSXT	7961-AUGUSTA	GA	152.
CSXT	7914-ATLANTA	GA	327.
CSXT	7907-MARIETTA	GA	337.
CSXT	7889-CARTERSVILLE	GA	369.
CSXT	7888-DALTON	GA	420.
CSXT	7235-CHATTANOOGA	TN	458.
CSXT	7187-TULLAHOMA	TN	539.
CSXT	7202-NASHVILLE	TN	618.
CSXT	7201-MAOISON	TN	628.
CSXT	7061-HOPKINSVILLE	KY	688.
CSXT	3839-HENDERSON	KY	775.
CSXT	3838-EVANSVILLE	IN	788.
CSXT	3812-VINCENNES	IN	838.
CSXT	4952-SALEM	IL	917.
CSXT	10859-EAST ST LOUIS	IL	982.
<TR>	10859-EAST ST LOUIS	IL	982.
<TR>	1085B-ST LOUIS	MO	988.
UP	10858-ST LOUIS	MO	988.
UP	10656-JEFFERSON CITY	MO	1110.
UP	10616-KANSAS CITY	MO	1286.
UP	10617-KANSAS CITY	KS	1289.
UP	11823-LAWRENCE	KS	1328.
UP	11697-TOPEKA	KS	1358.
UP	11696-MENOKEN	KS	1363.
UP	11681-MARYSVILLE	KS	1438.
UP	11405-HASTINGS	NE	1548.
UP	11410-GIBBON	NE	1574.
UP	11352-NORTH PLATTE	NE	1652.
UP	11358-O FALLONS	NE	1701.
UP	13703-JULESBURG	CO	1769.
UP	13465-CHEYENNE	WY	1915.
UP	13462-LARAMIE	WY	1967.
UP	13494-GRANGER	WY	2243.
UP	13369-MC CAMMON	ID	2435.
UP	13370-POCATELLO	ID	2458.
UP	13336-SCOVILLE	ID	2514.

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: USG 16212-HANFORD S 300, WA

RR	NOOE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7739-FAIRFAX	SC	94.
CSXT	7732-ROBBINS	SC	123.
CSXT	7961-AUGUSTA	GA	152.
CSXT	7914-ATLANTA	GA	327.
CSXT	7907-MARIETTA	GA	337.
CSXT	7889-CARTERSVILLE	GA	369.
CSXT	7888-DALTON	GA	420.
CSXT	7235-CHATTANOOGA	TN	458.
CSXT	7187-TULLAHOMA	TN	539.
CSXT	7202-NASHVILLE	TN	618.
CSXT	7201-MADISON	TN	628.
CSXT	7061-HOPKINSVILLE	KY	688.
CSXT	3839-HENDERSON	KY	775.
CSXT	3838-EVANSVILLE	IN	788.
CSXT	3812-VINCENNES	IN	838.
CSXT	4952-SALEM	IL	917.
CSXT	10859-EAST ST LOUIS	IL	982.
<TR>	10859-EAST ST LOUIS	IL	982.
<TR>	10858-ST LOUIS	MO	988.
UP	10858-ST LOUIS	MO	988.
UP	10656-JEFFERSON CITY	MO	1110.
UP	10616-KANSAS CITY	MO	1286.
UP	10617-KANSAS CITY	KS	1289.
UP	11823-LAWRENCE	KS	1328.
UP	11697-TOPEKA	KS	1358.
UP	11696-MENOKEN	KS	1363.
UP	11681-MARYSVILLE	KS	1438.
UP	11405-HASTINGS	NE	1548.
UP	11410-GI88ON	NE	1574.
UP	11352-NORTH PLATTE	NE	1652.
UP	11358-O FALLONS	NE	1701.
UP	13703-JULES8URG	CO	1769.
UP	13465-CHEYENNE	WY	1915.
UP	13462-LARAMIE	WY	1967.
UP	13494-GRANGER	WY	2243.
UP	13369-MC CAMMON	ID	2435.
UP	13370-POCATELLO	ID	2458.
UP	13412-NAMPA	ID	2700.
UP	14220-PENOLETON	OR	2968.
UP	14223-HINKLE	OR	2999.
UP	13894-WALLULA	WA	3028.
UP	13964-KENNEWICK	WA	3043.
UP	13941-RICHLAND	WA	3052.
USG	13941-RICHLAND	WA	3052.
USG	16212-HANFORD S 300	WA	3060.

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7675-FLORENCE	SC	98.
CSXT	7671-DILLON	SC	127.
CSXT	7470-HAMLET	NC	165.
CSXT	7472-WADESBORO	NC	190.
WSS	7472-WADESBORO	NC	190.
WSS	7462-LEXINGTON	NC	258.
NS	7462-LEXINGTON	NC	258.
NS	7478-SALISBURY	NC	275.
NS	7394-HICKORY	NC	332.
NS	7387-MARION	NC	374.
NS	7343-ASHEVILLE	NC	414.
NS	7318-MORRISTOWN	TN	494.
NS	7286-KNOXVILLE	TN	535.
NS	7288-OOSSETT	TN	560.
NS	15316-K-25	TN	581.

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7739-FAIRFAX	SC	94.
CSXT	7732-ROBBINS	SC	123.
CSXT	7961-AUGUSTA	GA	152.
CSXT	7914-ATLANTA	GA	327.
CSXT	7907-MARIETTA	GA	337.
CSXT	7889-CARTERSVILLE	GA	369.
CSXT	7888-DALTON	GA	420.
CSXT	7235-CHATTANOOGA	TN	458.
CSXT	7187-TULLAHOMA	TN	539.
CSXT	7202-NASHVILLE	TN	618.
CSXT	7201-MADISON	TN	628.
CSXT	7061-HOPKINSVILLE	KY	688.
CSXT	3839-HENDERSON	KY	775.
CSXT	3838-EVANSVILLE	IN	788.
CSXT	3812-VINCENNES	IN	838.
CSXT	4952-SALEM	IL	917.
CSXT	10859-EAST ST LOUIS	IL	982.
<TR>	10859-EAST ST LOUIS	IL	982.
<TR>	10858-ST LOUIS	MO	988.
UP	10858-ST LOUIS	MO	988.
UP	10656-JEFFERSON CITY	MO	1110.
UP	10616-KANSAS CITY	MO	1286.
UP	10617-KANSAS CITY	KS	1289.
UP	11823-LAWRENCE	KS	1328.
UP	11697-TOPEKA	KS	1358.
UP	11696-MENOKEN	KS	1363.
UP	11681-MARYSVILLE	KS	1438.
UP	11405-HASTINGS	NE	1548.
UP	11410-GIBBON	NE	1574.
UP	11352-NORTH PLATTE	NE	1652.
UP	11358-O FALLONS	NE	1701.
UP	13703-JULESBURG	CO	1769.
UP	13465-CHEYENNE	WY	1915.
UP	13462-LARAMIE	WY	1967.
UP	13494-GRANGER	WY	2243.
UP	13568-OGDEN	UT	2382.
UP	13595-SALT LAKE CITY	UT	2417.
UP	13630-LYNNDYL	UT	2529.
UP	14766-VALLEY	NV	2846.
USG	14766-VALLEY	NV	2846.
USG	16333-YUCCA MOUNTAIN	NV	2945.

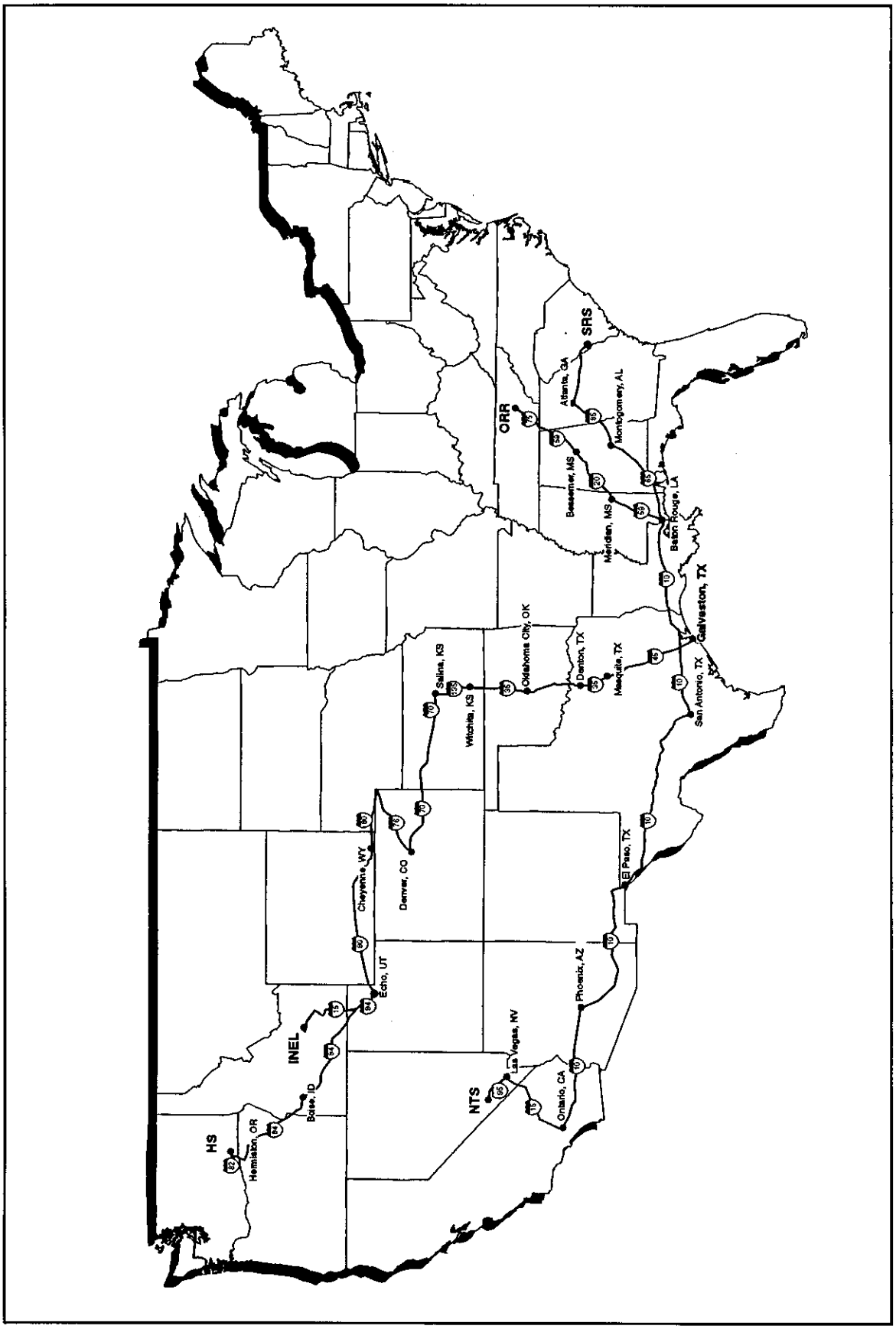

Figure E1-3 Representative Truck Routes from Galveston, TX to Department of Energy Management Sites

From: GALVESTON, TX
To : ID NATL ENG LAB

Routing through:

		GALVESTON				TX
I45		HOUSTON	SE	I45	I610	TX
I610		HOUSTON	N	I45	I610	TX
I45		DALLAS	SE	I45	I635	TX
I20	I635	MESQUITE	NW	I20	I635	TX
I635		DALLAS	NW	I35E	I635	TX
I35E		DENTON	S	I35E	I35W	TX
I35		OKLAHOMA CITY	S	I240	I35	OK
I240		OKLAHOMA CITY	SW	I240	I44	OK
I44		OKLAHOMA CITY	NE	I35	I44	OK
I35	I44	EDMOND	SE	I35	I44	OK
I35		SOUTH HAVEN	E	I35	U166	KS
I35 $	TKST$	WICHITA	S	I235	I35	KS
I235		WICHITA	N	I135	I235	KS
I135		SALINA	NW	I135	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LA8				ID

From: GALVESTON, TX
To : HANFORD, WA

Routing through:

		GALVESTON				TX
I45		HOUSTON	SE	I45	I610	TX
I610		HOUSTON	N	I45	I610	TX
I45		DALLAS	SE	I45	I635	TX
I20	I635	MESQUITE	NW	I20	I635	TX
I635		DALLAS	NW	I35E	I635	TX
I35E		DENTON	S	I35E	I35W	TX
I35		OKLAHOMA CITY	S	I240	I35	OK
I240		OKLAHOMA CITY	SW	I240	I44	OK
I44		OKLAHOMA CITY	NE	I35	I44	OK
I35	I44	EDMOND	SE	I35	I44	OK
I35		SOUTH HAVEN	E	I35	U166	KS
I35 $	TKST$	WICHITA	S	I235	I35	KS
I235		WICHITA	N	I135	I235	KS
I135		SALINA	NW	I135	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: GALVESTON, TX
To : K-25, TN

Routing through:

		GALVESTON				TX
I45		HOUSTON	SE	I45	I610	TX
I610		HOUSTON	E	I10	I610	TX
I10		BATON ROUGE	SE	I10	I12	LA
I12		SLIDELL	NE	I10	I12	LA
I59		MERIDIAN	W	I20	I59	MS
I20	I59	BESSEMER	SW	I20	I459	AL
yI459		CENTER POINT	SE	I459	I59	AL
I59		WILDWOOD	NW	I24	I59	GA
I24		EAST RIDGE	NE	I24	I75	TN
I75		OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: GALVESTON, TX
To : MERCURY, NV

Routing through:

		GALVESTON				TX
I45		HOUSTON	SE	I45	I610	TX
I610		HOUSTON	W	I10	I610	TX
I10		SAN ANTONIO	E	I10	I410	TX
I410		KIRBY	N	I35	I410	TX
I35	I410	SAN ANTONIO	NE	I35	I410	TX
I410		SAN ANTONIO	NW	I10	I410	TX
I10		PHOENIX	SE	I10	I17	AZ
I17		PHOENIX	W	I10	I17	AZ
I10		ONTARIO	E	I10	I15	CA
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

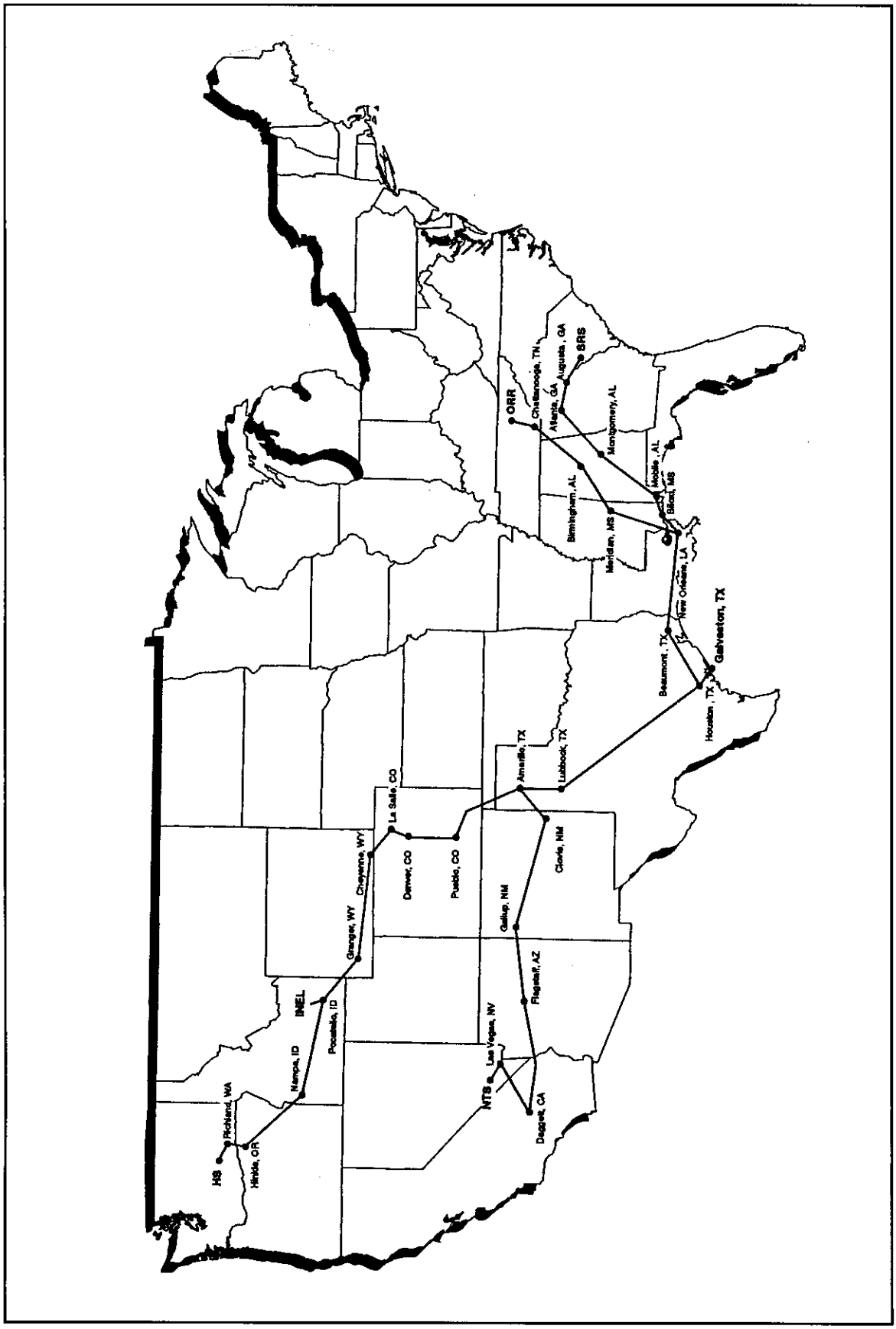

Figure E1-4 Representative Rail Routes from Galveston, TX to Department of Energy Management Sites

Route from: ATSF 13047-GALVESTON, TX
To: USG 15359-SRP, SC

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12399-HOUSTON	TX	68.
<TR>	12399-HOUSTON	TX	68.
UP	12399-HOUSTON	TX	68.
UP	12341-BEAUMONT	TX	156.
UP	9122-DE QUINCY	LA	203.
UP	9101-KINDER	LA	239.
UP	9112-LIVONIA	LA	317.
UP	8985-NEW ORLEANS	LA	446.
CSXT	8985-NEW ORLEANS	LA	446.
CSXT	8966-GULFPORT	MS	520.
CSXT	8926-BILOXI	MS	536.
CSXT	8967-PASCAGOULA	MS	553.
CSXT	8597-MOBILE	AL	592.
CSXT	8566-FLOMATON	AL	644.
CSXT	8657-MONTGOMERY	AL	769.
CSXT	8683-OPELIKA	AL	835.
CSXT	8142-LA GRANGE	GA	878.
CSXT	7914-ATLANTA	GA	954.
CSXT	7961-AUGUSTA	GA	1129.
CSXT	7732-ROBBINS	SC	1158.
CSXT	7717-DUNBARTON / WELL	SC	1167.
USG	7717-DUNBARTON / WELL	SC	1167.
USG	15359-SRP	SC	1175.

Route from: ATSF 13047-GALVESTON, TX
To: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12466-SOMERVILLE	TX	153.
ATSF	12480-TEMPLE	TX	230.
ATSF	12414-KILLEEN	TX	260.
ATSF	12725-BROWNWOOD	TX	353.
ATSF	12830-SWEETWATER	TX	468.
ATSF	12812-LUBBOCK	TX	582.
ATSF	12793-CANYON	TX	686.
ATSF	12792-AMARILLO	TX	703.
ATSF	13753-LA JUNTA	CO	963.
ATSF	13764-PUEBLO	CO	1018.
ATSF	13760-COLORADO SPRINGS	CO	1061.
ATSF	13727-DENVER	CO	1138.
UP	13727-DENVER	CO	1138.
UP	13712-LA SALLE	CO	1181.
UP	13709-GREELEY	CO	1188.
UP	13465-CHEYENNE	WY	1247.
UP	13462-LARAMIE	WY	1299.
UP	13494-GRANGER	WY	1575.
UP	13369-MC CAMMON	ID	1767.
UP	13370-POCATELLO	ID	1790.
UP	13336-SCOVILLE	ID	1846.

Route from: ATSF 13047-GALVESTON, TX
To: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12466-SOMERVILLE	TX	153.
ATSF	12480-TEMPLE	TX	230.
ATSF	12414-KILLEEN	TX	260.
ATSF	12725-BROWNWOOD	TX	353.
ATSF	12830-SWEETWATER	TX	468.
ATSF	12812-LUBBOCK	TX	582.
ATSF	12793-CANYON	TX	686.
ATSF	12792-AMARILLO	TX	703.
ATSF	13753-LA JUNTA	CO	963.
ATSF	13764-PUEBLO	CO	1018.
ATSF	13760-COLORADO SPRINGS	CO	1061.
ATSF	13727-DENVER	CO	1138.
UP	13727-DENVER	CO	1138.
UP	13712-LA SALLE	CO	1181.
UP	13709-GREELEY	CO	1188.
UP	13465-CHEYENNE	WY	1247.
UP	13462-LARAMIE	WY	1299.
UP	13494-GRANGER	WY	1575.
UP	13369-MC CAMMON	ID	1767.
UP	13370-POCATELLO	ID	1790.
UP	13412-NAMPA	ID	2032.
UP	14220-PENDLETON	OR	2301.
UP	14223-HINKLE	OR	2332.
UP	13894-WALLULA	WA	2361.
UP	13964-KENNEWICK	WA	2376.
UP	13941-RICHLAND	WA	2384.
USG	13941-RICHLAND	WA	2384.
USG	16212-HANFORD S 300	WA	2392.

Route From: ATSF 13D47-GALVESTON, TX
To: NS 15316-K-25, TN

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12399-HOUSTON	TX	68.
<TR>	12399-HOUSTON	TX	68.
UP	12399-HOUSTON	TX	68.
UP	12341-BEAUMONT	TX	156.
UP	9122-DE QUINCY	LA	203.
UP	9101-KINDER	LA	239.
UP	9112-LIVONIA	LA	317.
UP	8985-NEW ORLEANS	LA	446.
NS	8985-NEW ORLEANS	LA	446.
NS	8986-SLIDELL	LA	476.
NS	8963-HATTIESBURG	MS	558.
NS	8887-MERIDIAN	MS	641.
NS	8640-BOLIGEE	AL	693.
NS	8754-TUSCALOOSA	AL	737.
NS	8739-BIRMINGHAM	AL	807.
NS	8797-GADSDEN	AL	869.
NS	7235-CHATTANOOGA	TN	957.
NS	7260-HARRIMAN	TN	1038.
NS	15316-K-25	TN	1053.

Route from: ATSF 13047-GALVESTON, TX
To: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12466-SOMERVILLE	TX	153.
ATSF	12480-TEMPLE	TX	230.
ATSF	12414-KILLEEN	TX	260.
ATSF	12725-BROWNWOOD	TX	353.
ATSF	12830-SWEETWATER	TX	468.
ATSF	12812-LUBBOCK	TX	582.
ATSF	12806-FARWELL	TX	672.
ATSF	13025-CLOVIS	NM	683.
ATSF	12995-8ELEN	NM	924.
ATSF	12996-DALIES	NM	933.
ATSF	16077-GRANTS	NM	993.
ATSF	12999-GALLUP	NM	1070.
ATSF	12949-HOLBROOK	AZ	1181.
ATSF	12959-FLAGSTAFF	AZ	1276.
ATSF	12964-WILLIAMS	AZ	1305.
ATSF	12963-KINGMAN	AZ	1458.
ATSF	14663-DAGGETT	CA	1677.
UP	14663-DAGGETT	CA	1677.
UP	14762-LAS VEGAS	NV	1841.
UP	14766-VALLEY	NV	1856.
USG	14766-VALLEY	NV	1856.
USG	16333-YUCCA MOUNTAIN	NV	1955.

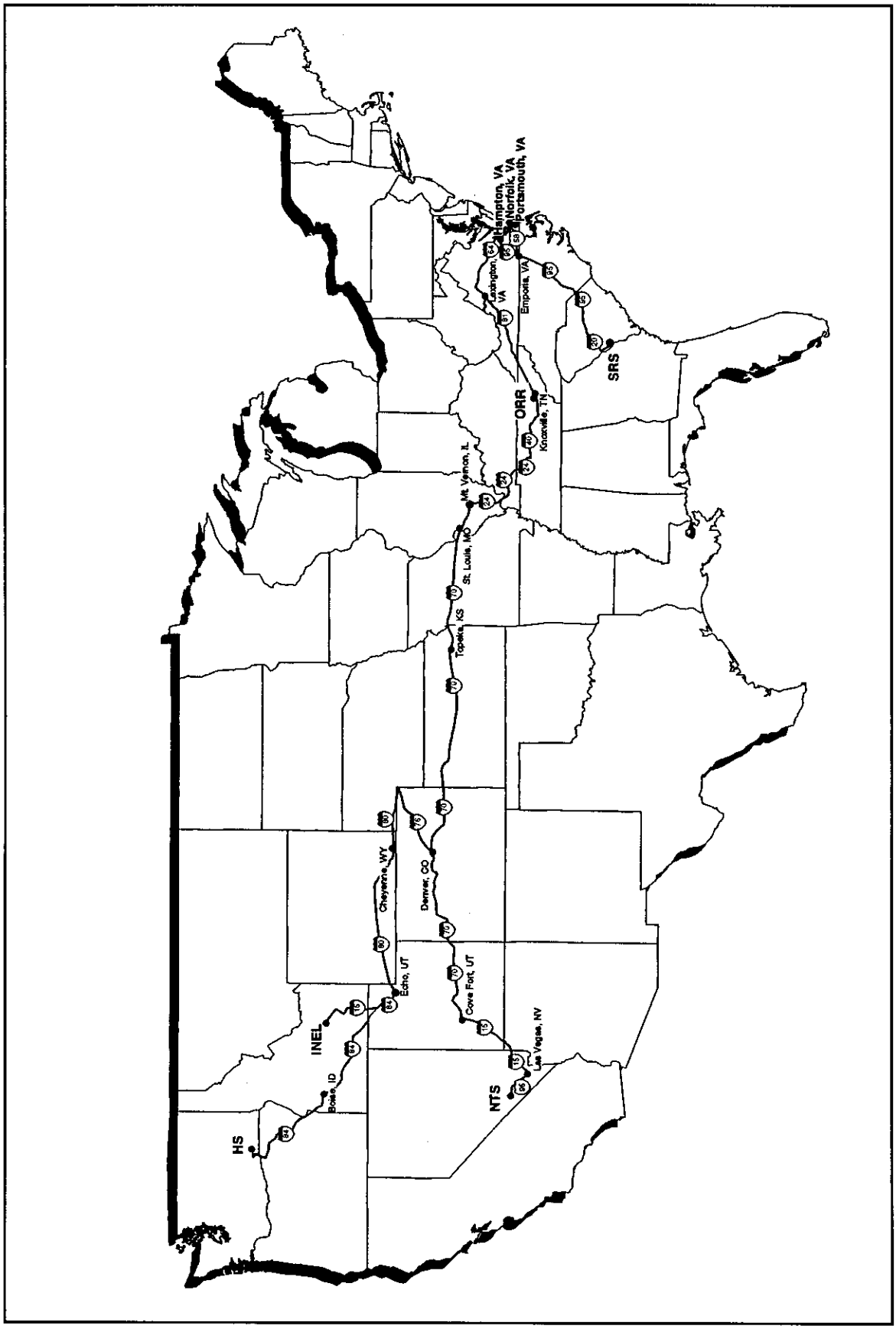

Figure E1-5 Representative Truck Routes from Hampton Roads Area Ports (Newport News, Norfolk, and Portsmouth, VA) to Department of Energy Management Sites

From: NEWPORT NEWS S I664 U60, VA
To : SRP, SC

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		HAMPTON	SE	I64	BRDG	VA
I64 #		WILLOGHBY BCH				VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U58		EMPORIA	N	I95	U58	VA
I95		FLORENCE	W	I20	I95	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
I230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: NEWPORT NEWS S I664 U60 VA
To : ID NATL ENG LAB ID

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		RICHMOND	N	I64	I95	VA
I64 $ I95 $		RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $ I77 $		CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $ TKST$		TOPEKA	E	I470	I70	KS
I470$ TKST$		TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: NEWPORT NEWS S I664 U60 VA
To : HANFORD, WA

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		RICHMOND	N	I64	I95	VA
I64 $ I95 $		RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $ I77 $		CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $ TKST$		TOPEKA	E	I470	I70	KS
I470$ TKST$		TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: NEWPORT NEWS S I664 U60 VA
To : K-25, TN

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		RICHMOND	N	I64	I95	VA
I64 $ I95 $		RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I81		FT CHISWELL	E	I77	I81	VA
I77	I81	WYTHEVILLE	E	I77	I81	VA
I81		DANDRIDGE	NE	I40	I81	TN
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: NEWPORT NEWS S I664 U60 VA
To : MERCURY, NV

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: NORFOLK E U13 U58 VA
To : SRP, SC

Routing through:

		NORFOLK	E	U13	U58	VA
U13		NORFOLK	NE	I64	U13	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U58		EMPORIA	N	I95	U58	VA
I95		FLORENCE	W	I20	I95	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: NORFOLK, E U13 U58 VA
To : ID NATL ENG LAB, ID

Routing through:

		NORFOLK	E	U13	U58	VA
U13		NORFOLK	NE	I64	U13	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U460		PETERSBURG				VA
I95 $	TRPT$	RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I95 $	TRPT$	RICHMOND	N	I95	U301	VA
I95		GLEN ALLEN	E	I295	I95	VA
I295		SHORT PUMP	NE	I295	I64	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I270		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: NORFOLK N I564 I64 VA
To : HANFORD, WA

Routing through:

		NORFOLK	N	I564	I64	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U460		PETERSBURG				VA
I95 $	TRPT$	RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I95 $	TRPT$	RICHMOND	N	I95	U301	VA
I95		GLEN ALLEN	E	I295	I95	VA
I295		SHORT PUMP	NE	I295	I64	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: NORFOLK E U13 U58 VA
To : K-25, TN

Routing through:

		NORFOLK	E	U13	U58	VA
U13		NORFOLK	NE	I64	U13	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U460		PETERSBURG				VA
I85 $		MATOACA	S	I85	U1	VA
I85		SUTHERLAND	E	I85	U460	VA
U460		BURKEVILLE	E	U360	U460	VA
U360	U460	BURKEVILLE	W	U360	U460	VA
U460		FARMVILLE	S	U15	U460	VA
U15	U460	FARMVILLE	W	U15	U460	VA
U460		BEDFORD				VA
U221	U460	BONSACK		U221	S220	VA
S220A		CLOVERDALE	N	I81	U220	VA
I81		FT CHISWELL	E	I77	I81	VA
I77	I81	WYTHEVILLE	E	I77	I81	VA
I81		DANORIDGE	NE	I40	I81	TN
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: NORFOLK E U13 U58 VA
To : MERCURY, NV

Routing through:

		NORFOLK	E	U13	U58	VA
U13		NORFOLK	NE	I64	U13	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U460		PETERSBURG				VA
I95 $	TRPT$	RICHMONO	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I95 $	TRPT$	RICHMOND	N	I95	U301	VA
I95		GLEN ALLEN	E	I295	I95	VA
I295		SHORT PUMP	NE	I295	I64	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: PORTSMOUTH, VA
To : SRP, SC

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U58		EMPORIA	N	I95	U58	VA
I95		FLORENCE	W	I20	I95	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: PORTSMOUTH, VA
To : ID NATL ENG LAB, ID

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		WILLOGHBY BCH				VA
I64 #		HAMPTON	SE	I64	BROG	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: PORTSMOUTH, VA
To : HANFORD, WA

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		WILLOGHBY BCH				VA
I64 #		HAMPTON	SE	I64	BRDG	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGOEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: PORTSMOUTH, VA
To : K-25, TN

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		WILLOGHBY BCH				VA
I64 #		HAMPTON	SE	I64	BRDG	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I81		FT CHISWELL	E	I77	I81	VA
I77	I81	WYTHEVILLE	E	I77	I81	VA
I81		DANDRIDGE	NE	I40	I81	TN
I4D		KNOXVILLE	NE	I40	I640	TN
I64D		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I4D	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: PORTSMOUTH, VA
To : MERCURY, NV

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		WILLOGHBY BCH				VA
I64 #		HAMPTON	SE	I64	BRDG	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EOWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

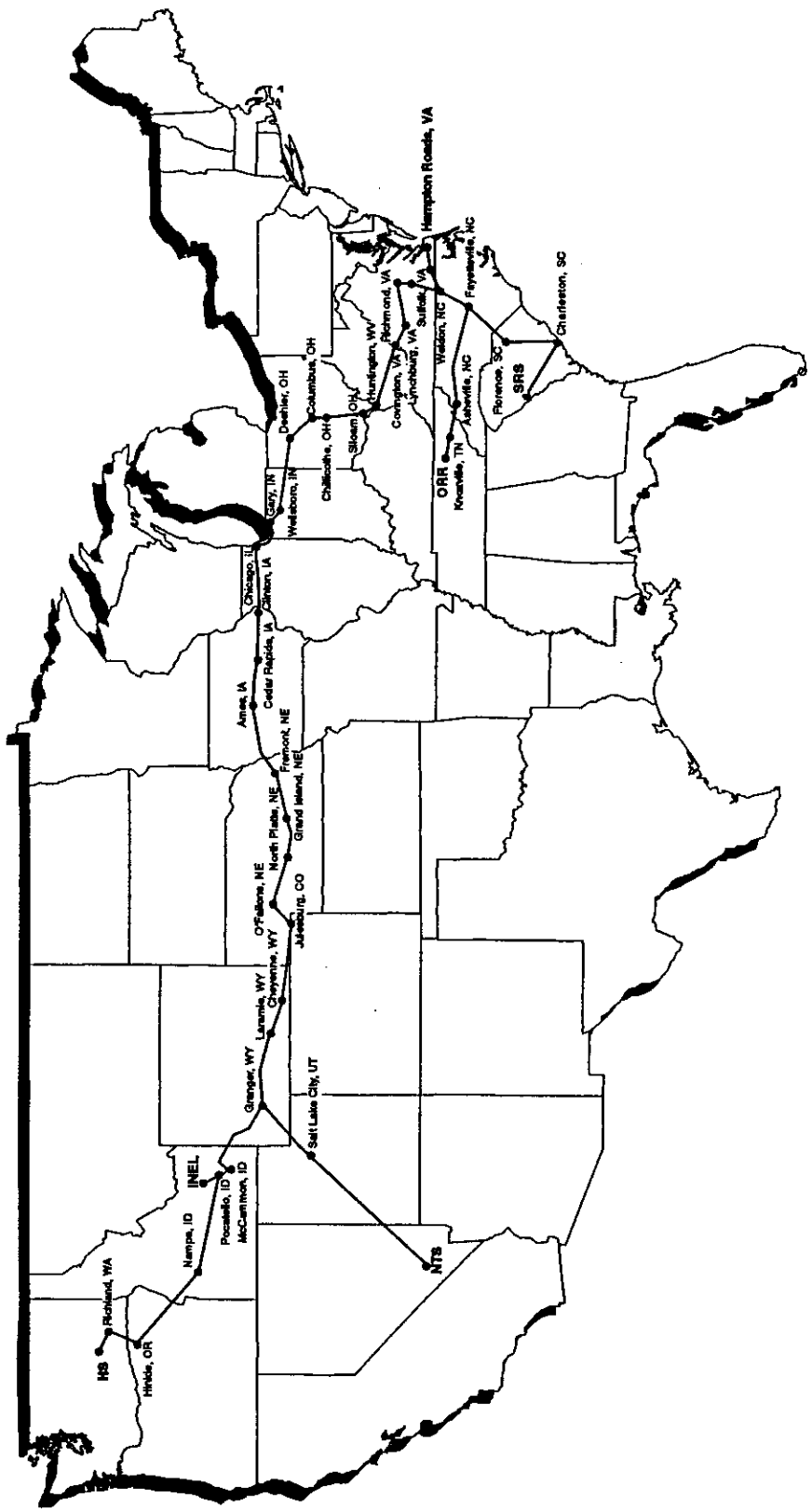

Figure E1-6 Representative Rail Routes from Hampton Roads Area Ports (Newport News, Norfolk, and Portsmouth, VA) to Department of Energy Management Sites

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6087-COLONIAL HEIGHTS	VA	97.
CSXT	6064-PETERSBURG	VA	103.
CSXT	7563-WELDON	NC	163.
CSXT	7565-ROCKY MOUNT	NC	200.
CSXT	7566-WILSON	NC	214.
CSXT	7606-FAYETTEVILLE	NC	288.
CSXT	7620-PEMBROKE	NC	317.
CSXT	7671-OILLON	SC	337.
CSXT	7675-FLORENCE	SC	366.
CSXT	7690-CHARLESTON	SC	464.
CSXT	7739-FAIRFAX	SC	558.
CSXT	7732-ROBBINS	SC	587.
CSXT	7717-DUNBARTON / WELL	SC	596.
USG	7717-DUNBARTON / WELL	SC	596.
USG	15359-SRP	SC	604.

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: UP 13336-SCOVILLE, IO

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6220-LYNCHBURG	VA	209.
CSXT	6198-GLASGOW	VA	235.
CSXT	6200-CLIFTON FORGE	VA	290.
CSXT	6189-COVINGTON	VA	317.
CSXT	6517-PRINCE	WV	384.
CSXT	6794-WEST CHARLESTOWN	WV	460.
CSXT	6795-ST ALBANS	WV	468.
CSXT	6806-BARBOURSVILLE	WV	499.
CSXT	6811-HUNTINGTON	WV	510.
CSXT	6808-KENOVA	WV	516.
CSXT	6807-CATLETTSBURG	KY	518.
CSXT	6809-ASHLAND	KY	524.
CSXT	6846-SILOAM	KY	547.
CSXT	3162-CHILLICOTHE	OH	609.
CSXT	3095-COLUMBUS (BROAD	OH	651.
CSXT	3402-MARION	OH	696.
CSXT	3002-FOSTORIA	OH	739.
CSXT	3484-DESHLER	OH	768.
CSXT	3993-WELLSBORO	IN	916.
CSXT	4070-GARY	IN	950.
CSXT	4073-CLARKE	IN	954.
CSXT	4074-INDIANA HARBOR	IN	957.
CSXT	4232-SOUTH CHICAGO	IL	965.
CSXT	4231-BURNSIDE	IL	968.
CSXT	4217-CHICAGO	IL	980.
CNW	4217-CHICAGO	IL	980.
CNW	4234-PROVISO	IL	994.
CNW	4311-DE KALB	IL	1036.
CNW	4324-NELSON	IL	1081.
CNW	10304-CLINTON	IA	1114.
CNW	10289-CEDAR RAPIDS	IA	1195.
CNW	10265-MARSHALLTOWN	IA	1262.
CNW	10246-NEVADA	IA	1289.
CNW	10271-AMES	IA	1300.
CNW	10176-MISSOURI VALLEY	IA	1433.
CNW	10198-CALIFORNIA JCT	IA	1439.
CNW	11340-FREMONT	NE	1467.
UP	11340-FREMONT	NE	1467.
UP	11406-GRAND ISLAND	NE	1576.
UP	11410-GIBBON	NE	1602.
UP	11352-NORTH PLATTE	NE	1680.
UP	11358-O FALLONS	NE	1729.
UP	13703-JULESBURG	CO	1797.
UP	13465-CHEYENNE	WY	1943.
UP	13462-LARAMIE	WY	1995.
UP	13494-GRANGER	WY	2271.
UP	13369-MC CAMMON	ID	2463.
UP	13370-POCATELLO	ID	2486.
UP	13336-SCOVILLE	ID	2542.

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: USG 16212-HANFORD S 300 WA

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6220-LYNCHBURG	VA	209.
CSXT	6198-GLASGOW	VA	235.
CSXT	6200-CLIFTON FORGE	VA	290.
CSXT	6189-COVINGTON	VA	317.
CSXT	6517-PRINCE	WV	384.
CSXT	6794-WEST CHARLESTOWN	WV	460.
CSXT	6795-ST ALBANS	WV	468.
CSXT	68D6-8ARBOURSVILLE	WV	499.
CSXT	6811-HUNTINGTON	WV	510.
CSXT	6808-KENOVA	WV	516.
CSXT	6807-CATLETTSBURG	KY	518.
CSXT	6809-ASHLAND	KY	524.
CSXT	6846-SILOAM	KY	547.
CSXT	3162-CHILLICOTHE	OH	609.
CSXT	3095-COLUMBUS (BROAD	OH	651.
CSXT	3402-MARION	OH	696.
CSXT	3002-FOSTORIA	OH	739.
CSXT	3484-DESHLER	OH	768.
CSXT	3993-WELLSBORO	IN	916.
CSXT	4070-GARY	IN	950.
CSXT	4073-CLARKE	IN	954.
CSXT	4074-INDIANA HARBOR	IN	957.
CSXT	4232-SOUTH CHICAGO	IL	965.
CSXT	4231-8URNSIDE	IL	968.
CSXT	4217-CHICAGO	IL	980.
CNW	4217-CHICAGO	IL	980.
CNW	4234-PROVISO	IL	994.
CNW	4311-DE KALB	IL	1036.
CNW	4324-NELSON	IL	1081.
CNW	10304-CLINTON	IA	1114.
CNW	10289-CEDAR RAPIDS	IA	1195.
CNW	10265-MARSHALLTOWN	IA	1262.
CNW	10246-NEVADA	IA	1289.
CNW	10271-AMES	IA	1300.
CNW	10176-MISSOURI VALLEY	IA	1433.
CNW	10198-CALIFORNIA JCT	IA	1439.
CNW	1134D-FREMONT	NE	1467.
UP	11340-FREMONT	NE	1467.
UP	11406-GRAND ISLAND	NE	1576.
UP	11410-GIBBON	NE	1602.
UP	11352-NORTH PLATTE	NE	1680.
UP	11358-O FALLONS	NE	1729.
UP	13703-JULESBURG	CO	1797.
UP	13465-CHEYENNE	WY	1943.
UP	13462-LARAMIE	WY	1995.
UP	13494-GRANGER	WY	2271.
UP	13369-MC CAMMON	ID	2463.
UP	13370-POCATELLO	ID	2486.
UP	13412-NAMPA	ID	2728.
UP	14220-PENDLETON	OR	2997.
UP	14223-HINKLE	OR	3028.
UP	13894-WALLULA	WA	3057.
UP	13964-KENNEWICK	WA	3072.
UP	13941-RICHLAND	WA	3080.
USG	13941-RICHLAND	WA	3080.
USG	16212-HANFORD S 300	WA	3088.

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6087-COLONIAL HEIGHTS	VA	97.
CSXT	6064-PETERSBURG	VA	103.
CSXT	7563-WELDON	NC	163.
CSXT	7565-ROCKY MOUNT	NC	200.
CSXT	7566-WILSON	NC	214.
CSXT	7606-FAYETTEVILLE	NC	288.
CSXT	7620-PEMBROKE	NC	317.
CSXT	7470-HAMLET	NC	348.
CSXT	7472-WADESBORO	NC	373.
WSS	7472-WADESBORO	NC	373.
WSS	7462-LEXINGTON	NC	441.
NS	7462-LEXINGTON	NC	441.
NS	7478-SALISBURY	NC	458.
NS	7394-HICKORY	NC	515.
NS	7387-MARION	NC	557.
NS	7343-ASHEVILLE	NC	597.
NS	7318-MORRISTOWN	TN	677.
NS	7286-KNOXVILLE	TN	718.
NS	7288-DOSSETT	TN	743.
NS	15316-K-25	TN	764.

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6220-LYNCHBURG	VA	209.
CSXT	6198-GLASGOW	VA	235.
CSXT	6200-CLIFTON FORGE	VA	290.
CSXT	6189-COVINGTON	VA	317.
CSXT	6517-PRINCE	WV	384.
CSXT	6794-WEST CHARLESTOWN	WV	460.
CSXT	6795-ST ALBANS	WV	468.
CSXT	6806-BARBOURSVILLE	WV	499.
CSXT	6811-HUNTINGTON	WV	510.
CSXT	6808-KENOVA	WV	516.
CSXT	6807-CATLETTSBURG	KY	518.
CSXT	6809-ASHLAND	KY	524.
CSXT	6846-SILOAM	KY	547.
CSXT	3162-CHILLICOTHE	OH	609.
CSXT	3095-COLUMBUS (BROAD	OH	651.
CSXT	3402-MARION	OH	696.
CSXT	3002-FOSTORIA	OH	739.
CSXT	3484-DESHLER	OH	768.
CSXT	3993-WELLSBORO	IN	916.
CSXT	4070-GARY	IN	950.
CSXT	4073-CLARKE	IN	954.
CSXT	4074-INDIANA HARBOR	IN	957.
CSXT	4232-SOUTH CHICAGO	IL	965.
CSXT	4231-BURNSIDE	IL	968.
CSXT	4217-CHICAGO	IL	980.
CNW	4217-CHICAGO	IL	980.
CNW	4234-PROVISO	IL	994.
CNW	4311-DE KALB	IL	1036.
CNW	4324-NELSON	IL	1081.
CNW	10304-CLINTON	IA	1114.
CNW	10289-CEDAR RAPIDS	IA	1195.
CNW	10265-MARSHALLTOWN	IA	1262.
CNW	10246-NEVADA	IA	1289.
CNW	10271-AMES	IA	1300.

ROUTE FROM : CSXT 6024-NEWPORT NEWS, VA
TO : USG 16333-YUCCA MOUNTAIN, NV -
Continued from Column 1

CNW	10176-MISSOURI VALLEY	IA	1433.
CNW	10198-CALIFORNIA JCT	IA	1439.
CNW	11340-FREMONT	NE	1467.
UP	11340-FREMONT	NE	1467.
UP	11406-GRAND ISLAND	NE	1576.
UP	11410-GIBBON	NE	1602.
UP	11352-NORTH PLATTE	NE	1680.
UP	1135B-O FALLONS	NE	1729.
UP	13703-JULESBURG	CO	1797.
UP	13465-CHEYENNE	WY	1943.
UP	13462-LARAMIE	WY	1995.
UP	13494-GRANGER	WY	2271.
UP	13568-OGDEN	UT	2410.
UP	13595-SALT LAKE CITY	UT	2445.
UP	13630-LYNNDYL	UT	2558.
UP	14766-VALLEY	NV	2875.
USG	14766-VALLEY	NV	2875.
USG	16333-YUCCA MOUNTAIN	NV	2974.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	6064-PETERSBURG	VA	148.
CSXT	6087-COLONIAL HEIGHTS	VA	154.
CSXT	6082-RICHMOND	VA	175.
CSXT	6220-LYNCHBURG	VA	308.
CSXT	6198-GLASGOW	VA	334.
CSXT	6200-CLIFTON FORGE	VA	389.
CSXT	6189-COVINGTON	VA	416.
CSXT	6517-PRINCE	WV	483.
CSXT	6794-WEST CHARLESTOWN	WV	559.
CSXT	6795-ST ALBANS	WV	567.
CSXT	6806-BARBOURSVILLE	WV	598.
CSXT	6811-HUNTINGTON	WV	609.
CSXT	6808-KENOVA	WV	615.
CSXT	6807-CATLETTSBURG	KY	617.
CSXT	6809-ASHLAND	KY	623.
CSXT	6846-SILOAM	KY	646.
CSXT	3162-CHILLICOTHE	OH	709.
CSXT	3095-COLUMBUS (BROAD	OH	751.
CSXT	3402-MARION	OH	796.
CSXT	3002-FOSTORIA	OH	839.
CSXT	3484-DESHLER	OH	867.
CSXT	3993-WELLSBORO	IN	1015.
CSXT	4070-GARY	IN	1049.
CSXT	4073-CLARKE	IN	1053.
CSXT	4074-INDIANA HARBOR	IN	1056.
CSXT	4232-SOUTH CHICAGO	IL	1064.
CSXT	4231-BURNSIDE	IL	1067.
CSXT	4217-CHICAGO	IL	1079.
CNW	4217-CHICAGO	IL	1079.
CNW	4234-PROVISO	IL	1094.
CNW	4311-DE KALB	IL	1136.
CNW	4324-NELSON	IL	1181.
CNW	10304-CLINTON	IA	1213.
CNW	10289-CEDAR RAPIDS	IA	1294.
CNW	10265-MARSHALLTOWN	IA	1361.
CNW	10246-NEVADA	IA	1388.
CNW	10271-AMES	IA	1399.
CNW	10176-MISSOURI VALLEY	IA	1532.
CNW	10198-CALIFORNIA JCT	IA	1538.
CNW	11340-FREMONT	NE	1566.
UP	11340-FREMONT	NE	1566.
UP	11406-GRAND ISLAND	NE	1675.
UP	11410-GIBBON	NE	1701.
UP	11352-NORTH PLATTE	NE	1779.
UP	11358-O FALLONS	NE	1828.
UP	13703-JULESBURG	CO	1896.
UP	13465-CHEYENNE	WY	2042.
UP	13462-LARAMIE	WY	2094.
UP	13494-GRANGER	WY	2370.
UP	13369-MC CAMMON	ID	2562.
UP	13370-POCATELLO	ID	2585.
UP	13336-SCOVILLE	ID	2641.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: USG 16212-HANFORD S 300 WA

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	6064-PETERSBURG	VA	148.
CSXT	6087-COLONIAL HEIGHTS	VA	154.
CSXT	6082-RICHMOND	VA	175.
CSXT	6220-LYNCHBURG	VA	308.
CSXT	6198-GLASGOW	VA	334.
CSXT	6200-CLIFTON FORGE	VA	389.
CSXT	6189-COVINGTON	VA	416.
CSXT	6517-PRINCE	WV	483.
CSXT	6794-WEST CHARLESTOWN	WV	559.
CSXT	6795-ST ALBANS	WV	567.
CSXT	6806-BARBOURSVILLE	WV	598.
CSXT	6811-HUNTINGTON	WV	609.
CSXT	6808-KENOVA	WV	615.
CSXT	6807-CATLETTSBURG	KY	617.
CSXT	6809-ASHLAND	KY	623.
CSXT	6846-SILOAM	KY	646.
CSXT	3162-CHILLICOTHE	OH	709.
CSXT	3095-COLUMBUS (BROAD	OH	751.
CSXT	3402-MARION	OH	796.
CSXT	3002-FOSTORIA	OH	839.
CSXT	3484-DESHLER	OH	867.
CSXT	3993-WELLSBORO	IN	1015.
CSXT	4070-GARY	IN	1049.
CSXT	4073-CLARKE	IN	1053.
CSXT	4074-INDIANA HARBOR	IN	1056.
CSXT	4232-SOUTH CHICAGO	IL	1064.
CSXT	4231-BURNSIDE	IL	1067.
CSXT	4217-CHICAGO	IL	1079.
CNW	4217-CHICAGO	IL	1079.
CNW	4234-PROVISO	IL	1094.
CNW	4311-DE KALB	IL	1136.
CNW	4324-NELSON	IL	1181.
CNW	10304-CLINTON	IA	1213.
CNW	10289-CEDAR RAPIDS	IA	1294.
CNW	10265-MARSHALLTOWN	IA	1361.
CNW	10246-NEVADA	IA	1388.
CNW	10271-AMES	IA	1399.
CNW	10176-MISSOURI VALLEY	IA	1532.
CNW	10198-CALIFORNIA JCT	IA	1538.
CNW	11340-FREMONT	NE	1566.
UP	11340-FREMONT	NE	1566.
UP	11406-GRAND ISLAND	NE	1675.
UP	11410-GIBBON	NE	1701.
UP	11352-NORTH PLATTE	NE	1779.
UP	11358-O FALLONS	NE	1828.
UP	13703-JULESBURG	CO	1896.
UP	13465-CHEYENNE	WY	2042.
UP	13462-LARAMIE	WY	2094.
UP	13494-GRANGER	WY	2370.
UP	13369-MC CAMMON	ID	2562.
UP	13370-POCATELLO	ID	2585.
UP	13412-NAMPA	ID	2827.
UP	14220-PENDLETON	OR	3096.
UP	14223-HINKLE	OR	3127.
UP	13894-WALLULA	WA	3156.
UP	13964-KENNEWICK	WA	3171.
UP	13941-RICHLAND	WA	3179.
USG	13941-RICHLAND	WA	3179.
USG	16212-HANFORD S 300	WA	3187.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	7565-ROCKY MOUNT	NC	125.
CSXT	7566-WILSON	NC	139.
CSXT	7606-FAYETTEVILLE	NC	213.
CSXT	7620-PEMBROKE	NC	242.
CSXT	7470-HAMLET	NC	273.
CSXT	7472-WADESBORO	NC	298.
WSS	7472-WADESBORO	NC	298.
WSS	7462-LEXINGTON	NC	366.
NS	7462-LEXINGTON	NC	366.
NS	7478-SALISBURY	NC	383.
NS	7394-HICKORY	NC	440.
NS	7387-MARION	NC	482.
NS	7343-ASHEVILLE	NC	522.
NS	7318-MORRISTOWN	TN	602.
NS	7286-KNOXVILLE	TN	643.
NS	7288-DOSSETT	TN	668.
NS	15316-K-25	TN	689.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	7565-ROCKY MOUNT	NC	125.
CSXT	7566-WILSON	NC	139.
CSXT	7606-FAYETTEVILLE	NC	213.
CSXT	7620-PEMBROKE	NC	242.
CSXT	7671-DILLON	SC	262.
CSXT	7675-FLORENCE	SC	291.
CSXT	7690-CHARLESTON	SC	389.
CSXT	7739-FAIRFAX	SC	483.
CSXT	7732-ROBBINS	SC	512.
CSXT	7717-DUNBARTON / WELL	SC	521.
USG	7717-DUNBARTON / WELL	SC	521.
USG	15359-SRP	SC	529.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	6064-PETERSBURG	VA	148.
CSXT	6087-COLONIAL HEIGHTS	VA	154.
CSXT	6082-RICHMOND	VA	175.
CSXT	6220-LYNCHBURG	VA	308.
CSXT	6198-GLASGOW	VA	334.
CSXT	6200-CLIFTON FORGE	VA	389.
CSXT	6189-COVINGTON	VA	416.
CSXT	6517-PRINCE	WV	483.
CSXT	6794-WEST CHARLESTOWN	WV	559.
CSXT	6795-ST ALBANS	WV	567.
CSXT	6806-BARBOURSVILLE	WV	598.
CSXT	6811-HUNTINGTON	WV	609.
CSXT	6808-KENOVA	WV	615.
CSXT	6807-CATLETTSBURG	KY	617.
CSXT	6809-ASHLAND	KY	623.
CSXT	6846-SILOAM	KY	646.
CSXT	3162-CHILLICOTHE	OH	709.
CSXT	3095-COLUMBUS (BROAD	OH	751.
CSXT	3402-MARION	OH	796.
CSXT	3002-FOSTORIA	OH	839.
CSXT	3484-DESHLER	OH	867.
CSXT	3993-WELLSBORO	IN	1015.
CSXT	4070-GARY	IN	1049.
CSXT	4073-CLARKE	IN	1053.
CSXT	4074-INDIANA HARBOR	IN	1056.
CSXT	4232-SOUTH CHICAGO	IL	1064.
CSXT	4231-BURNSIDE	IL	1067.
CSXT	4217-CHICAGO	IL	1079.
CNW	4217-CHICAGO	IL	1079.
CNW	4234-PROVISO	IL	1094.
CNW	4311-DE KALB	IL	1136.
CNW	4324-NELSON	IL	1181.
CNW	10304-CLINTON	IA	1213.
CNW	10289-CEDAR RAPIDS	IA	1294.
CNW	10265-MARSHALLTOWN	IA	1361.
CNW	10246-NEVADA	IA	1388.
CNW	10271-AMES	IA	1399.
CNW	10176-MISSOURI VALLEY	IA	1532.
CNW	10198-CALIFORNIA JCT	IA	1538.
CNW	11340-FREMONT	NE	1566.
UP	11340-FREMONT	NE	1566.
UP	11406-GRAND ISLAND	NE	1675.
UP	11410-GIBBON	NE	1701.
UP	11352-NORTH PLATTE	NE	1779.
UP	11358-O FALLONS	NE	1828.
UP	13703-JULESBURG	CO	1896.
UP	13465-CHEYENNE	WY	2042.
UP	13462-LARAMIE	WY	2094.
UP	13494-GRANGER	WY	2370.
UP	13568-OGDEN	UT	2509.
UP	13595-SALT LAKE CITY	UT	2545.
UP	13630-LYNNDYL	UT	2657.
UP	14766-VALLEY	NV	2974.
USG	14766-VALLEY	NV	2974.
USG	16333-YUCCA MOUNTAIN	NV	3073.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	7565-ROCKY MOUNT	NC	95.
CSXT	7566-WILSON	NC	109.
CSXT	7606-FAYETTEVILLE	NC	183.
CSXT	7620-PEMBROKE	NC	212.
CSXT	7671-DILLON	SC	232.
CSXT	7675-FLORENCE	SC	261.
CSXT	7690-CHARLESTON	SC	359.
CSXT	7739-FAIRFAX	SC	453.
CSXT	7732-ROBBINS	SC	482.
CSXT	7717-DUNBARTON / WELL	SC	491.
USG	7717-DUNBARTON / WELL	SC	491.
USG	15359-SRP	SC	499.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	6064-PETERSBURG	VA	118.
CSXT	6087-COLONIAL HEIGHTS	VA	124.
CSXT	6082-RICHMOND	VA	145.
CSXT	6220-LYNCHBURG	VA	278.
CSXT	6198-GLASGOW	VA	304.
CSXT	6200-CLIFTON FORGE	VA	359.
CSXT	6189-COVINGTON	VA	386.
CSXT	6517-PRINCE	WV	453.
CSXT	6794-WEST CHARLESTOWN	WV	529.
CSXT	6795-ST ALBANS	WV	537.
CSXT	6806-BARBOURSVILLE	WV	568.
CSXT	6811-HUNTINGTON	WV	579.
CSXT	6808-KENOVA	WV	585.
CSXT	6807-CATLETTSBURG	KY	587.
CSXT	6809-ASHLAND	KY	593.
CSXT	6846-SILOAM	KY	616.
CSXT	3162-CHILLICOTHE	OH	679.
CSXT	3095-COLUMBUS (BROAD	OH	721.
CSXT	3402-MARION	OH	766.
CSXT	3002-FOSTORIA	OH	809.
CSXT	3484-DESHLER	OH	837.
CSXT	3993-WELLSBORO	IN	985.
CSXT	4070-GARY	IN	1019.
CSXT	4073-CLARKE	IN	1023.
CSXT	4074-INDIANA HARBOR	IN	1026.
CSXT	4232-SOUTH CHICAGO	IL	1034.
CSXT	4231-BURNSIDE	IL	1037.
CSXT	4217-CHICAGO	IL	1049.
CNW	4217-CHICAGO	IL	1049.
CNW	4234-PROVISO	IL	1064.
CNW	4311-DE KALB	IL	1106.
CNW	4324-NELSON	IL	1151.
CNW	10304-CLINTON	IA	1183.
CNW	10289-CEDAR RAPIDS	IA	1264.
CNW	10265-MARSHALLTOWN	IA	1331.
CNW	10246-NEVADA	IA	1358.
CNW	10271-AMES	IA	1369.
CNW	10176-MISSOURI VALLEY	IA	1502.
CNW	10198-CALIFORNIA JCT	IA	1508.
CNW	11340-FREMONT	NE	1536.
UP	11340-FREMONT	NE	1536.
UP	11406-GRAND ISLAND	NE	1645.
UP	11410-GIBBON	NE	1671.
UP	11352-NORTH PLATTE	NE	1749.
UP	11358-O FALLONS	NE	1798.
UP	13703-JULESBURG	CO	1866.
UP	13465-CHEYENNE	WY	2012.
UP	13462-LARAMIE	WY	2064.
UP	13494-GRANGER	WY	2340.
UP	13369-MC CAMMON	ID	2532.
UP	13370-POCATELLO	ID	2555.
UP	13336-SCOVILLE	ID	2611.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	6064-PETERSBURG	VA	118.
CSXT	6087-COLONIAL HEIGHTS	VA	124.
CSXT	6082-RICHMOND	VA	145.
CSXT	6220-LYNCHBURG	VA	278.
CSXT	6198-GLASGOW	VA	304.
CSXT	6200-CLIFTON FORGE	VA	359.
CSXT	6189-COVINGTON	VA	386.
CSXT	6517-PRINCE	WV	453.
CSXT	6794-WEST CHARLESTOWN	WV	529.
CSXT	6795-ST ALBANS	WV	537.
CSXT	6806-BARBOURSVILLE	WV	568.
CSXT	6811-HUNTINGTON	WV	579.
CSXT	6808-KENOVA	WV	585.
CSXT	6807-CATLETTSBURG	KY	587.
CSXT	6809-ASHLAND	KY	593.
CSXT	6846-SILOAM	KY	616.
CSXT	3162-CHILLICOTHE	OH	679.
CSXT	3095-COLUMBUS (BROAD	OH	721.
CSXT	3402-MARION	OH	766.
CSXT	3002-FOSTORIA	OH	809.
CSXT	3484-DESHLER	OH	837.
CSXT	3993-WELLSBORO	IN	985.
CSXT	4070-GARY	IN	1019.
CSXT	4073-CLARKE	IN	1023.
CSXT	4074-INDIANA HARBOR	IN	1026.
CSXT	4232-SOUTH CHICAGO	IL	1034.
CSXT	4231-BURNSIDE	IL	1037.
CSXT	4217-CHICAGO	IL	1049.
CNW	4217-CHICAGO	IL	1049.
CNW	4234-PROVISO	IL	1064.
CNW	4311-DE KALB	IL	1106.
CNW	4324-NELSON	IL	1151.
CNW	10304-CLINTON	IA	1183.
CNW	10289-CEDAR RAPIDS	IA	1264.
CNW	10265-MARSHALLTOWN	IA	1331.
CNW	10246-NEVADA	IA	1358.
CNW	10271-AMES	IA	1369.
CNW	10176-MISSOURI VALLEY	IA	1502.
CNW	10198-CALIFORNIA JCT	IA	1508.
CNW	11340-FREMONT	NE	1536.
UP	11340-FREMONT	NE	1536.
UP	11406-GRAND ISLAND	NE	1645.
UP	1141D-GIBBDN	NE	1671.
UP	11352-NORTH PLATTE	NE	1749.
UP	11358-O FALLDNS	NE	1798.
UP	13703-JULESBURG	CO	1866.
UP	13465-CHEYENNE	WY	2012.
UP	13462-LARAMIE	WY	2064.
UP	13494-GRANGER	WY	2340.
UP	13369-MC CAMMON	ID	2532.
UP	13370-PDCATELLO	ID	2555.
UP	13412-NAMPA	ID	2797.
UP	14220-PENDLETON	OR	3066.
UP	14223-HINKLE	OR	3097.
UP	13894-WALLULA	WA	3126.
UP	13964-KENNEWICK	WA	3141.
UP	13941-RICHLAND	WA	3149.
USG	13941-RICHLAND	WA	3149.
USG	16212-HANFORD S 300	WA	3157.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	7565-ROCKY MOUNT	NC	95.
CSXT	7566-WILSON	NC	109.
CSXT	7606-FAYETTEVILLE	NC	183.
CSXT	7620-PEMBROKE	NC	212.
CSXT	7470-HAMLET	NC	243.
CSXT	7472-WADESBORO	NC	268.
WSS	7472-WADESBORO	NC	268.
WSS	7462-LEXINGTON	NC	336.
NS	7462-LEXINGTON	NC	336.
NS	7478-SALISBURY	NC	353.
NS	7394-HICKORY	NC	410.
NS	7387-MARION	NC	452.
NS	7343-ASHEVILLE	NC	492.
NS	7318-MORRISTOWN	TN	572.
NS	7286-KNOXVILLE	TN	613.
NS	7288-DOSSETT	TN	638.
NS	15316-K-25	TN	659.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	6064-PETERSBURG	VA	118.
CSXT	6087-COLONIAL HEIGHTS	VA	124.
CSXT	6082-RICHMOND	VA	145.
CSXT	6220-LYNCHBURG	VA	278.
CSXT	6198-GLASGOW	VA	304.
CSXT	6200-CLIFTON FORGE	VA	359.
CSXT	6189-COVINGTON	VA	386.
CSXT	6517-PRINCE	WV	453.
CSXT	6794-WEST CHARLESTOWN	WV	529.
CSXT	6795-ST ALBANS	WV	537.
CSXT	6806-BARBOURSVILLE	WV	568.
CSXT	6811-HUNTINGTON	WV	579.
CSXT	6808-KENOVA	WV	585.
CSXT	6807-CATLETTSBURG	KY	587.
CSXT	6809-ASHLAND	KY	593.
CSXT	6846-SILOAM	KY	616.
CSXT	3162-CHILLICOTHE	OH	679.
CSXT	3095-COLUMBUS (BROAD	OH	721.
CSXT	3402-MARION	OH	766.
CSXT	3002-FOSTORIA	OH	809.
CSXT	3484-DESHLER	OH	837.
CSXT	3993-WELLSBORO	IN	985.
CSXT	4070-GARY	IN	1019.
CSXT	4073-CLARKE	IN	1023.
CSXT	4074-INDIANA HARBOR	IN	1026.
CSXT	4232-SOUTH CHICAGO	IL	1034.
CSXT	4231-BURNSIDE	IL	1037.
CSXT	4217-CHICAGO	IL	1049.
CNW	4217-CHICAGO	IL	1049.
CNW	4234-PROVISO	IL	1064.
CNW	4311-DE KALB	IL	1106.
CNW	4324-NELSON	IL	1151.
CNW	10304-CLINTON	IA	1183.
CNW	10289-CEDAR RAPIDS	IA	1264.
CNW	10265-MARSHALLTOWN	IA	1331.
CNW	10246-NEVADA	IA	1358.
CNW	10271-AMES	IA	1369.
CNW	10176-MISSOURI VALLEY	IA	1502.
CNW	10198-CALIFORNIA JCT	IA	1508.
CNW	11340-FREMONT	NE	1536.
UP	11340-FREMONT	NE	1536.
UP	11406-GRAND ISLAND	NE	1645.
UP	11410-GIBBON	NE	1671.
UP	11352-NORTH PLATTE	NE	1749.
UP	11358-O FALLONS	NE	1798.
UP	13703-JULESBURG	CO	1866.
UP	13465-CHEYENNE	WY	2012.
UP	13462-LARAMIE	WY	2064.
UP	13494-GRANGER	WY	2340.
UP	13568-OGDEN	UT	2479.
UP	13595-SALT LAKE CITY	UT	2515.
UP	13630-LYNNDYL	UT	2627.
UP	14766-VALLEY	NV	2944.
USG	14766-VALLEY	NV	2944.
USG	16333-YUCCA MOUNTAIN	NV	3043.

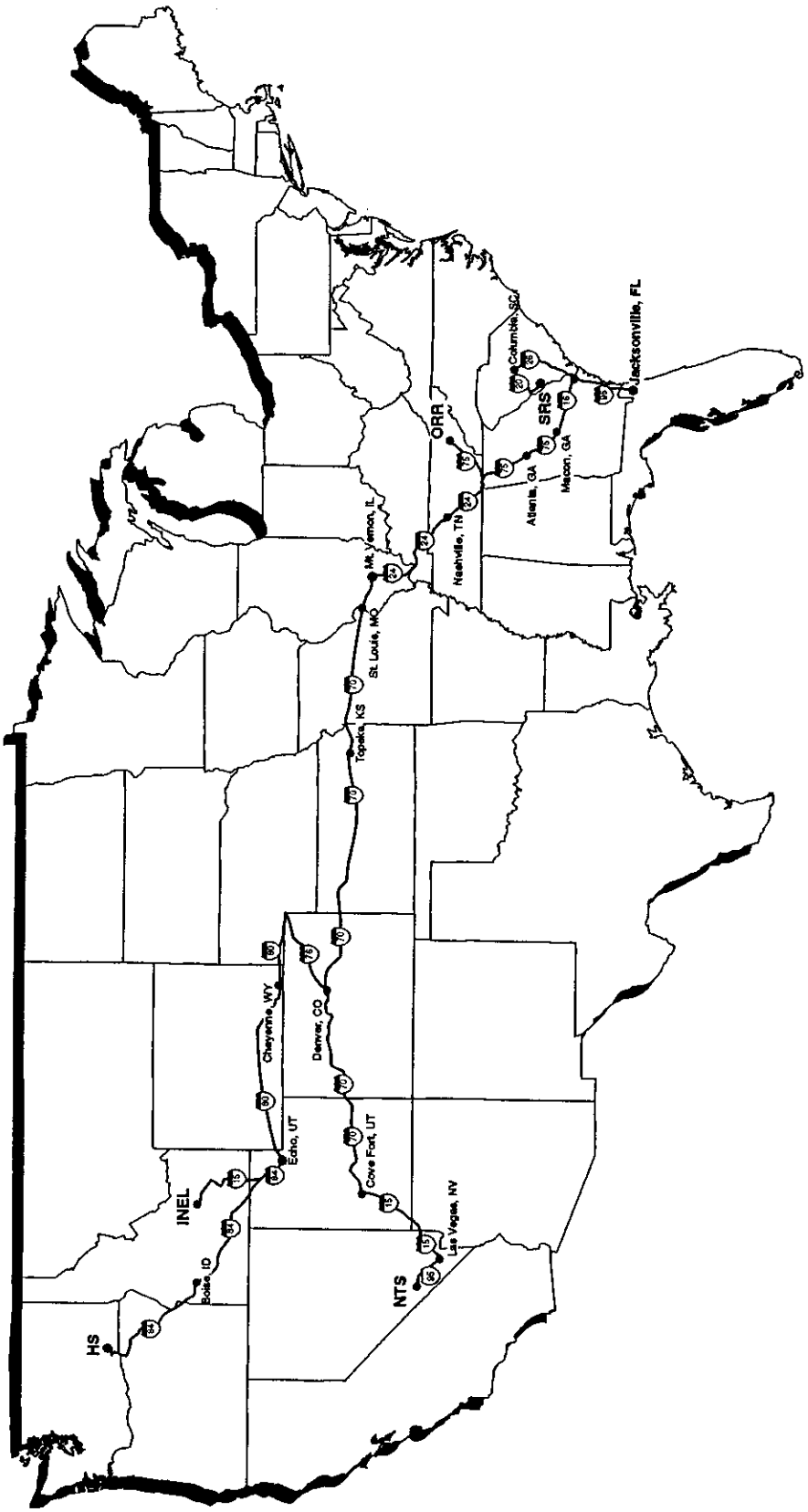

Figure E1-7 Representative Truck Routes from Jacksonville, FL to Department of Energy Management Sites

From: JACKSONVILLE, FL
To : SRP, SC

Routing through:

	JACKSONVILLE				FL
U23	JACKSONVILLE	W	I95	U23	FL
I95	JACKSONVILLE	NW	I95	U1	FL
I95 #	JACKSONVILLE	N	I295	I95	FL
I95	ROSINVILLE	N	I26	I95	SC
I26	COLUMBIA	NW	I20	I26	SC
I20	NORTH AUGUSTA	NW	I20	S230	SC
S230	NORTH AUGUSTA				SC
S125	CLEARWATER	W	U1	U278	SC
U278	BEECH ISLAND		U278	S125	SC
S125	JACKSON	SE	S125	LSRP	SC
LSRP	SRP				SC

From: JACKSONVILLE, FL
To : ID NATL ENG LAB, ID

Routing through:

	JACKSONVILLE				FL
U23	JACKSONVILLE	W	I95	U23	FL
I95	JACKSONVILLE	NW	I95	U1	FL
I95 #	JACKSONVILLE	N	I295	I95	FL
1295	JACKSONVILLE	W	I10	I295	FL
I10	WINFIELD	W	I10	I75	FL
I75	MACON	S	I475	I75	GA
I475	SMARR	E	I475	I75	GA
I75	HAPEVILLE	S	I285	I75	GA
I285	COLLEGE PARK	S	I285	I85	GA
I285 I85	RED OAK	E	I285	I85	GA
I285	ATLANTA	NW	I285	I75	GA
I75	EAST RIDGE	NE	I24	I75	TN
I24	NASHVILLE	E	I24	I40	TN
I24 I40	NASHVILLE	SE	I24	I40	TN
I24 I65	INGLEWOOD	W	I24	I65	TN
I24	PULLEYS MILL	W	I24	I57	IL
I57	MT VERNON	SW	I57	I64	IL
I57 I64	MT VERNON	NW	I57	I64	IL
I64	WASHINGTON PK	SE	I255	I64	IL
I255	EDWARDSVILLE	SW	I255	I270	IL
I270	ST LOUIS	NW	I270	I70	MO
I70	KANSAS CITY	SE	I435	I70	MO
I435	KANSAS CITY	W	I435	I70	KS
I70 $ TKST$	TOPEKA	E	I470	I70	KS
I470$ TKST$	TOPEKA	S	I335	I470	KS
I470	TOPEKA	W	I470	I70	KS
I70	DENVER	NE	I270	I70	CO
I270	COMMERCE CITY	NW	I270	I76	CO
I76	COMMERCE CITY	W	I25	I76	CO
I25	CHEYENNE	SW	I25	I80	WY
I80	ECHO		I80	I84	UT
I84	OGDEN	S	I15	I84	UT
I15 I84	TREMONTON	W	I15	I84	UT
I15	BLACKFOOT	NW	I15	X92	ID
U26	ATOMIC CITY	NW	U20	U26	ID
U20 U26	IO NATL ENG LAB				ID

From: JACKSONVILLE, FL
To : HANFORD, WA

Routing through:

	JACKSONVILLE				FL
U23	JACKSONVILLE	W	I95	U23	FL
I95	JACKSONVILLE	NW	I95	U1	FL
I95 #	JACKSONVILLE	N	I295	I95	FL
I295	JACKSONVILLE	W	I10	I295	FL
I10	WINFIELO	W	I10	I75	FL
I75	MACON	S	I475	I75	GA
I475	SMARR	E	I475	I75	GA
I75	HAPEVILLE	S	I285	I75	GA
I285	COLLEGE PARK	S	I285	I85	GA
I285 I85	RED OAK	E	I285	I85	GA
I285	ATLANTA	NW	I285	I75	GA
I75	EAST RIDGE	NE	I24	I75	TN
I24	NASHVILLE	E	I24	I40	TN
I24 I40	NASHVILLE	SE	I24	I40	TN
I24 I65	INGLEWOOD	W	I24	I65	TN
I24	PULLEYS MILL	W	I24	I57	IL
I57	MT VERNON	SW	I57	I64	IL
I57 I64	MT VERNON	NW	I57	I64	IL
I64	WASHINGTON PK	SE	I255	I64	IL
I255	EDWARDSVILLE	SW	I255	I270	IL
I270	ST LOUIS	NW	I270	I70	MO
I70	KANSAS CITY	SE	I435	I70	MO
I435	KANSAS CITY	W	I435	I70	KS
I70 $ TKST$	TOPEKA	E	I470	I70	KS
I470$ TKST$	TOPEKA	S	I335	I470	KS
I470	TOPEKA	W	I470	I70	KS
I70	OENVER	NE	I270	I70	CO
I270	COMMERCE CITY	NW	I270	I76	CO
I76	COMMERCE CITY	W	I25	I76	CO
I25	CHEYENNE	SW	I25	I80	WY
I80	ECHO		I80	I84	UT
I84	OGOEN	S	I15	I84	UT
I15 I84	TREMONTON	W	I15	I84	UT
I84	HERMISTON	SW	I82	I84	OR
I82	WEST RICHLANO	S	I182	I82	WA
I182	RICHLAND	SE	I182	S240	WA
S240	RICHLAND	N	S240	LR4S	WA
LR4S	HANFORD				WA

From: JACKSONVILLE, FL
To : K-25, TN

Routing through:

	JACKSONVILLE				FL
U23	JACKSONVILLE	W	I95	U23	FL
I95	JACKSONVILLE	NW	I95	U1	FL
I95 #	JACKSONVILLE	N	I295	I95	FL
I295	JACKSONVILLE	W	I10	I295	FL
I10	WINFIELD	W	I10	I75	FL
I75	MACON	S	I475	I75	GA
I475	SMARR	E	I475	I75	GA
I75	HAPEVILLE	S	I285	I75	GA
I285	COLLEGE PARK	S	I285	I85	GA
I285 I85	RED OAK	E	I285	I85	GA
I285	ATLANTA	NW	I285	I75	GA
I75	OAK RIDGE	S	I40	I75	TN
I40	KINGSTON	E	I40	S58	TN
S58	K-25				TN

From: JACKSONVILLE, FL
To : MERCURY, NV

Routing through:

		JACKSONVILLE				FL
U23		JACKSONVILLE	W	I95	U23	FL
I95		JACKSONVILLE	NW	I95	U1	FL
I95 #		JACKSONVILLE	N	I295	I95	FL
I295		JACKSONVILLE	W	I10	I295	FL
I10		WINFIELO	W	I10	I75	FL
I75		MACON	S	I475	I75	GA
I475		SMARR	E	I475	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

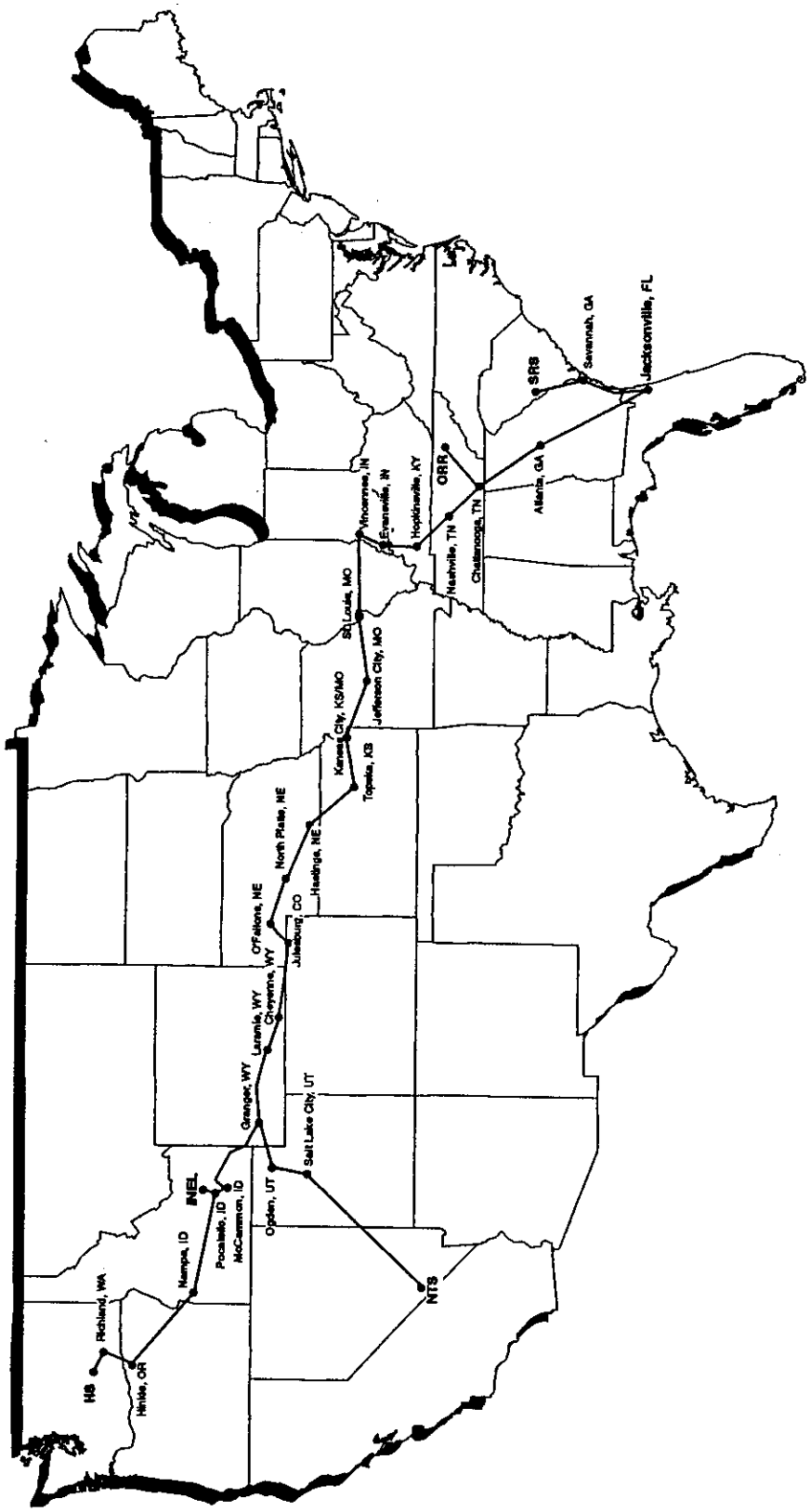

Figure E1-8 Representative Rail Routes from Jacksonville, FL to Department of Energy Management Sites

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8080-NAHUNTA	GA	64.
CSXT	8007-SAVANNAH	GA	145.
CSXT	7739-FAIRFAX	SC	213.
CSXT	7732-ROBBINS	SC	242.
CSXT	7717-DUNBARTON / WELL	SC	251.
USG	7717-DUNBARTON / WELL	SC	251.
USG	15359-SRP	SC	259.

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8079-WAYCROSS	GA	71.
CSXT	8069-CORDELE	GA	179.
CSXT	8144-MANCHESTER	GA	260.
CSXT	7914-ATLANTA	GA	337.
CSXT	7907-MARIETTA	GA	347.
CSXT	7889-CARTERSVILLE	GA	379.
CSXT	7888-DALTON	GA	430.
CSXT	7235-CHATTANOOGA	TN	468.
CSXT	7187-TULLAHOMA	TN	549.
CSXT	7202-NASHVILLE	TN	628.
CSXT	7201-MADISON	TN	638.
CSXT	7061-HOPKINSVILLE	KY	698.
CSXT	3839-HENDERSON	KY	785.
CSXT	3838-EVANSVILLE	IN	798.
CSXT	3812-VINCENNES	IN	848.
CSXT	4952-SALEM	IL	927.
CSXT	10859-EAST ST LOUIS	IL	992.
<TR>	10859-EAST ST LOUIS	IL	992.
<TR>	10858-ST LOUIS	MO	998.
UP	10858-ST LOUIS	MO	998.
UP	10656-JEFFERSON CITY	MO	1120.
UP	10616-KANSAS CITY	MO	1296.
UP	10617-KANSAS CITY	KS	1299.
UP	11823-LAWRENCE	KS	1338.
UP	11697-TOPEKA	KS	1368.
UP	11696-MENOKEN	KS	1373.
UP	11681-MARYSVILLE	KS	1448.
UP	11405-HASTINGS	NE	1558.
UP	11410-GIBBON	NE	1584.
UP	11352-NORTH PLATTE	NE	1662.
UP	11358-O FALLONS	NE	1711.
UP	13703-JULESBURG	CO	1779.
UP	13465-CHEYENNE	WY	1925.
UP	13462-LARAMIE	WY	1977.
UP	13494-GRANGER	WY	2253.
UP	13369-MC CAMMON	ID	2445.
UP	13370-POCATELLO	ID	2468.
UP	13336-SCOVILLE	ID	2524.

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8079-WAYCROSS	GA	71.
CSXT	8069-CORDELE	GA	179.
CSXT	8144-MANCHESTER	GA	260.
CSXT	7914-ATLANTA	GA	337.
CSXT	7907-MARIETTA	GA	347.
CSXT	7889-CARTERSVILLE	GA	379.
CSXT	7888-DALTON	GA	430.
CSXT	7235-CHATTANOOGA	TN	468.
CSXT	7187-TULLAHOMA	TN	549.
CSXT	7202-NASHVILLE	TN	628.
CSXT	7201-MADISON	TN	638.
CSXT	7061-HOPKINSVILLE	KY	698.
CSXT	3839-HENDERSON	KY	785.
CSXT	3838-EVANSVILLE	IN	798.
CSXT	3812-VINCENNES	IN	848.
CSXT	4952-SALEM	IL	927.
CSXT	10859-EAST ST LOUIS	IL	992.
<TR>	10859-EAST ST LOUIS	IL	992.
<TR>	10858-ST LOUIS	MO	998.
UP	10858-ST LOUIS	MO	998.
UP	10656-JEFFERSON CITY	MO	1120.
UP	10616-KANSAS CITY	MO	1296.
UP	10617-KANSAS CITY	KS	1299.
UP	11823-LAWRENCE	KS	1338.
UP	11697-TOPEKA	KS	1368.
UP	11696-MENOKEN	KS	1373.
UP	11681-MARYSVILLE	KS	1448.
UP	11405-HASTINGS	NE	1558.
UP	11410-GIBBON	NE	1584.
UP	11352-NORTH PLATTE	NE	1662.
UP	11358-O FALLONS	NE	1711.
UP	13703-JULESBURG	CO	1779.
UP	13465-CHEYENNE	WY	1925.
UP	13462-LARAMIE	WY	1977.
UP	13494-GRANGER	WY	2253.
UP	13369-MC CAMMON	ID	2445.
UP	13370-POCATELLO	ID	2468.
UP	13412-NAMPA	ID	2710.
UP	14220-PENDLETON	OR	2979.
UP	14223-HINKLE	OR	3010.
UP	13894-WALLULA	WA	3039.
UP	13964-KENNEWICK	WA	3054.
UP	13941-RICHLAND	WA	3062.
USG	13941-RICHLAND	WA	3062.
USG	16212-HANFORD S 300	WA	3070.

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8079-WAYCROSS	GA	71.
CSXT	8069-CORDELE	GA	179.
CSXT	8144-MANCHESTER	GA	260.
CSXT	7914-ATLANTA	GA	337.
CSXT	7907-MARIETTA	GA	347.
CSXT	7889-CARTERSVILLE	GA	379.
CSXT	7888-DALTON	GA	430.
CSXT	7235-CHATTANOOGA	TN	468.
NS	7235-CHATTANOOGA	TN	468.
NS	7260-HARRIMAN	TN	550.
NS	15316-K-25	TN	565.

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8079-WAYCROSS	GA	71.
CSXT	8069-CORDELE	GA	179.
CSXT	8144-MANCHESTER	GA	260.
CSXT	7914-ATLANTA	GA	337.
CSXT	7907-MARIETTA	GA	347.
CSXT	7889-CARTERSVILLE	GA	379.
CSXT	7888-DALTON	GA	430.
CSXT	7235-CHATTANOOGA	TN	468.
CSXT	7187-TULLAHOMA	TN	549.
CSXT	7202-NASHVILLE	TN	628.
CSXT	7201-MADISON	TN	638.
CSXT	7061-HOPKINSVILLE	KY	698.
CSXT	3839-HENDERSON	KY	785.
CSXT	3838-EVANSVILLE	IN	798.
CSXT	3812-VINCENNES	IN	848.
CSXT	4952-SALEM	IL	927.
CSXT	10859-EAST ST LOUIS	IL	992.
<TR>	10859-EAST ST LOUIS	IL	992.
<TR>	10858-ST LOUIS	MO	998.
UP	10858-ST LOUIS	MO	998.
UP	10656-JEFFERSON CITY	MO	1120.
UP	10616-KANSAS CITY	MO	1296.
UP	10617-KANSAS CITY	KS	1299.
UP	11823-LAWRENCE	KS	1338.
UP	11697-TOPEKA	KS	1368.
UP	11696-MENOKEN	KS	1373.
UP	11681-MARYSVILLE	KS	1448.
UP	11405-HASTINGS	NE	1558.
UP	11410-GIBBON	NE	1584.
UP	11352-NORTH PLATTE	NE	1662.
UP	11358-O FALLONS	NE	1711.
UP	13703-JULESBURG	CO	1779.
UP	13465-CHEYENNE	WY	1925.
UP	13462-LARAMIE	WY	1977.
UP	13494-GRANGER	WY	2253.
UP	13568-OGDEN	UT	2392.
UP	13595-SALT LAKE CITY	UT	2427.
UP	13630-LYNNDYL	UT	2540.
UP	14766-VALLEY	NV	2857.
USG	14766-VALLEY	NV	2857.
USG	16333-YUCCA MOUNTAIN	NV	2956.

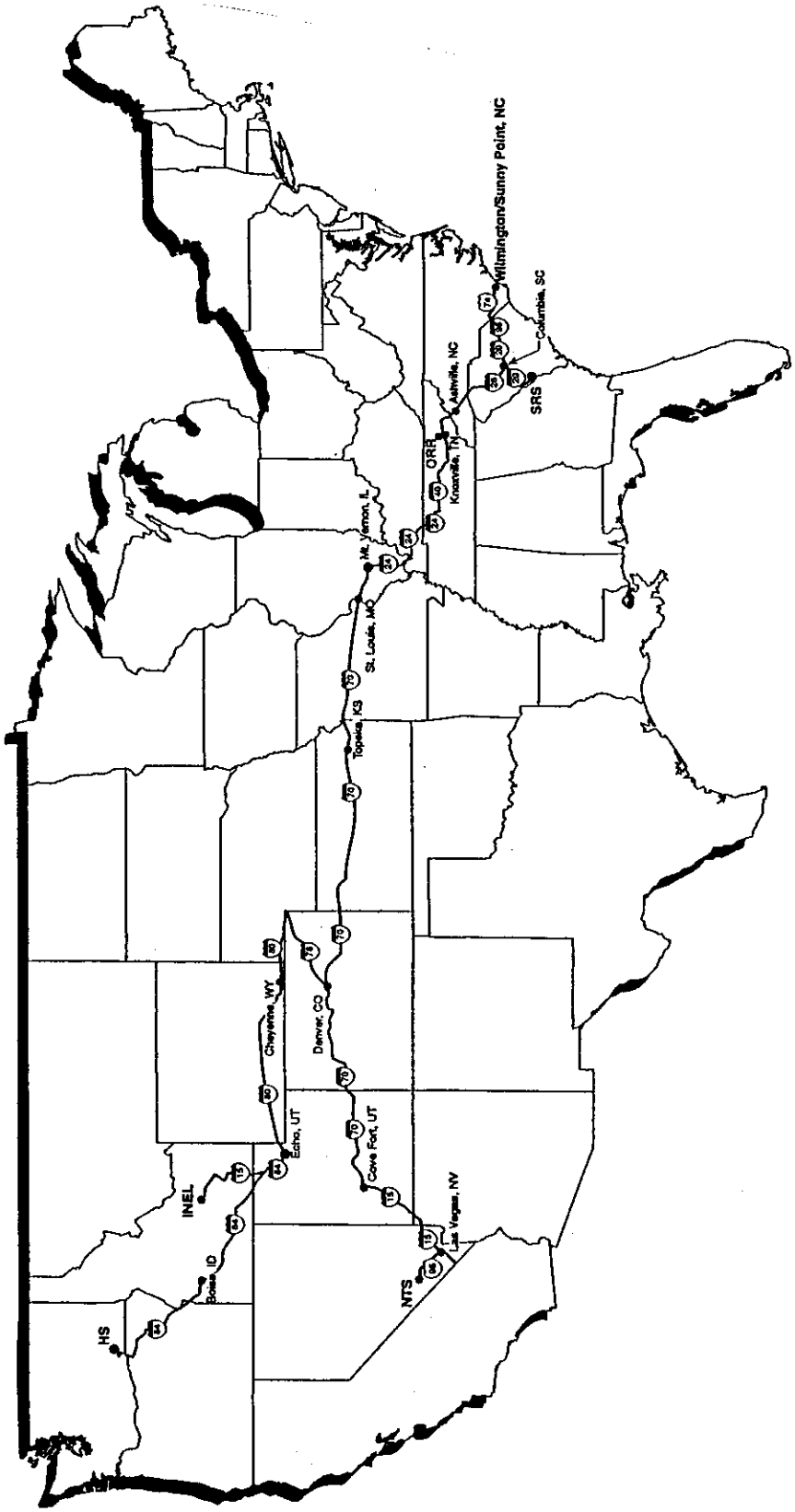

Figure E1-9 Representative Truck Routes from MOTSU, NC to Department of Energy Management Sites

From: SUNNY POINT TM, NC
To : SRP, SC

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		BENSON	NE	I40	I95	NC
I95		FLORENCE	W	I20	I95	SC
I20		MARTINEZ	E	I20	X65	GA
S28		BEECH ISLAND		U278	S125	SC
S125		SRP BARCD 1		S125	LC	SC

From: SUNNY POINT TM, NC
To : ID NATL ENG LAB, ID

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		RALEIGH	SE	I40	I440	NC
I40		HILLSBOROUGH	S	I40	I85	NC
I40	I85	GREENSBORO	S	I40	I85	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB		U20	LOCL	ID

From: SUNNY POINT TM, NC
To : HANFORD, WA

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		RALEIGH	SE	I40	I440	NC
I40	I440	RALEIGH	SW	I40	I440	NC
I40		HILLSBOROUGH	S	I40	I85	NC
I40	I85	GREENSBORO	S	I40	I85	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNDN	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	X5	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: SUNNY POINT TM, NC
To : K-25, TN

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		RALEIGH	SE	I40	I440	NC
I40	I440	RALEIGH	SW	I40	I440	NC
I40		HILLSBOROUGH	S	I40	I85	NC
I40	I85	GREENSBORO	S	I40	I85	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		KINGSTON	E	I40	X356	TN
S58		K-25				TN

From: SUNNY POINT TM, NC
To : MERCURY, NV

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELANO	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		RALEIGH	SE	I40	I440	NC
I40	I440	RALEIGH	SW	I40	I440	NC
I40		HILLSBOROUGH	S	I40	I85	NC
I40	I85	GREENSBORO	S	I40	I85	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		LAS VEGAS	W	U95	U95B	NV
U95BU		LAS VEGAS	NW	U95	U95B	NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

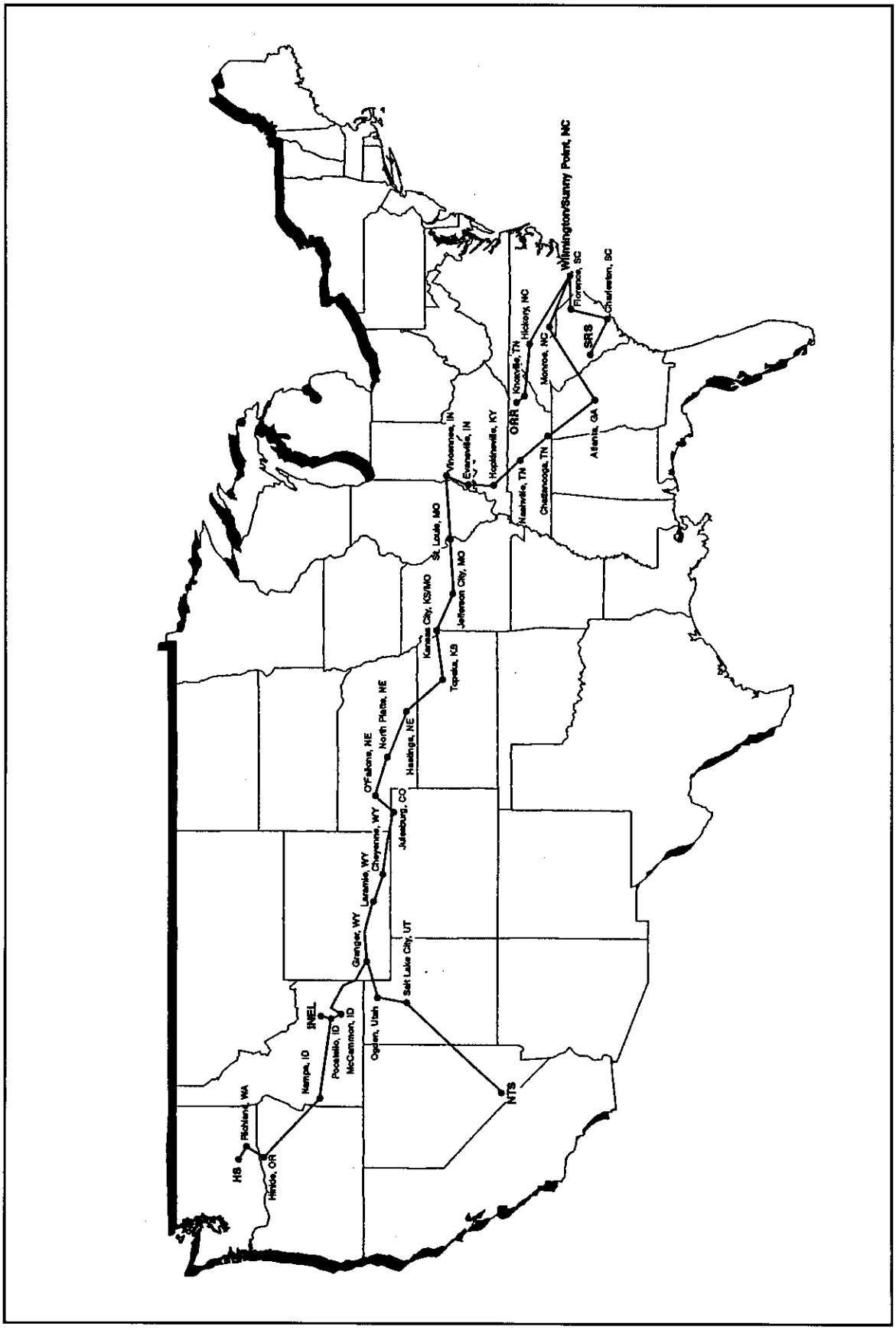

Figure E1-10 Representative Rail Routes from MOTSU, NC to Department of Energy Management Sites

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7671-DILLON	SC	115.
CSXT	7675-FLORENCE	SC	144.
CSXT	7690-CHARLESTON	SC	242.
CSXT	7739-FAIRFAX	SC	336.
CSXT	7732-ROBBINS	SC	365.
CSXT	7717-OUNBARTON / WELL	SC	374.
USG	7717-OUNBARTON / WELL	SC	374.
USG	15359-SRP	SC	382.

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7470-HAMLET	NC	126.
CSXT	7407-MONROE	NC	179.
CSXT	7834-CLINTON	SC	271.
CSXT	7838-GREENWOOD	SC	299.
CSXT	7956-ATHENS	GA	380.
CSXT	17420-ATLANTA BELT JCT	GA	442.
CSXT	17424-TILFORD YARD (S.	GA	450.
CSXT	7907-MARIETTA	GA	465.
CSXT	7889-CARTERSVILLE	GA	493.
CSXT	7888-DALTON	GA	544.
CSXT	7235-CHATTANOOGA	TN	581.
CSXT	7224-WAUHATCHIE	TN	587.
CSXT	7187-TULLAHOMA	TN	663.
CSXT	7201-MADISON	TN	751.
CSXT	7061-HOPKINSVILLE	KY	811.
CSXT	3839-HENDERSON	KY	876.
CSXT	3838-EVANSVILLE	IN	889.
CSXT	3812-VINCENNES	IN	939.
CSXT	4952-SALEM	IL	1017.
CSXT	10859-EAST ST LOUIS	IL	1084.
TRRA	10859-EAST ST LOUIS	IL	1084.
UP	10859-EAST ST LOUIS	IL	1084.
UP	10858-ST LOUIS	MO	1085.
UP	10860-PACIFIC	MO	1112.
UP	10656-JEFFERSON CITY	MO	1210.
UP	10616-KANSAS CITY	MO	1387.
UP	10617-KANSAS CITY	KS	1389.
UP	11823-LAWRENCE	KS	1428.
UP	11697-TOPEKA	KS	1458.
UP	11696-MENOKEN	KS	1463.
UP	11681-MARYSVILLE	KS	1538.
UP	11405-HASTINGS	NE	1646.
UP	11410-GIBBON	NE	1672.
UP	11352-NORTH PLATTE	NE	1791.
UP	11358-O FALLONS	NE	1802.
UP	13703-JULESBURG	CO	1870.
UP	13465-CHEYENNE	WY	2016.
UP	13462-LARAMIE	WY	2068.
UP	13494-GRANGER	WY	2344.
UP	13369-MC CAMMON	ID	2536.
UP	13370-POCATELLO	ID	2559.
UP	13336-SCOVILLE	ID	2615.

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7470-HAMLET	NC	126.
CSXT	7407-MONROE	NC	179.
CSXT	7406-CHARLOTTE	NC	202.
CSXT	7395-MOUNT HOLLY	NC	213.
CSXT	7388-BOSTIC	NC	276.
CSXT	7387-MARION	NC	303.
CSXT	7311-JOHNSON CITY	TN	407.
CSXT	7312-KINGSPORT	TN	438.
CSXT	7313-FRISCO	TN	444.
CSXT	6946-BEAVER JCT	KY	570.
CSXT	6807-CATLETTSBURG	KY	654.
CSXT	6809-ASHLAND	KY	660.
CSXT	6846-SILOAM	KY	684.
CSXT	3162-CHILLICOTHE	OH	746.
CSXT	3095-COLUMBUS (BROAD	OH	788.
CSXT	3402-MARION	OH	833.
CSXT	3002-FOSTORIA	OH	876.
CSXT	3484-OESHLER	OH	904.
CSXT	3993-WELLSBORO	IN	1053.
CSXT	4070-GARY	IN	1087.
CSXT	4073-CLARKE	IN	1091.
CSXT	4076-HAMMONO	IN	1097.
CSXT	4228-BURNHAM / CALUME	IL	1099.
CSXT	4163-BLUE ISLAND	IL	1107.
IHB	4163-BLUE ISLAND	IL	1107.
IHB	4172-ARGO	IL	1119.
IHB	4170-LA GRANGE	IL	1123.
BN	4170-LA GRANGE	IL	1123.
BN	4190-AURORA	IL	1148.
BN	4317-SAVANNA	IL	1239.
BN	4327-EAST DUBUQUE	IL	1279.
BN	5736-LA CROSSE	WI	1390.
BN	9830-ST PAUL	MN	1511.
BN	9793-SOO LINE JCT	MN	1514.
BN	15603-EAST MINNEAPOLIS	MN	1521.
BN	9798-NORTHTOWN	MN	1527.
BN	9826-COON CREEK	MN	1532.
BN	9671-SAUK RAPIDS	MN	1582.
BN	9663-STAPLES	MN	1647.
BN	11131-MOORHEAD	MN	1761.
BN	11132-FARGO	NO	1764.
BN	11134-CASSELTON	NO	1784.
BN	10935-SURREY	NO	1999.
BN	10936-MINOT	NO	2005.
BN	15740-WILLISTON	NO	2117.
BN	13168-HAVRE	MT	2436.
BN	13089-SHELBY	MT	2537.
BN	13300-SANDPOINT	ID	2874.
BN	13828-SPOKANE	WA	2937.
BN	13890-PASCO	WA	3088.
BN	13964-KENNEWICK	WA	3090.
WCRC	13964-KENNEWICK	WA	3090.
UP	13964-KENNEWICK	WA	3090.
UP	13941-RICHLAND	WA	3098.
USG	13941-RICHLAND	WA	3098.
USG	16212-HANFORD S 300	WA	3106.

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7470-HAMLET	NC	126.
CSXT	7407-MONROE	NC	179.
CSXT	7406-CHARLOTTE	NC	202.
CSXT	7395-MOUNT HOLLY	NC	213.
CSXT	7388-BOSTIC	NC	276.
CSXT	7387-MARION	NC	303.
NS	7387-MARION	NC	303.
NS	7343-ASHEVILLE	NC	343.
NS	7318-MORRISTOWN	TN	423.
NS	16403-SEVIER YARD W ENT	TN	456.
NS	16401-COSTER	TN	464.
NS	7288-OOSSETT	TN	4BB.
NS	15316-K-25	TN	509.

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NOOE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7470-HAMLET	NC	126.
CSXT	7407-MONROE	NC	179.
CSXT	7834-CLINTON	SC	271.
CSXT	7838-GREENWOOD	SC	299.
CSXT	7956-ATHENS	GA	380.
CSXT	17420-ATLANTA BELT JCT	GA	442.
CSXT	17424-TILFORD YARD (S.	GA	450.
CSXT	7907-MARIETTA	GA	465.
CSXT	7889-CARTERSVILLE	GA	493.
CSXT	7888-DALTON	GA	544.
CSXT	7235-CHATTANOOGA	TN	581.
CSXT	7224-WAUHATCHIE	TN	587.
CSXT	7187-TULLAHOMA	TN	663.
CSXT	7201-MADISON	TN	751.
CSXT	7061-HOPKINSVILLE	KY	811.
CSXT	3839-HENDERSON	KY	876.
CSXT	3838-EVANSVILLE	IN	889.
CSXT	3812-VINCENNES	IN	939.
CSXT	4952-SALEM	IL	1017.
CSXT	1D859-EAST ST LOUIS	IL	1084.
TRRA	10859-EAST ST LOUIS	IL	1084.
UP	10859-EAST ST LOUIS	IL	1084.
UP	10858-ST LOUIS	MO	1085.
UP	10860-PACIFIC	MO	1112.
UP	10656-JEFFERSON CITY	MO	1210.
UP	1D616-KANSAS CITY	MO	1387.
UP	10617-KANSAS CITY	KS	1389.
UP	11823-LAWRENCE	KS	1428.
UP	11697-TOPEKA	KS	1458.
UP	11696-MENOKEN	KS	1463.
UP	11681-MARYSVILLE	KS	1538.
UP	11405-HASTINGS	NE	1646.
UP	1141D-GIBBON	NE	1672.
UP	11352-NORTH PLATTE	NE	1791.
UP	11358-O FALLONS	NE	1802.
UP	13703-JULESBURG	CO	1870.
UP	13465-CHEYENNE	WY	2016.
UP	13462-LARAMIE	WY	2068.
UP	13494-GRANGER	WY	2344.
UP	13568-OGDEN	UT	2487.
UP	13595-SALT LAKE CITY	UT	2523.
UP	13630-LYNNDYL	UT	2635.
UP	14766-VALLEY	NV	2952.
USG	14766-VALLEY	NV	2952.
USG	16333-YUCCA MOUNTAIN	NV	3051.

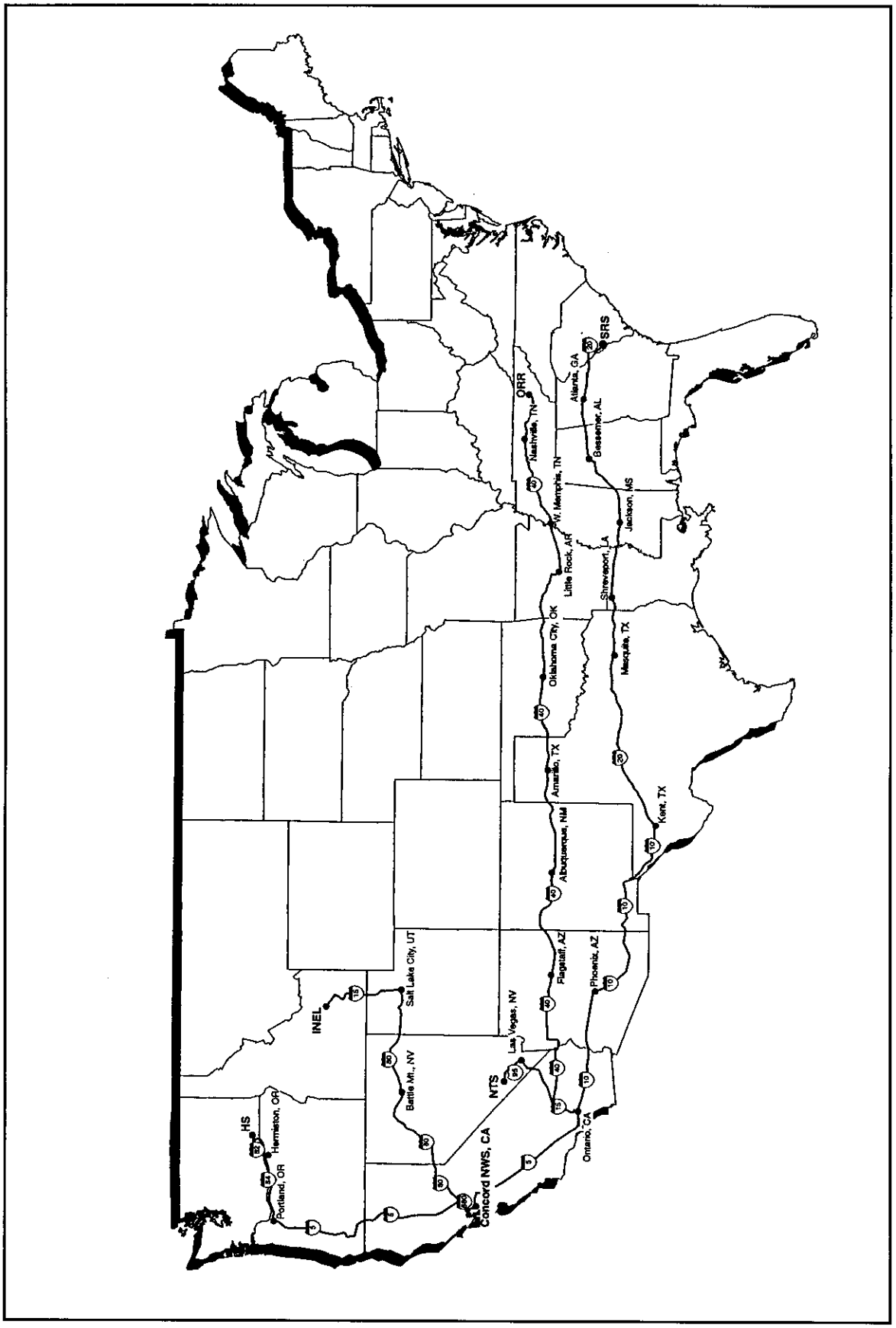

Figure E1-11 Representative Truck Routes from Naval Weapons Station, Concord, CA to Department of Energy Management Sites

From: CONCORD NW I680 S4 CA
To : SRL, SC

Routing through:

		CONCORD	NW	I680	S4	CA
I680		DUBLIN	SE	I580	I680	CA
I580		VERNALIS	W	I5	I580	CA
I5		SAN FERNANDO	NW	I210	I5	CA
I210		POMONA	NW	I10	I210	CA
I10		PHOENIX	W	I10	I17	AZ
I17		PHOENIX	SE	I10	I17	AZ
I10		KENT	E	I10	I20	TX
I20		DUNCANVILLE	NE	I20	U67	TX
I20	I635	MESQUITE	NW	I20	I635	TX
I20		SHREVEPORT	W	I20	I220	LA
I220		BOSSIER CITY	E	I20	I220	LA
I20		JACKSON	SW	I20	I55	MS
I20	I55	JACKSON	S	I20	I55	MS
I20		MERIDIAN	W	I20	I59	MS
I20	I59	BESSEMER	SW	I20	I459	AL
I459		IRONDALE	E	I20	I459	AL
I20		ATLANTA	W	I20	I285	GA
I285		RED OAK	E	I285	I85	GA
I285	I85	COLLEGE PARK	S	I285	I85	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: CONCORD NW I680 S4 CA
To : ID NATL ENG LAB, ID

Routing through:

		CONCORD	NW	I680	S4	CA
I680		MARTINEZ	E	I680	LOCL	CA
I680#		BENICIA	E	I680	I780	CA
I680		ROCKVILLE	SW	I680	I80	CA
I80		SALT LAKE CITY	W	I215	I80	UT
I215		N SALT LAKE		I15	I215	UT
I15		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: CONCORD NW I680 S4 CA
To : HANFORD, WA

Routing through:

		CONCORD	NW	I680	S4	CA
I680		MARTINEZ	E	I680	LOCL	CA
I680#		BENICIA	E	I680	I780	CA
I680		ROCKVILLE	SW	I680	I80	CA
I80		VACAVILLE	E	I505	I80	CA
I505		DUNNIGAN	S	I5	I505	CA
I5		TUALATIN	S	I205	I5	OR
I205		PORTLAND	E	I205	I84	OR
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORO				WA

From: CONCORD NW I680 S4 CA
To : K-25, TN

Routing through:

		CONCORD	NW	I680	S4	CA
I680		DUBLIN	SE	I580	I680	CA
I580		VERNALIS	W	I5	I580	CA
I5		SAN FERNANDO	NW	I210	I5	CA
I210		POMONA	NW	I10	I210	CA
I10		ONTARIO	E	I10	I15	CA
I15		BARSTOW		I15	I40	CA
I40		OKLAHOMA CITY	W	I40	I44	OK
I44		OKLAHOMA CITY	SW	I240	I44	OK
I240		OKLAHOMA CITY	SE	I240	I40	OK
I40		WEST MEMPHIS	N	I40	I55	AR
I40	I55	WEST MEMPHIS	E	I40	I55	AR
I40		NASHVILLE	W	I40	I440	TN
I440		NASHVILLE	SE	I24	I440	TN
I24		NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: CONCORD NW I680 S4 CA
To : MERCURY, NV

Routing through:

		CONCORD	NW	I680	S4	CA
I680		DUBLIN	SE	I580	I680	CA
I580		VERNALIS	W	I5	I580	CA
I5		SAN FERNANDO	NW	I210	I5	CA
I210		POMONA	NW	I10	I210	CA
I10		ONTARIO	E	I10	I15	CA
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

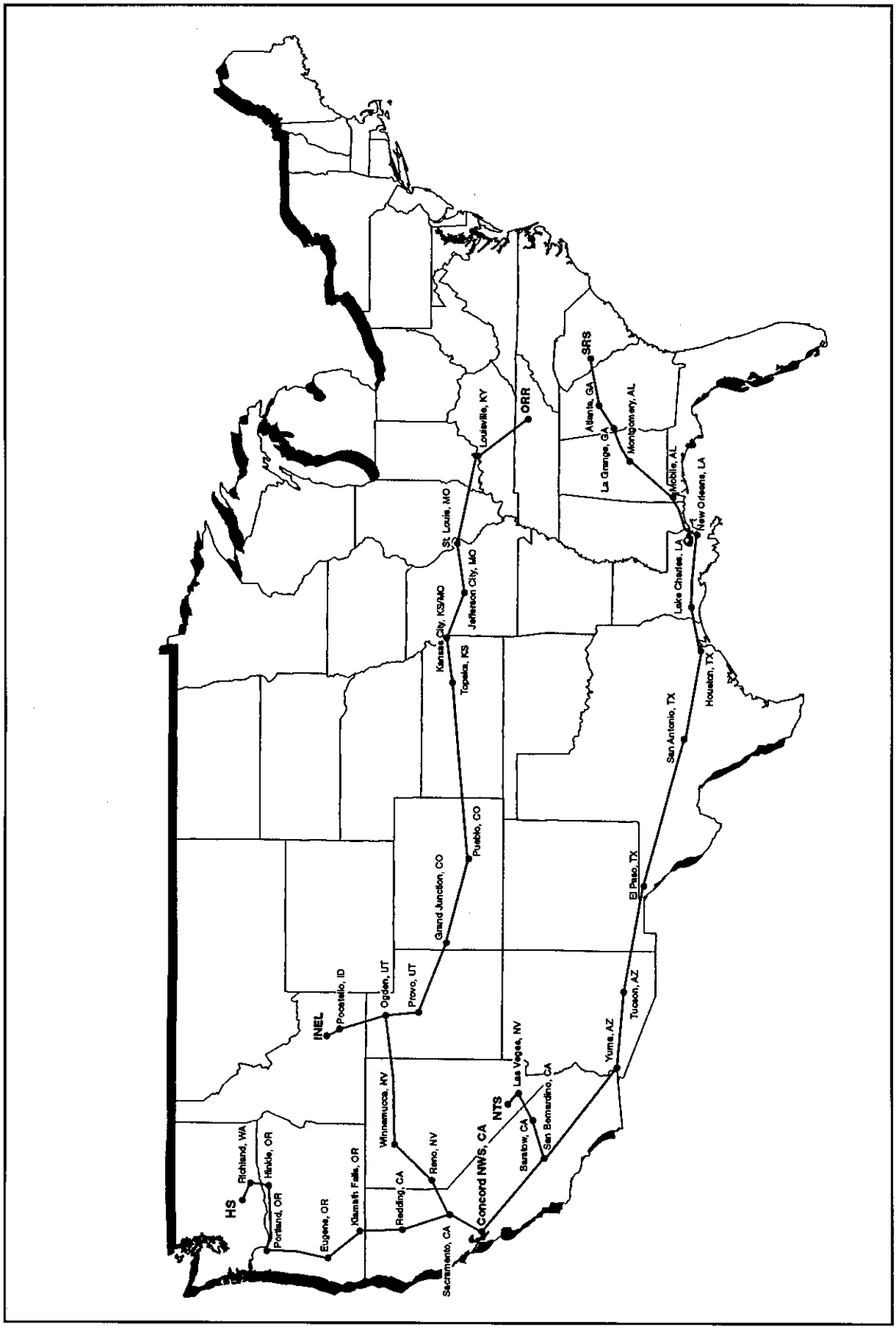

Figure E1-12 Representative Rail Routes from Naval Weapons Stations, Concord, CA to Department of Energy Management Sites

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14470-PITTSBURG	CA	14.
SP	14498-LATHRUP	CA	79.
SP	14529-MODESTO	CA	106.
SP	14570-FRESNO	CA	204.
SP	14622-BAKERSFIELD	CA	309.
SP	14621-MOJAVE	CA	374.
SP	14692-PALMDALE	CA	408.
SP	14666-SAN BERNARDINO	CA	485.
SP	12971-YUMA	AZ	679.
SP	12972-WELLTON	AZ	720.
SP	12938-PICACHO	AZ	889.
SP	12942-TUCSON	AZ	938.
SP	12877-EL PASO	TX	1248.
SP	12611-SAN ANTONIO	TX	1866.
SP	12538-FLATONIA	TX	1958.
SP	12536-EAGLE LAKE	TX	2009.
SP	12399-HOUSTON	TX	2078.
SP	12341-BEAUMONT	TX	2165.
SP	12348-ORANGE	TX	2188.
SP	9124-LAKE CHARLES	LA	2231.
SP	9113-LAFAYETTE	LA	2297.
SP	9038-THIBODAUX JCT	LA	2394.
SP	8985-NEW ORLEANS	LA	2454.
CSXT	8985-NEW ORLEANS	LA	2454.
CSXT	8966-GULFPORT	MS	2528.
CSXT	8926-BILOXI	MS	2545.
CSXT	8967-PASCAGOULA	MS	2561.
CSXT	8597-MOBILE	AL	2600.
CSXT	8566-FLOMATON	AL	2653.
CSXT	8657-MONTGOMERY	AL	2777.
CSXT	8683-OPELIKA	AL	2843.
CSXT	8142-LA GRANGE	GA	2886.
CSXT	7914-ATLANTA	GA	2962.
CSXT	7961-AUGUSTA	GA	3137.
CSXT	7732-ROBBINS	SC	3166.
CSXT	7717-DUNBARTON / WELL	SC	3175.
USG	7717-DUNBARTON / WELL	SC	3175.
USG	15359-SRP	SC	3183.

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14431-SUISUN-FAIRFIELD	CA	21.
SP	14409-DAVIS	CA	48.
SP	14411-SACRAMENTO	CA	62.
SP	14415-ROSEVILLE	CA	77.
SP	14821-RENO	NV	194.
SP	14816-SPARKS	NV	209.
SP	14812-HAZEN	NV	242.
SP	14813-WINNEMUCCA	NV	376.
SP	14793-ELKO	NV	500.
SP	13568-OGDEN	UT	723.
UP	13568-OGDEN	UT	723.
UP	13369-MC CAMMON	ID	836.
UP	13370-POCATELLO	ID	859.
UP	13336-SCOVILLE	ID	915.

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14431-SUISUN-FAIRFIELD	CA	21.
SP	14409-DAVIS	CA	48.
SP	14411-SACRAMENTO	CA	62.
SP	14415-ROSEVILLE	CA	77.
SP	14383-MARYSVILLE	CA	108.
SP	14385-CHICO	CA	145.
SP	14387-TEHAMA	CA	177.
SP	14313-REDDING	CA	224.
SP	14250-KLAMATH FALLS	OR	386.
SP	14271-EUGENE	OR	579.
SP	14287-ALBANY	OR	623.
SP	14141-SALEM	OR	651.
SP	14179-PORTLAND	OR	704.
UP	14179-PORTLAND	OR	704.
UP	14197-OREGON TRUNK JCT	OR	799.
UP	14223-HINKLE	OR	891.
UP	13894-WALLULA	WA	920.
UP	13964-KENNEWICK	WA	935.
UP	13941-RICHLAND	WA	944.
USG	13941-RICHLAND	WA	944.
USG	16212-HANFORD S 300	WA	951.

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14431-SUISUN-FAIRFIELD	CA	21.
SP	14409-DAVIS	CA	48.
SP	14411-SACRAMENTO	CA	62.
SP	14415-ROSEVILLE	CA	77.
SP	14821-RENO	NV	194.
SP	14816-SPARKS	NV	209.
SP	14812-HAZEN	NV	242.
SP	14813-WINNEMUCCA	NV	376.
SP	14793-ELKO	NV	500.
SP	13568-OGDEN	UT	723.
SP	13595-SALT LAKE CITY	UT	758.
SP	13610-PROVO	UT	803.
SP	13611-SPRINGVILLE	UT	808.
SP	13613-THISTLE	UT	822.
SP	13646-GRAND JCT	CO	1048.
SP	13645-GLENWOOD SPRINGS	CO	1138.
SP	13673-DOTSERO	CO	1153.
SP	13764-PUEBLO	CO	1377.
SP	11673-HERINGTON	KS	1837.
SP	11697-TOPEKA	KS	1914.
SP	11823-LAWRENCE	KS	1944.
SP	10617-KANSAS CITY	KS	1982.
SP	10616-KANSAS CITY	MO	1985.
SP	10627-PLEASANT HILL	MO	2017.
SP	10656-JEFFERSON CITY	MO	2133.
SP	10858-ST LOUIS	MO	2255.
SP	10859-EAST ST LOUIS	IL	2261.
<TR>	10859-EAST ST LOUIS	IL	2261.
NS	10859-EAST ST LOUIS	IL	2261.
NS	4953-CENTRALIA	IL	2319.
NS	4954-MOUNT VERNON	IL	2341.
NS	4797-MOUNT CARMEL	IL	2404.
NS	7009-JEFFERSONVILLE	IN	2529.
NS	7008-LOUISVILLE	KY	2533.
NS	6979-DANVILLE	KY	2633.
NS	7260-HARRIMAN	TN	2795.
NS	15316-K-25	TN	2810.

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14470-PITTSBURG	CA	14.
SP	14498-LATHRUP	CA	79.
SP	14529-MODESTO	CA	106.
SP	14570-FRESNO	CA	204.
SP	14622-BAKERSFIELD	CA	309.
SP	14621-MOJAVE	CA	374.
SP	14692-PALMDALE	CA	408.
SP	14666-SAN BERNARDINO	CA	485.
UP	14666-SAN BERNARDINO	CA	485.
UP	14664-BARSTOW	CA	563.
UP	14762-LAS VEGAS	NV	736.
UP	14766-VALLEY	NV	751.
USG	14766-VALLEY	NV	751.
USG	16333-YUCCA MOUNTAIN	NV	850.

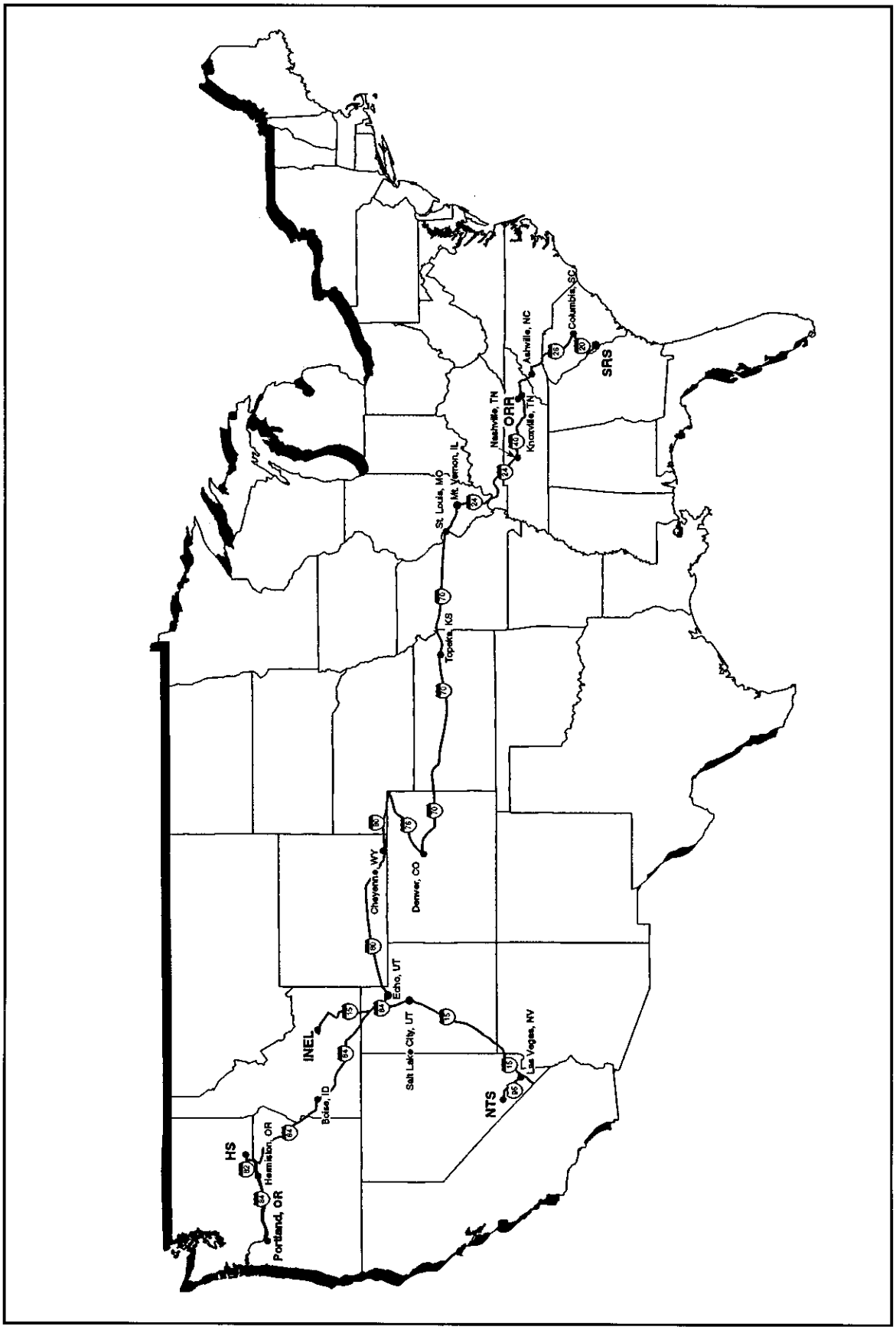

Figure E1-13 Representative Truck Routes from Portland, OR to Department of Energy Management Sites

From: PORTLAND, OR
To : SRL, SC

Routing through:

		PORTLAND				OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWAROSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: PORTLAND, OR
To : ID NATL ENG LAB, ID

Routing through:

		PORTLAND				OR
I84		RAFT RIVER	W	I84	I86	ID
I86		CHUBBUCK	E	I15	I86	ID
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LA8				ID

From: PORTLAND, OR
To : HANFORD, WA

Routing through:

		PORTLAND				OR
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: PORTLAND, OR
To : K-25, TN

Routing through:

		PORTLAND				OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		OENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: PORTLAND, OR
To : MERCURY, NV

Routing through:

		PORTLAND				OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		SALT LAKE CITY	W	I15	I80	UT
I15	I80	SALT LAKE CITY	S	I15	I80	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

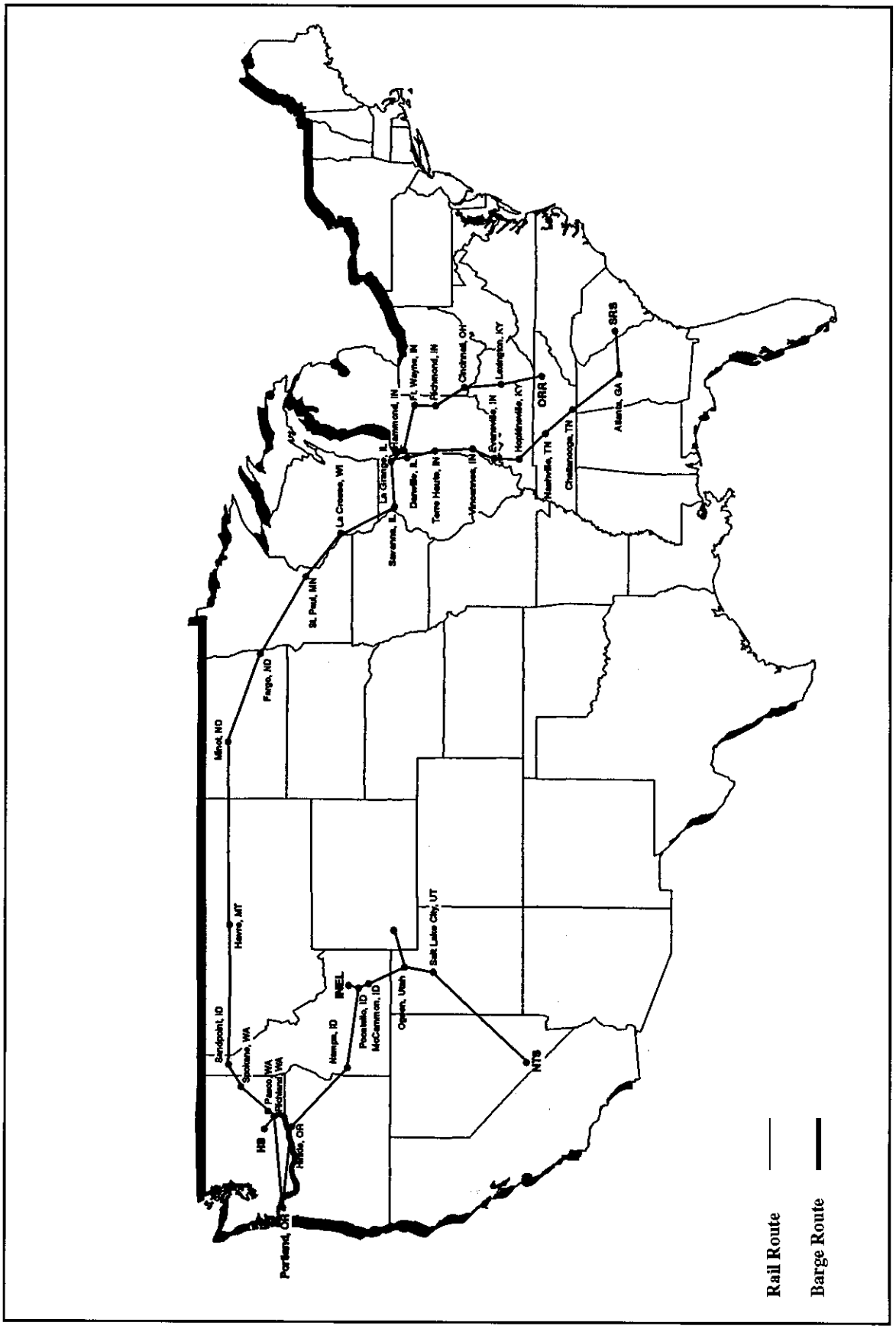

Figure E1-14 Representative Rail and Barge Routes from Portland, OR to Department of Energy Management Sites

ROUTE FROM: BN 14179-PORTLAND, OR
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
BN	14180-VANCOUVER	WA	7.
BN	13964-KENNEWICK	WA	220.
BN	13890-PASCO	WA	222.
BN	13828-SPOKANE	WA	373.
BN	13300-SANDPOINT	ID	436.
BN	13089-SHELBY	MT	773.
BN	13168-HAVRE	MT	874.
BN	15740-WILLISTON	ND	1193.
BN	10936-MINOT	ND	1305.
BN	10935-SURREY	ND	1311.
BN	11134-CASSELTON	ND	1526.
BN	11132-FARGO	ND	1546.
BN	11131-MOORHEAD	MN	1549.
BN	9663-STAPLES	MN	1663.
BN	9671-SAUK RAPIOS	MN	1728.
BN	9826-COON CREEK	MN	1778.
BN	9798-NORTHTOWN	MN	1787.
BN	9830-ST PAUL	MN	1799.
8N	5736-LA CROSSE	WI	1920.
BN	4327-EAST DUBUQUE	IL	2031.
BN	4317-SAVANNA	IL	2071.
BN	4190-AURORA	IL	2162.
BN	4170-LA GRANGE	IL	2187.
IHB	4170-LA GRANGE	IL	2187.
IHB	4172-ARGO	IL	2191.
IHB	4163-BLUE ISLAND	IL	2203.
IHB	4223-DOLTON / RIVERDA	IL	2207.
CSXT	4223-DOLTON / RIVERDA	IL	2207.
CSXT	4206-CHICAGO HEIGHTS	IL	2217.
CSXT	4636-WATSEKA	IL	2268.
CSXT	4642-OANVILLE	IL	2313.
CSXT	3863-TERRE HAUTE	IN	2370.
CSXT	3812-VINCENNES	IN	2423.
CSXT	3838-EVANSVILLE	IN	2473.
CSXT	3839-HENDERSON	KY	2486.
CSXT	7061-HOPKINSVILLE	KY	2573.
CSXT	7201-MADISON	TN	2634.
CSXT	7202-NASHVILLE	TN	2644.
CSXT	7187-TULLAHOMA	TN	2723.
CSXT	7235-CHATTANOOGA	TN	2803.
CSXT	7888-DALTON	GA	2841.
CSXT	7889-CARTERSVILLE	GA	2892.
CSXT	7907-MARIETTA	GA	2924.
CSXT	7914-ATLANTA	GA	2934.
CSXT	7961-AUGUSTA	GA	3109.
CSXT	7732-ROBBINS	SC	3138.
CSXT	7717-DUN8ARTON / WELL	SC	3147.
USG	7717-DUNBARTON / WELL	SC	3147.
USG	15359-SRP	SC	3155.

ROUTE FROM: BN 14179-PORTLAND, OR
TO: UP 13336-SCOVILLE, ID

RR	NOOE	STATE	DIST
BN	14179-PORTLAND	OR	0.
UP	14179-PORTLAND	OR	0.
UP	14197-OREGON TRUNK JCT	OR	95.
UP	14223-HINKLE	OR	187.
UP	14220-PENOLETON	OR	218.
UP	13412-NAMPA	ID	487.
UP	13370-POCATELLO	ID	729.
UP	13336-SCOVILLE	ID	785.

ROUTE FROM: BN 14179-PORTLAND, OR
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
BN	14180-VANCOUVER	WA	7.
BN	13964-KENNEWICK	WA	220.
BN	13890-PASCO	WA	222.
WCRC	13890-PASCO	WA	222.
WCRC	13964-KENNEWICK	WA	223.
WCRC	13941-RICHLAND	WA	231.
USG	13941-RICHLAND	WA	231.
USG	16212-HANFORD S 300	WA	239.

ROUTE FROM: BN 14179-PORTLAND, OR
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
BN	14180-VANCOUVER	WA	7.
BN	13964-KENNEWICK	WA	220.
BN	13890-PASCO	WA	222.
BN	13828-SPOKANE	WA	373.
BN	13300-SANDPOINT	ID	436.
BN	13089-SHELBY	MT	773.
BN	13168-HAVRE	MT	874.
BN	15740-WILLISTON	ND	1193.
BN	10936-MINOT	ND	13D5.
BN	10935-SURREY	ND	1311.
BN	11134-CASSELTON	ND	1526.
BN	11132-FARGO	ND	1546.
BN	11131-MOORHEAD	MN	1549.
BN	9663-STAPLES	MN	1663.
BN	9671-SAUK RAPIDS	MN	1728.
BN	9826-COON CREEK	MN	1778.
BN	9798-NORTHTOWN	MN	1787.
BN	9830-ST PAUL	MN	1799.
BN	5736-LA CROSSE	WI	1920.
BN	4327-EAST DUBUQUE	IL	2031.
BN	4317-SAVANNA	IL	2071.
BN	4190-AURORA	IL	2162.
BN	4170-LA GRANGE	IL	2187.
IHB	4170-LA GRANGE	IL	2187.
IH8	4172-ARGO	IL	2191.
IH8	4163-8LUE ISLANO	IL	2203.
IHB	4228-BURNHAM / CALUME	IL	2211.
NS	4228-BURNHAM / CALUME	IL	2211.
NS	4076-HAMMOND	IN	2213.
NS	4064-HOBART	IN	2229.
NS	4020-ARGOS	IN	2292.
NS	3548-FORT WAYNE	IN	2351.
NS	3650-MUNCIE	IN	2415.
NS	3688-RICHMOND	IN	2460.
NS	3251-HAMILTON	OH	2515.
NS	3234-IVORYDALE	OH	2532.
NS	3228-CINCINNATI	OH	2539.
NS	6850-LEXINGTON	KY	2613.
NS	6979-OANVILLE	KY	2650.
NS	7260-HARRIMAN	TN	2812.
NS	15316-K-25	TN	2827.

ROUTE FROM: BN 14179-PORTLAND, OR
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
UP	14179-PORTLAND	OR	0.
UP	14197-OREGON TRUNK JCT	OR	95.
UP	14223-HINKLE	OR	187.
UP	14220-PENDLETON	OR	218.
UP	13412-NAMPA	ID	487.
UP	13370-POCATELLO	ID	729.
UP	13369-MC CAMMON	ID	752.
UP	13568-OGDEN	UT	865.
UP	13595-SALT LAKE CITY	UT	901.
UP	13630-LYNNDYL	UT	1013.
UP	14766-VALLEY	NV	1330.
USG	14766-VALLEY	NV	1330.
USG	16333-YUCCA MOUNTAIN	NV	1429.

ROUTE FROM: BRG 16859-PORTLAND; PORT OF OR
TO: BRG 16851-RICHLAND; PORT OF WA

RR	NODE	STATE	DIST
BRG	16859-PORTLAND; PORT	OR	0.
BRG	16852-COLUMBIA/WILLAM	WA	10.
BRG	16853-VANCOUVER; PORT	WA	14.
BRG	16848-COLUMBIA/SNAKE	WA	231.
BRG	16851-RICHLAND; PORT	WA	243.

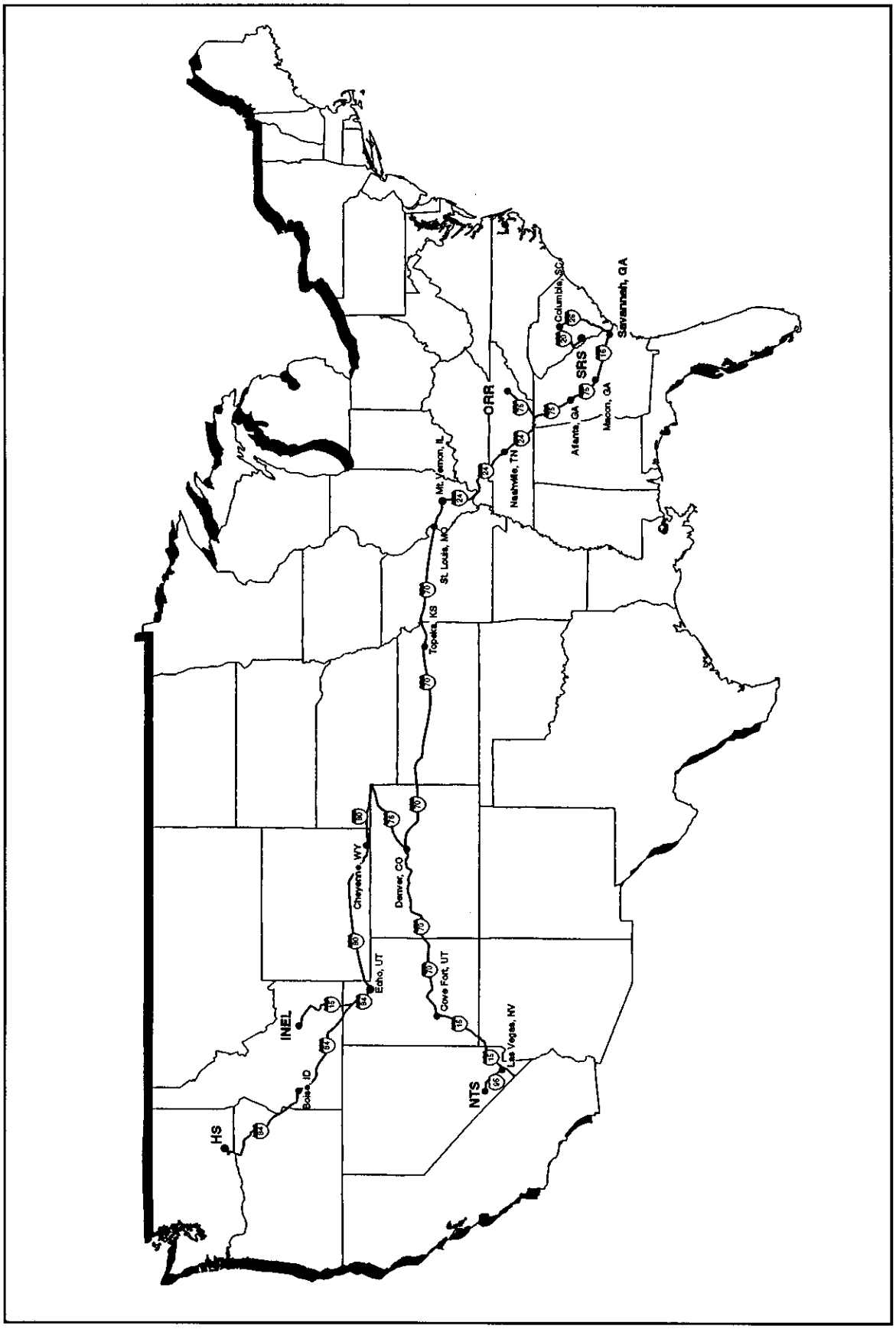

Figure E1-15 Representative Truck Routes from Savannah, GA to Department of Energy Management Sites

From: SAVANNAH, GA
To : SRP, SC

Routing through:

	SAVANNAH				GA
I16	POOLER	S	I16	I95	GA
I95	ROSINVILLE	N	I26	I95	SC
I26	COLUMBIA	NW	I20	I26	SC
I20	NORTH AUGUSTA	NW	I20	S230	SC
S230	NORTH AUGUSTA				SC
S125	CLEARWATER	W	U1	U278	SC
U278	BEECH ISLAND		U278	S125	SC
S125	JACKSON	SE	S125	LSRP	SC
LSRP	SRP				SC

From: SAVANNAH, GA
To : ID NATL ENG LAB, ID

Routing through:

		SAVANNAH				GA
I16		MACON	NW	I16	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		CDMMERCE CITY	NW	I270	I76	CO
I76		CDMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: SAVANNAH, GA
To : HANFORD, WA

Routing through:

		SAVANNAH				GA
I16		MACON	NW	I16	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		CDMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: SAVANNAH, GA
To : K-25, TN

Routing through:

		SAVANNAH				GA
I16		MACON	NW	I16	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: SAVANNAH, GA
To : MERCURY, NV

Routing through:

		SAVANNAH				GA
I16		MACON	NW	I16	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	REO OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

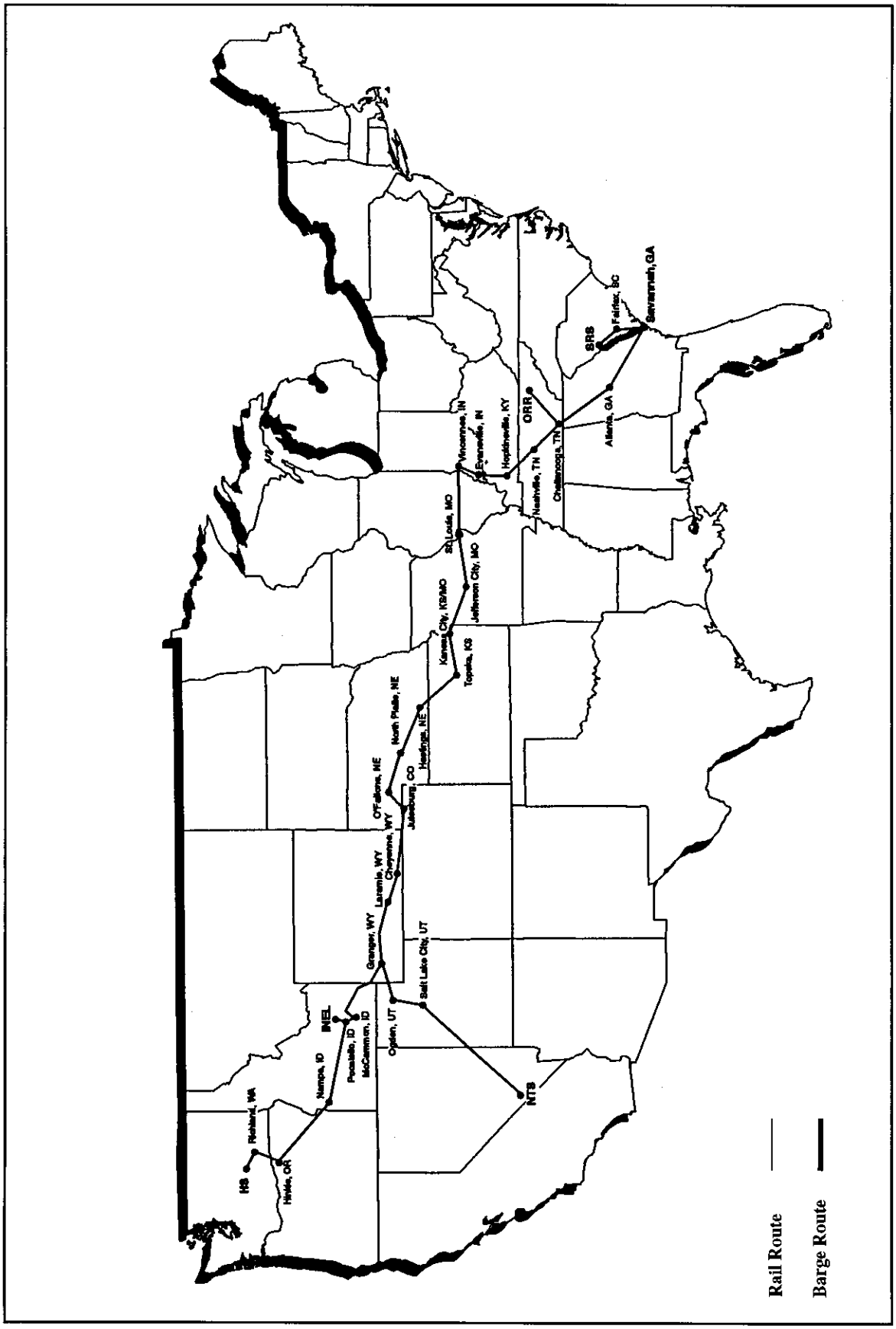

Figure E1-16 Representative Rail and Barge Routes from Savannah, GA to Department of Energy Management Sites

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	7739-FAIRFAX	SC	68.
CSXT	7732-ROBBINS	SC	97.
CSXT	7717-DUNBARTON / WELL	SC	106.
USG	7717-DUN8ARTON / WELL	SC	106.
USG	15359-SRP	SC	114.

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: UP 13336-SCOVILLE, IO

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	8079-WAYCROSS	GA	93.
CSXT	8069-CORDELE	GA	201.
CSXT	8144-MANCHESTER	GA	282.
CSXT	7914-ATLANTA	GA	359.
CSXT	7907-MARIETTA	GA	369.
CSXT	7889-CARTERSVILLE	GA	401.
CSXT	7888-DALTON	GA	452.
CSXT	7235-CHATTANOOGA	TN	490.
CSXT	7187-TULLAHOMA	TN	571.
CSXT	7202-NASHVILLE	TN	650.
CSXT	7201-MADISON	TN	660.
CSXT	7061-HOPKINSVILLE	KY	720.
CSXT	3839-HENDERSON	KY	807.
CSXT	3838-EVANSVILLE	IN	820.
CSXT	3812-VINCENNES	IN	870.
CSXT	4952-SALEM	IL	949.
CSXT	10859-EAST ST LOUIS	IL	1014.
<TR>	10859-EAST ST LOUIS	IL	1014.
<TR>	10858-ST LOUIS	MO	1020.
UP	10858-ST LOUIS	MO	1020.
UP	10656-JEFFERSON CITY	MO	1142.
UP	10616-KANSAS CITY	MO	1318.
UP	10617-KANSAS CITY	KS	1321.
UP	11823-LAWRENCE	KS	1360.
UP	11697-TOPEKA	KS	1390.
UP	11696-MENOKEN	KS	1395.
UP	11681-MARYSVILLE	KS	1470.
UP	11405-HASTINGS	NE	1580.
UP	11410-GIBBON	NE	1606.
UP	11352-NORTH PLATTE	NE	1684.
UP	11358-O FALLONS	NE	1733.
UP	13703-JULESBURG	CO	1801.
UP	13465-CHEYENNE	WY	1947.
UP	13462-LARAMIE	WY	1999.
UP	13494-GRANGER	WY	2275.
UP	13369-MC CAMMON	ID	2467.
UP	13370-POCATELLO	ID	2490.
UP	13336-SCOVILLE	ID	2546.

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	8079-WAYCROSS	GA	93.
CSXT	8069-CORDELE	GA	201.
CSXT	8144-MANCHESTER	GA	282.
CSXT	7914-ATLANTA	GA	359.
CSXT	7907-MARIETTA	GA	369.
CSXT	7889-CARTERSVILLE	GA	401.
CSXT	7888-DALTON	GA	452.
CSXT	7235-CHATTANOOGA	TN	490.
CSXT	7187-TULLAHOMA	TN	571.
CSXT	7202-NASHVILLE	TN	650.
CSXT	7201-MADISON	TN	660.
CSXT	7061-HOPKINSVILLE	KY	720.
CSXT	3839-HENDERSON	KY	807.
CSXT	3838-EVANSVILLE	IN	820.
CSXT	3812-VINCENNES	IN	870.
CSXT	4952-SALEM	IL	949.
CSXT	10859-EAST ST LOUIS	IL	1014.
<TR>	10859-EAST ST LOUIS	IL	1014.
<TR>	10858-ST LOUIS	MO	1020.
UP	10858-ST LOUIS	MO	1020.
UP	10656-JEFFERSON CITY	MO	1142.
UP	10616-KANSAS CITY	MO	1318.
UP	10617-KANSAS CITY	KS	1321.
UP	11823-LAWRENCE	KS	1360.
UP	11697-TOPEKA	KS	1390.
UP	11696-MENOKEN	KS	1395.
UP	11681-MARYSVILLE	KS	1470.
UP	11405-HASTINGS	NE	1580.
UP	11410-GIBBON	NE	1606.
UP	11352-NORTH PLATTE	NE	1684.
UP	11358-O FALLONS	NE	1733.
UP	13703-JULESBURG	CO	1801.
UP	13465-CHEYENNE	WY	1947.
UP	13462-LARAMIE	WY	1999.
UP	13494-GRANGER	WY	2275.
UP	13369-MC CAMMON	ID	2467.
UP	13370-POCATELLO	ID	2490.
UP	13412-NAMPA	ID	2732.
UP	14220-PENDLETON	OR	3000.
UP	14223-HINKLE	OR	3031.
UP	13894-WALLULA	WA	3060.
UP	13964-KENNEWICK	WA	3075.
UP	13941-RICHLAND	WA	3084.
USG	13941-RICHLAND	WA	3084.
USG	16212-HANFORD S 300	WA	3092.

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	8079-WAYCROSS	GA	93.
CSXT	8069-CORDELE	GA	201.
CSXT	8144-MANCHESTER	GA	282.
CSXT	7914-ATLANTA	GA	359.
CSXT	7907-MARIETTA	GA	369.
CSXT	7889-CARTERSVILLE	GA	401.
CSXT	7888-DALTON	GA	452.
CSXT	7235-CHATTANOOGA	TN	490.
NS	7235-CHATTANOOGA	TN	490.
NS	7260-HARRIMAN	TN	572.
NS	15316-K-25	TN	587.

ROUTE FROM: BRG 16923-SAVANNAH RIV/ICW, GA
TO: BRG 16932-SAVANNAH BLF L/D, GA

RR	NODE	STATE	DIST
BRG	16923-SAVANNAH RIV/ICW	GA	0.
BRG	16929-SAVANNAH PORT O	GA	10.
BRG	16932-SAVANNAH BLF L/O	GA	146.

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	8079-WAYCROSS	GA	93.
CSXT	8069-CORDELE	GA	201.
CSXT	8144-MANCHESTER	GA	282.
CSXT	7914-ATLANTA	GA	359.
CSXT	7907-MARIETTA	GA	369.
CSXT	7889-CARTERSVILLE	GA	401.
CSXT	7888-DALTON	GA	452.
CSXT	7235-CHATTANODGA	TN	490.
CSXT	7187-TULLAHOMA	TN	571.
CSXT	7202-NASHVILLE	TN	650.
CSXT	7201-MADISON	TN	660.
CSXT	7061-HOPKINSVILLE	KY	720.
CSXT	3839-HENDERSON	KY	807.
CSXT	3838-EVANSVILLE	IN	820.
CSXT	3812-VINCENNES	IN	870.
CSXT	4952-SALEM	IL	949.
CSXT	10859-EAST ST LOUIS	IL	1014.
<TR>	10859-EAST ST LOUIS	IL	1014.
<TR>	10858-ST LOUIS	MO	1020.
UP	10858-ST LOUIS	MO	1020.
UP	10656-JEFFERSON CITY	MO	1142.
UP	10616-KANSAS CITY	MO	1318.
UP	10617-KANSAS CITY	KS	1321.
UP	11823-LAWRENCE	KS	1360.
UP	11697-TOPEKA	KS	1390.
UP	11696-MENOKEN	KS	1395.
UP	11681-MARYSVILLE	KS	1470.
UP	11405-HASTINGS	NE	1580.
UP	11410-GIBBON	NE	1606.
UP	11352-NORTH PLATTE	NE	1684.
UP	11358-O FALLONS	NE	1733.
UP	13703-JULESBURG	CO	1801.
UP	13465-CHEYENNE	WY	1947.
UP	13462-LARAMIE	WY	1999.
UP	13494-GRANGER	WY	2275.
UP	13568-OGDEN	UT	2414.
UP	13595-SALT LAKE CITY	UT	2449.
UP	13630-LYNNDYL	UT	2561.
UP	14766-VALLEY	NV	2878.
USG	14766-VALLEY	NV	2878.
USG	16333-YUCCA MOUNTAIN	NV	2977.

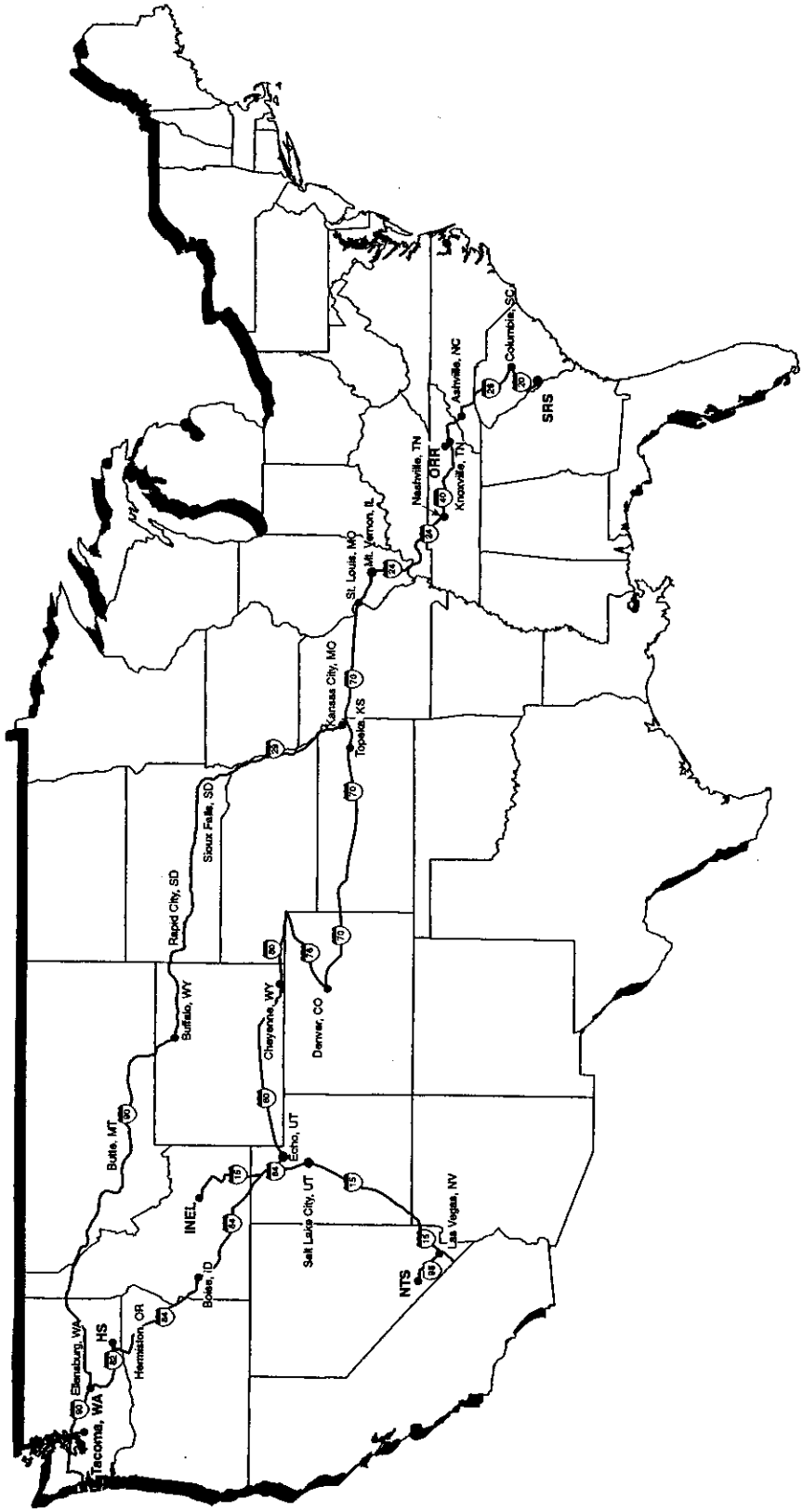

Figure E1-17 Representative Truck Routes from Tacoma, WA to Department of Energy Management Sites

From: TACOMA E I5 X135 WA
To : SRP, SC

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		BUTTE	W	I15	I90	MT
I15	I90	BUTTE	E	I15	I90	MT
I90		SIOUX FALLS	NW	I29	I90	SD
I29		LOVELAND	SW	I29	I680	IA
I680		MINDEN	NW	I680	I80	IA
I80		COUNCIL BLUFFS	SE	I29	I80	IA
I29		KANSAS CITY	NW	I29	I435	MO
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: TACOMA E I5 X135 WA
To : ID NATL ENG LAB, ID

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		ELLENSBURG	SE	I82	I90	WA
I82		HERMISTON	SW	I82	I84	OR
I84		RAFT RIVER	W	I84	I86	ID
I86		CHUBBUCK	E	I15	I86	ID
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: TACOMA E I5 X135 WA
To : HANFORD, WA

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		ELLENSBURG	SE	I82	I90	WA
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: TACOMA E I5 X135 WA
To : K-25, TN

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		BUTTE	W	I15	I90	MT
I15	I90	BUTTE	E	I15	I90	MT
I90		SIOUX FALLS	NW	I29	I90	SD
I29		LOVELAND	SW	I29	I680	IA
I680		MINDEN	NW	I680	I80	IA
I80		COUNCIL BLUFFS	SE	I29	I80	IA
I29		KANSAS CITY	NW	I29	I435	MO
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: TACOMA E I5 X135 WA
To : MERCURY, NV

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		ELLENSBURG	SE	I82	I90	WA
I82		HERMISTON	SW	I82	I84	OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		SALT LAKE CITY	W	I15	I80	UT
I15	I80	SALT LAKE CITY	S	I15	I80	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

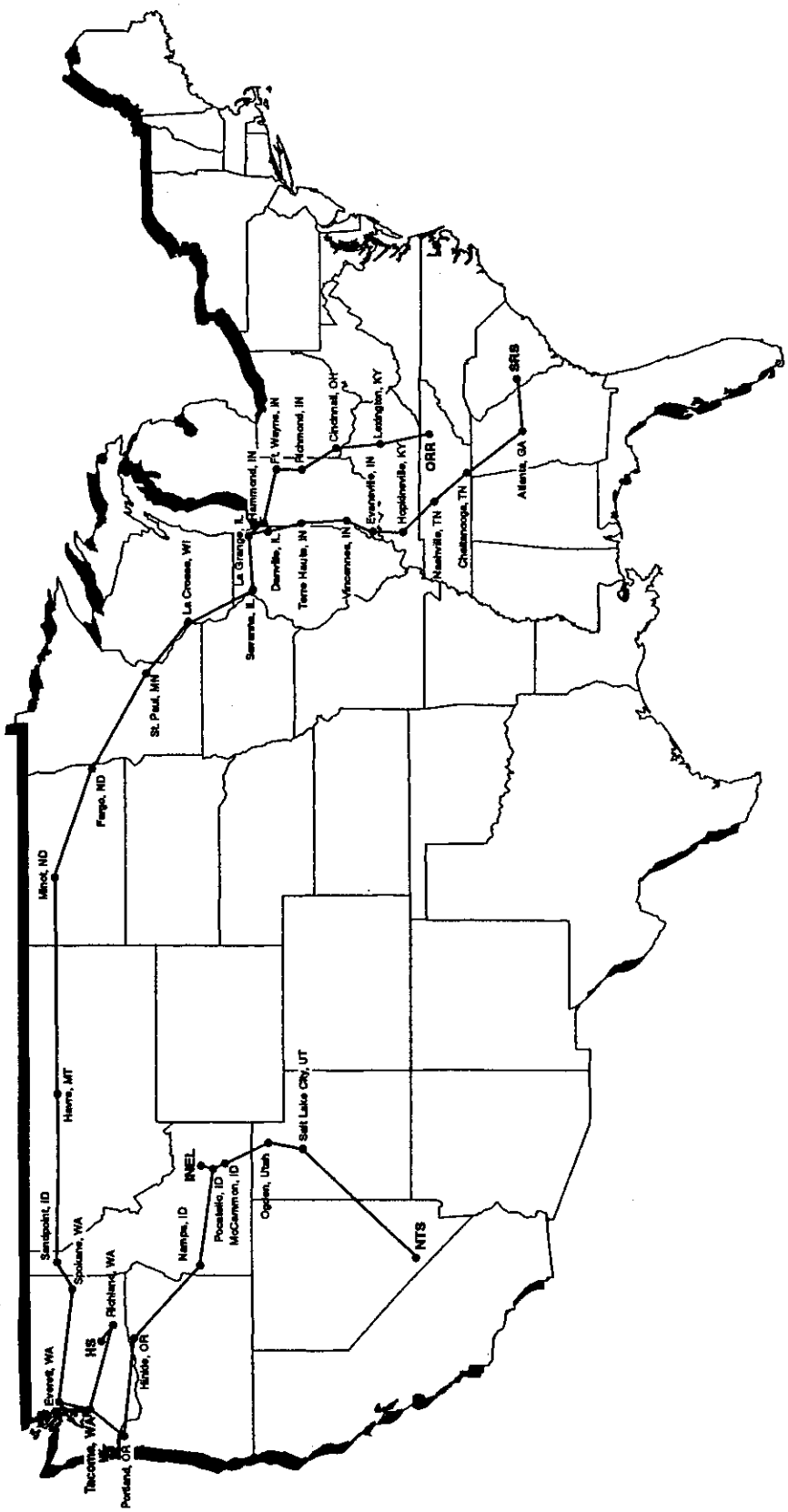

Figure E1-18 Representative Rail Routes from Tacoma, WA to Department of Energy Management Sites

ROUTE FROM: BN 14100-TACOMA, WA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14011-SEATTLE	WA	42.
BN	14008-EVERETT	WA	74.
BN	13828-SPOKANE	WA	383.
BN	13300-SANOPOINT	ID	447.
BN	13089-SHELBY	MT	784.
BN	13168-HAVRE	MT	885.
BN	15740-WILLISTON	NO	1204.
BN	10936-MINOT	NO	1316.
BN	10935-SURREY	NO	1322.
BN	11134-CASSELTON	ND	1537.
BN	11132-FARGO	ND	1557.
BN	11131-MOORHEAD	MN	1560.
BN	9663-STAPLES	MN	1674.
BN	9671-SAUK RAPIDS	MN	1739.
BN	9826-COON CREEK	MN	1789.
BN	9798-NORTHTOWN	MN	1797.
BN	9830-ST PAUL	MN	1809.
BN	5736-LA CROSSE	WI	1930.
BN	4327-EAST DUBUQUE	IL	2041.
BN	4317-SAVANNA	IL	2081.
BN	4190-AURORA	IL	2172.
BN	4170-LA GRANGE	IL	2197.
IHB	4170-LA GRANGE	IL	2197.
IHB	4172-ARGO	IL	2201.
IHB	4163-8LUE ISLAND	IL	2213.
IHB	4223-DOLTON / RIVERDA	IL	2217.
CSXT	4223-DOLTON / RIVERDA	IL	2217.
CSXT	4206-CHICAGO HEIGHTS	IL	2227.
CSXT	4636-WATSEKA	IL	2278.
CSXT	4642-DANVILLE	IL	2323.
CSXT	3863-TERRE HAUTE	IN	2380.
CSXT	3812-VINCENNES	IN	2433.
CSXT	3838-EVANSVILLE	IN	2483.
CSXT	3839-HENDERSON	KY	2496.
CSXT	7061-HOPKINSVILLE	KY	2584.
CSXT	7201-MADISON	TN	2644.
CSXT	7202-NASHVILLE	TN	2654.
CSXT	7187-TULLAHOMA	TN	2733.
CSXT	7235-CHATTANOOGA	TN	2813.
CSXT	7888-DALTON	GA	2851.
CSXT	7889-CARTERSVILLE	GA	2902.
CSXT	7907-MARIETTA	GA	2935.
CSXT	7914-ATLANTA	GA	2944.
CSXT	7961-AUGUSTA	GA	3119.
CSXT	7732-ROBBINS	SC	3148.
CSXT	7717-DUN8ARTON / WELL	SC	3157.
USG	7717-DUNBARTON / WELL	SC	3157.
USG	15359-SRP	SC	3165.

ROUTE FROM: BN 14100-TACOMA, WA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14180-VANCOUVER	WA	142.
BN	14179-PORTLAND	OR	150.
UP	14179-PORTLAND	OR	150.
UP	14197-OREGON TRUNK JCT	OR	245.
UP	14223-HINKLE	OR	337.
UP	14220-PENOLETON	OR	368.
UP	13412-NAMPA	ID	636.
UP	13370-POCATELLO	ID	878.
UP	13336-SCOVILLE	ID	934.

ROUTE FROM: BN 14100-TACOMA, WA
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14180-VANCOUVER	WA	142.
BN	13964-KENNEWICK	WA	355.
BN	13890-PASCO	WA	357.
WCRC	13890-PASCO	WA	357.
WCRC	13964-KENNEWICK	WA	358.
WCRC	13941-RICHLAND	WA	366.
USG	13941-RICHLAND	WA	366.
USG	16212-HANFORO S 300	WA	374.

ROUTE FROM: BN 14100-TACOMA, WA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14011-SEATTLE	WA	42.
BN	14008-EVERETT	WA	74.
BN	13828-SPOKANE	WA	383.
BN	13300-SANDPOINT	ID	447.
BN	13089-SHELBY	MT	784.
BN	13168-HAVRE	MT	885.
BN	15740-WILLISTON	ND	1204.
BN	10936-MINOT	ND	1316.
BN	10935-SURREY	ND	1322.
BN	11134-CASSELTON	ND	1537.
BN	11132-FARGO	ND	1557.
BN	11131-MOORHEAD	MN	1560.
BN	9663-STAPLES	MN	1674.
BN	9671-SAUK RAPIDS	MN	1739.
BN	9826-COON CREEK	MN	1789.
BN	9798-NORTHTOWN	MN	1797.
BN	9830-ST PAUL	MN	1809.
BN	5736-LA CROSSE	WI	1930.
BN	4327-EAST OUBUQUE	IL	2041.
BN	4317-SAVANNA	IL	2081.
BN	4190-AURORA	IL	2172.
BN	4170-LA GRANGE	IL	2197.
IHB	4170-LA GRANGE	IL	2197.
IHB	4172-ARGO	IL	2201.
IHB	4163-BLUE ISLAND	IL	2213.
IHB	4228-8URNHAM / CALUME	IL	2221.
NS	4228-BURNHAM / CALUME	IL	2221.
NS	4076-HAMMOND	IN	2223.
NS	4064-HOBART	IN	2240.
NS	4020-ARGOS	IN	2303.
NS	3548-FORT WAYNE	IN	2362.
NS	3650-MUNCIE	IN	2426.
NS	3688-RICHMOND	IN	2471.
NS	3251-HAMILTON	OH	2525.
NS	3234-IVORYDALE	OH	2542.
NS	3228-CINCINNATI	OH	2549.
NS	6850-LEXINGTON	KY	2623.
NS	6979-DANVILLE	KY	2660.
NS	7260-HARRIMAN	TN	2822.
NS	15316-K-25	TN	2837.

ROUTE FROM: BN 14100-TACOMA, WA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14180-VANCOUVER	WA	142.
BN	14179-PORTLAND	OR	150.
UP	14179-PORTLAND	OR	150.
UP	14197-OREGON TRUNK JCT	OR	245.
UP	14223-HINKLE	OR	337.
UP	14220-PENDLETON	OR	368.
UP	13412-NAMPA	ID	636.
UP	13370-POCATELLO	ID	878.
UP	13369-MC CAMMON	ID	901.
UP	13568-OGDEN	UT	1015.
UP	13595-SALT LAKE CITY	UT	1050.
UP	13630-LYNNDYL	UT	1162.
UP	14766-VALLEY	NV	1479.
USG	14766-VALLEY	NV	1479.
USG	16333-YUCCA MOUNTAIN	NV	1578.

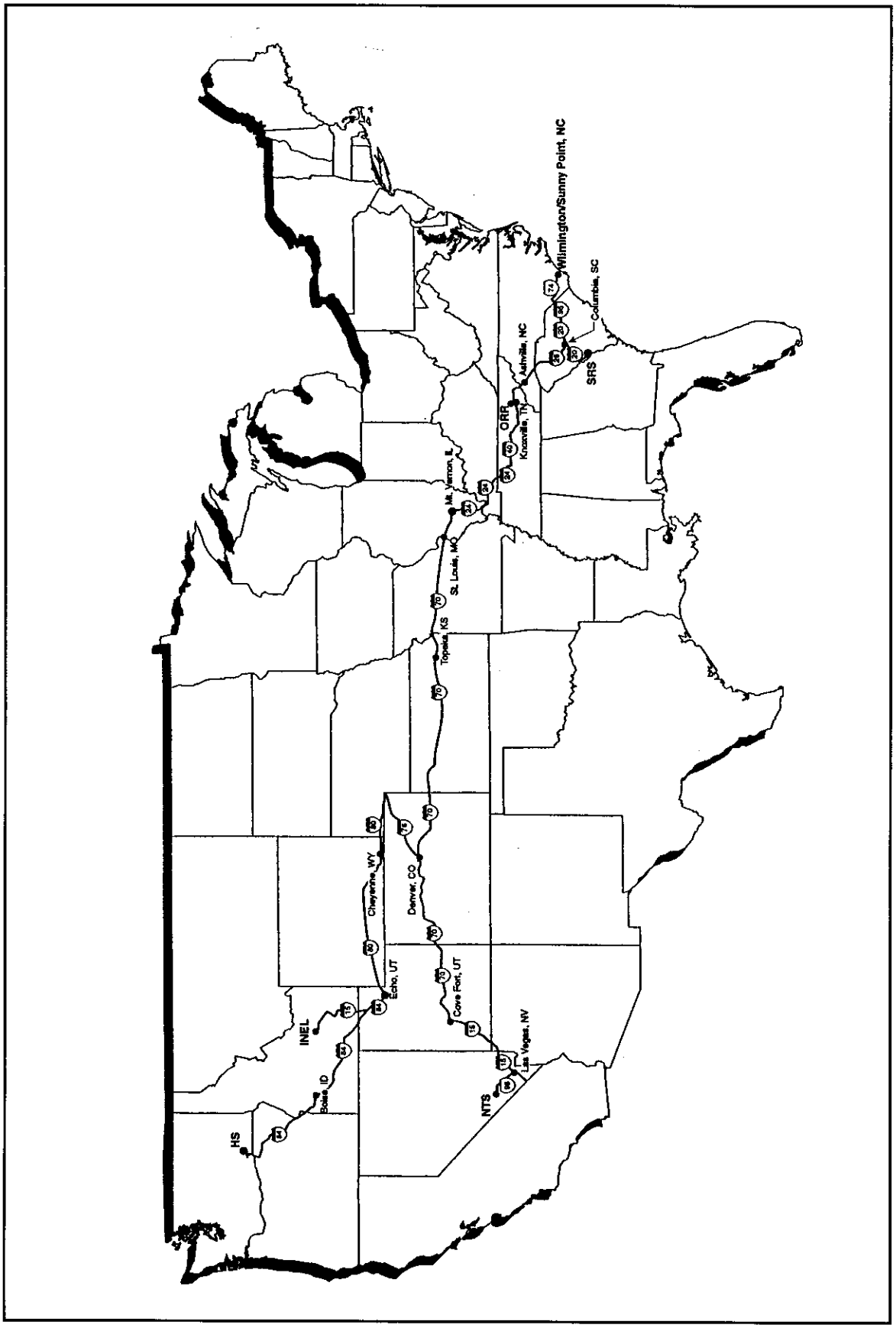

Figure E1-19 Representative Truck Routes from Wilmington, NC to Department of Energy Management Sites

From: WILMINGTON, NC
To : SRP, SC

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELAND	SE	U17	U74	NC
U74	U76	CHADBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: WILMINGTON, NC
To : ID NATL ENG LAB, ID

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELAND	SE	U17	U74	NC
U74	U76	CHAOBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		COLUMBIA	NW	I20	I26	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	IO NATL ENG LAB				ID

From: WILMINGTON, NC
To : HANFORD, WA

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELAND	SE	U17	U74	NC
U74	U76	CHADBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		COLUMBIA	NW	I20	I26	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		OENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORO				WA

From: WILMINGTON, NC
To : K-25, TN

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELANO	SE	U17	U74	NC
U74	U76	CHAOBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		COLUMBIA	NW	I20	I26	SC
I26		ASHEVILLE	SW	I26	I40	NC
140		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: WILMINGTON, NC
To : MERCURY, NV

Routing through:

		WILMINGTON					NC
LOCAL		WILMINGTON	N	U117	LOCL		NC
U117		WILMINGTON	NW	U117	U421		NC
U421		WILMINGTON	W	U17	U421		NC
U17	U74	LELAND	SE	U17	U74		NC
U74	U76	CHADBOURN	NE	U74	U76		NC
U74		LUMBERTON	SW	I95	U74		NC
I95		FLORENCE	W	I20	I95		SC
I20		COLUMBIA	NW	I20	I26		SC
I26		ASHEVILLE	SW	I26	I40		NC
I40		KNOXVILLE	NE	I40	I640		TN
I640		KNOXVILLE	NW	I640	I75		TN
I640	I75	KNOXVILLE	W	I40	I640		TN
I40	I75	OAK RIDGE	S	I40	I75		TN
I40		NASHVILLE	E	I24	I40		TN
I24	I40	NASHVILLE	SE	I24	I40		TN
I24	I65	INGLEWOOD	W	I24	I65		TN
I24		PULLEYS MILL	W	I24	I57		IL
I57		MT VERNON	SW	I57	I64		IL
I57	I64	MT VERNON	NW	I57	I64		IL
I64		WASHINGTON PK	SE	I255	I64		IL
I255		EDWARDSVILLE	SW	I255	I27D		IL
I270		ST LOUIS	NW	I270	I70		MO
I70		KANSAS CITY	SE	I435	I70		MO
I435		KANSAS CITY	W	I435	I70		KS
I70 $	TKST$	TOPEKA	E	I470	I70		KS
I470$	TKST$	TOPEKA	S	I335	I470		KS
I470		TOPEKA	W	I470	I70		KS
I70		COVE FORT	W	I15	I70		UT
I15		LAS VEGAS					NV
U95		MERCURY	S	U95	LOCL		NV
LOCAL		MERCURY					NV

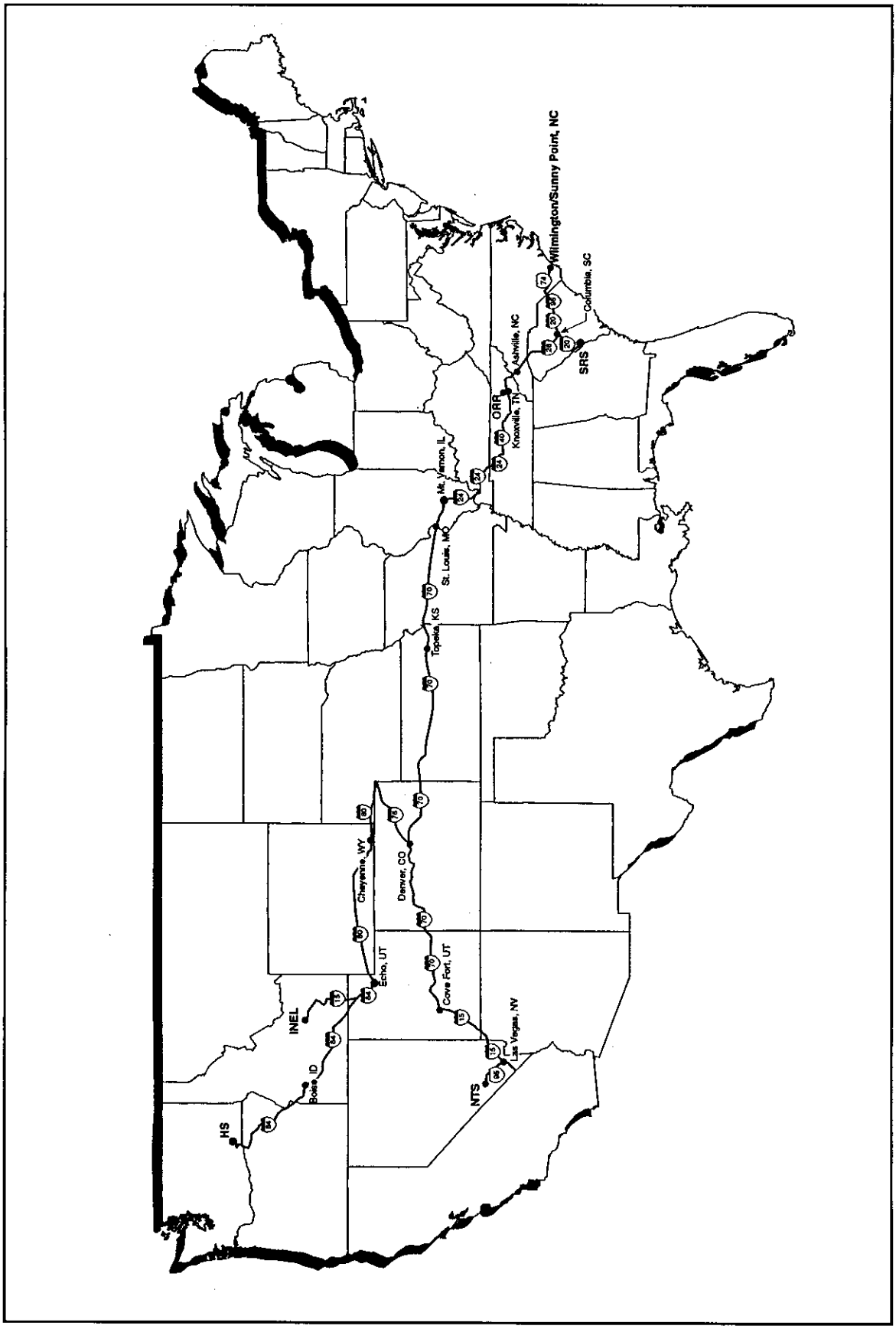

Figure E1-20 Representative Rail Routes from Wilmington, NC to Department of Energy Management Sites

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7671-OILLON	SC	106.
CSXT	7675-FLORENCE	SC	135.
CSXT	7690-CHARLESTON	SC	233.
CSXT	7739-FAIRFAX	SC	327.
CSXT	7732-ROBBINS	SC	356.
CSXT	7717-DUNBARTON / WELL	SC	365.
USG	7717-DUNBARTON / WELL	SC	365.
USG	15359-SRP	SC	373.

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7470-HAMLET	NC	117.
CSXT	74D7-MONROE	NC	170.
CSXT	7834-CLINTON	SC	262.
CSXT	7838-GREENWOOD	SC	290.
CSXT	7956-ATHENS	GA	371.
CSXT	7914-ATLANTA	GA	462.
CSXT	7907-MARIETTA	GA	472.
CSXT	7889-CARTERSVILLE	GA	504.
CSXT	7888-DALTON	GA	555.
CSXT	7235-CHATTANOOGA	TN	593.
CSXT	7187-TULLAHOMA	TN	674.
CSXT	7202-NASHVILLE	TN	753.
CSXT	7201-MADISON	TN	763.
CSXT	7061-HOPKINSVILLE	KY	823.
CSXT	3839-HENDERSON	KY	910.
CSXT	3838-EVANSVILLE	IN	923.
CSXT	3812-VINCENNES	IN	973.
CSXT	4952-SALEM	IL	1052.
CSXT	10859-EAST ST LOUIS	IL	1117.
TRRA	10859-EAST ST LOUIS	IL	1117.
TRRA	10858-ST LOUIS	MO	1123.
UP	10858-ST LOUIS	MO	1123.
UP	10656-JEFFERSON CITY	MO	1245.
UP	10616-KANSAS CITY	MO	1421.
UP	10617-KANSAS CITY	KS	1424.
UP	11823-LAWRENCE	KS	1463.
UP	11697-TOPEKA	KS	1493.
UP	11696-MENOKEN	KS	1498.
UP	11681-MARYSVILLE	KS	1573.
UP	11405-HASTINGS	NE	1683.
UP	11410-GIBBON	NE	1709.
UP	11352-NORTH PLATTE	NE	1787.
UP	11358-O FALLONS	NE	1836.
UP	13703-JULESBURG	CO	1904.
UP	13465-CHEYENNE	WY	2050.
UP	13462-LARAMIE	WY	2102.
UP	13494-GRANGER	WY	2378.
UP	13369-MC CAMMON	ID	2570.
UP	13370-POCATELLO	ID	2593.
UP	13336-SCOVILLE	ID	2649.

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: USG 16212-HANFORD S 300, WA

RR	NOOE	STATE	OIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEM8ROKE	NC	86.
CSXT	7470-HAMLET	NC	117.
CSXT	7407-MONROE	NC	170.
CSXT	7834-CLINTON	SC	262.
CSXT	7838-GREENWOOD	SC	290.
CSXT	7956-ATHENS	GA	371.
CSXT	7914-ATLANTA	GA	462.
CSXT	7907-MARIETTA	GA	472.
CSXT	7889-CARTERSVILLE	GA	504.
CSXT	7888-DALTON	GA	555.
CSXT	7235-CHATTANOOGA	TN	593.
CSXT	7187-TULLAHOMA	TN	674.
CSXT	7202-NASHVILLE	TN	753.
CSXT	7201-MADISON	TN	763.
CSXT	7061-HOPKINSVILLE	KY	823.
CSXT	3839-HENDERSON	KY	910.
CSXT	3838-EVANSVILLE	IN	923.
CSXT	3812-VINCENNES	IN	973.
CSXT	4952-SALEM	IL	1052.
CSXT	10859-EAST ST LOUIS	IL	1117.
TRRA	10859-EAST ST LOUIS	IL	1117.
TRRA	10858-ST LOUIS	MO	1123.
UP	10858-ST LOUIS	MO	1123.
UP	10656-JEFFERSON CITY	MO	1245.
UP	10616-KANSAS CITY	MO	1421.
UP	10617-KANSAS CITY	KS	1424.
UP	11823-LAWRENCE	KS	1463.
UP	11697-TOPEKA	KS	1493.
UP	11696-MENOKEN	KS	1498.
UP	11681-MARYSVILLE	KS	1573.
UP	11405-HASTINGS	NE	1683.
UP	11410-GIBBON	NE	1709.
UP	11352-NORTH PLATTE	NE	1787.
UP	11358-O FALLONS	NE	1836.
UP	13703-JULESBURG	CO	1904.
UP	13465-CHEYENNE	WY	2050.
UP	13462-LARAMIE	WY	2102.
UP	13494-GRANGER	WY	2378.
UP	13369-MC CAMMON	ID	2570.
UP	13370-POCATELLO	ID	2593.
UP	13412-NAMPA	ID	2835.
UP	14220-PENDLETON	OR	3103.
UP	14223-HINKLE	OR	3134.
UP	13894-WALLULA	WA	3163.
UP	13964-KENNEWICK	WA	3178.
UP	13941-RICHLAND	WA	3187.
USG	13941-RICHLAND	WA	3187.
USG	16212-HANFORD S 3D0	WA	3195.

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7470-HAMLET	NC	117.
CSXT	7472-WADESBORO	NC	142.
WSS	7472-WADESBORO	NC	142.
WSS	7462-LEXINGTON	NC	210.
NS	7462-LEXINGTON	NC	210.
NS	7478-SALISBURY	NC	227.
NS	7394-HICKORY	NC	284.
NS	7387-MARION	NC	326.
NS	7343-ASHEVILLE	NC	366.
NS	7318-MORRISTOWN	TN	446.
NS	7286-KNOXVILLE	TN	487.
NS	7288-DOSSETT	TN	512.
NS	15316-K-25	TN	533.

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7470-HAMLET	NC	117.
CSXT	7407-MONROE	NC	170.
CSXT	7834-CLINTON	SC	262.
CSXT	7838-GREENWOOD	SC	290.
CSXT	7956-ATHENS	GA	371.
CSXT	7914-ATLANTA	GA	462.
CSXT	7907-MARIETTA	GA	472.
CSXT	7889-CARTERSVILLE	GA	504.
CSXT	7888-DALTON	GA	555.
CSXT	7235-CHATTANOOGA	TN	593.
CSXT	7187-TULLAHOMA	TN	674.
CSXT	7202-NASHVILLE	TN	753.
CSXT	7201-MADISON	TN	763.
CSXT	7061-HOPKINSVILLE	KY	823.
CSXT	3839-HENDERSON	KY	910.
CSXT	3838-EVANSVILLE	IN	923.
CSXT	3812-VINCENNES	IN	973.
CSXT	4952-SALEM	IL	1052.
CSXT	10859-EAST ST LOUIS	IL	1117.
TRRA	10859-EAST ST LOUIS	IL	1117.
TRRA	10858-ST LOUIS	MO	1123.
UP	10858-ST LOUIS	MO	1123.
UP	10656-JEFFERSON CITY	MO	1245.
UP	10616-KANSAS CITY	MO	1421.
UP	10617-KANSAS CITY	KS	1424.
UP	11823-LAWRENCE	KS	1463.
UP	11697-TOPEKA	KS	1493.
UP	11696-MENOKEN	KS	1498.
UP	11681-MARYSVILLE	KS	1573.
UP	11405-HASTINGS	NE	1683.
UP	11410-GIBBON	NE	1709.
UP	11352-NORTH PLATTE	NE	1787.
UP	11358-O FALLONS	NE	1836.
UP	13703-JULESBURG	CO	1904.
UP	13465-CHEYENNE	WY	2050.
UP	13462-LARAMIE	WY	2102.
UP	13494-GRANGER	WY	2378.
UP	13568-OGDEN	UT	2517.
UP	13595-SALT LAKE CITY	UT	2552.
UP	13630-LYNNDYL	UT	2664.
UP	14766-VALLEY	NV	2981.
USG	14766-VALLEY	NV	2981.
USG	16333-YUCCA MOUNTAIN	NV	3080.

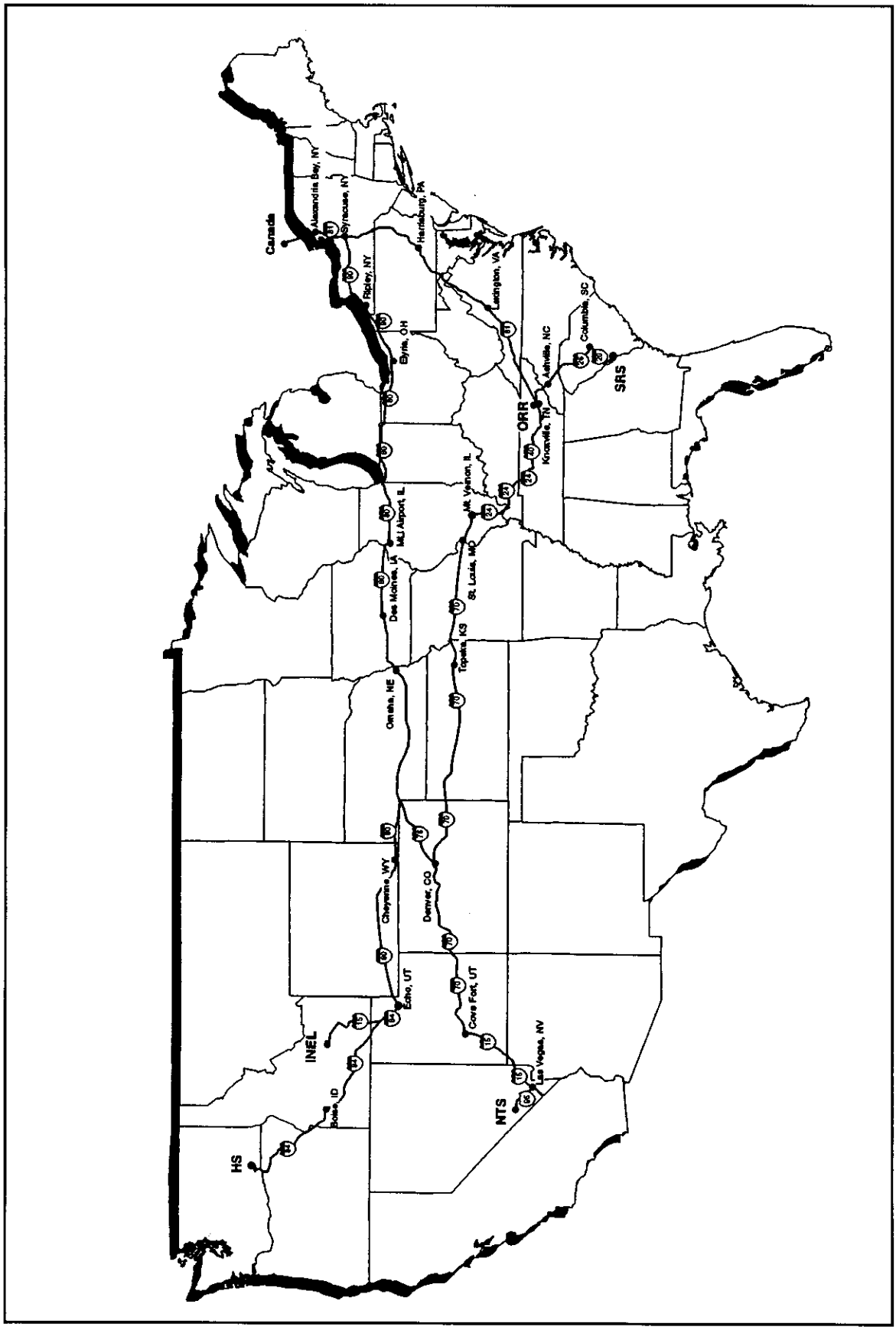

Figure E1-21 Representative Truck Routes from Eastern Canada to Department of Energy Management Sites

From: ALEXANDRIA BAY SW I81 S12, NY
To : SRP, SC

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		CICERO	S	I481	I81	NY
I481		SYRACUSE	S	I481	I81	NY
I81		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I81		FT CHISWELL	E	I77	I81	VA
I77		COLUMBIA	NE	I20	I77	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: ALEXANDRIA BAY SW I81 S12, NY
To : ID NATL ENG LAB, ID

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		SYRACUSE	N	I81	I90	NY
I90 $	TNYT$	BUFFALO	NE	I290	I90	NY
I90	TNYT	LACKAWANNA	E	I90	X55	NY
I90 $	TNYT$	RIPLEY	W	I90	X61	NY
I90		WILLOUGH8Y HLS	W	I271	I90	OH
I271		BEDFORD	NE	I271	I480	OH
I480		N RIDGEVILLE	S	I80	X9A	OH
I80 $		ELYRIA	NW	I80	I90	OH
I80 $	I90 $	PDRTAGE	W	I80	I90	IN
I80	I94	LANSING	W	I294	I80	IL
I294$	I80 $	HOMEWOOD	NW	I294	I80	IL
I80		GREEN ROCK	SE	I74	I80	IL
I280	I74	MLI AIRPORT		I280	I74	IL
I280		DAVENPORT	NW	I280	I80	IA
I8D		DES MOINES	N	I235	I35	IA
I35	I80	DES MOINES	W	I235	I35	IA
I80		MINDEN	NW	I680	I80	IA
I680		LOVELAND	SW	I29	I680	IA
I29	I680	CRESCENT	W	I29	I680	IA
I680		OMAHA	SW	I680	I80	NE
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		8LACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: ALEXANDRIA BAY SW I81 S12, NY
To : HANFORD, WA

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		SYRACUSE	N	I81	I90	NY
I90 $	TNYT$	BUFFALO	NE	I290	I90	NY
I90	TNYT	LACKAWANNA	E	I90	X55	NY
I90 $	TNYT$	RIPLEY	W	I90	X61	NY
I90		WILLOUGHBY HLS	W	I271	I90	OH
I271		BEDFORD	NE	I271	I480	OH
I480		N RIDGEVILLE	S	I80	X9A	OH
I80 $		ELYRIA	NW	I80	I90	OH
I80 $	I90 $	PORTAGE	W	I80	I90	IN
I80	I94	LANSING	W	I294	I80	IL
I294$	I80 $	HOMEWOOD	NW	I294	I80	IL
I80		GREEN ROCK	SE	I74	I80	IL
I280	I74	MLI AIRPORT		I280	I74	IL
I280		DAVENPORT	NW	I280	I80	IA
I80		DES MOINES	N	I235	I35	IA
I35	I80	DES MOINES	W	I235	I35	IA
I80		MINDEN	NW	I680	I80	IA
I680		LOVELAND	SW	I29	I68D	IA
I29	I680	CRESCENT	W	I29	I680	IA
I680		OMAHA	SW	I680	I80	NE
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFDRD				WA

From: ALEXANDRIA BAY SW I81 S12, NY
To : K-25, TN

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		CICERO	S	I481	I81	NY
I481		SYRACUSE	S	I481	I81	NY
I81		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I81		FT CHISWELL	E	I77	I81	VA
I77	I81	WYTHEVILLE	E	I77	I81	VA
I81		DANDRIDGE	NE	I40	I81	TN
I4D		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: ALEXANDRIA BAY SW I81 S12, NY
To : YUCCA MOUNTAIN, NV

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		SYRACUSE	N	I81	I90	NY
I90 $	TNYT$	8UFFALO	NE	I290	I90	NY
I90	TNYT	LACKAWANNA	E	I90	X55	NY
I90 $	TNYT$	RIPLEY	W	I90	X61	NY
I90		WILLOUGHBY HLS	W	I271	I90	OH
I271		BEDFORD	NE	I271	I480	OH
I480		N RIDGEVILLE	S	I80	X9A	OH
I80 $		ELYRIA	NW	I80	I90	OH
I80 $	I90 $	PORTAGE	W	I80	I90	IN
I80	I94	LANSING	W	I294	I80	IL
I294$	I80 $	HOMEWOOD	NW	I294	I80	IL
I80		GREEN ROCK	SE	I74	I80	IL
I280	I74	MLI AIRPORT		I280	I74	IL
I280		DAVENPORT	NW	I280	I80	IA
I80		DES MOINES	N	I235	I35	IA
I35	I80	OES MOINES	W	I235	I35	IA
I80		MINOEN	NW	I680	I80	IA
I68D		LOVELANO	SW	I29	I680	IA
I29	I680	CRESCENT	W	I29	I680	IA
I680		OMAHA	SW	I680	I80	NE
I80		BIG SPRINGS	SW	I76	I80	NE
I76		COMMERCE CITY	W	I25	I76	CO
I25		DENVER	N	I25	I70	CO
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		AMARGOSA VALLY		U95	S373	NV
LOCAL		YUCCA MOUNTAIN				NV

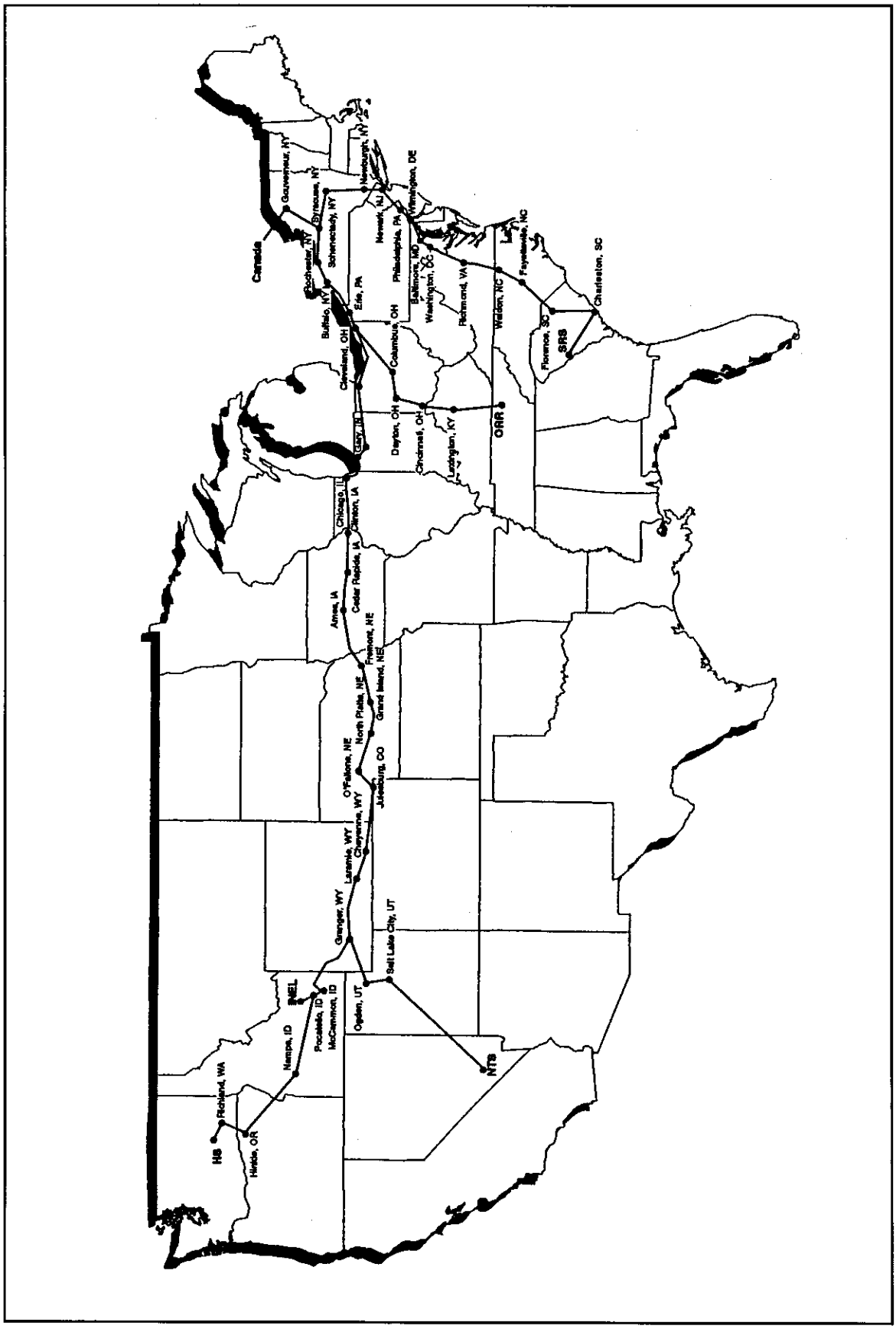

Figure E1-22 Representative Rail Routes from Eastern Canada to Department of Energy Management Sites

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	755-ROME	NY	145.
CR	756-UTICA	NY	159.
CR	706-SCHENECTADY	NY	237.
CR	700-SELKIRK	NY	259.
CR	1094-NEWBURGH	NY	334.
CR	1215-JERSEY CITY	NJ	400.
CR	1183-NEWARK	NJ	405.
CR	1230-ALDENE	NJ	411.
CR	1311-BOUND BROOK	NJ	430.
CR	1447-PHILADELPHIA	PA	492.
CR	2456-WILMINGTON	DE	529.
CR	2516-BALTIMORE	MD	593.
CR	2596-WASHINGTON	DC	627.
CR	2595-ALEXANDRIA	VA	637.
CSXT	2595-ALEXANDRIA	VA	637.
CSXT	6082-RICHMOND	VA	752.
CSXT	6087-COLONIAL HEIGHTS	VA	774.
CSXT	6064-PETERSBURG	VA	779.
CSXT	7563-WELDON	NC	839.
CSXT	7565-ROCKY MOUNT	NC	876.
CSXT	7566-WILSON	NC	890.
CSXT	7606-FAYETTEVILLE	NC	964.
CSXT	7620-PEMBROKE	NC	993.
CSXT	7671-DILLON	SC	1013.
CSXT	7675-FLORENCE	SC	1042.
CSXT	7690-CHARLESTON	SC	1140.
CSXT	7739-FAIRFAX	SC	1234.
CSXT	7732-ROBBINS	SC	1263.
CSXT	7717-DUNBARTON / WELLS	SC	1272.
USG	7717-DUNBARTON / WELLS	SC	1272.
USG	15359-SRP	SC	1280.

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	780-SOLVAY	NY	103.
CR	817-ROCHESTER	NY	179.
CR	880-BUFFALO	NY	241.
CR	968-ERIE	PA	333.
CR	2649-ASHTABULA	OH	372.
CR	2728-CLEVELAND	OH	429.
CR	2633-ELYRIA	OH	456.
CR	3442-TOLEDO	OH	535.
CR	3526-GOSHEN	IN	657.
CR	3525-ELKHART	IN	667.
CR	4022-SOUTH BEND	IN	682.
CR	4067-PORTER	IN	727.
CR	4070-GARY	IN	742.
CR	4073-CLARKE	IN	746.
CR	4074-INDIANA HARBOR	IN	749.
CR	4232-SOUTH CHICAGO	IL	757.
CR	4217-CHICAGO	IL	770.
CNW	4217-CHICAGO	IL	770.
CNW	4234-PROVISO	IL	784.
CNW	4311-DE KALB	IL	826.
CNW	4324-NELSON	IL	871.
CNW	10304-CLINTON	IA	903.
CNW	10289-CEDAR RAPIDS	IA	984.
CNW	10265-MARSHALLTOWN	IA	1051.
CNW	10246-NEVADA	IA	1078.
CNW	10271-AMES	IA	1089.
CNW	10176-MISSOURI VALLEY	IA	1222.
CNW	10198-CALIFORNIA JCT	IA	1228.
CNW	11340-FREMONT	NE	1256.
UP	11340-FREMONT	NE	1256.
UP	11406-GRAND ISLAND	NE	1365.
UP	11410-GIBBON	NE	1391.
UP	11352-NORTH PLATTE	NE	1469.
UP	11358-O FALLONS	NE	1518.
UP	13703-JULESBURG	CO	1586.
UP	13465-CHEYENNE	WY	1732.
UP	13462-LARAMIE	WY	1784.
UP	13494-GRANGER	WY	2060.
UP	13369-MC CAMMON	ID	2252.
UP	13370-POCATELLO	ID	2275.
UP	13336-SCOVILLE	ID	2331.

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	780-SOLVAY	NY	103.
CR	817-ROCHESTER	NY	179.
CR	880-BUFFALO	NY	241.
CR	968-ERIE	PA	333.
CR	2649-ASHTABULA	OH	372.
CR	2728-CLEVELANO	OH	429.
CR	2633-ELYRIA	OH	456.
CR	3442-TOLEDO	OH	535.
CR	3526-GOSHEN	IN	657.
CR	3525-ELKHART	IN	667.
CR	4022-SOUTH BEND	IN	682.
CR	4067-PORTER	IN	727.
CR	4070-GARY	IN	742.
CR	4073-CLARKE	IN	746.
CR	4074-INOIANA HARBOR	IN	749.
CR	4232-SOUTH CHICAGO	IL	757.
CR	4217-CHICAGO	IL	770.
CNW	4217-CHICAGO	IL	770.
CNW	4234-PROVISO	IL	784.
CNW	4311-OE KALB	IL	826.
CNW	4324-NELSON	IL	871.
CNW	10304-CLINTON	IA	903.
CNW	10289-CEOAR RAPIDS	IA	984.
CNW	10265-MARSHALLTOWN	IA	1051.
CNW	10246-NEVAOA	IA	1078.
CNW	10271-AMES	IA	1089.
CNW	10176-MISSOURI VALLEY	IA	1222.
CNW	10198-CALIFORNIA JCT	IA	1228.
CNW	11340-FREMONT	NE	1256.
UP	11340-FREMONT	NE	1256.
UP	11406-GRAND ISLANO	NE	1365.
UP	11410-GIBBON	NE	1391.
UP	11352-NORTH PLATTE	NE	1469.
UP	11358-O FALLONS	NE	1518.
UP	13703-JULES8URG	CO	1586.
UP	13465-CHEYENNE	WY	1732.
UP	13462-LARAMIE	WY	1784.
UP	13494-GRANGER	WY	2060.
UP	13369-MC CAMMON	ID	2252.
UP	13370-POCATELLO	IO	2275.
UP	13412-NAMPA	ID	2517.
UP	14220-PENOLETON	OR	2786.
UP	14223-HINKLE	OR	2817.
UP	13894-WALLULA	WA	2846.
UP	13964-KENNEWICK	WA	2861.
UP	13941-RICHLANO	WA	2870.
USG	13941-RICHLANO	WA	2870.
USG	16212-HANFORO S 30O	WA	2877.

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	780-SOLVAY	NY	103.
CR	817-ROCHESTER	NY	179.
CR	880-BUFFALO	NY	241.
CR	968-ERIE	PA	333.
CR	2649-ASHTABULA	OH	372.
CR	2728-CLEVELANO	OH	429.
CR	2629-WELLINGTON	OH	466.
CR	3399-CRESTLINE	OH	504.
CR	3094-COLUMBUS (4TH STOH		566.
CR	3095-COLUMBUS (BROAD OH		567.
CR	14993-COLUMBUS (BUCKEYOH		569.
CR	3300-SPRINGFIELD	OH	611.
CR	3282-OAYTON	OH	632.
CR	3250-MIDDLETOWN	OH	652.
CR	3234-IVORYDALE	OH	677.
CR	3228-CINCINNATI	OH	684.
NS	3228-CINCINNATI	OH	684.
NS	6850-LEXINGTON	KY	758.
NS	6979-OANVILLE	KY	795.
NS	7260-HARRIMAN	TN	957.
NS	15316-K-25	TN	972.

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	780-SOLVAY	NY	103.
CR	817-ROCHESTER	NY	179.
CR	880-BUFFALO	NY	241.
CR	968-ERIE	PA	333.
CR	2649-ASHTA8ULA	OH	372.
CR	2728-CLEVELAND	OH	429.
CR	2633-ELYRIA	OH	456.
CR	3442-TOLEDO	OH	535.
CR	3526-GOSHEN	IN	657.
CR	3525-ELKHART	IN	667.
CR	4022-SOUTH BEND	IN	682.
CR	4067-PORTER	IN	727.
CR	4070-GARY	IN	742.
CR	4073-CLARKE	IN	746.
CR	4074-INDIANA HARBOR	IN	749.
CR	4232-SOUTH CHICAGO	IL	757.
CR	4217-CHICAGO	IL	770.
CNW	4217-CHICAGO	IL	770.
CNW	4234-PROVISO	IL	784.
CNW	4311-DE KALB	IL	826.
CNW	4324-NELSON	IL	871.
CNW	10304-CLINTON	IA	903.
CNW	10289-CEDAR RAPIDS	IA	984.
CNW	10265-MARSHALLTOWN	IA	1051.
CNW	10246-NEVAOA	IA	1078.
CNW	10271-AMES	IA	1089.
CNW	10176-MISSOURI VALLEY	IA	1222.
CNW	10198-CALIFORNIA JCT	IA	1228.
CNW	11340-FREMONT	NE	1256.
UP	11340-FREMONT	NE	1256.
UP	11406-GRAND ISLAND	NE	1365.
UP	1141D-GIBBON	NE	1391.
UP	11352-NORTH PLATTE	NE	1469.
UP	11358-O FALLONS	NE	1518.
UP	13703-JULESBURG	CO	1586.
UP	13465-CHEYENNE	WY	1732.
UP	13462-LARAMIE	WY	1784.
UP	13494-GRANGER	WY	2060.
UP	13568-OGDEN	UT	2199.
UP	13595-SALT LAKE CITY	UT	2235.
UP	13630-LYNNDYL	UT	2347.
UP	14766-VALLEY	NV	2664.
USG	14766-VALLEY	NV	2664.
USG	16333-YUCCA MOUNTAIN	NV	2763.

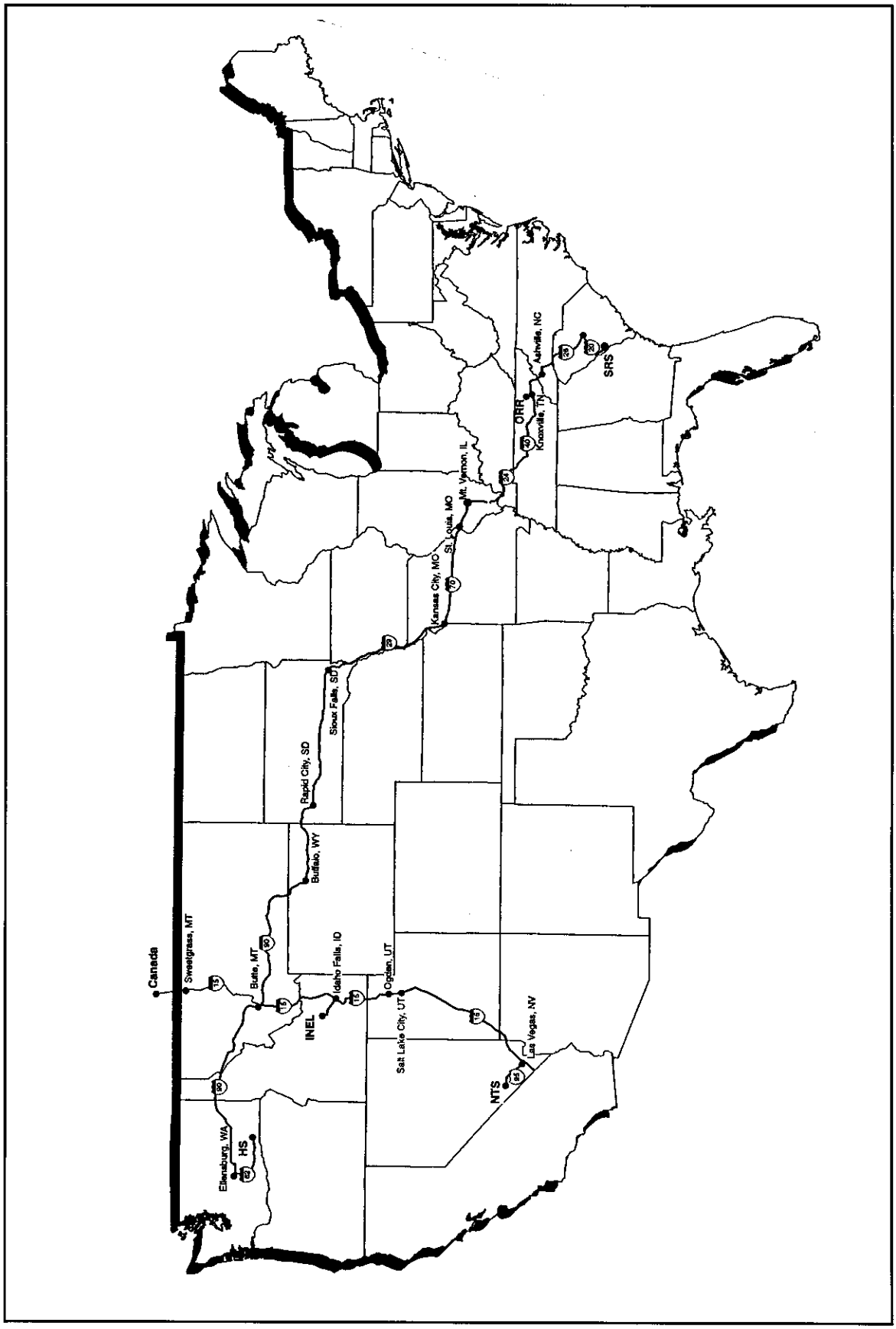

Figure E1-23 Representative Truck Routes from Western Canada to Department of Energy Management Sites

From: SWEETGRASS I15 X394 MT
To : SRL, SC

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I90		SIOUX FALLS	NW	I29	I90	SD
I29		LOVELAND	SW	I29	I680	IA
I680		MINDEN	NW	I680	I80	IA
I80		COUNCIL BLUFFS	SE	I29	I80	IA
I29		KANSAS CITY	NW	I29	I435	MO
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	N	I24	I265	TN
I265		NASHVILLE	W	I265	I40	TN
I40		NASHVILLE	W	I40	I440	TN
I440		NASHVILLE	SE	I24	I440	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: SWEETGRASS I15 X394 MT
To : ID NATL ENG LAB, ID

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I15	I90	BUTTE	W	I15	I90	MT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: SWEETGRASS I15 X394 MT
To : HANFORD, WA

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I15	I90	BUTTE	W	I15	I90	MT
I90		ELLENSBURG	SE	I82	I90	WA
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: SWEETGRASS I15 X394 MT
To : K-25, TN

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I90		SIOUX FALLS	NW	I29	I90	SD
I29		LOVELAND	SW	I29	I680	IA
I680		MINDEN	NW	I680	I80	IA
I80		COUNCIL BLUFFS	SE	I29	I80	IA
I29		KANSAS CITY	NW	I29	I435	MO
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	N	I24	I265	TN
I265		NASHVILLE	W	I265	I40	TN
I40		NASHVILLE	W	I40	I440	TN
I440		NASHVILLE	SE	I24	I440	TN
I24		NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: SWEETGRASS I15 X394 MT
To : MERCURY, NV

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I15	I90	BUTTE	W	I15	I90	MT
I15		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		N SALT LAKE		I15	I215	UT
I215		MIDVALE		I15	I215	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

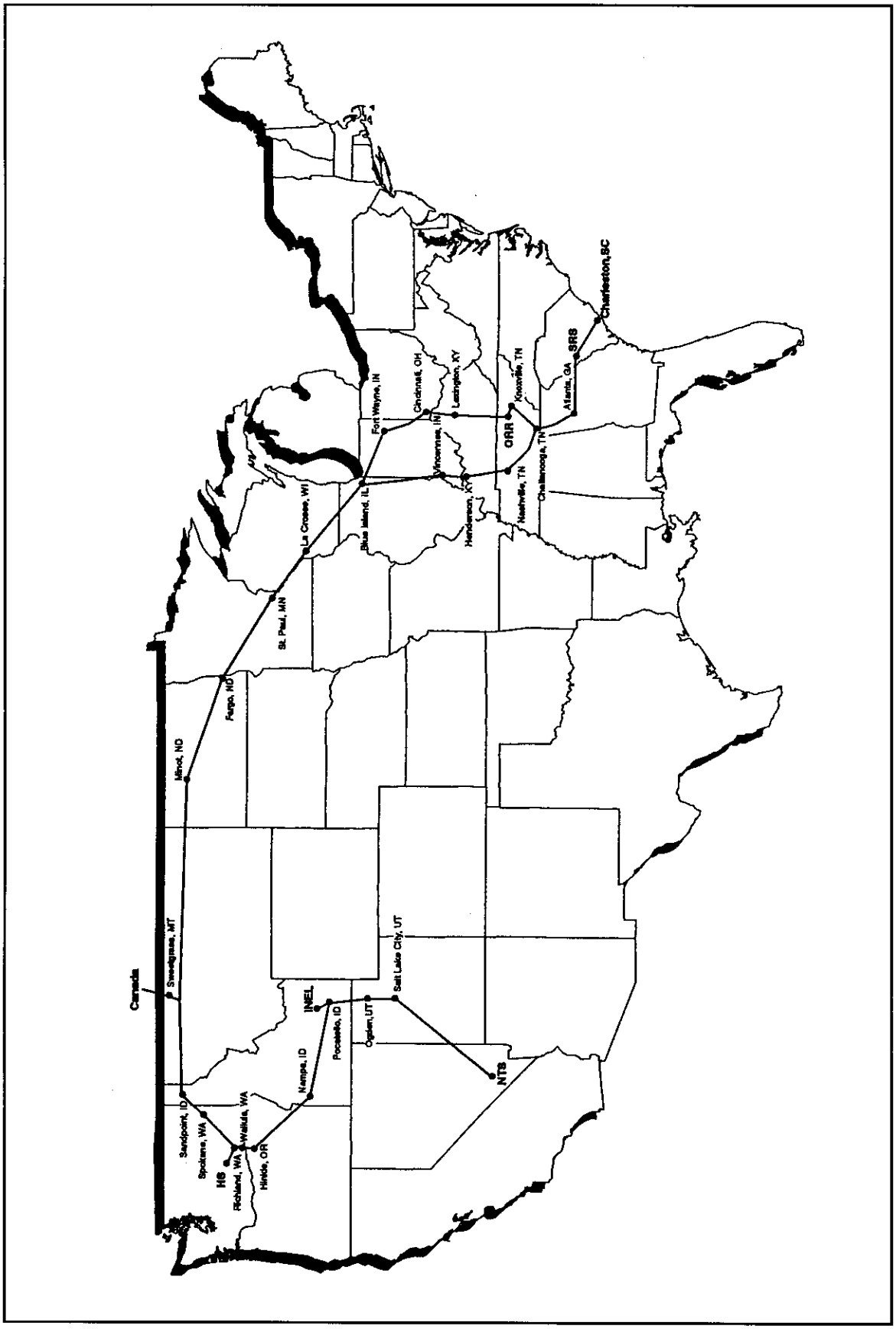

Figure E1-24 Representative Rail Routes from Western Canada to Department of Energy Management Sites

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
8N	13089-SHELBY	MT	43.
8N	13168-HAVRE	MT	144.
BN	15740-WILLISTON	ND	463.
BN	10936-MINOT	ND	575.
BN	10935-SURREY	ND	581.
BN	11134-CASSELTON	ND	796.
BN	11132-FARGO	ND	816.
BN	11131-MOORHEAD	MN	819.
BN	9663-STAPLES	MN	933.
BN	9671-SAUK RAPIDS	MN	998.
BN	9826-COON CREEK	MN	1048.
BN	9798-NORTHTOWN	MN	1056.
BN	9830-ST PAUL	MN	1069.
BN	5736-LA CROSSE	WI	1190.
BN	4327-EAST DUBUQUE	IL	1301.
BN	4317-SAVANNA	IL	1341.
BN	4190-AURORA	IL	1432.
BN	4170-LA GRANGE	IL	1457.
IHB	4170-LA GRANGE	IL	1457.
IHB	4172-ARGO	IL	1461.
IHB	4163-BLUE ISLAND	IL	1473.
IHB	4223-OOLTON / RIVERDA	IL	1477.
CSXT	4223-OOLTON / RIVERDA	IL	1477.
CSXT	4206-CHICAGO HEIGHTS	IL	1487.
CSXT	4636-WATSEKA	IL	1538.
CSXT	4642-DANVILLE	IL	1583.
CSXT	3863-TERRE HAUTE	IN	1640.
CSXT	3812-VINCENNES	IN	1693.
CSXT	3838-EVANSVILLE	IN	1743.
CSXT	3B39-HENDERSON	KY	1756.
CSXT	7061-HOPKINSVILLE	KY	1843.
CSXT	7201-MADISON	TN	1903.
CSXT	7202-NASHVILLE	TN	1913.
CSXT	7187-TULLAHOMA	TN	1992.
CSXT	7235-CHATTANOOGA	TN	2073.
CSXT	7888-OALTON	GA	2111.
CSXT	7889-CARTERSVILLE	GA	2162.
CSXT	7907-MARIETTA	GA	2194.
CSXT	7914-ATLANTA	GA	2204.
CSXT	7961-AUGUSTA	GA	2379.
CSXT	7732-ROBBINS	SC	2408.
CSXT	7717-DUNBARTON / WELL	SC	2417.
USG	7717-DUNBARTON / WELL	SC	2417.
USG	15359-SRP	SC	2425.

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13300-SANDPOINT	IO	374.
BN	13828-SPOKANE	WA	437.
BN	13890-PASCO	WA	589.
BN	13B94-WALLULA	WA	605.
UP	13894-WALLULA	WA	605.
UP	14223-HINKLE	OR	634.
UP	14220-PENDLETON	OR	665.
UP	13412-NAMPA	ID	933.
UP	13370-POCATELLO	ID	1175.
UP	13336-SCOVILLE	ID	1231.

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13300-SANDPOINT	ID	374.
BN	13828-SPOKANE	WA	437.
BN	13890-PASCO	WA	589.
WCRC	13890-PASCO	WA	589.
WCRC	13964-KENNEWICK	WA	590.
WCRC	13941-RICHLANO	WA	598.
USG	13941-RICHLAND	WA	598.
USG	16212-HANFORO S 300	WA	606.

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13089-SHELBY	MT	43.
BN	13168-HAVRE	MT	144.
BN	15740-WILLISTON	ND	463.
BN	10936-MINOT	ND	575.
BN	10935-SURREY	ND	5B1.
BN	11134-CASSELTON	ND	796.
BN	11132-FARGO	ND	816.
BN	11131-MOORHEAO	MN	819.
BN	9663-STAPLES	MN	933.
8N	9671-SAUK RAPIDS	MN	998.
BN	9826-COON CREEK	MN	1048.
BN	9798-NORTHTOWN	MN	1056.
BN	9830-ST PAUL	MN	1069.
BN	5736-LA CROSSE	WI	1190.
BN	4327-EAST DUBUQUE	IL	1301.
BN	4317-SAVANNA	IL	1341.
BN	4190-AURORA	IL	1432.
BN	4170-LA GRANGE	IL	1457.
IHB	4170-LA GRANGE	IL	1457.
IHB	4172-ARGO	IL	1461.
IHB	4163-BLUE ISLAND	IL	1473.
IHB	4228-8URNHAM / CALUME	IL	1481.
NS	4228-BURNHAM / CALUME	IL	1481.
NS	4076-HAMMOND	IN	1483.
NS	4064-HOBART	IN	1499.
NS	4020-ARGOS	IN	1562.
NS	3548-FORT WAYNE	IN	1621.
NS	3650-MUNCIE	IN	1685.
NS	3688-RICHMOND	IN	1730.
NS	3251-HAMILTON	OH	1785.
NS	3234-IVORYDALE	OH	1802.
NS	3228-C1NCINNATI	OH	1809.
NS	6850-LEXINGTON	KY	1883.
NS	6979-DANVILLE	KY	1920.
NS	7260-HARRIMAN	TN	2082.
NS	15316-K-25	TN	2097.

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13300-SANOPOINT	ID	374.
BN	13828-SPOKANE	WA	437.
BN	13890-PASCO	WA	589.
BN	13894-WALLULA	WA	605.
UP	13894-WALLULA	WA	605.
UP	14223-HINKLE	OR	634.
UP	14220-PENDLETON	OR	665.
UP	13412-NAMPA	ID	933.
UP	13370-POCATELLO	IO	1175.
UP	13369-MC CAMMON	ID	1198.
UP	13568-OGDEN	UT	1312.
UP	13595-SALT LAKE CITY	UT	1347.
UP	13630-LYNNDYL	UT	1459.
UP	14766-VALLEY	NV	1776.
USG	14766-VALLEY	NV	1776.
USG	16333-YUCCA MOUNTAIN	NV	1875.

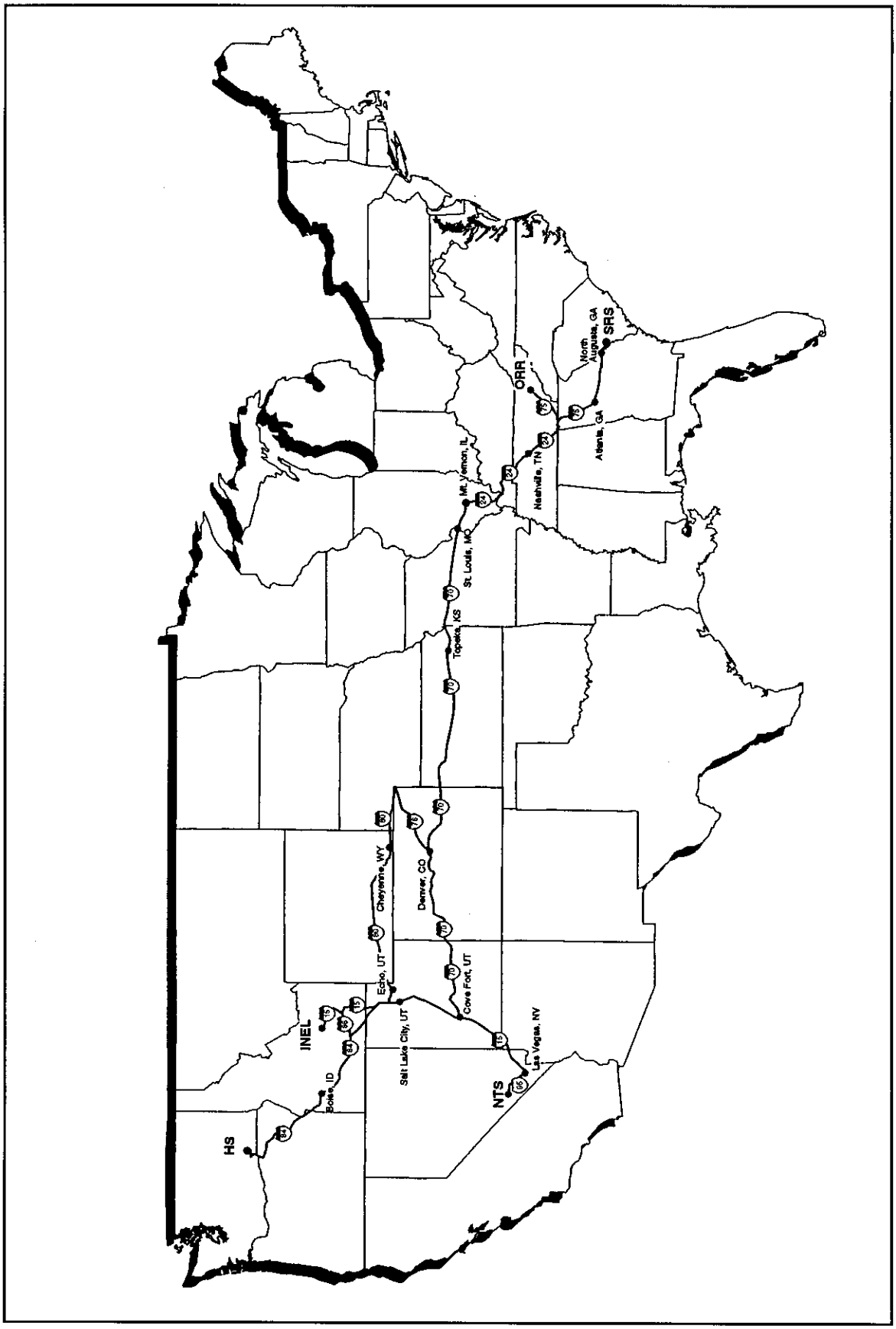

Figure E1-25 Representative Truck Routes Between Department of Energy Management Sites

From: ID NATL ENG LAB, ID
To : HANFORD, WA

Routing through:

		ID NATL ENG LAB				ID
U20	U26	ATOMIC CITY	NW	U20	U26	ID
U26		BLACKFOOT	NW	I15	X92	ID
I15		CHUBBUCK	E	I15	I86	ID
I86		RAFT RIVER	W	I84	I86	ID
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: ID NATL ENG LAB, ID
To : MERCURY, NV

Routing through:

		IO NATL ENG LAB				ID
U20	U26	ATOMIC CITY	NW	U20	U26	ID
U26		BLACKFOOT	NW	I15	X92	IO
I15		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		SALT LAKE CITY	W	I15	I80	UT
I15	I80	SALT LAKE CITY	S	I15	I80	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: ID NATL ENG LAB, ID
To : SRL, SC

Routing through:

		ID NATL ENG LAB				ID
U20	U26	ATOMIC CITY	NW	U20	U26	ID
U26		BLACKFOOT	NW	I15	X92	ID
I15		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		OENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: ID NATL ENG LAB, ID
To : K-25, TN

Routing through:

		ID NATL ENG LAB				ID
U20	U26	ATOMIC CITY	NW	U20	U26	ID
U26		BLACKFOOT	NW	I15	X92	ID
I15		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: HANFORD, WA
To : MERCURY, NV

Routing through:

		HANFORD				WA
LR4S		RICHLAND	N	S240	LR4S	WA
S240		RICHLAND	SE	I182	S240	WA
I182		WEST RICHLAND	S	I182	I82	WA
I82		HERMISTON	SW	I82	I84	OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGOEN	S	I15	I84	UT
I15		SALT LAKE CITY	W	I15	I80	UT
I15	I80	SALT LAKE CITY	S	I15	I80	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: HANFORD, WA
To : SRL, SC

Routing through:

		HANFORD				WA
LR4S		RICHLANO	N	S240	LR4S	WA
S240		RICHLAND	SE	I182	S240	WA
I182		WEST RICHLAND	S	I182	I82	WA
I82		HERMISTON	SW	I82	I84	OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I8D		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: HANFORD, WA
To : K-25, TN

Routing through:

		HANFORD				WA
LR4S		RICHLAND	N	S240	LR4S	WA
S240		RICHLAND	SE	I182	S240	WA
I182		WEST RICHLAND	S	I182	I82	WA
I82		HERMISTON	SW	I82	I84	OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		OENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: K-25, TN
To : SRL, SC

Routing through:

	K-25				TN
S58	KINGSTON	E	I40	S58	TN
I40	OAK RIDGE	S	I40	I75	TN
I75	ATLANTA	NW	I285	I75	GA
I285	ATLANTA	E	I20	I285	GA
I20	NORTH AUGUSTA	NW	I20	S230	SC
S230	NORTH AUGUSTA				SC
S125	CLEARWATER	W	U1	U278	SC
U278	BEECH ISLAND		U278	S125	SC
S125	JACKSON	SE	S125	LSRP	SC
LSRP	SRL				SC

From: MERCURY, NV
To : SRL, SC

Routing through:

		MERCURY				NV
LOCAL		MERCURY	S	U95	LOCL	NV
U95		LAS VEGAS				NV
I15		COVE FORT	W	I15	I70	UT
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: MERCURY, NV
To : K-25, TN

Routing through:

		MERCURY				NV
LOCAL		MERCURY	S	U95	LOCL	NV
U95		LAS VEGAS				NV
I15		COVE FORT	W	I15	I70	UT
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

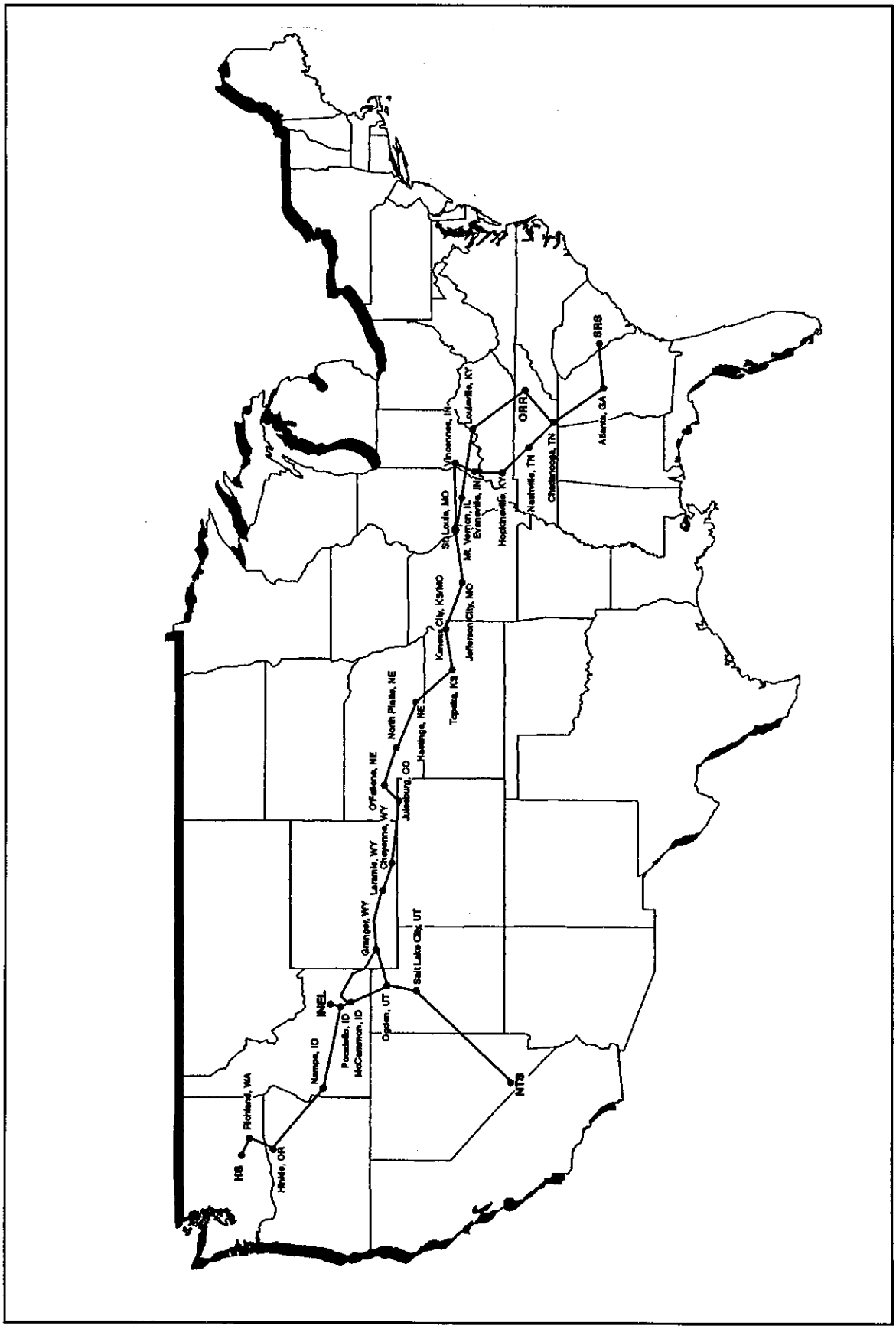

Figure E1-26 Representative Rail Routes Between Department of Energy Management Sites

ROUTE FROM: USG 15359-SRP, SC
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
USG	15359-SRP	SC	0.
USG	7717-DUNBARTON / WELL	SC	8.
CSXT	7717-DUNBARTON / WELL	SC	8.
CSXT	7732-ROBBINS	SC	17.
CSXT	7961-AUGUSTA	GA	46.
CSXT	7914-ATLANTA	GA	221.
CSXT	7907-MARIETTA	GA	231.
CSXT	7889-CARTERSVILLE	GA	263.
CSXT	7888-DALTON	GA	314.
CSXT	7235-CHATTANOOGA	TN	352.
CSXT	7187-TULLAHOMA	TN	433.
CSXT	7202-NASHVILLE	TN	512.
CSXT	7201-MADISON	TN	522.
CSXT	7061-HOPKINSVILLE	KY	582.
CSXT	3839-HENDERSON	KY	669.
CSXT	3838-EVANSVILLE	IN	682.
CSXT	3812-VINCENNES	IN	732.
CSXT	4952-SALEM	IL	811.
CSXT	10859-EAST ST LOUIS	IL	876.
<TR>	10859-EAST ST LOUIS	IL	876.
<TR>	10858-ST LOUIS	MO	882.
UP	10858-ST LOUIS	MO	882.
UP	10656-JEFFERSON CITY	MO	1004.
UP	10616-KANSAS CITY	MO	1180.
UP	10617-KANSAS CITY	KS	1183.
UP	11823-LAWRENCE	KS	1222.
UP	11697-TOPEKA	KS	1252.
UP	11696-MENOKEN	KS	1257.
UP	11681-MARYSVILLE	KS	1332.
UP	11405-HASTINGS	NE	1442.
UP	11410-GIBBON	NE	1468.
UP	11352-NORTH PLATTE	NE	1546.
UP	11358-O FALLONS	NE	1595.
UP	13703-JULESBURG	CO	1663.
UP	13465-CHEYENNE	WY	1809.
UP	13462-LARAMIE	WY	1861.
UP	13494-GRANGER	WY	2137.
UP	13369-MC CAMMON	ID	2329.
UP	13370-POCATELLO	ID	2352.
UP	13412-NAMPA	ID	2594.
UP	14220-PENDLETON	OR	2862.
UP	14223-HINKLE	OR	2893.
UP	13894-WALLULA	WA	2922.
UP	13964-KENNEWICK	WA	2937.
UP	13941-RICHLAND	WA	2946.
USG	13941-RICHLAND	WA	2946.
USG	16212-HANFORD S 300	WA	2954.

ROUTE FROM: USG 15359-SRP, SC
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
USG	15359-SRP	SC	0.
USG	7717-DUNBARTON / WELL	SC	8.
CSXT	7717-DUNBARTON / WELL	SC	8.
CSXT	7732-ROBBINS	SC	17.
CSXT	7961-AUGUSTA	GA	46.
CSXT	7914-ATLANTA	GA	221.
CSXT	7907-MARIETTA	GA	231.
CSXT	7889-CARTERSVILLE	GA	263.
CSXT	7888-DALTON	GA	314.
CSXT	7235-CHATTANOOGA	TN	352.
CSXT	7187-TULLAHOMA	TN	433.
CSXT	7202-NASHVILLE	TN	512.
CSXT	7201-MADISON	TN	522.
CSXT	7061-HOPKINSVILLE	KY	582.
CSXT	3839-HENDERSON	KY	669.
CSXT	3838-EVANSVILLE	IN	682.
CSXT	3812-VINCENNES	IN	732.
CSXT	4952-SALEM	IL	811.
CSXT	10859-EAST ST LOUIS	IL	876.
<TR>	10859-EAST ST LOUIS	IL	876.
<TR>	10858-ST LOUIS	MO	882.
UP	10858-ST LOUIS	MO	882.
UP	10656-JEFFERSON CITY	MO	1004.
UP	10616-KANSAS CITY	MO	1180.
UP	10617-KANSAS CITY	KS	1183.
UP	11823-LAWRENCE	KS	1222.
UP	11697-TOPEKA	KS	1252.
UP	11696-MENOKEN	KS	1257.
UP	11681-MARYSVILLE	KS	1332.
UP	11405-HASTINGS	NE	1442.
UP	11410-GIBBON	NE	1468.
UP	11352-NORTH PLATTE	NE	1546.
UP	11358-O FALLONS	NE	1595.
UP	13703-JULESBURG	CO	1663.
UP	13465-CHEYENNE	WY	1809.
UP	13462-LARAMIE	WY	1861.
UP	13494-GRANGER	WY	2137.
UP	13369-MC CAMMON	ID	2329.
UP	13370-POCATELLO	ID	2352.
UP	13336-SCOVILLE	ID	2408.

ROUTE FROM: USG 15359-SRP, SC
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
USG	15359-SRP	SC	0.
USG	7717-DUNBARTON / WELL	SC	8.
CSXT	7717-DUNBARTON / WELL	SC	8.
CSXT	7732-ROBBINS	SC	17.
CSXT	7961-AUGUSTA	GA	46.
CSXT	7914-ATLANTA	GA	221.
CSXT	7907-MARIETTA	GA	231.
CSXT	7889-CARTERSVILLE	GA	263.
CSXT	7888-DALTON	GA	314.
CSXT	7235-CHATTANOOGA	TN	352.
CSXT	7187-TULLAHOMA	TN	433.
CSXT	7202-NASHVILLE	TN	512.
CSXT	7201-MADISON	TN	522.
CSXT	7061-HOPKINSVILLE	KY	582.
CSXT	3839-HENDERSON	KY	669.
CSXT	3838-EVANSVILLE	IN	682.
CSXT	3812-VINCENNES	IN	732.
CSXT	4952-SALEM	IL	811.
CSXT	10859-EAST ST LOUIS	IL	876.
<TR>	10859-EAST ST LOUIS	IL	876.
<TR>	10858-ST LOUIS	MO	882.
UP	10858-ST LOUIS	MO	882.
UP	10656-JEFFERSON CITY	MO	1004.
UP	10616-KANSAS CITY	MO	1180.
UP	10617-KANSAS CITY	KS	1183.
UP	11823-LAWRENCE	KS	1222.
UP	11697-TOPEKA	KS	1252.
UP	11696-MENOKEN	KS	1257.
UP	11681-MARYSVILLE	KS	1332.
UP	11405-HASTINGS	NE	1442.
UP	11410-GIBBON	NE	1468.
UP	11352-NORTH PLATTE	NE	1546.
UP	11358-O FALLONS	NE	1595.
UP	13703-JULESBURG	CO	1663.
UP	13465-CHEYENNE	WY	1809.
UP	13462-LARAMIE	WY	1861.
UP	13494-GRANGER	WY	2137.
UP	13568-OGDEN	UT	2276.
UP	13595-SALT LAKE CITY	UT	2311.
UP	13630-LYNNDYL	UT	2423.
UP	14766-VALLEY	NV	2740.
USG	14766-VALLEY	NV	2740.
USG	16333-YUCCA MOUNTAIN	NV	2839.

ROUTE FROM: UP 13336-SCOVILLE, ID
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
UP	13336-SCOVILLE	ID	0.
UP	13370-POCATELLO	ID	56.
UP	13412-NAMPA	ID	298.
UP	14220-PENDLETON	OR	567.
UP	14223-HINKLE	OR	598.
UP	13894-WALLULA	WA	627.
UP	13964-KENNEWICK	WA	642.
UP	13941-RICHLAND	WA	650.
USG	13941-RICHLAND	WA	650.
USG	16212-HANFORD S 300	WA	658.

ROUTE FROM: UP 13336-SCOVILLE, ID
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
UP	13336-SCOVILLE	ID	0.
UP	13370-POCATELLO	ID	56.
UP	13369-MC CAMMON	ID	79.
UP	13568-OGDEN	UT	193.
UP	13595-SALT LAKE CITY	UT	228.
UP	13630-LYNNDYL	UT	340.
UP	14766-VALLEY	NV	657.
USG	14766-VALLEY	NV	657.
USG	16333-YUCCA MOUNTAIN	NV	756.

ROUTE FROM: USG 16212-HANFORD S 300, WA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
USG	16212-HANFORD S 300	WA	0.
USG	13941-RICHLAND	WA	8.
UP	13941-RICHLAND	WA	8.
UP	13964-KENNEWICK	WA	16.
UP	13894-WALLULA	WA	31.
UP	14223-HINKLE	OR	60.
UP	14220-PENDLETON	OR	91.
UP	13412-NAMPA	ID	360.
UP	13370-POCATELLO	ID	602.
UP	13369-MC CAMMON	ID	625.
UP	13568-OGDEN	UT	738.
UP	13595-SALT LAKE CITY	UT	774.
UP	13630-LYNNDYL	UT	886.
UP	14766-VALLEY	NV	1203.
USG	14766-VALLEY	NV	1203.
USG	16333-YUCCA MOUNTAIN	NV	1302.

ROUTE FROM: NS 15316-K-25, TN
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
NS	15316-K-25	TN	0.
NS	7260-HARRIMAN	TN	15.
NS	6979-DANVILLE	KY	177.
NS	7008-LOUISVILLE	KY	277.
NS	7009-JEFFERSONVILLE	IN	281.
NS	4797-MOUNT CARMEL	IL	406.
NS	4954-MOUNT VERNON	IL	469.
NS	4953-CENTRALIA	IL	491.
NS	10859-EAST ST LOUIS	IL	549.
NS	10858-ST LOUIS	MO	555.
NS	10494-CENTRALIA	MO	674.
NS	10498-MOBERLY	MO	697.
NS	10616-KANSAS CITY	MO	828.
UP	10616-KANSAS CITY	MO	828.
UP	10617-KANSAS CITY	KS	831.
UP	11823-LAWRENCE	KS	869.
UP	11697-TOPEKA	KS	899.
UP	11696-MENOKEN	KS	904.
UP	11681-MARYSVILLE	KS	979.
UP	11405-HASTINGS	NE	1089.
UP	11410-GIBBON	NE	1115.
UP	11352-NORTH PLATTE	NE	1193.
UP	11358-O FALLONS	NE	1242.
UP	13703-JULESBURG	CO	1310.
UP	13465-CHEYENNE	WY	1456.
UP	13462-LARAMIE	WY	1508.
UP	13494-GRANGER	WY	1784.
UP	13369-MC CAMMON	ID	1976.
UP	13370-POCATELLO	IO	1999.
UP	13412-NAMPA	IO	2241.
UP	14220-PENOLETON	OR	2510.
UP	14223-HINKLE	OR	2541.
UP	13894-WALLULA	WA	2570.
UP	13964-KENNEWICK	WA	2585.
UP	13941-RICHLAND	WA	2594.
USG	13941-RICHLAND	WA	2594.
USG	16212-HANFORD S 300	WA	2601.

ROUTE FROM: NS 15316-K-25, TN
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
NS	15316-K-25	TN	0.
NS	7260-HARRIMAN	TN	15.
NS	6979-DANVILLE	KY	177.
NS	7008-LOUISVILLE	KY	277.
NS	7009-JEFFERSONVILLE	IN	281.
NS	4797-MOUNT CARMEL	IL	406.
NS	4954-MOUNT VERNON	IL	469.
NS	4953-CENTRALIA	IL	491.
NS	10859-EAST ST LOUIS	IL	549.
NS	10858-ST LOUIS	MO	555.
NS	10494-CENTRALIA	MO	674.
NS	10498-MOBERLY	MO	697.
NS	10616-KANSAS CITY	MO	828.
UP	10616-KANSAS CITY	MO	828.
UP	10617-KANSAS CITY	KS	831.
UP	11823-LAWRENCE	KS	869.
UP	11697-TOPEKA	KS	899.
UP	11696-MENOKEN	KS	904.
UP	11681-MARYSVILLE	KS	979.
UP	11405-HASTINGS	NE	1089.
UP	11410-GIBBON	NE	1115.
UP	11352-NORTH PLATTE	NE	1193.
UP	11358-O FALLONS	NE	1242.
UP	13703-JULESBURG	CO	1310.
UP	13465-CHEYENNE	WY	1456.
UP	13462-LARAMIE	WY	1508.
UP	13494-GRANGER	WY	1784.
UP	13369-MC CAMMON	ID	1976.
UP	13370-POCATELLO	ID	1999.
UP	13336-SCOVILLE	ID	2055.

ROUTE FROM: NS 15316-K-25, TN
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
NS	15316-K-25	TN	0.
NS	7260-HARRIMAN	TN	15.
NS	6979-DANVILLE	KY	177.
NS	7008-LOUISVILLE	KY	277.
NS	7009-JEFFERSONVILLE	IN	281.
NS	4797-MOUNT CARMEL	IL	406.
NS	4954-MOUNT VERNON	IL	469.
NS	4953-CENTRALIA	IL	491.
NS	10859-EAST ST LOUIS	IL	549.
NS	10858-ST LOUIS	MO	555.
NS	10494-CENTRALIA	MO	674.
NS	10498-MOBERLY	MO	697.
NS	10616-KANSAS CITY	MO	828.
UP	10616-KANSAS CITY	MO	828.
UP	10617-KANSAS CITY	KS	831.
UP	11823-LAWRENCE	KS	869.
UP	11697-TOPEKA	KS	899.
UP	11696-MENOKEN	KS	904.
UP	11681-MARYSVILLE	KS	979.
UP	11405-HASTINGS	NE	1089.
UP	11410-GIBBON	NE	1115.
UP	11352-NORTH PLATTE	NE	1193.
UP	11358-O FALLONS	NE	1242.
UP	13703-JULESBURG	CO	1310.
UP	13465-CHEYENNE	WY	1456.
UP	13462-LARAMIE	WY	1508.
UP	13494-GRANGER	WY	1784.
UP	13568-OGDEN	UT	1923.
UP	13595-SALT LAKE CITY	UT	1959.
UP	13630-LYNNDYL	UT	2071.
UP	14766-VALLEY	NV	2388.
USG	14766-VALLEY	NV	2388.
USG	16333-YUCCA MOUNTAIN	NV	2487.

ROUTE FROM: NS 15316-K-25, TN
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
NS	15316-K-25	TN	0.
NS	7288-DOSSETT	TN	21.
NS	7286-KNOXVILLE	TN	46.
NS	7318-MORRISTOWN	TN	87.
NS	7343-ASHEVILLE	NC	167.
NS	7814-SPARTANBURG	SC	234.
CSXT	7814-SPARTANBURG	SC	234.
CSXT	7838-GREENWOOD	SC	300.
CSXT	7961-AUGUSTA	GA	371.
CSXT	7732-ROBBINS	SC	400.
CSXT	7717-DUNBARTON / WELL	SC	409.
USG	7717-DUNBARTON / WELL	SC	409.
USG	15359-SRP	SC	417.

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix F
Description and Impacts of Storage Technology Alternatives

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

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Appendix F
Description and Impacts of Storage Technology Alternatives

Summary

This appendix presents a description and evaluation of currently available spent nuclear fuel storage technologies, and their applicability to foreign research reactor spent nuclear fuel. These technologies represent the range of alternatives that would be available to implement the proposed action. Some of these technologies are currently in use at U.S. Department of Energy (DOE) facilities. Several dry storage cask and/or building designs have been licensed by the U.S. Nuclear Regulatory Commission (NRC) and are operational with commercial nuclear power plant spent fuel at several locations.

This appendix also discusses potential storage sites and impacts of foreign research reactor spent nuclear fuel storage at these locations. The major sections in this appendix are:

Section F.1 Description of Existing and Proposed Technologies for Storage of Spent Nuclear Fuel

Section F.2 Storage Technology Evaluation Methodology

Section F.3 Selection of Storage Technologies for Further Evaluation

Section F.4 Environmental Impacts at Foreign Research Reactor Spent Nuclear Fuel Management Sites

Section F.5 Occupational Radiation Impacts from Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel

Section F.6 Evaluation Methodologies and Assumptions for Incident-Free Operations and Hypothetical Accidents at Management Sites

Section F.7 Economic Evaluation of Foreign Research Reactor Spent Nuclear Fuel Storage and Related Management Alternatives

Figure F-1 presents the different spent nuclear fuel storage technologies, which are divided into wet and dry systems and further classified by their materials of construction (i.e., concrete, metal), location (i.e., aboveground or belowground), and size (i.e., cask versus vault building or pool). The final level of detail is the specific design with 12 specific vendors' designs displayed in this figure. The following specific designs are of U.S. origin: Nuclear Assurance Corporation, MC-10, NUHOMS, and Ventilated Storage Cask-24 (Section F.1 describes these in more detail). The others are designed by foreign companies, but many of these companies, such as Transnuclear Inc., have U.S. affiliates. The principal categories of spent nuclear fuel storage technology are dry vault (building), dry cask, and wet pool.

This appendix discusses the aforementioned designs in terms of their shielding, criticality, thermal, structural, cost, and ease of use features. Some numerical design parameters are presented for comparison. Advantages and vulnerabilities of each design are also presented. Since none of these designs have been specifically designed or licensed for foreign research reactor spent nuclear fuel and related research reactor type fuel, some extrapolation has been made in this comparative assessment. All of the existing commercial designs and proposed new designs appear to be adaptable to foreign research reactor spent

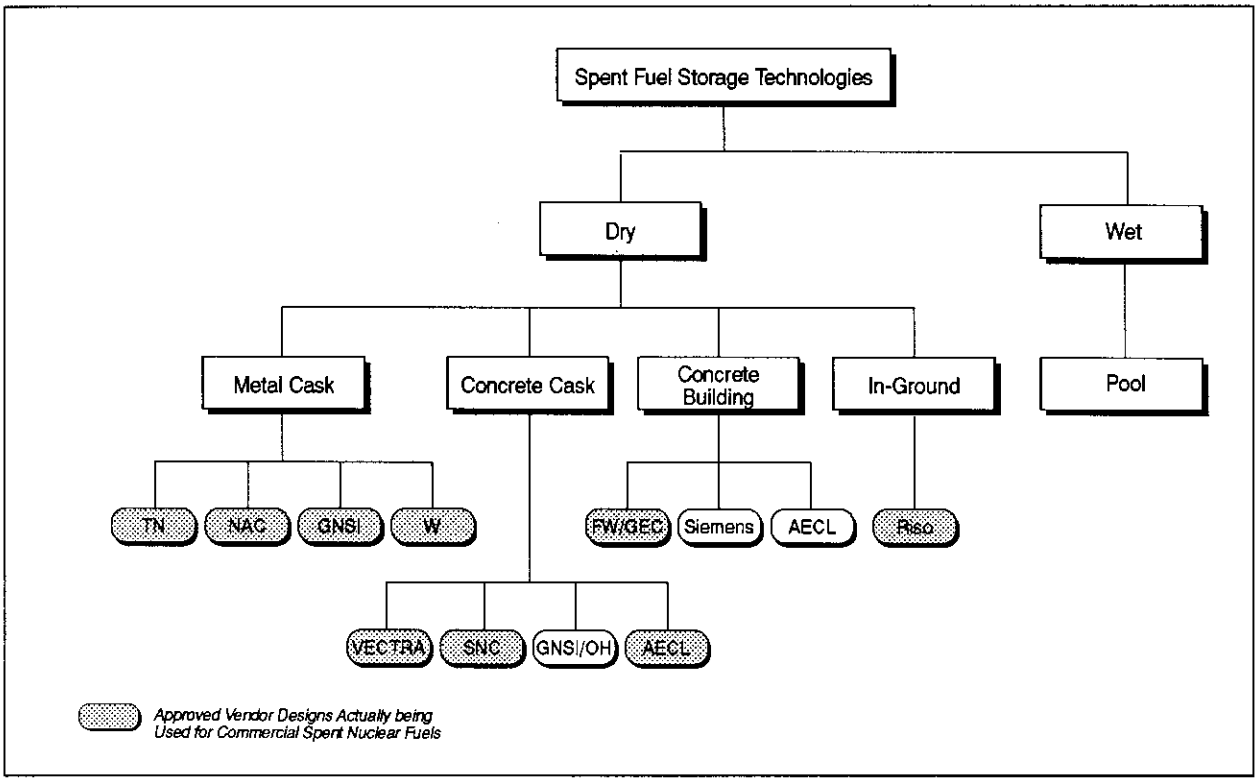


Figure F-1 Spent Nuclear Fuel Storage Technologies and Vendors

nuclear fuel with only minor, easily implemented modifications, such as interior baskets for holding the spent nuclear fuel. Use of existing facilities at a site for staging and characterization favors a cask storage approach, while a stand-alone, separate spent nuclear fuel storage approach requires a vault and other support facilities. Schedule and monetary considerations favor casks over the vault for sites with existing facilities, and this is why most domestic utilities are pursuing dry casks for long-term storage of spent nuclear fuel. Casks are the only independent spent nuclear fuel storage installation designs that have received certification by the NRC in accordance with 10 Code of Federal Regulations (CFR) 72 Appendix K.

The evaluation indicates that both wet and dry storage of foreign research reactor spent nuclear fuel appear acceptable for the time periods envisioned for the proposed action (i.e., through 2036). Commercial spent nuclear fuel dry storage systems require a minimum wet pool storage time or cooldown period of approximately 5 years after discharge from the nuclear reactor prior to emplacement into dry storage. In actual practice, this usually averages around an 8-year average cooldown period and, frequently, the commercial spent nuclear fuel has had over a 10-year cooldown period in a wet pool prior to emplacement into dry storage. This cooldown period ensures that licensed conditions for cladding temperatures (based upon potential corrosion, and usually around 350°C, or 630°F) are not exceeded. Foreign research reactor spent nuclear fuel has a lower cladding temperature limit based upon a phase transition in the aluminum metal cladding; this aluminum cladding limit has been identified as 175°C (315°F). Thus, a maximum cooldown period of 3 years of wet pool storage after irradiation has been identified for the foreign research reactor spent nuclear fuel prior to emplacement into dry storage. This would ensure that all foreign research reactor spent nuclear fuel elements were below a heat load of 40 Watts each, and most elements would be 10 Watts or less. The majority of the currently available foreign research reactor spent nuclear fuel already meets this requirement. Most of the existing dry storage designs appear acceptable for foreign

research reactor spent nuclear fuel, without any clear preference. It should be noted that a research and development project to examine the applicability of aluminum-clad spent nuclear fuel dry storage at the Savannah River Site was initiated in Fiscal Year (FY) 1994.

The utilization of dry storage methods for foreign research reactor spent nuclear fuel requires the acquisition of racks, baskets, storage canisters, and/or casks. New construction would be required for dry vaults, except for several existing facilities at the Nevada Test Site and Idaho National Engineering Laboratory.

The utilization of wet storage methods requires a lined basin within a seismically qualified facility with the ability to maintain water chemistry and handle liquid radioactive waste. Currently, there are few existing DOE facilities in this category, and none have sufficient capacity to accommodate all of the foreign research reactor spent nuclear fuel. Thus, the selection of wet storage would require DOE acquisition of a facility, either by new construction or purchase of an existing facility such as the Barnwell Nuclear Fuels Plant (BNFP) that is owned by Allied General Nuclear Services. A summary of storage technology characteristics is given Table F-1. Sections F.4, F.5, and F.6 address environmental impacts, occupational dose, and accident consequences for storage. Section F.7 discusses costs in detail.

Table F-1 Summary of Storage Technology Characteristics for Commercial and Foreign Research Reactor Spent Nuclear Fuel

Storage Technology	*DOE Site Status (New or Existing)*	*Land Use Ha (Ac)*	*Annual Low-level Waste (m^3)*[a]	*Potential Annual Spent Nuclear Fuel Storage Public Impact (LCFs)*	*Lead Time until Spent Nuclear Fuel Storage (Years)*
Dry Vault - Utility Fuel	New	4 (5)	1-4	NA	2-3
Dry Cask (Concrete) - Utility Fuel	New	4 (5)	1-4	NA	2-3
Wet Pool - Utility Fuel	New	2 (3)	1-4	NA	3-5
Dry Vault - Savannah River Site Research Reactor Fuel	New	4 (8)	16	0	5-10[b]
Dry Cask - Savannah River Site Research Reactor Fuel	New	4 (5)	16	0	3-5[b]
Wet Pool - Idaho National Engineering Laboratory Research Reactor Fuel	New	2 (3)	12	2.4×10^{-12} to 2.5×10^{-10}	5-10[b]

NA = Not Available; LCF = Latent Cancer Fatality

[a] *Low-Level Waste generation decreases significantly if spent nuclear fuel is only being stored, without additional spent nuclear fuel receipts. To convert to ft^3, multiply by 35.3.*

[b] *To allow for extended periodic examination and characterization of fuel.*

DOE currently has pilot-scale experience with dry storage of spent nuclear fuel, and there are no identified technical constraints that would prevent dry storage of foreign research reactor spent nuclear fuel. There would be some need, however, for characterization, canning, and periodic inspection and monitoring. Both NRC-licensed and not yet licensed dry storage designs are readily available from commercial vendors. NRC-licensed designs have the following advantages:

- specific NRC requirements have been met that are equivalent to DOE requirements and guidance,
- extensive, interactive technical safety reviews have already been conducted between the supplier and the regulator,
- peer and public review has occurred as part of the licensing process,
- proven applications are in operation at commercial nuclear power plant sites, and
- documentation and quality assurance requirements have been satisfied.

For sites with an existing spent nuclear fuel infrastructure that includes facilities for spent nuclear fuel receipt, examination, and loading, a modular approach based upon casks can be implemented rapidly to meet Phase 1 requirements using standard funding and procurement capital appropriation methods. The casks could also be used for Phase 2, and their usage would avoid additional procurement. A modular dry vault approach represents an integrated self-contained, stand-alone facility, and can be used at any of the proposed management sites. However, construction of the vault could represent a major project or major systems acquisition under DOE management requirements, which may require a 7 to 10 year period for completion. Thus, a vault dry storage approach probably could not be available immediately. Metal cask development programs, such as dual- and multi-purpose casks, eliminate many storage site handling requirements and may provide future improvements.

Section F.7 evaluates the economics of the entire (40-year plus) foreign research reactor spent nuclear fuel program, including transportation, receipt/handling/inspection, storage, preparation for disposal, transportation to the repository, and disposal for the storage/disposal and chemical separation/vitrification alternatives. Costs are presented as rough-order-of-magnitude net present values, using a 4.9 percent real discount rate. In 1996 dollars, minimum total program costs for the storage alternative are about $800 million. This total divides into four very roughly equal parts: shipping to the United States and program management; receiving and storage at existing facilities; receiving, storage, and fuels qualification at not-yet existing facilities; and repository disposal. Other cost factors would be expected to add as much as $500 million to the program costs. Among the other cost factors are systems integration and logistics contingencies ($75 to $100 million), risks associated with limited characterization of the spent nuclear fuel ($100 million), risks associated with direct disposal of HEU ($50 to $100 million) and the probability that future discount rates will be lower than the current 4.9 percent rate ($200 million or more). Total costs, including all contingencies and risks could thus be in the $1.3 billion range.

For chemical separation alternatives, minimum total program costs are about $700 million. Savings in chemical separation and disposal of high-level waste versus storage and disposal of spent nuclear fuel account for the bulk of the difference between the costs in the chemical separation case and the storage case. Other cost factors would be expected to add as much as $250 million to the program costs. The key cost factors are systems integration and logistics contingencies ($75 to $100 million) and the probability that future discount rates will be lower than the current 4.9 percent rate ($100 million or more). If part of the material shipped to the Savannah River Site was chemically separated and part was stored, costs would typically be between the boundaries for all-separation and all-storage.

Hybrid alternatives that ship about 1/4 of the foreign research reactor spent nuclear fuel to the United Kingdom Atomic Energy Authority's Dounreay facility and manage the remainder as in the U.S. chemical separation case generate minimum total program costs of about $650 million. Other cost factors would be about the same as in the chemical separation case.

At the level of accuracy in the costs presented here, alternatives based on chemical separation of aluminum-based spent nuclear fuel in the United States are likely to cost about the same as alternatives that divert a significant fraction of the spent nuclear fuel (aluminum-based and TRIGA) to Dounreay. Alternatives based on storage and direct disposal of spent nuclear fuel or some non-separation processing approach (e.g., melt and dilute) are likely to cost several hundred million dollars more.

F.1 Description of Existing and Proposed Technologies for Storage of Spent Nuclear Fuel

In this section, two major generic technologies will be presented. International and domestic types of each technology will be addressed. Section F.1.1 will discuss the dry storage designs available. Section F.1.2 will address wet storage technology types. The range of alternatives available to each site for the implementation of the proposed action is presented in Section F.1.3.

F.1.1 Dry Storage Designs

F.1.1.1 Overview of Dry Storage Approaches

There are several types of dry storage technology currently in use or proposed by various vendors at DOE sites as well as at commercial nuclear power facilities. These include:

- aboveground free-standing metal casks,
- aboveground free-standing concrete casks,
- aboveground free-standing dry storage buildings (vaults),
- inground lined and unlined holes or wells with or without casks,
- hot cells (buildings), and
- aboveground free-standing multi-purpose or dual-purpose casks.

It should be noted that additional support facilities for transfer and staging operations may be required in order to use the aforementioned dry storage technologies. A short discussion of the advantages and disadvantages of all dry storage technologies is given in the following sections.

It is important to appreciate the different approaches to handling weight and shielding. Today, most spent nuclear fuel facilities utilize a wet pool environment for handling, storing, and transferring spent nuclear fuel to transportation casks. The pool water provides shielding [usually a 3-meter (m) or 10-ft water cover is the minimum requirement], confinement of contamination, decay heat removal, and thermal capacity. All spent nuclear fuel elements weigh less than 0.9 metric tons (1 ton) and are readily moved within the pool by a crane of that capacity. Transportation containers (casks) for highway transport weigh between 18 and 36 metric tons (20 and 40 tons), and rail casks can weigh up to 91 metric tons (100 tons). Thus, most wet pool facilities have a bridge crane spanning the storage areas and the receiving bay(s) with a capacity of 45 to 91 metric tons (50 to 100 tons). Economical dry storage requires that a large number of elements be stored in each cask. Cask weights exceeding 91 metric tons (100 tons) are possible.

Dry storage manufacturers have overcome this problem by using metallic "transfer" canisters. These transfer canisters are considerably lighter than transportation casks, and usually weigh in the 9 to 27 metric tons (10 to 30 ton) range before loading. The transfer canister provides some shielding, but is principally for confinement of the spent nuclear fuel. They are loaded in the same manner as transportation casks.

For dry cask storage, the canister is loaded onto a truck and transferred to a previously constructed, shielded concrete cask away from the wet pool. With a vault, the canister is moved by a crane within a concrete shielded facility and placed in a storage tube within concrete shielding.

Radioactive materials in spent nuclear fuel require two levels of confinement for dry storage. These are usually the cladding material and the metal container (or transfer canister) within the metal cask or concrete structure (cask or vault). Leaking fuel elements can be dry stored provided they are placed within a separate metal container (i.e., "can") within the canister. This is relatively easy to accomplish, but can consume additional storage space. For foreign research reactor spent nuclear fuel, the impacts of canned fuel upon storage capacity should be minimal. The amount of canning expected for foreign research reactor spent nuclear fuel is not yet determined.

F.1.1.1.1 Aboveground Free-Standing Metal Casks

Metal storage casks are generally robust and some may even have been originally designed to meet transportation requirements. They are resistant to seismic loads, high winds, design basis tornado missiles, and accidental drops. The mechanism for heat removal is simple, using direct metal conduction to the external surface which is cooled by natural convection. They are not subject to air pathway blockage by snow, ice, or flooding. The shielding is accomplished by various means, primarily thick steel, lead, or cast iron wall sections. The dry metal casks are passive, requiring minimal surveillance. There are no high-temperature thermal limits on cask material; however, if the material is cast iron or ferritic steel, there may be low-temperature thermal limits to prevent brittle fracture. Brittle fracture is a phenomenon that occurs in some materials such as glass at normal temperatures, or in cast iron or some steels (ferritic) at low temperatures. Fracture requires a stress to initiate. Thermal limits always apply to the fuel cladding. This type of dry storage has a proven track record in the United States and overseas.

The disadvantages of the metal cask designs are the following. Frequently, metal cask designs are more expensive than concrete/metal hybrid designs or dry vault storage designs. The current metal cask designs use dual compressible "O" rings with a pressure gauge to monitor the confinement seal. "O" rings are gaskets which, when compressed, form a gas-tight seal. Seal leakage is a possible event which must be considered for this design. The metal cask may be very heavy, thus imposing a limiting factor for cranes at existing facilities.

F.1.1.1.2 Aboveground Free-Standing Concrete Casks

The advantages of concrete casks, as compared to all other storage technologies, are given below. Concrete cask systems are inexpensive relative to metal casks. The concrete casks require no active systems because they are totally passive. They consist of a welded cylindrical container or basket enclosing the spent nuclear fuel which is then placed inside either a vertical or horizontal concrete structure. The concrete shielding structure may be fabricated onsite. This type of dry storage has been utilized at commercial nuclear power plant facilities, for example: H.B. Robinson, Oconee, Calvert Cliffs, and Palisades. Concrete casks have also been licensed for use at the Brunswick plant. Many other utilities are already committed to taking this route for the interim storage of their commercial spent nuclear fuel.

The disadvantages for concrete cask systems are: (1) more surveillance is needed than with metal casks to verify no blockage of air passages, (2) they are not licensed for transportation over public roads, (3) they require a special purpose onsite shielded transportation cask, and (4) the long-term concrete temperature limit restricts the heat load of the spent fuel. However, for the foreign research reactor spent nuclear fuel, heat loads and fuel cladding temperature limits are a small fraction of the commercial spent nuclear fuel values. Therefore, high concrete temperatures are expected to be avoided.

F.1.1.1.3 Aboveground Free-Standing Dry Storage Building (Vault)

Vault storage consists of a large concrete aboveground building enclosing a vertical or horizontal array of spent nuclear fuel storage metal tubes and support systems, The advantages for the vault type of dry storage, as compared to all other storage technologies, are the following. For large quantities of spent nuclear fuel assemblies, the vault may have economic advantages when compared with either type of cask system. The heat removal is passive. The heat removal capacity for a properly designed vault is large, and therefore, there should be little concern for thermal limits being imposed (although there may be individual fuel decay heat limits). The vault which is licensed in the United States and abroad, has no high temperature limit associated with concrete. However, there is a low temperature limit because the secondary fuel confinement barrier is ferritic steel. To comply with current NRC 10 CFR 72 regulations, all spent nuclear fuel storage systems must have two confinement barriers. The intact fuel cladding is considered the first confinement barrier, and the cask or vessel is considered the secondary confinement barrier. The vault has a major advantage over all other types of dry storage because it provides a shielded means for loading the spent nuclear fuel on the vault premises. Another important advantage of the vault is the ease of spent nuclear fuel retrieval and monitoring while in storage. The vault includes facilities for inspection, placement in containers, and drying of wet fuel. The weight/volume of stored fuel is not a limiting factor. This type of system is currently in use at Fort St. Vrain in Colorado, at Wylfa, Wales, and is under construction at the PAKs nuclear power plant in Hungary.

The disadvantage is that, for small quantities of spent nuclear fuel, the cost may be higher than either the metal or concrete cask systems since a vault requires greater capital outlays.

F.1.1.1.4 Inground Lined and Unlined Wells With or Without Casks

The RISO National Laboratory's inground concrete block design relies on forced air convection heat transfer from the existing handling bay ventilation system, which includes High Efficiency Particulate Air filters and an air humidity monitoring system. Forced air heat removal is accomplished by directing the air around spent nuclear fuel containers and out through tubes embedded in the concrete. Like the pool storage systems, the RISO National Laboratory's system relies on active heat removal systems.

F.1.1.1.5 Hot Cell Facilities

Although hot cells are available at many facilities, including the Savannah River Site, the Idaho National Engineering Laboratory, and the Nevada Test Site, they can best be considered for small quantities of spent nuclear fuel for very short periods of time. Hot cells are basically set up to perform various operations on hazardous materials, and are generally not spacious enough to store materials on an indefinite or long-term basis. Furthermore, hot cells are frequently contaminated; this contamination may pose problems when it is time to transfer the foreign research reactor spent nuclear fuel to the repository. It is important to note that DOE possesses several unique hot cells that may be capable of foreign research reactor spent nuclear fuel storage due to their large, vault-like design.

F.1.1.1.6 Aboveground Free-Standing Multi-Purpose or Dual-Purpose Casks

The dual-purpose cask combines the functions of interim storage at a designated site and transportation on public roads, rail systems, or waterways. A multi-purpose cask may also add a third function of a repository canister; i.e., the cask and its contents need no further processing, characterization or identification in order to be compatible with the final repository. A dual-purpose or multi-purpose cask has attractive possibilities for the storage of spent nuclear fuel, regardless of the type of reactor (i.e., commercial or research reactor). Dual-purpose designs would satisfy the two functions of storage and

transportation. For commercial utilities, this implies satisfaction of 10 CFR 71 requirements for transportation and 10 CFR 72 requirements for storage. By minimizing fuel handling operations, the dose for workers can be reduced, and the number of additional low-level waste products can be reduced. Minimization of fuel handling may also result in cost reductions, although this case has not been made. For a multi-purpose cask, satisfaction of 10 CFR 60 requirements is also necessary. DOE's Office of Civilian Radioactive Waste Management was actively pursuing a program to develop multi-purpose canister for domestic use (EG&G, 1994b; DOE, 1994f; DOE, 1994b; DOE, 1994c). However, DOE has decided in November 1995 to withdraw its proposal to prepare the EIS for this project.

The dual-purpose cask systems that are currently proposed offer a reduction in handling of the spent nuclear fuel in the storage to transportation operations, and multi-purpose cask systems offer an even greater reduction in handling in the operations involved at the repository site. However, no detailed cost/benefit analyses have been undertaken for foreign research reactor spent nuclear fuel. Furthermore, there is no basis at this time for concluding that either the "waste form" (intact spent nuclear fuel assemblies) or the sealed container will be compatible with the repository requirements. It is premature to draw any conclusion on the desirability to proceed with a multi-purpose cask system for foreign research reactor spent nuclear fuel use.

F.1.1.2 Specific Dry Storage Designs

There are no currently licensed dry storage systems specifically for foreign research reactor spent nuclear fuel in the United States. There are, however, many examples of dry spent nuclear fuel systems licensed by the NRC for commercial fuel. Table F-2 provides an overview of current manufacturers of dry storage systems. Table F-3 is a listing of dry storage systems currently licensed in the United States.

Dry storage systems must meet many design criteria, such as protection of fuel from degradation, shielding, thermal, criticality safety, structural integrity of confinement vessel, structural integrity of shielding, mechanical handling of fuel assemblies or canisters, containment and operational aspects. Some of these criteria are interrelated. For example, thermal criteria are designed to maintain fuel and cladding structural integrity. Shielding, thermal, and criticality parameters are the most important and are discussed in the following sections.

F.1.1.2.1 Shielding Design Comparisons

A spent nuclear fuel storage system must provide for adequate shielding of both the gamma and neutron radiation that emanate from irradiated nuclear fuel. The shielding must be designed to reduce the combined gamma and neutron dose rate to values that are below the limits for the public at the site boundary, collocated workers, and workers at the fuel storage facility. These limits are determined by Federal regulations such as 10 CFR 72 and 10 CFR 20. Shielding is designed for the maximum expected gamma and neutron source term, which is determined by performing computer code analyses of the nuclear fuel that account for the initial fuel fissile material inventory, its burnup in the reactor core, and the time after removal from the reactor (i.e., decay time) prior to its anticipated placement in the storage facility. The selection of a bounding and conservative set of these parameters results in the calculation of the highest possible gamma and neutron source term to be used in shielding design and analyses.

Shielding for gamma radiation relies on the use of high atomic weight or density materials, which attenuate and absorb gamma rays. The material selection depends on design limitations regarding shield thickness, cost, strength, and weight. The five materials which are almost always used in spent nuclear fuel storage facilities for gamma shielding are water, lead, steel, ductile iron or concrete. Lead and steel, having much higher densities and atomic weights than concrete and water, can provide relatively more

Table F-2 Dry Storage Technology Systems

Company	*Metal Cask*	*Concrete Cask*	*Building (Vault)*
AECL/Transnuclear	---	SILO	MACSTOR/CANSTOR
FW/GEC	---	---	MDV
GNS/GNSI	CASTOR	---	---
GNSI/OH	---	HDC	---
Nuclear Assurance Corp.	NAC	---	---
VECTRA	---	NUHOMS	---
Sierra Nuclear Corporation	---	VSC	---
RISO National Laboratory	---	DR3	---
Siemens Power Corporation	---	---	FUELSTOR
Transnuclear, Inc.	TN	---	---
Westinghouse Electric Corporation	MC-10	---	---
Atomic Energy of Canada, Ltd.	---	SILO	CANSTOR
Total Design:	**4**	**6**	**4**

FUELSTOR = Fuel Encapsulation and Lag Storage; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); GNSI = General Nuclear Systems, Inc.; GNSI/OH = GNSI of Ontario Hydro; VSC = Ventilated Storage Cask; MDV = Modular Dry Vault

Table F-3 Dry Storage Systems Currently Licensed in the United States

Manufacturer	*System*	*Location*
General Nuclear Systems, Inc. (Chem-Nuclear)	CASTOR V/21	Surry
VECTRA	NUHOMS-7P, NUHOMS-24P, & NUHOMS-52B	Robinson, Oconee, & Calvert Cliffs
Westinghouse Electric Corporation	MC-10	Surry
Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom)	MDV	Wylfa, Wales UK PAKS (Hungary) Fort St. Vrain (USA)
Transnuclear	TN-24 & TN-40	Surry, Prairie Island
Sierra Nuclear Corporation	Ventilated Storage Cask-24	Palisades
Nuclear Assurance Corporation	NAC-C28, NAC-I28, NAC S/T, & NAC S/TC	Surry

effective gamma shielding with a smaller thickness of material. However, using steel and/or lead imposes a design penalty of increased cost. Concrete and water are much less expensive and may reduce overall shielding costs. It should also be noted that lead can be categorized as a Resource Conservation and Recovery Act waste, which restricts future decommissioning and disposal options. Concrete and water may also present unique safety problems, such as leakage (for water) or cracking (for concrete) during postulated accidents.

Neutron shielding requires low atomic weight materials because the uncharged neutron can only be absorbed by reducing its energy in collisions with nuclei similar in mass. Since the mass of a neutron is approximately one atomic mass unit, low atomic mass elements such as hydrogen, inert gas, lithium, carbon, and boron are suitable shields. Hydrogenous materials such as concrete and water are typically used in neutron shielding. It should be noted, however, that a sufficient thickness of heavier materials (such as high carbon steels) can provide neutron shielding. Also, shielding manufacturers offer products that have been artificially fortified in their hydrogen content, such as special forms of concrete, borated

resin, and polyethylene. As in the case for gamma shielding, design factors in material selection include cost, density, weight, and safety.

The design of spent nuclear fuel storage facility shielding must also incorporate other factors along with cost, density, weight, and safety. Shielding usually performs a second function as a heat transfer medium from the spent nuclear fuel decay heat to the environment, and must therefore be able to effectively remove heat without exceeding fuel and shielding storage temperature limits. In some instances, the shielding also performs a structural function, either in handling or support.

Table F-4 shows a comparison of specific designs with a view toward shielding considerations. All of these designs will be discussed in more detail in subsequent sections of this appendix.

F.1.1.2.2 Thermal Design Comparisons

Spent nuclear fuel storage facilities are designed to effectively remove spent nuclear fuel decay heat during both incident-free operation and postulated accident conditions. Thermal design limits include long-term fuel storage cladding temperature to maintain cladding integrity and, in some cases, temperature limits of structural and/or shielding materials. Unlike pool storage systems, most of the dry storage systems emphasize passive heat removal. In contrast, active systems in wet pools include pumps, make-up water systems, filtration and water treatment systems, and heat exchangers.

All dry storage designs encapsulate the fuel, after it is dried, in a metal canister or tube that is evacuated (vacuum dried) and then filled with an inert gas such as helium. Helium is frequently used for its relatively high thermal conductivity that enhances heat conduction and heat transfer from the fuel to the encapsulating metal canister. Helium's inert properties also inhibit cladding corrosion. Since all the dry fuel storage technologies utilize a metal canister to enclose spent nuclear fuel, the first modes of heat transfer from the fuel to this canister's walls are heat conduction and radiation from the fuel cladding surface through the inert gas to the inside wall of the metal canister. Decay heat transfer from the encapsulating canister to the environment is accomplished by several different mechanisms dependent upon the specific storage design technology.

The dry metal cask design relies on its solid thick metal wall for conduction heat transfer from the fuel storage cavity to the atmosphere. Metal cask conduction heat transfer is not susceptible to any accident or degradation. This thermal design is inherently easy to analyze because conduction is a well-known heat transfer mechanism, and the thermal conductivity of such metal cask materials as carbon steel, stainless steel, and ductile cast iron is well known over the range of temperatures and conditions that are expected in the cask while storing spent nuclear fuel. With known design fuel decay heat, cask geometry (i.e., cask wall thickness), conduction material composition, and suitably conservative heat transfer assumptions from the cask metal surface to the ambient air, the temperature distribution within the cask and maximum fuel cladding temperature can be calculated with a high degree of certainty.

The dry concrete cask design uses a combination of conduction, natural convection, and radiation heat transfer to remove decay heat from the stored spent nuclear fuel and maintain acceptable operating temperatures. An air passageway around the storage canister is provided in this design because the relatively low thermal conductivity and allowable operating temperature limit of concrete, as compared to metal, prevent the concrete shield walls from serving as the primary means of decay heat removal. Radiation streaming requires that the inlet and outlet air passages to the cavity surrounding the canister be designed as a geometric labyrinth with suitable bends. One concrete cask design, the Atomic Energy of Canada, Ltd. SILO, does not have air passages but instead relies solely on conduction through solid

Table F-4 Comparison of Shield Design Parameters for Spent Nuclear Fuel Dry Storage Systems Currently Licensed in the United States

Manufacturer	*Model*	*Shield Material*	*Shield Thickness*	*Design Limit Surface Dose Rate*[a]
Nuclear Assurance Corporation	S/T	S.S., Lead, NS4FR	Radial: 20.3 cm (8 in) S.S. & 17.8 cm (7 in) NS4FR Axial: 12.7 cm (5 in) S.S.-PB & 7.6 cm (3 in) NS4FR	1 milliSievert/hr (100 mrem/hr)
Transnuclear, Inc.	TN-24	Borated Resin, C.S.	NA	Side: 0.57 milliSievert/hr (57 mrem/hr) Top: 0.11 milliSievert/hr (11 mrem/hr) Bottom: 0.45 milliSievert/hr (45 mrem/hr)
	TN-40	Borated Resin, C.S.	Radial: 21.6 cm (8.5 in) C.S. 11.4 cm (4.5 in) Resin Bottom: 22.2 cm (8.75 in) C.S. Top: 15.9 cm (6.25 in) Cast Iron	Side: 0.58 milliSievert/hr (58 mrem/hr) Top: 0.26 milliSievert/hr (26 mrem/hr) Bottom: 12.75 milliSievert/hr (1,275 mrem/hr)
Westinghouse Electric Corporation	MC-10	NS-3, C.S.	Radial: 25.4 cm (10 in) Steel, 7.6 cm (3 in) NS-3 Bottom: 25.4 cm (10 in) steel	2 milliSievert/hr (200 mrem/hr)
General Nuclear Systems, Inc.	CASTOR V21	Cast Iron, S.S., Polyethylene Rods	Radial: 30.5 cm (12 in) Bottom: 27.9 cm (11 in) Top: 39.1 cm (15.4 in) Rods Radial: 72-6.1 cm (2.4 in) Diameter	2 milliSievert/hr (200 mrem/hr)
VECTRA	NUHOMS 7P , 24P, and 52B	Concrete, S.S.	Side: 45.7/60.1 cm (18/24 in) Rear: 60.1 cm (24 in) Roof: 91.4 cm (36 in)	2 milliSievert/hr (200 mrem/hr) (at air inlet)
Sierra Nuclear Corporation	Ventilated Storage Cask-24	Concrete RX-277, Hydrogenated Concrete, C.S.	Radial: Steel & Concrete Top: RX-277 & Steel Bottom: Steel & Concrete	Side: 0.20 milliSievert/hr (20 mrem/hr) Top: 0.50 milliSievert/hr (50 mrem/hr) Air Inlet or Outlet: 1 milliSievert/hr (100 mrem/hr)
FW/GEC	MDV	Concrete	106.7 cm (42 in)	0.21 milliSievert/hr (21 mrem/hr)

NA = Not Available; C.S. = Carbon Steel; Pb = Lead; S.S. = Stainless Steel; NS-3 = Concrete; NS4FR = Special Fire-Resistant Castable Resin; RX-277 = Special Concrete with Extra Hydrogen; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); MDV = Modular Dry Vault

[a] *These are limits established for commercial spent nuclear fuel assemblies. The dose rate expected from storage of foreign research reactor spent nuclear fuel is likely to be lower.*

concrete. The SILO's thermal design is acceptable only because it is limited to a much smaller total decay heat power than the air passage concrete casks.

Since a passive design is an underlying requirement of all dry concrete cask designs, the total airflow path from the cask air inlet to its outlet must include a sufficient elevation change to ensure natural convection airflow under all expected meteorological and heat load conditions.

The heat transfer from the canister follows two parallel paths: (1) convection from the surface of the canister to the naturally-induced airflow through the canister cavity, and (2) radiation and conduction heat transfer from the canister across the air in the cavity to the concrete shield and then conduction through the

concrete shield wall thickness to the ambient air outside the concrete. Natural convection air heat removal is greater than the radiation and conduction through the air layer and concrete shield.

The heat transfer design of the concrete cask is vulnerable to accidents in which significant blockage of the air inlets and/or outlets restricts or prevents sufficient airflow into the canister cavity. Multiple inlets and outlets at different, and sometimes diametrically opposed, locations around the cask are used to reduce the likelihood of such an accident. Conservative adiabatic heatup analyses for these designs with commercial spent nuclear fuel have shown that temperature limits are not approached in more than 24 hours, even if the airflow inlets and outlets are completely blocked. Therefore, concrete cask sites have included a daily visual surveillance frequency for inspection of air inlets and outlets to ensure that they are not blocked. The adiabatic heatup for foreign research reactor spent nuclear fuel and its concomitant surveillance frequency may be different.

The concrete cask thermal design also requires more complex analyses for temperature distribution in both the fuel and the concrete due to the complex multidimensional and combined conduction-radiation-convection modes of heat transfer. An important thermal design issue for the concrete casks is proof that natural convection buoyancy-driven airflow will be induced through the inlet-cavity-outlet path under the entire range of expected wind and decay heat conditions, including the possibility of partial blockage that may be obscured from outside visual inspections. Unlike metal casks, which only have the fuel cladding temperature as a thermal limit, concrete casks are also limited by both the absolute magnitude and gradients of temperature within the concrete.

The concrete vault storage building represents a larger version of the concrete cask design in the realm of heat transfer. An array of vertically or horizontally oriented metal canisters enclosing spent nuclear fuel is surrounded by a concrete building with labyrinth air inlet and outlet passages. With the exception of size, this design utilizes the same modes of heat transfer as the concrete cask. Its inherently larger flow areas for inlets and outlets and typically larger elevation from inlet to outlet provide a greater natural convection airflow and reduce vulnerability to airflow passage blockage.

Specific Thermal Features

Thermal design performance parameters of specific manufacturers' dry storage technologies are presented in Table F-5. This table shows that all dry spent nuclear fuel storage technologies use radiation and conduction as heat transfer mechanisms, and that concrete-based systems also rely on internal air passage natural convection heat transfer. All the systems have fuel cladding temperature limits, but systems relying on concrete also have concrete temperature limits.

Pool storage systems utilize an active cooling system with pumps and heat exchangers that remove decay heat transferred to the pool water from stored fuel via conduction and natural convection. The relatively large mass and heat capacity of the pool water provide a significant margin of time before the pool water reaches its boiling temperature in the event of a cooling system failure.

The RISO National Laboratory's inground concrete block design relies on forced air convection heat transfer from the existing handling bay ventilation system, which includes High Efficiency Particulate Air filters and an air humidity monitoring system. Forced-air heat removal is accomplished by directing the heating, ventilation, and air conditioning air around the stored fuel, and then out through separate tubes embedded in the concrete. Like the pool, the RISO National Laboratory's system relies on active heat removal systems.

The Atomic Energy of Canada, Ltd., SILO is an exception to the previously discussed concrete cask designs because it relies solely on conduction through a solid concrete structure for decay heat removal,

Table F-5 Comparison of Thermal Design Parameters for Spent Nuclear Fuel Dry Storage Systems Currently Licensed in the United States

Manufacturer	*Model*	*Design Heat Load (KW)*	*Thermal Limits[a] (°C)*	*Heat Transfer Mode(s)[b]*
Nuclear Assurance Corp.	S/T	26	NA	Conduction, Radiation
	S/TC	22	NA	Conduction, Radiation
Transnuclear, Inc.	TN 24	24	149 - Resin	Conduction, Radiation
	TN 40		NA	Conduction, Radiation
Westinghouse Electric Corp.	MC-10	13.5	340 - LWR Cladding	Conduction, Radiation
GNSI	CASTOR V21	21	370 - LWR Cladding	Conduction, Radiation
VECTRA	NUHOMS 7P and 24P	24	340 - Fuel Clad Normal 570 - Fuel Clad Accident	Conduction, Radiation, Natural Convection
	Standardized 24P and 52B	24 and 19	225 - Concrete Accident 177 - Concrete Normal 578 - Fuel Clad Accident 378 - Fuel Clad Normal	Conduction, Radiation, Natural Convection
Sierra Nuclear Corporation	Ventilated Storage Cask-24	24	93 - Concrete Normal 177 - Concrete Accident 570 - Fuel Clad Accident 378 - Fuel Clad Normal	Conduction, Radiation, Natural Convection
FW/GEC	Modular Dry Vault	0.15 per HTGR[c] Canister	399 - for Fort St. Vrain Type Fuel	Conduction, Radiation, Natural Convection

NA = Not Available; LWR = Light Water Reactor; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); GNSI = General Nuclear Systems, Inc.

[a] *Fuel Limits are for Commercial Light Water Reactor Zircaloy Clad Fuel Type or for HTGR fuel.*

[b] *Heat transfer modes are for commercial spent nuclear fuel.*

[c] *HTGR= High Temperature Gas Reactor Type Fuel from Fort St. Vrain.*

without internal natural convection airflow around the canister. The SILO can maintain acceptable concrete and fuel cladding temperatures without internal airflow passages by limiting its contained total fuel heat load to about 4 kilowatts, as compared to the 24 kilowatts typical of other concrete casks with airflow passages. This lower heat load may not be limiting for the storage of foreign research reactor spent nuclear fuel since it does not produce decay heat as high as that for commercial nuclear power plant fuel for the same decay time.

F.1.1.2.3 Criticality Prevention Design of Spent Nuclear Fuel Storage Technology

A self-sustaining nuclear fission process is called criticality. Unlike the previously discussed thermal and shielding designs, criticality prevention design for spent nuclear fuel storage facilities does not rely on materials outside of the fuel storage basket or canister. Instead, the canister interior fuel support structure and fuel specifications for storage are the determining factors in criticality control.

Spent nuclear fuel storage facilities are shown to meet specific regulatory subcriticality requirements by conservative criticality analyses. These analyses conservatively assume that the spent nuclear fuel has its original enrichment of fissile material [e.g., fresh unirradiated fuel weight percent uranium-235 (^{235}U)]. In reality, the fuel has been irradiated and the initial concentration of fissile material is reduced from its original value through fission reactions producing numerous fission products.

Suitable criteria for establishing nuclear criticality safety have been documented (ANSI, 1984b, 1983, and 1975b). These documents deal specifically with, respectively, the storage of commercial spent nuclear fuel outside of the reactor and in dry storage installations.

Another conservative aspect of these criticality analyses is the requirement that a sensitivity study be performed that varies the water concentration within the canister free volume from no water to 100 percent water, to optimize moderation density. These analyses usually show that the most reactive (i.e., closest to critical conditions) configuration occurs with a water density less than that equivalent to a fully flooded canister (related to enrichment).

The criticality analyses explicitly model the fuel geometry, all materials present in the fuel, and the structural spacer design within the canister. Center-to-center distance for the fuel in the canister is another important parameter in determining the reactivity of the stored fuel.

In summary, the criticality prevention design of spent nuclear fuel storage facilities ensures that each canister will remain subcritical throughout the entire operation, during both incident-free and accident conditions. The criticality prevention design incorporates the following features:

- fuel specifications, including type of fuel, maximum initial fresh fuel ^{235}U enrichment, and number of fuel assemblies to be stored in a single canister,
- fuel assembly spacing inside the canister as set by the presence of structural support and spacing members, and
- the presence and composition of any neutron absorbing material between adjacent fuel assemblies inside the canister.

F.1.1.2.4 Current NRC-Licensed, Dry Storage Technologies for Commercial Spent Nuclear Fuel

The technologies discussed in this section are described in terms of their use for storage of commercial spent nuclear fuel. Foreign research reactor spent nuclear fuel storage design parameters will be different for each technology.

F.1.1.2.4.1 Nuclear Assurance Corporation S/T, NAC-C28 S/T, NAC-I28

Description of Nuclear Assurance Corporation S/T, NAC-C28 S/T and NAC-I28

Two of the Nuclear Assurance Corporation metal casks for the storage of spent nuclear fuel are in use at the Surry Nuclear Power Plant in Virginia. The Nuclear Assurance Corporation S/T design uses a combination of stainless steel and lead for gamma shielding and NS4FR, which is a fire-resistant castable resin, for neutron shielding (NRC, 1988a). To ensure a surface contact dose rate of less than 100 mrem/hr, 20.3 cm (8 in) of stainless steel and lead and 17.8 cm (7 in) of NS4FR are used in the cylindrical wall, while the top and bottom shields are composed of 7.6 cm (3 in) of NS4FR and about 12 cm (5 in) of steel and lead. Total weight of the loaded cask is either 91 metric tons (100 tons) for 26 intact Pressurized Water Reactor fuel assemblies, or 112 metric tons (124 tons), which is just under the 125-ton limit of many loading cranes for the 56 consolidated fuel assembly model. The Nuclear Assurance Corporation S/T models have been licensed by the NRC. The Nuclear Assurance Corporation S/T is shown in Figure F-2.

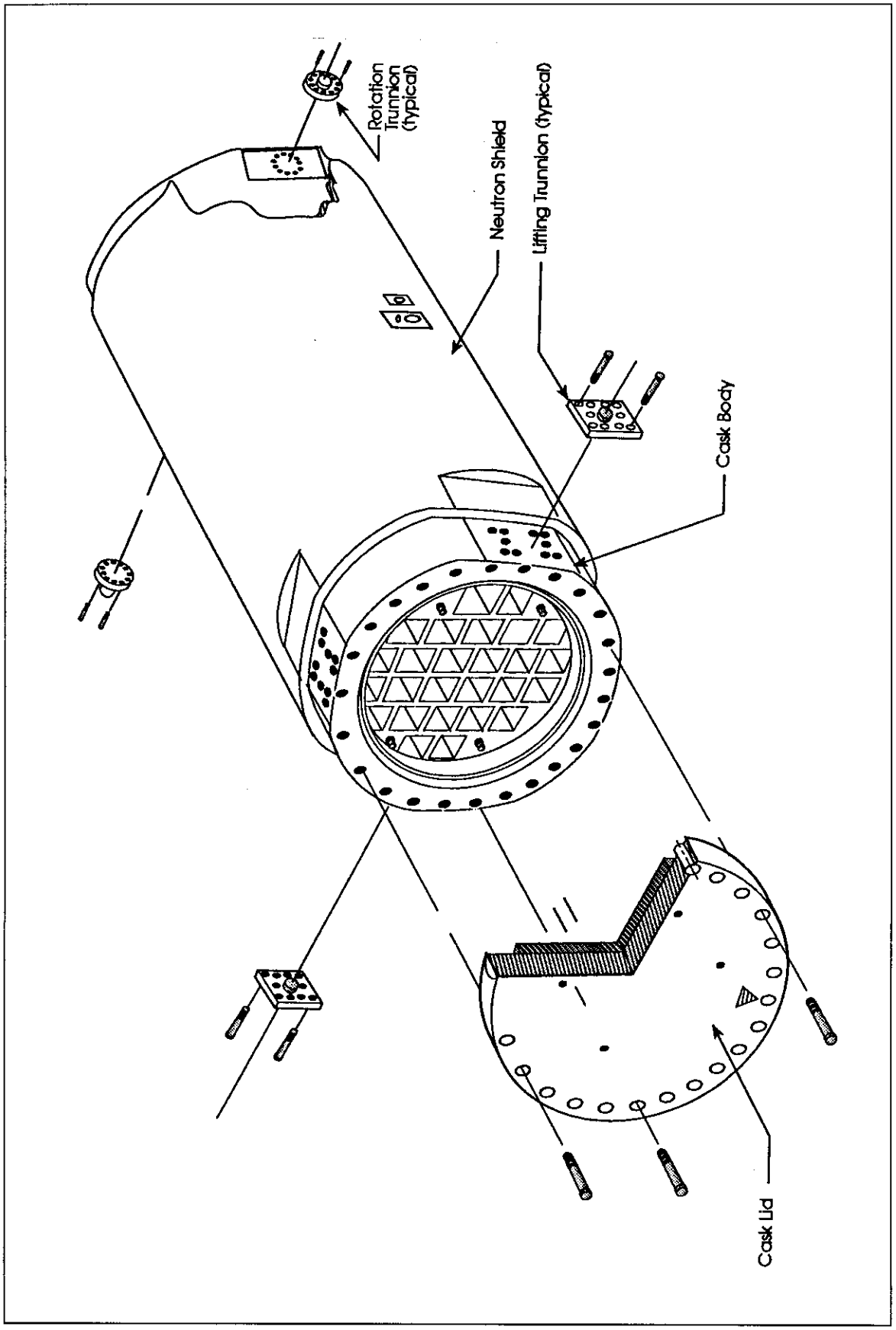


Figure F-2 Nuclear Assurance Corporation Spent Nuclear Fuel Storage Cask Body NAC S/T

NRC Certification or Basis for License

The NRC has granted a Certificate of Compliance to Model Nuclear Assurance Corporation S/T (Certificate Number 1002). Nuclear Assurance Corporation Model NAC-C28 S/T is also certified with Certificate Number 1003. The basis for these certificates is 10 CFR 72 Subparts K and L. Nuclear Assurance Corporation model NAC-I28 is currently licensed on a site-specific basis at Surry Nuclear Power Plant based on 10 CFR 72 Subparts A through I.

F.1.1.2.4.2 General Nuclear Systems, Inc. CASTOR V/21

Description of General Nuclear Systems, Inc. CASTOR V/21

In the United States, the General Nuclear Systems, Inc. CASTOR V/21 has been approved by the NRC and is in use at the Surry Nuclear Power Plant. This design relies on thick ductile cast iron and polyethylene as both its gamma and neutron shields. Ductile cast iron contains significant quantities of nodular graphite, which is essentially carbon, a good neutron shield. Polyethylene is a form of plastic that is high in hydrogen. The ductile cast iron shield is 30.5 cm (12 in) thick. Additional neutron shielding is provided by seventy-two 6.1 cm (2.4 in) diameter polyethylene rods placed in axial holes in the cast iron wall. The top lid shielding is 39.1 cm (15.4 in) of stainless steel, and the bottom lid shielding is 27.9 cm (11 in) of ductile cast iron. The V/21, holding 21 Pressurized Water Reactor fuel assemblies at Surry, weighs 96 metric tons (106 tons) fully loaded. A sketch of the CASTOR V/21 is presented in Figure F-3. The shielding design basis is for a surface contact dose rate less than 200 mrem/hr. There is a wide range of CASTOR designs for a variety of fuel types, including test reactor fuel. A conceptual design [CASTOR Material Test Reactor (MTR) 2] for a dual-purpose, transport/storage cask for research reactor fuel has been developed. This cask uses the same basic ductile cast iron body for shielding.

NRC Certification or Basis for License

The NRC has granted Certificate of Compliance Number 1000 for the General Nuclear Systems, Inc. model CASTOR V/21 under the terms of 10 CFR 72 Subparts L and K (Models X/28 and X/33 are not currently licensed, but are being reviewed by the NRC).

F.1.1.2.4.3 Westinghouse Electric Corporation MC-10

Description of Westinghouse Electric Corporation MC-10

The Westinghouse Electric Corporation MC-10 metal cask has been approved by the NRC and is in use at the Surry Nuclear Power Plant site (NRC, 1987). This cask design utilizes thick carbon steel and BISCO NS-3 hydrogenated concrete for shielding. The NS-3 provides neutron shielding, while the carbon steel is used for gamma shielding. Total radial neutron and gamma shielding is approximately 33 cm (13 in), while axial shielding is about 25.4 cm (10 in). The design surface contact dose rate is 200 mrem/hr, which bounds the actual vendor-calculated maximum surface contact dose rates of 7, 38, and 57 mrem/hr at the top, side, and bottom of the cask. The MC-10 was designed to hold 24 Pressurized Water Reactor fuel assemblies and weighs 103 metric tons (113.3 tons) fully loaded.

NRC Certification or Basis for License

The NRC has issued Certificate of Compliance Number 1001 for the metal cask model MC-10 in accordance with the terms of 10 CFR 72 Subparts L and K.

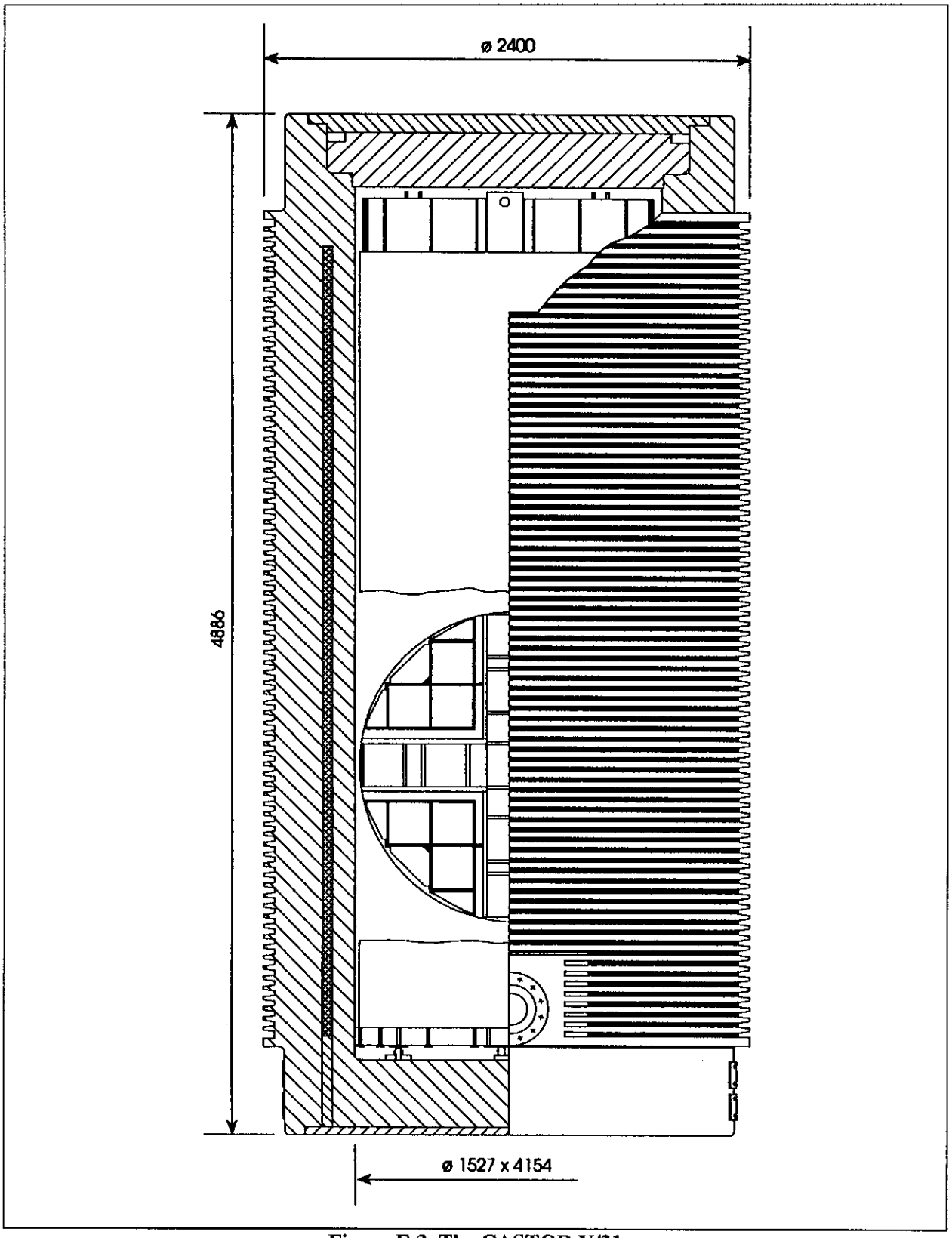


Figure F-3 The CASTOR V/21

F.1.1.2.4.4 Transnuclear, Inc. TN-24 and TN-40

Description of Transnuclear, Inc. TN-24, TN-40

The Transnuclear, Inc. design has been developed and produced for a large number of storage and transportation systems for radioactive materials, including spent nuclear fuel. The TN-24 and TN-40 models store 24 and 40 spent Pressurized Water Reactor fuel assemblies, respectively. TN systems feature metal casks for both transportation and storage of spent fuel. The TN-24 is an NRC-licensed storage cask that uses carbon steel for gamma shielding and a borated resin for neutron shielding (NRC, 1989). The TN-40 is a newer model that uses a two-metal shell design, with the inner shell consisting of high quality carbon steel for containment and the outer shell providing shielding and heat transfer, but of a lower quality steel. For the two models, top and side contact dose rate limits are less than 100 mrem per hour, but the bottom of the cask may have a contact dose rate limit as high as 1,275 mrem/hr. It should be noted that the normal configuration for these casks is to be standing upright on their bottoms, thereby precluding exposure to this relatively higher dose rate. A sketch of a Transnuclear, Inc. TN cask is shown in Figure F-4.

NRC Certification or Basis for License

The TN-24 model has been issued NRC Certificate of Compliance Number 1005 and is licensed according to 10 CFR 72 Subparts L and K. The TN-40 model is licensed on a site-specific basis at the Prairie Island Nuclear Power Plant in Minnesota (owned by Northern States Power) under the provisions of 10 CFR 72 Subparts A through I. The Transnuclear, Inc. Model TN-32 is not yet approved.

F.1.1.2.4.5 VECTRA Design NUHOMS-7P, -24P, and -52B

Description of VECTRA NUHOMS-7P, -24P, and -52B

VECTRA's NUHOMS designs utilize a horizontal concrete dry storage system for spent nuclear fuel (NUTECH, 1988). The NUHOMS-7P and NUHOMS-24P designs have been approved by the NRC for Pressurized Water Reactor spent nuclear fuel and are in use at the Robinson, Oconee, and Calvert Cliffs Nuclear Power Plant sites. The NRC approved the use of NUHOMS-52B for the Brunswick power plant, but the utility shipped this spent nuclear fuel to its Robinson plant. The NUHOMS design uses concrete as both gamma and neutron shielding. The requirement for internal air passages to allow natural convection heat removal from the metal storage canister placed within the concrete structure required 90 degree bends in the concrete shield for air passages to avoid radiation streaming and more detailed shielding analyses. The reinforced side wall concrete shield thickness is 45.7 or 61 cm (18 or 24 in) depending on location in the array, while the rear wall is 61 cm (24 in) thick and the roof is 91.4 cm (36 in) thick. The maximum surface contact dose rate limit at the air inlet is 200 mrem/hr. A sketch of the NUHOMS-24P system is shown in Figure F-5.

NRC Certification or Basis for License

NUHOMS models 7P and 24P are licensed at specific sites under the provisions of 10 CFR 72. Vectra has also received a license from the NRC for their standardized NUHOMS-24P and -52B models for use by the light water reactor utilities.

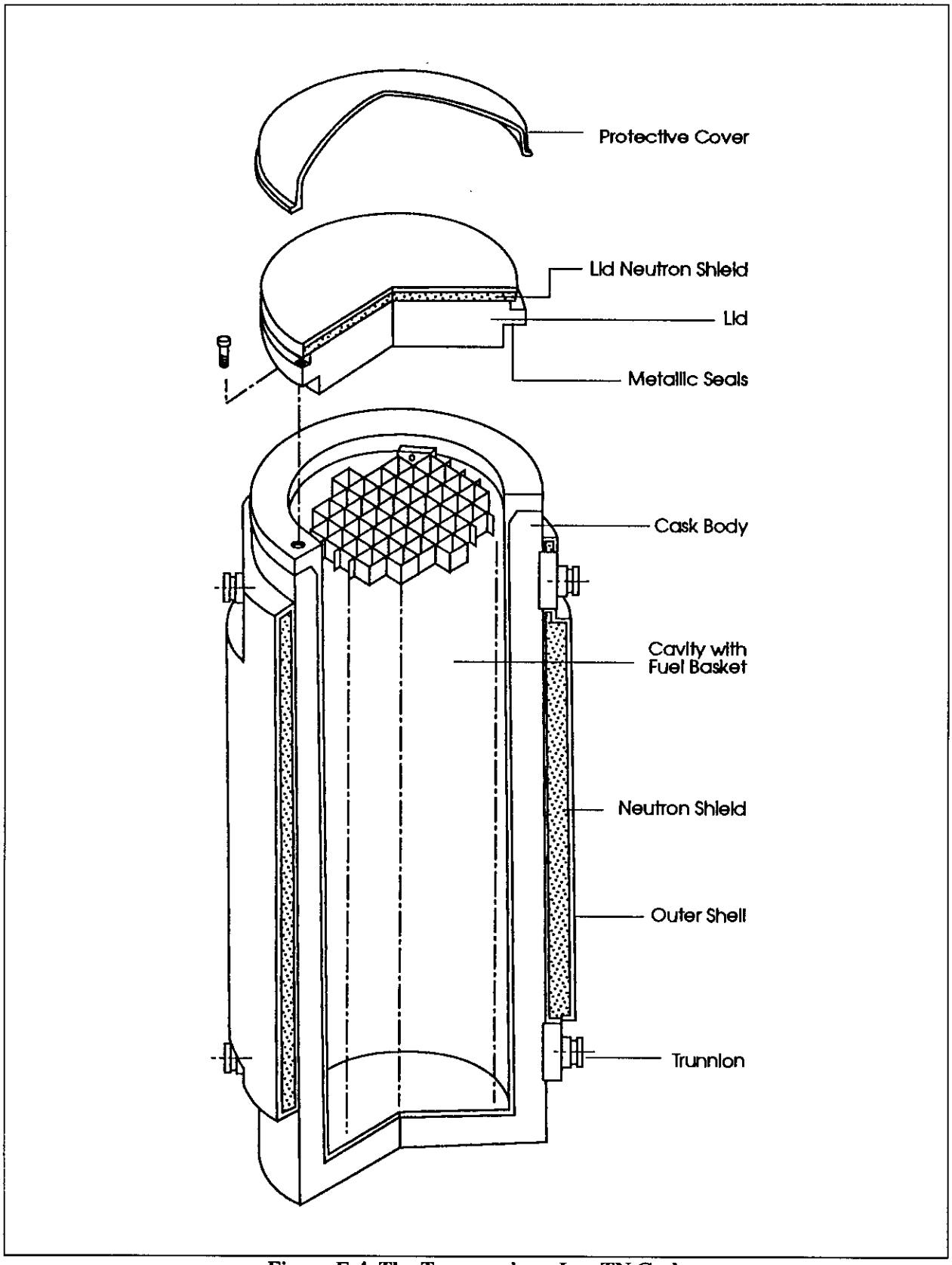


Figure F-4 The Transnuclear, Inc. TN Cask

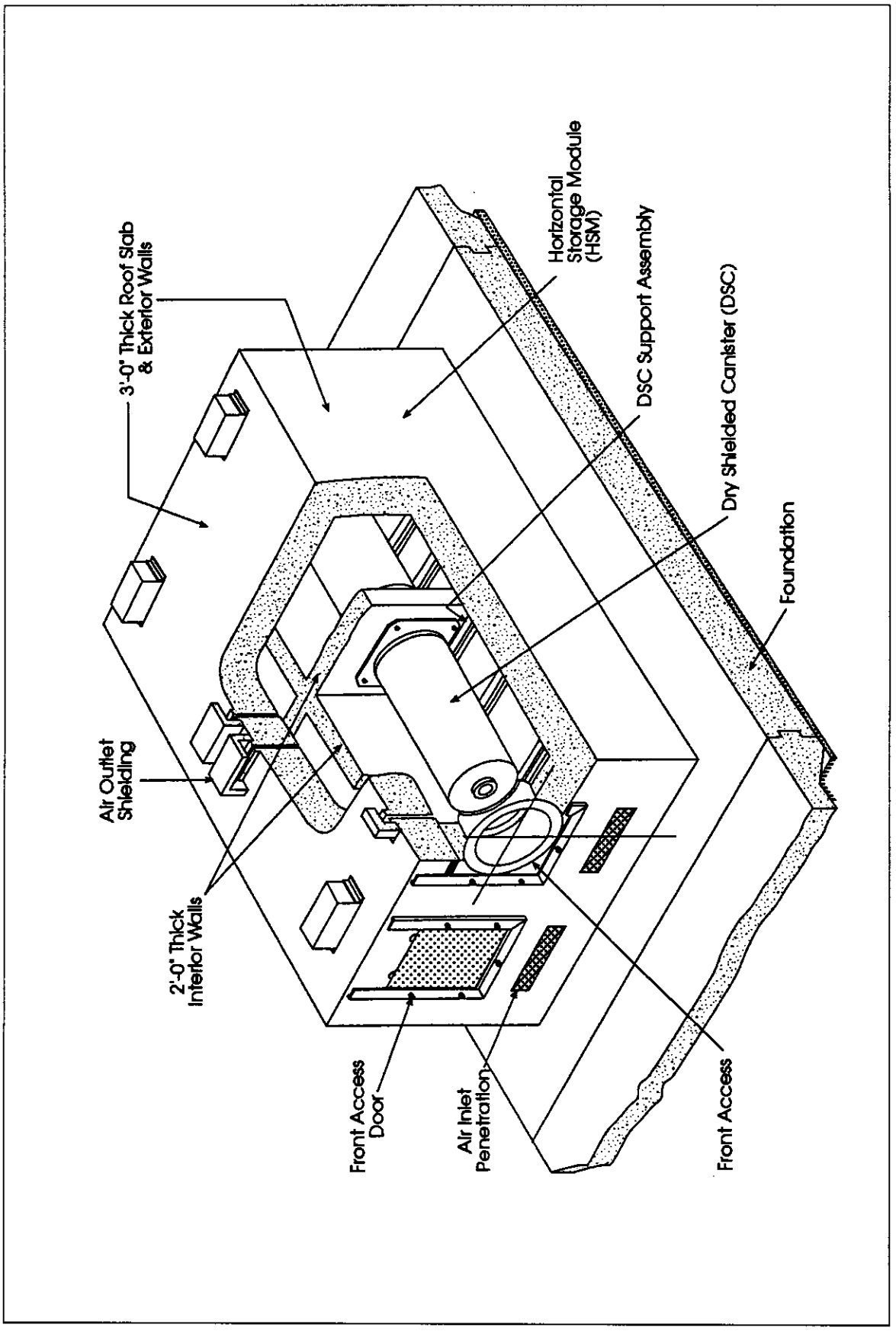


Figure F-5 The NUHOMS-24P Horizontal Storage Module Components

F.1.1.2.4.6 Modular Dry Vault

Description of Modular Dry Vault

The modular dry vault spent nuclear fuel storage system [designed by Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom)] is the only vault system in the United States that has been approved by the NRC and is in operation at the Fort St. Vrain nuclear power plant site. The modular dry vault places spent nuclear fuel in vertically oriented cylindrical steel fuel storage containers which are then inserted into a steel charge face structure within the thick concrete structure. A labyrinth airflow passage system provides natural convection airflow for decay heat removal. The shielding is provided by the 106.7 cm (42 in) thick concrete walls and the labyrinth airflow passages. For the Fort St. Vrain fuel, maximum design modular dry vault surface dose rate is 21 mrem/hr. A picture of the cross section of the modular dry vault is shown in Figure F-6.

NRC Certification or Basis for License

The modular dry vault model has been approved by the NRC for the site-specific application at Fort St. Vrain. The basis for the license is 10 CFR 72.

F.1.1.2.4.7 Ventilated Storage Cask System (VSC-24)

Description of VSC-24

The Ventilated Storage Cask, designed by Sierra Nuclear Corporation, is a vertical concrete cask design that has been approved by the NRC and is in use at the Palisades nuclear power plant site. As with the NUHOMS design, this system relies on concrete for both neutron and gamma shielding and incorporates internal airflow passages requiring detailed shielding analyses to demonstrate acceptable streaming doses. The Ventilated Storage Cask design dose rates are 20 mrem/hr side contact and 50 mrem/hr top contact. A sketch of the Ventilated Storage Cask is shown in Figure F-7.

NRC Certification or Basis for License

The Sierra Nuclear Corporation's Model VSC-24 has been granted Certificate of Compliance Number 1004 by the NRC. The basis for this certificate is 10 CFR 72 Subparts L and K.

F.1.1.2.5 Manufacturers of Commercial Nuclear Fuel Dry Storage Systems Not Currently Licensed by the NRC in the United States

In addition to the above examples of dry cask storage systems licensed in the United States, there are other systems either licensed outside the United States or in the design and/or licensing stage (Table F-6). Tables F-7 and F-8 show shielding and thermal related parameters of the various dry cask models that are not currently licensed in the United States.

F.1.1.2.5.1 Description of MACSTOR

The MACSTOR system (designed by Atomic Energy of Canada, Ltd. and Transnuclear, Inc.), representing a synthesis of both metal and concrete casks in a modular dry vault, is being reviewed for use in Canada (AECLT, 1994). Spent nuclear fuel is placed in 0.95 cm (0.375 in) thick carbon steel canisters or baskets that are then placed (in a vertical position) in concrete modules. Air labyrinth passages into and

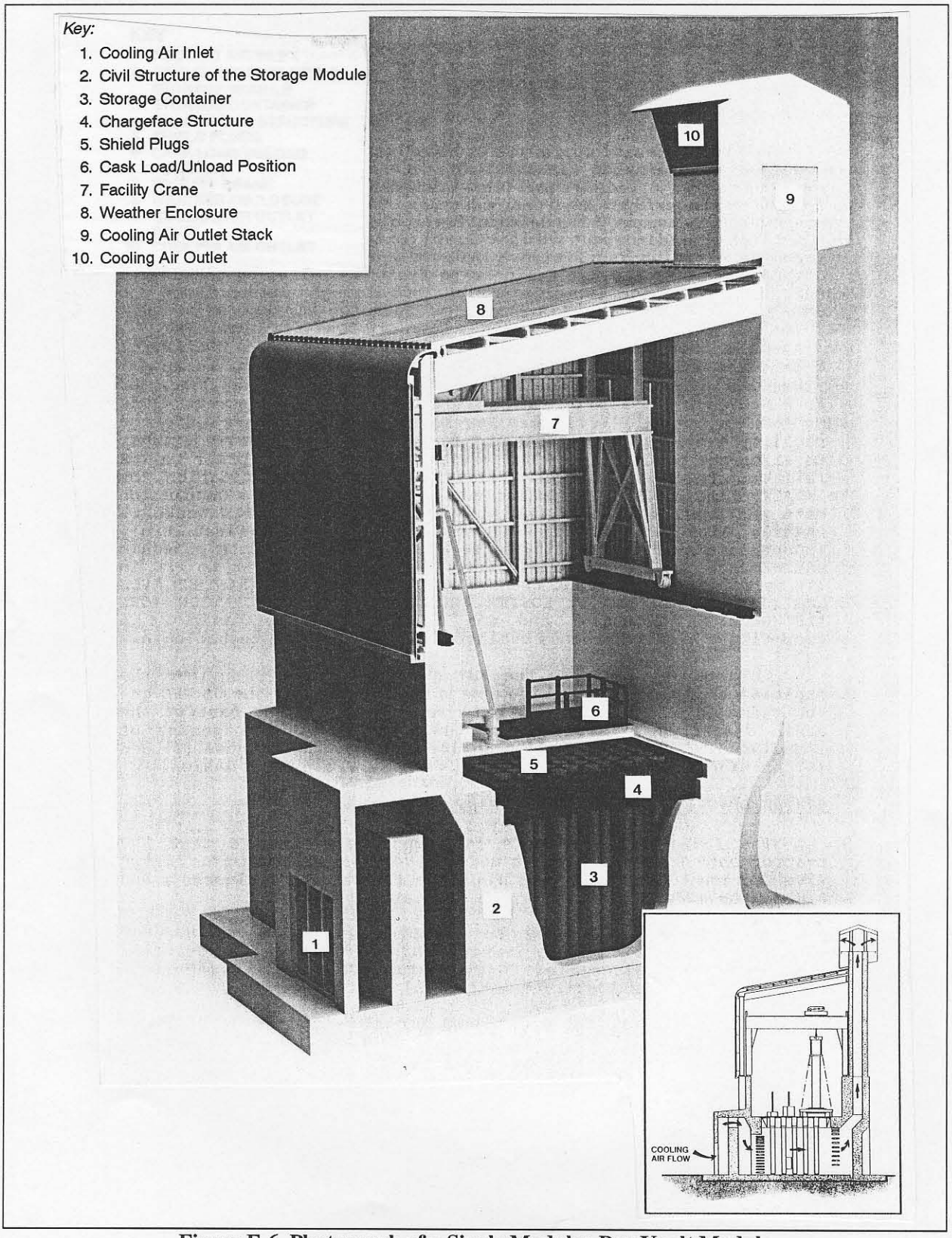


Figure F-6 Photograph of a Single Modular Dry Vault Module

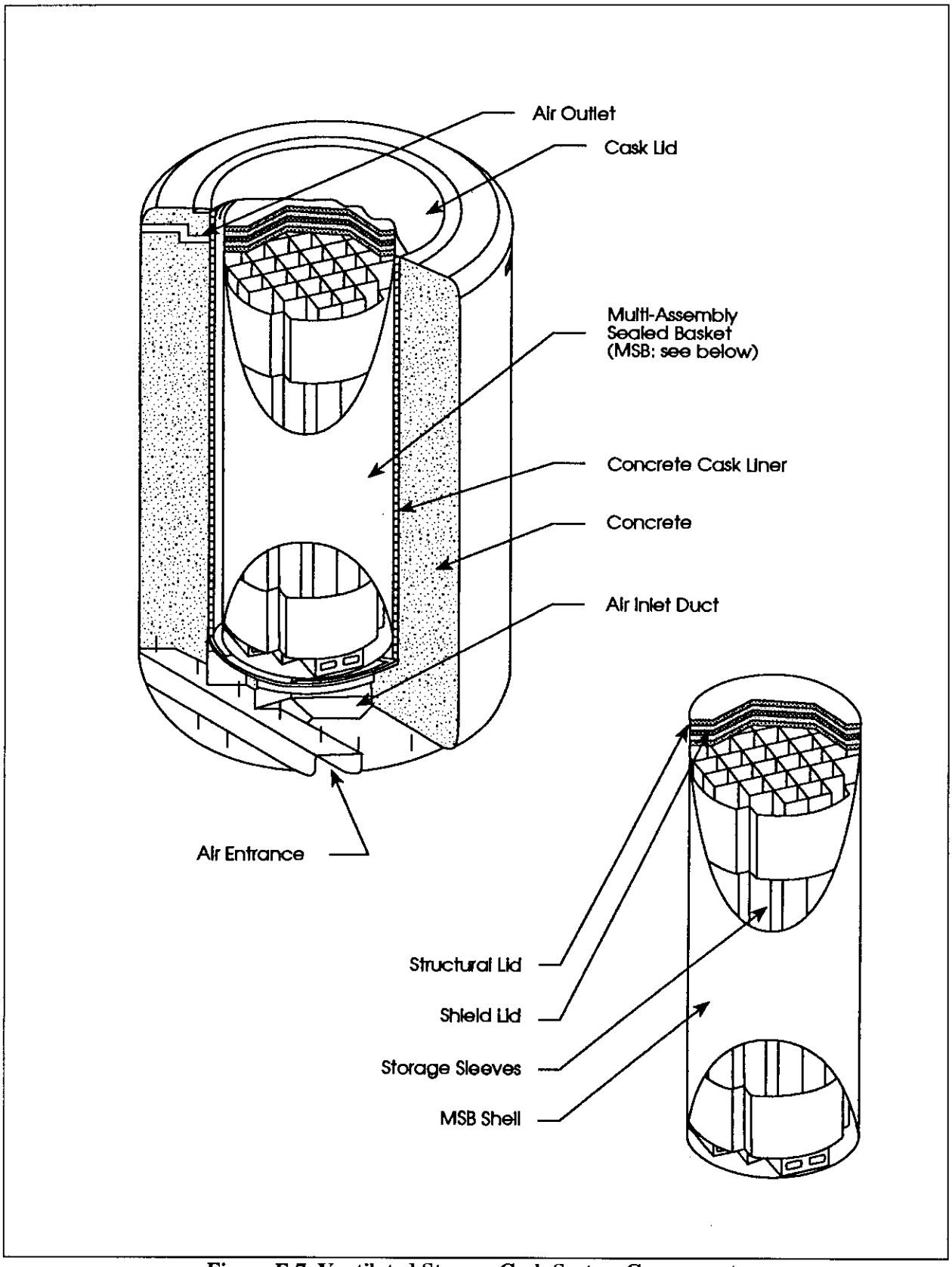


Figure F-7 Ventilated Storage Cask System Components

Table F-6 Manufacturers of Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	*Facility or Model*	*Location or Status*
General Nuclear Systems, Inc. of Ontario Hydro	HDC	Canada
Atomic Energy of Canada, Ltd./Transnuclear Inc.	MACSTOR	Canada
Siemens Power Corporation	FUELSTOR	Germany
RISO National Laboratory	DR3	Denmark
Atomic Energy of Canada, Ltd.	SILO, Canister	Canada
Nuclear Assurance Corporation	NAC-26	10 CFR 72 License Pending
General Nuclear Systems, Inc.	X28 and X33	10 CFR 72 License Pending
Transnuclear, Inc.	TN-32	10 CFR 72 License Pending

FUELSTOR = Fuel Encapsulation and Lag Storage

Table F-7 Comparison of Shield Design Parameters for Spent Nuclear Fuel Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	*Model*	*Shield Material*	*Shield Thickness in cm (in)*	*Design Limit Surface Dose Rate milliSievert/hr (mrem/hr)*
GNSI/OH	HDC-36	High Density Carbon Steel, Concrete	2.5 (1) CS, 45.7 (18) High-Density Concrete	NA
Siemens Power Corporation	FUELSTOR	Concrete	304.8 (120)	0.00001 (0.001)
Transnuclear/Atomic Energy of Canada Ltd.	MACSTOR	Concrete	96.5 (38)	0.025 (2.5)
RISO National Laboratory	DR3	Carbon Steel, Concrete, Earth	76.2 (30) Carbon Steel, Axial 15.2 (6) Concrete, Radial	NA
Atomic Energy of Canada Ltd.	SILO	Concrete	91.4 (36) Concrete	0.025 (2.5)
Atomic Energy of Canada Ltd.	CANSTOR	Concrete	NA	NA
Transnuclear	TN-32	Borated Resin Carbon Steel	Radial: 21.6 (8.5) 11.4 (4.5) Resin Bottom: 22.2 (8.75) Carbon Steel Top: 15.9 (6.25) Cast Iron	Side: 0.86 (86) Top: 0.18 (18) Bottom: 3.15 (315)
General Nuclear Systems, Inc.	CASTOR X28 and X33	Cast Iron, Stainless Steel Polyethylene Rods	Radial: 30.5 (12) Bottom: 27.9 (11) Top: 39.1 (15.4) Rods Radial: 72- 6.1 (2.4) Dia.	2 (200)

NA = Not Available; FUELSTOR = Fuel Encapsulation and Lag Storage ; GNSI/OH = General Nuclear Systems, Inc. of Ontario Hydro; Transnuclear = Transnuclear, Inc.

out of the module provide a flow path for natural convection airflow to remove decay heat. The vault concrete walls are 96.5 cm (38 in) thick and designed to reduce dose rates to less than 2.5 mrem/hr on contact. This concrete thickness is maintained even where the airflow passage labyrinth is located. A cross section of the MACSTOR is given in Figure F-8.

F.1.1.2.5.2 Description of a Fuel Encapsulation and Lag Storage Facility

The Fuel Encapsulation and Lag Storage system is similar to the modular dry vault and MACSTOR/CANSTOR in that it is a stand-alone concrete building with interior steel storage containers. Unlike the modular dry vault and CANSTOR/MACSTOR, the Fuel Encapsulation and Lag Storage system

Table F-8 Comparison of Thermal Design Parameters for Spent Nuclear Fuel Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	*Model*	*Design Heat Load (KW)*	*Thermal Limits (°C)*	*Heat Transfer Mode(s)*
GNSI/OH	HDC	---	---	Conduction, Radiation
Siemens Power Corporation	FUELSTOR	Up to 2 KW Per Canister	380/2 KW 250-1KW-Clad	Conduction, Radiation, Natural Convection
Transnuclear Inc./Atomic Energy of Canada Ltd.	MACSTOR CANSTOR	240 (20 Canisters at 12 each)	93 Concrete 340 LWR Clad	Conduction, Radiation
RISO National Laboratory	DR3	---	NA	Conduction, Radiation, Forced Convection
Atomic Energy of Canada Ltd.	SILO	4	NA	Convection
Transnuclear Inc.	TN-32	---	NA	Conduction, Radiation
General Nuclear Systems, Inc.	CASTOR X28 and X33	19.2 and 20.9	370 LWR Cladding	Conduction, Radiation

NA = Not Available; LWR = Light Water Reactor; FUELSTOR = Fuel Encapsulation and Lag Storage; GNSI/OH = General Nuclear Systems, Inc. of Ontario Hydro

stores spent nuclear fuel containers in a horizontal position. The Fuel Encapsulation and Lag Storage system is designed for a surface contact dose rate of 0.001 mrem/hr, and relies on its combined 304.8 cm (120 in) thick inner and outer shield concrete walls and labyrinth airflow passages for shielding. The Fuel Encapsulation and Lag Storage system is not licensed by the NRC or in use in the United States A cross section of the Fuel Encapsulation and Lag Storage system is shown in Figure F-9.

F.1.1.2.5.3 Description of a RISO National Laboratory Facility

In Denmark, the RISO National Laboratory has designed and constructed a dry storage facility at its DR3 PLUTO type research reactor to store MTR spent nuclear fuel from this reactor. This facility was installed under the floor of the active handling bay at the reactor and consists of four prefabricated octagonal concrete blocks placed in a vertical position into steel lined holes in the earth. Each block contains 12 storage holes in a triangular mesh, with a carbon steel form forming each hole and a separate stainless steel tube containing the spent nuclear fuel. Axial shielding is provided by a 76.2 cm (30 in) thick carbon steel plug. Radial shielding is provided by the surrounding earth and concrete of the octagonal block with the minimum concrete thickness of 15.2 cm (6 in). A sketch of the RISO National Laboratory's design is shown in Figure F-10.

F.1.1.2.5.4 Description of SILO

The SILO has been designed and licensed in Canada, and over 180 concrete SILOs have been built for the storage of Canadian research reactor and CANDU-commercial reactor spent fuel (AECLT, 1994). The SILO consists of a carbon steel-lined cylindrical hole inside a 91.4 cm (36 in) thick vertical concrete cylinder without any labyrinth airflow passage for heat removal. Carbon or stainless steel canisters containing the spent nuclear fuel are placed inside the SILO and stacked up to nine high before being covered by a steel and concrete plug. The surface SILO dose rate limit is 2.5 mrem/hr. The unique design aspect of the SILO is that it is the only concrete cask without airflow passages for natural convection heat removal. It has been used for short-length low decay heat fuel which is dimensionally similar to foreign

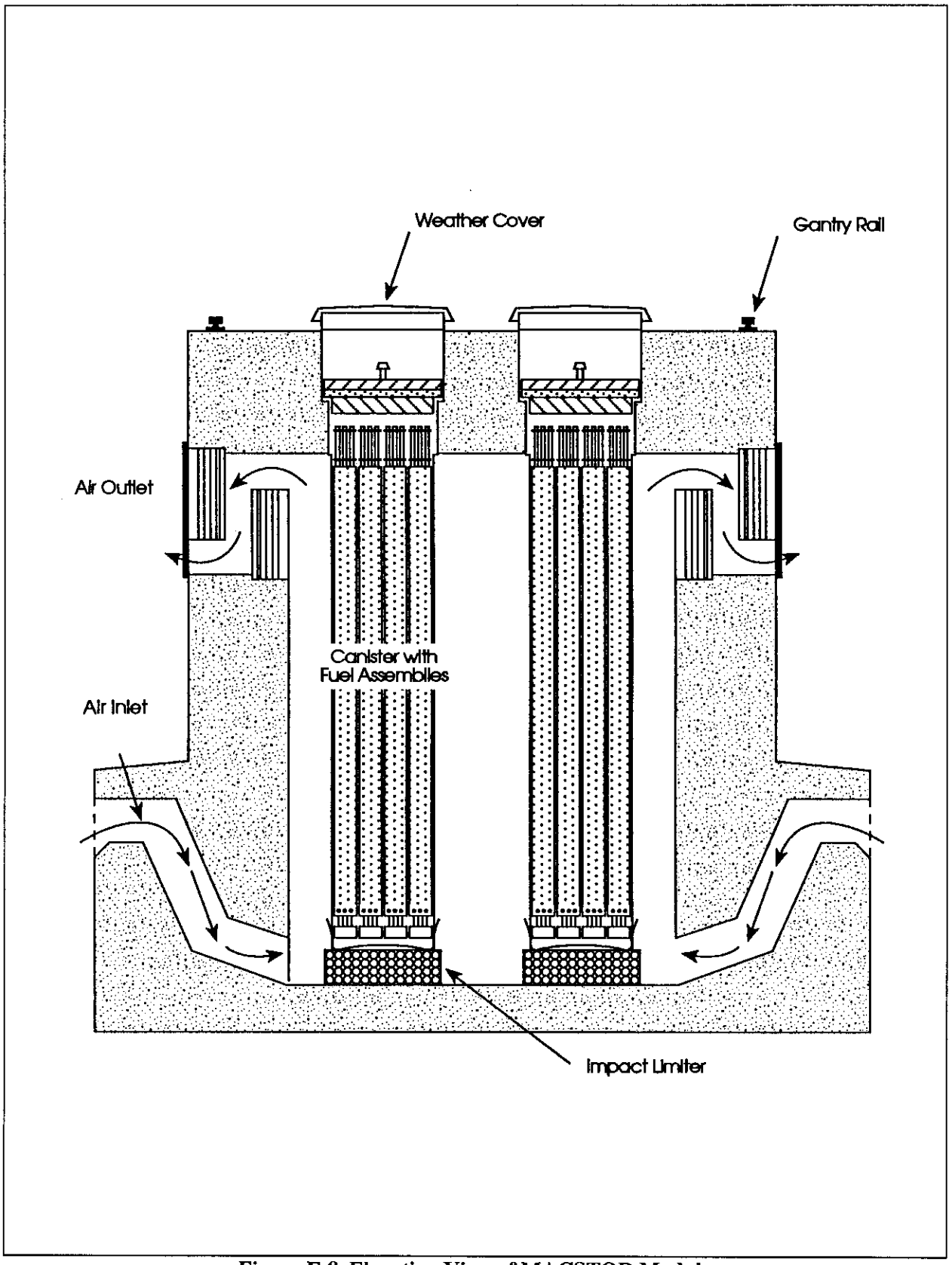


Figure F-8 Elevation View of MACSTOR Module

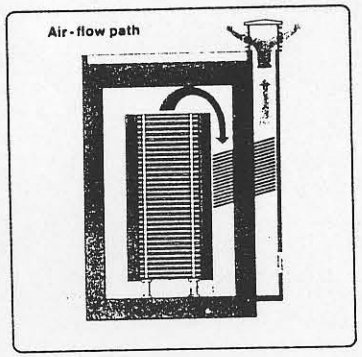
FUELSTOR

FUEL ENCAPSULATION AND LAG STORAGE FACILITY

For at-reactor dry storage of spent fuel

Main technical features:

- Closed-cycle vault resulting in triple barrier system
- Passive cooling by natural convection ensuring low fuel temperatures
- Low initial storage temperature resulting in no restrictions regarding the length of the storage period
- Sealed canister storage
- Horizontal placement of helium-filled canisters in storage vault
- Reinforced concrete construction for safe shielding resulting in radiation levels outside the building below permissible dose of unrestricted area (ALARA)
- Reinforced concrete construction to withstand the effects of natural phenomena and man-induced events such as earthquakes, tornado missiles and aircraft crashes
- Nuclear criticality safety is ensured by the building design as well as spent fuel canister arrangement
- Store is easily expandable by modules
- Low land consumption due to compact spent fuel storage
- Nuclear safeguard provisions



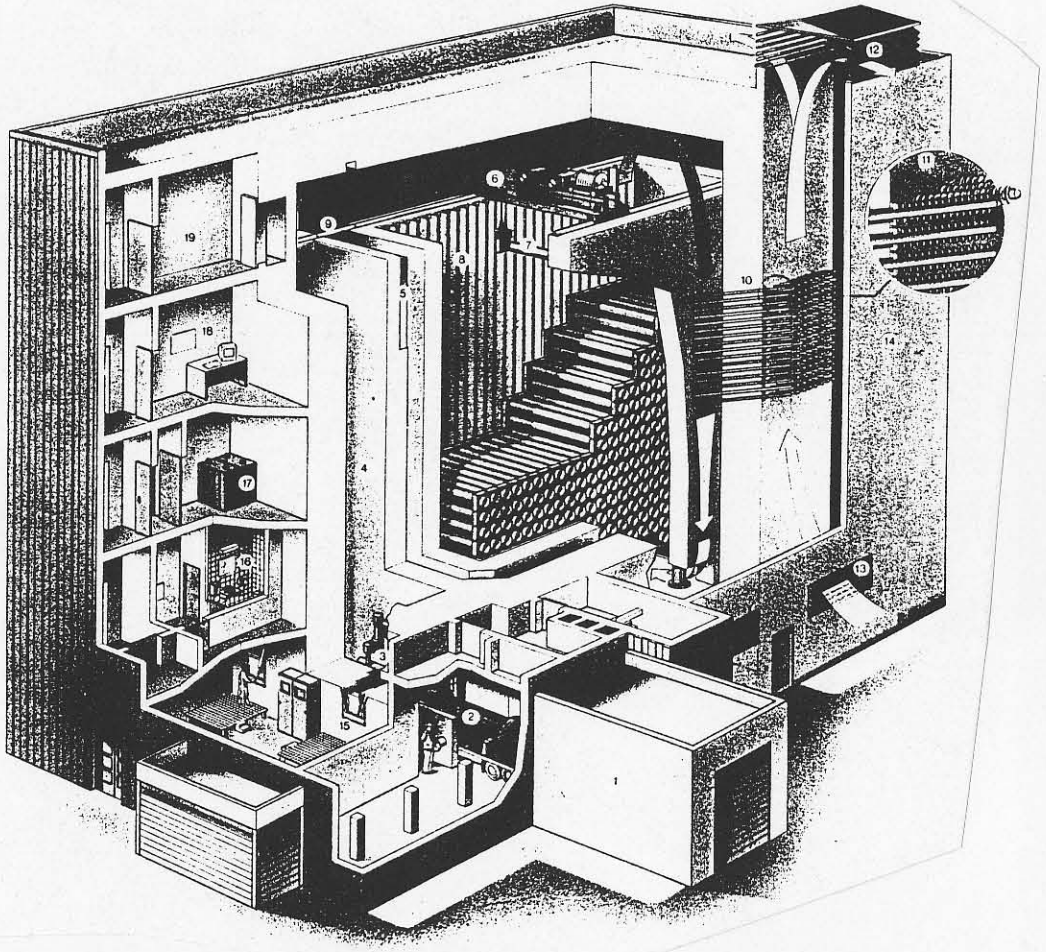


Figure F-9 Fuel Encapsulation and Lag Storage System Cross Section

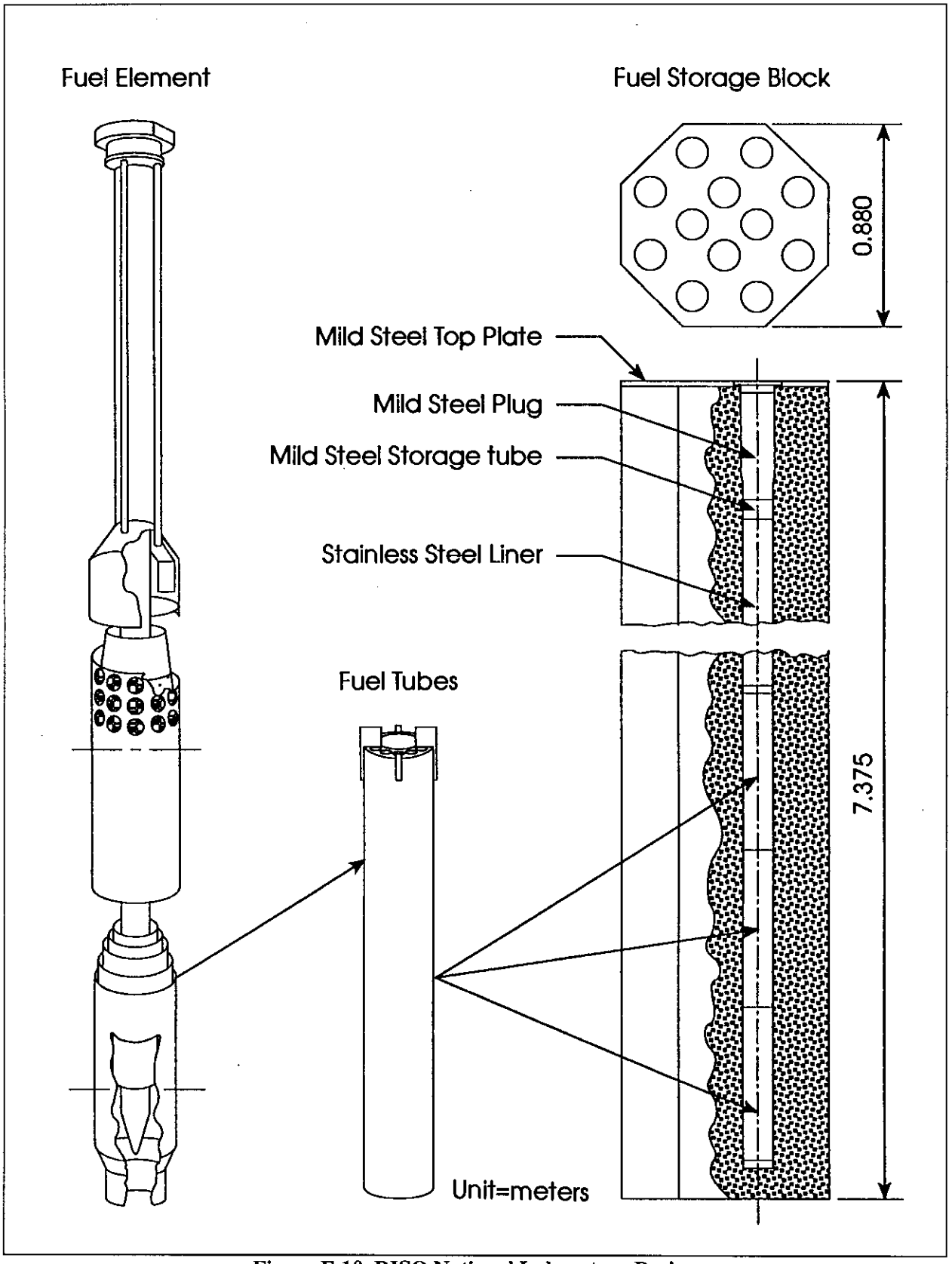


Figure F-10 RISO National Laboratory Design

research reactor spent nuclear fuel. The SILO has been licensed in Canada and is currently undergoing license approval in South Korea, which uses the same regulations as the NRC. A sketch of the SILO is given in Figure F-11.

F.1.1.2.5.5 Dual-Purpose Cask and Canister Systems

Dual-purpose designs must satisfy NRC requirements for both storage and transportation (10 CFR Parts 72 and 71, respectively). It is believed that such dual-purpose designs would reduce incident-free handling of individual spent nuclear fuel assemblies, reduce the volume of low-level radioactive waste that would otherwise be generated from using a single-purpose cask system (one cask for storage with subsequent transfer of individual assemblies to transport casks and disposal packages), and may play a role in reducing overall worker radiation exposures over a single-purpose cask system.

At the present time, there are two dual-purpose casks for light water reactor fuel use: Nuclear Assurance Corporation's Storage/Transport Cask and Vectra's dual-purpose canister system (MP-187). Nuclear Assurance Corporation's Storage/Transport Cask has received NRC approval. The NRC is expected to approve Vectra's MP-187 in the near future.

The VECTRA MP-187 is a derivative of a design approved earlier by the NRC. The MP-187 design includes a stainless steel confinement canister, a horizontal reinforced concrete module for storing the canister, and a special onsite/offsite transportation cask system that may also be used to store the canister in a vertical orientation. This system is currently being evaluated by the NRC for the Rancho Seco nuclear power plant. The applicant also has a variation for the canister design to accommodate canned spent nuclear fuel for damaged spent nuclear fuel assemblies, and which cannot be stored without a second confinement barrier.

DOE had proposed expanding the role of a dual-purpose system to that of a multi-purpose canister-based system (DOE, 1994f; DOE, 1994b; DOE, 1994c). Fuel would be loaded into a canister at the reactor site. The canister could then be placed into unique, specially designed overpacks for storage at the reactor site, transportation to a federal facility, or disposal in a repository. Final NRC approval for use of the multi-purpose canister as a component of the disposal package requires that the multi-purpose canister and its surrounding overpack meet 10 CFR Part 60 requirements. The fact that no site has been chosen yet for a repository adds an element of uncertainty to the third function: disposal. DOE has decided in November 1995 to withdraw its proposal to prepare the EIS for this canister. The Department of the Navy, however, will complete this EIS and will limit its scope to the storage and transport of Navy spent nuclear fuel.

F.1.2 Wet Storage Designs

In addition to the previous examples of dry storage technology, there are several types of wet storage systems currently in use at DOE sites and at commercial nuclear power facilities. These include aboveground pools (lined or unlined), inground pools (lined or unlined), and shutdown reactor vessels. For the purposes of this appendix, a pool refers to a canal or a basin.

Description of Wet Pool Spent Nuclear Fuel Storage Technology

The storage of spent nuclear fuel in pools (i.e., wet storage technology) has been in use for over 40 years (since the early water-cooled reactors began operating). The basic concept underlying wet storage is analogous to the development of light water-cooled nuclear reactors for defense, research, and electric power production purposes.

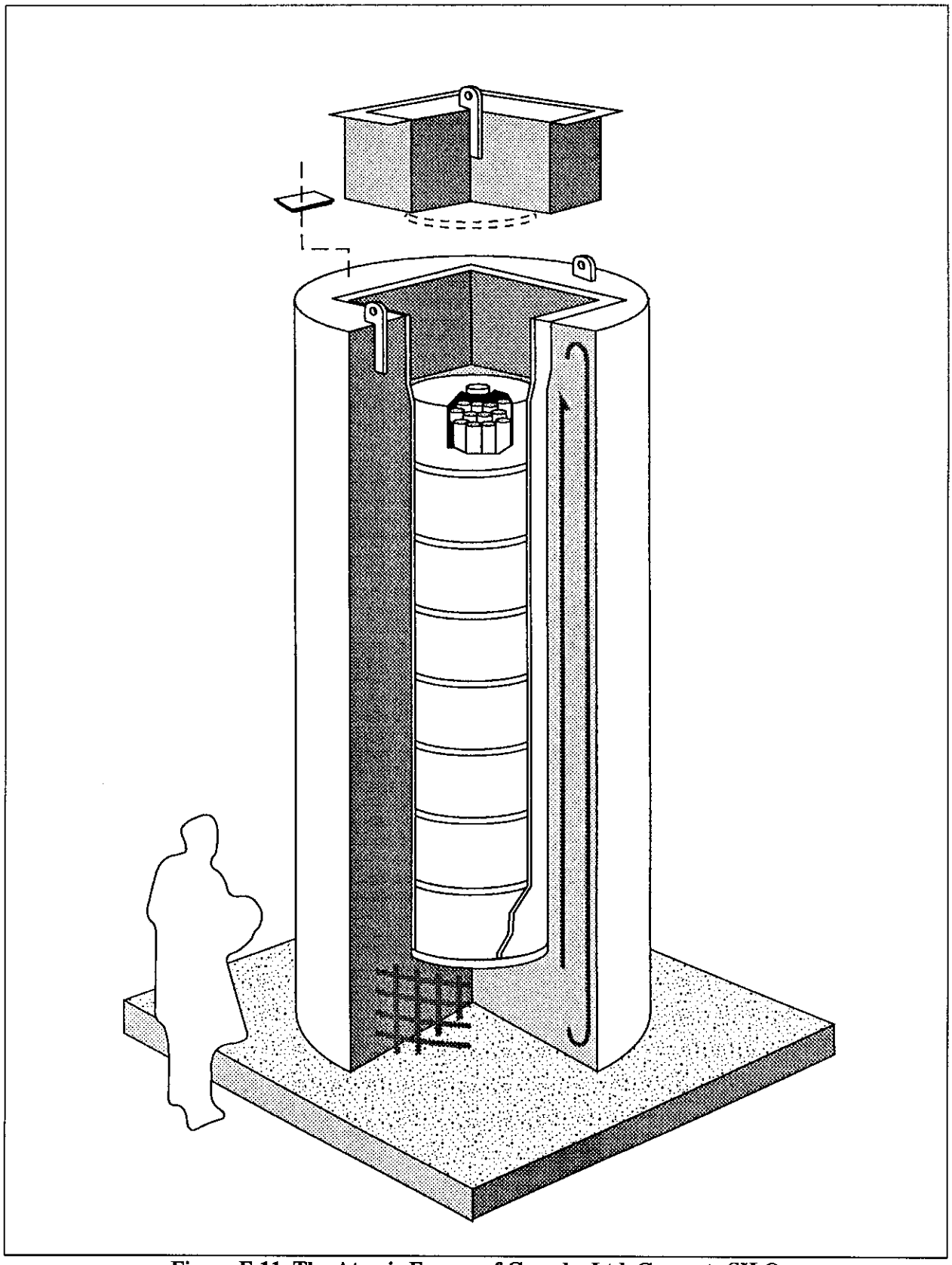


Figure F-11 The Atomic Energy of Canada, Ltd. Concrete SILO

In terms of spent nuclear fuel storage, water offers several distinct advantages, which can be summarized as:

- low cost for shielding and coolant medium,
- visual confirmation of fuel location and ease of handling,
- high heat capacity allowing for a large time period before thermal limits are exceeded,
- multi-purpose shield for both neutrons and gamma rays,
- inherent ability to retain many fission products which could leak from failed spent nuclear fuel, and
- insusceptibility to degradation from spent nuclear fuel radiation.

Water pool storage also has some shortcomings. These are:

- the need to maintain high purity water to prevent corrosion,
- the requirement for active safety systems connected to the water for heat removal, purity control, and water makeup,
- extensive lined and reinforced concrete walls for ensuring no leakage of water under all accident conditions,
- generation of radioactive waste from degraded fuel which is collected by the water purification systems, and
- groundwater monitoring to detect any leakage of radioactive pool water into the environment.

For every water-cooled reactor in the world, the decision has always been made to construct an adjoining or integral spent nuclear fuel storage pool. Currently, over 600 water cooled electric power-, research-, and defense-related reactors are operating in the world, each with its own wet storage pool for spent nuclear fuel (Nuclear Engineering International, 1993). Experience has shown that this technology is safe and effective.

At commercial nuclear power plants, the pool storage is located in a structure adjacent to a containment building that is capable of direct hydraulic connection to the reactor core through a system of canals, gates, and pools. The spent nuclear fuel pool building is designed and built to withstand all the accidents and dynamic loads required of other safety-related structures at nuclear power plants. It has its own crane and fuel handling equipment, and a separate heating, ventilation, and air conditioning system to mitigate radioactive releases to the environment. The nuclear power plant control room includes monitors and controls for the spent nuclear fuel pool. Redundant separate trains of equipment are used to fulfill the requirements of heat removal from the spent nuclear fuel pool water, removal of impurities and radioactive materials from the water, and maintenance of the water level to ensure adequate shielding above the spent nuclear fuel. At commercial power plants, such parameters as water level, water temperature, flow and temperature difference across heat exchangers used to cool the water, water purity, activity levels, and radiation dose rates are all monitored and measured.

All U.S. commercial nuclear power plant pools are stainless steel lined and use racks made of stainless steel to store spent nuclear fuel. Stainless steel is used to line the pool walls and floor to help maintain high water purity by preventing the release of chemicals from unlined concrete and to simplify decontamination at the end of the facility's life. The racks provide support and spacing for each fuel assembly, thus controlling criticality and maintaining fuel structural integrity.

Detailed criticality and thermal-hydraulic analyses are performed to demonstrate to the licensing authorities (the NRC in the United States) that fuel can never become critical, and that the assembly spacing in the racks allows for adequate cooling so as to prevent nucleate and bulk boiling in the pool or on any fuel surfaces. Shielding analyses substantiate the adequacy of the water depth above the fuel in the pool (usually at least 6.1 m or 20 ft), and the thickness of concrete pool walls and piping routing for systems connected to the pool water. This piping may contain pool water that is contaminated with radioisotopes released from spent nuclear fuel in the pool, and must be considered in dose rate evaluations. The shielding analyses provide assurances that the dose rate levels are acceptably low to workers around the spent nuclear fuel pool. Accident analyses are performed to show that the most conservative effects to the public of a postulated release of failed spent nuclear fuel fission products in the pool meet all regulatory dose rate limits.

One difference between nuclear power plant spent nuclear fuel wet storage and that which would be used for foreign research reactor spent nuclear fuel is that at nuclear plants, the pools include soluble boron in the water as a means of controlling criticality. Boron is a powerful neutron absorber, and as such prevents the approach to criticality since neutrons are needed to initiate and maintain a uranium fission chain reaction. Soluble boron would not be used in a wet storage facility for foreign research reactor spent nuclear fuel because it would corrode the aluminum cladding materials present in most foreign research reactor spent nuclear fuel. If neutron-absorbing materials were deemed desirable for foreign research reactor spent nuclear fuel wet storage, they could be incorporated as solid boron-aluminum alloy plates encased in aluminum or stainless steel that are integral to the storage rack design so that boron is physically present between each fuel assembly. This design has been successfully licensed and operated at many commercial nuclear power plants to allow for a higher density or tighter packing of the spent nuclear fuel assemblies.

Figure F-12 illustrates a typical wet pool storage facility for spent nuclear fuel. The operational experience of wet storage facilities is excellent, with no significant accidents or events. In the few instances when fuel was damaged while being moved into or out of the pool, the water mitigated any radiological consequences to workers, the public, or the environment. Some events involved temporary loss of pool water heat-removal systems. The large heat capacity of the water in the pool reduced any increase in fuel temperature so that no harmful effects resulted from such a loss in cooling capacity. These two types of wet storage facility events emphasize the principal benefit of water as a coolant and shielding medium, namely its very large thermal inertia and shielding/radionuclide retention capacity.

There is a long and successful history of safe operation for wet storage of spent nuclear fuel in the commercial power, research, and defense sectors of the nuclear industry. The technology is well known, licensed, and offers extensive operational experience.

The same arguments that apply to dry storage of spent nuclear fuel also apply to wet storage facilities. 10 CFR 72 applies to both dry and wet storage facilities for spent nuclear fuel at commercial licensees. DOE Order 6430.1A (DOE, 1989a) applies to the general design of nuclear facilities on DOE sites, including those for spent nuclear fuel storage. This DOE order references 10 CFR 72 for most of the specific details on spent nuclear fuel storage.

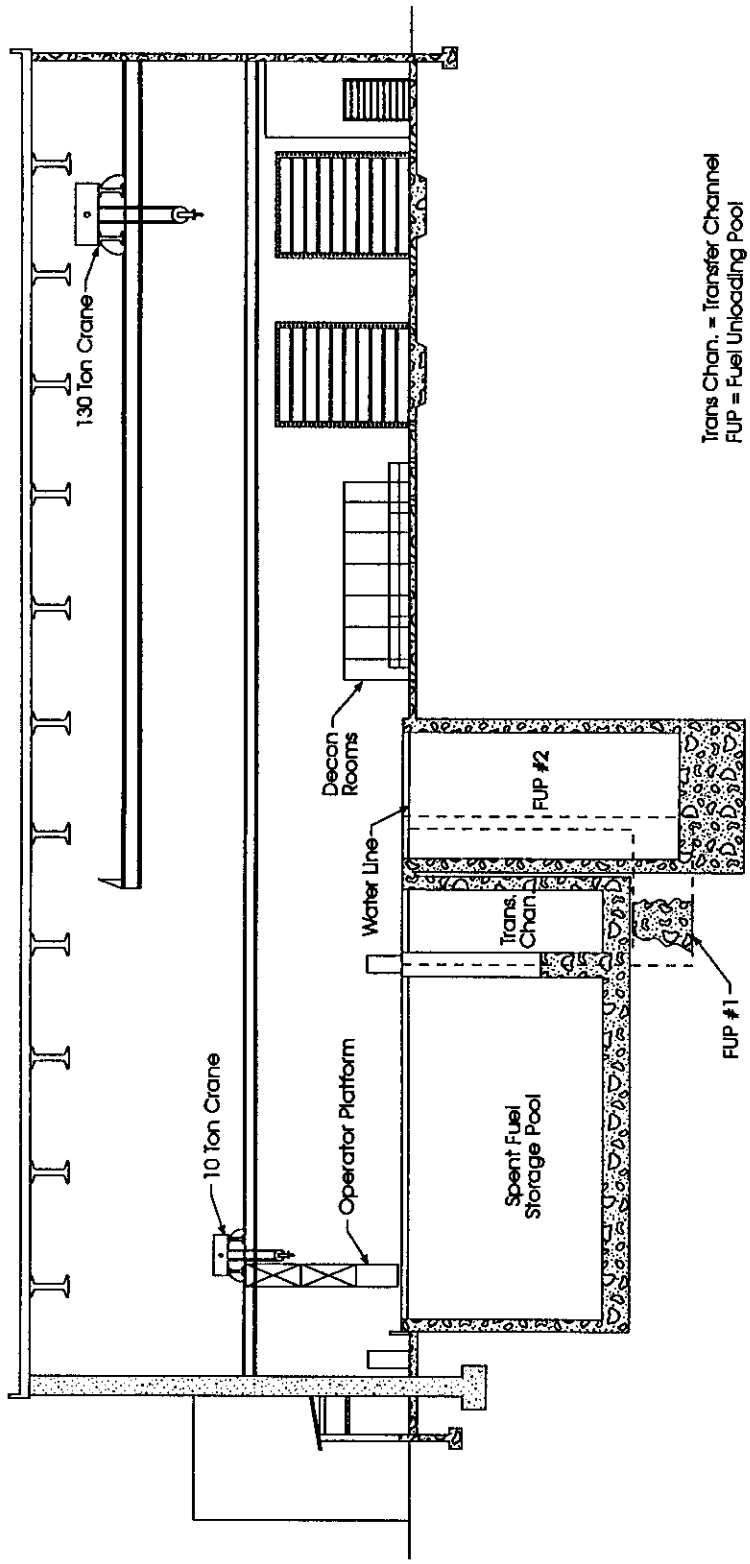


Figure F-12 Typical Wet Pool Storage Facility for Spent Nuclear Fuel

F.1.3 Summary of DOE Spent Nuclear Fuel Locations and Activities

DOE currently has about 2,700 MTHM of spent nuclear fuel in its storage facilities across the DOE complex (DOE, 1994h). Additional generation of about only 100 MTHM is anticipated during the next 40 years. Most of the spent nuclear fuel storage occurs at three sites: Hanford Site (77 percent), Idaho National Engineering Laboratory (10.9 percent), and Savannah River Site (7.3 percent) (Table F-9). Note that the quantities of DOE spent nuclear fuel completely dwarf the expected amount of foreign research reactor spent nuclear fuel (about 19 MTHM) on an MTHM basis (i.e., foreign research reactor spent nuclear fuel is less than 1 percent of the total). However, on a volume basis, foreign research reactor spent nuclear fuel represents about 10 percent of the total and, thus, their storage facilities would be of a significant size. Predominantly wet storage is used at DOE sites, although some limited experience exists with dry storage (e.g., Los Alamos National Laboratory and Idaho National Engineering Laboratory).

Table F-9 DOE Spent Nuclear Fuel Inventory[a, b]

Generator or Storage Site[c]	*Existing (1995)*		*Future Increases (through 2035)*		*Total (2035)*	
	MTHM[d]	*Percent*	*MTHM*[d]	*Percent*	*MTHM*[d]	*Percent*
Hanford Site	2,132.44	80.6	0.00	0.0	2,132.44	77.8
Idaho National Engineering Laboratory[e]	261.23	9.9	12.92	13.5	274.14	10.0
Savannah River Site	206.27	7.8	0.00	0.0	206.27	7.5
Naval Nuclear Propulsion Reactors	0.00[f]	0.0	55.00	57.6	55.0	2.0
Oak Ridge Reservation	0.65	<0.1	1.13	1.2	1.78	<0.1
Other DOE Sites	0.78	<0.1	1.50	1.6	2.28	<0.1
Non-DOE Domestic Research Reactors[g]	2.22	<0.1	3.28	3.4	5.50	0.2
Special-Case Commercial Reactors[h]	42.69	1.6	0	0	42.69	1.6
Foreign Research Reactors[i]	0	0	21.7	22.7	21.70	0.8
Total	**2,646.27**		**95.53**		**2,741.80**	
Percent of 2035 Total	**96.5**		**3.5**		**100.00**	

[a] *Source: DOE, 1995g*

[b] *Numbers may not sum due to rounding.*

[c] *The Nevada Test Site does not currently store or generate spent nuclear fuel and is not expected to generate spent nuclear fuel through 2035. However, in the 2010-2020 timeframe, a repository may open, with annual capacity over 1,000 MTHM.*

[d] *One MTHM equals approximately 2,200 pounds.*

[e] *Sum of fuel located at the Idaho National Engineering Laboratory.*

[f] *Existing inventory of Naval spent nuclear fuel is included in the Idaho National Engineering Laboratory totals (9.95 MTHM).*

[g] *Includes research reactors at commercial, university, and Government facilities.*

[h] *This total is just that stored at non-DOE facilities (Babcock & Wilcox Research Center and Fort St. Vrain). The total inventory of spent nuclear fuel from special-case commercial reactors is 186.41 MTHM. This fuel is also stored at the Idaho National Engineering Laboratory, the Oak Ridge Reservation, the Hanford Site, the Savannah River Site, and the West Valley Demonstration Project.*

[i] *At the Savannah River Site and the Idaho National Engineering Laboratory.*

Wet Storage

DOE spent nuclear fuel pools are in many cases more than 20 years old and were originally unlined, due to simplicity and the relatively short planned duration (3 to 6 months) of spent nuclear fuel storage prior to reprocessing. The spent nuclear fuel storage basins are concrete with 30 to 90 cm (1 to 3 ft) thick walls, and the bottom is usually thicker than the sides. For shielding purposes, the pool maintains a minimum of 3 m (10 ft) of water over the spent nuclear fuel at all times. Thus, total water depth typically ranges between 4.5 to 6.1 m (15 to 20 ft), although some facilities extend to 9.1 m (30 ft). Steel, stainless steel, or aluminum racks are affixed to the bottom of the pool for holding the spent nuclear fuel in a vertical configuration. The spent nuclear fuel basins provide for recirculation and heat removal capabilities, but limited water clarification and purification. Chemicals from the exposed concrete increase pool turbidity and tend to accelerate corrosion phenomena, particularly for aluminum-clad fuels. Some of the DOE spent nuclear fuel wet storage facilities do not meet the present construction requirements.

Wet storage still remains the predominant technology for storing irradiated materials (DOE, 1993b; Taylor et al., 1994). Currently, there are some 29 DOE spent nuclear fuel storage pool facilities in the complex, ranging in age from 10 years to more than 40 years. Facilities built more than 30 years ago were constructed to standards far less rigorous than exist today. Several DOE orders address spent nuclear fuel storage facilities indirectly, while DOE Order 6430.1A specifically sets the design criteria that addresses storage facilities (spent nuclear fuel facilities that are part of a reactor facility are covered by DOE Order 5480.6). Most DOE storage pools were not designed for long-term storage of spent nuclear fuel and targets and have very limited space available for consolidation.

Most of the storage pool surfaces are bare concrete A few are lined with stainless steel, and some are coated with epoxy or vinyl. The unlined pools are more susceptible to leakage and to increased contamination by soluble radionuclides. The unlined bare concrete storage pools do not have effective leak-detection systems to detect and capture potential leaks. To help identify pool leakage, more than 50 percent of DOE storage pools have had groundwater monitoring wells installed.

Severe corrosion of materials within many of the DOE storage pools has occurred. Corrosion has been generally attributed to poor water quality control and material incompatibilities, which has led to pitting and galvanic corrosion of spent nuclear fuel and storage equipment. This could potentially create a problem when the spent nuclear fuel materials have to be moved. In some cases, equipment failure could cause fissile material reconfiguration, which could increase nuclear criticality concerns. As a result of corrosion, release of radionuclides and fissile material to the pools has occurred. Corrosion also creates handling, packaging, inventory control, waste generation, and cleanup problems with the storage pools.

Savannah River Site and DOE (Taylor et al., 1994) consider the following facilities potentially suitable for near-term future wet spent nuclear fuel storage (in some cases with facility upgrades):

- Idaho National Engineering Laboratory
  - Power Burst Facility Canal
  - Idaho Chemical Processing Plant (ICPP)-666 Pool
  - Expended Core Facility
- Savannah River Site
  - 105-K Disassembly Basin

- 105-L Disassembly Basin
- 105-C Disassembly Basin
- 105-P Disassembly Basin
- Receiving Basin for Offsite Fuels (RBOF) Facility (244-H)
- BNFP (acquisition required).

However, only the ICPP-666 pool and the BNFP were found to meet all current standards, and, thus, be considered suitable for long-term storage.

DOE has improved some of its spent nuclear fuel facilities and has plans for additional upgrades (DOE, 1993b; DOE, 1995g). Typical upgrades include:

- installation and operation of water purification equipment, such as demineralizer columns and filters,
- reracking and fuel consolidation to increase fuel storage space, and
- improving seismic resistance (where possible, via additional supports).

These upgrades would extend the life of existing facilities and allow safe storage of spent nuclear fuel until new facilities are constructed or the spent nuclear fuel is chemically separated. In addition, spent nuclear fuel suspect of leaking during this interim period would be removed and canned to extend its safe storage.

Dry Storage

DOE has fewer dry storage facilities, and these range from approximately 1 to 50 years in age. There are many different types and applications of dry storage used throughout the DOE complex. Spent nuclear fuel is sorted in steel structures; lined and unlined concrete hot cells; steel-lined; concrete; below-grade vaults; reprocessing canyon dissolver cells; cans contained in steel wells; and large, above-grade storage casks. Spent nuclear fuel has been characterized and stored in dry configurations within hot cell facilities since the 1950s. Most DOE hot cells were not designed and built for long-term storage of spent nuclear fuel. Their primary mission was to conduct tests and basic research on irradiated fuels resulting in very limited capacity for storage of spent nuclear fuel.

Since the 1970s, spent nuclear fuel has been stored in facilities specifically engineered for longer-term dry storage. Modern dry storage methods in newer facilities provide low corrosion environments within sealed barriers for monitored interim retrievable storage. A few examples of dry storage confinement methods include sealed canisters in wells surrounded by concrete and extensive release protection incorporating High Efficiency Particulate Air-filtered ventilation systems. By using current dry storage technology, dry storage facilities could be engineered to withstand severe natural phenomena hazards, fires, and explosions without damage to the fuel or release of radionuclides. Dry storage technologies can be adapted to store many types of damaged and undamaged DOE-owned spent nuclear fuel.

The application of dry storage technologies generally results in fewer environmental, safety, and health issues as compared with wet storage. However, DOE has limited experience with aluminum-clad, high decay heat fuels in dry storage facilities.

Some quantities of spent nuclear fuel may be in dry storage facilities for much longer than originally anticipated. Over the years, several inground steel-lined storage well barriers have had the potential for severe corrosion, which could result in undetected releases to the environment. This is particularly important, because of the inaccessibility of these facilities for inspection and characterization (e.g., Argonne National Laboratory Radioactive Scrap and Waste Facility).

The Savannah River Site and DOE (Taylor et al., 1994) consider the following facilities suitable for near-term dry storage of spent nuclear fuel.[1]

- Argonne National Laboratory-West
  - Hot Fuel Examination Facility
  - Radioactive Scrap and Waste Facility
- Idaho National Engineering Laboratory
  - Test Area North Test Pad
  - ICPP Irradiated Fuel Storage Facility (IFSF)
  - ICPP-749 (Drywells, Second Generation)
- Savannah River Site
  - 221-H (Extensive modification required)
  - 221-F (Extensive modification required).

However, certain DOE requirements, such as DOE Orders 6430.1A and 5480.6, make it likely that these facilities could not qualify for future, long-term, dry storage of spent nuclear fuel. Excluding the Savannah River Site facilities (because of the extensive required modifications), none of these facilities appear to be very useful for long-term spent nuclear fuel storage. Extensive modifications to the facilities would be required to meet seismic criteria and increase the storage capacity or convert existing facilities (e.g., F- and H-Canyons at the Savannah River Site) into suitable dry storage facilities. However, facilities such as the Hot Fuel Examination Facility and Test Area North appear suitable as possible staging and characterization facilities in a dry cask storage approach, based upon the presented information (Taylor et al., 1994).

F.1.3.1 Savannah River Site

The Savannah River Site occupies an area of approximately 800 km^2 (310 mi^2) in South Carolina, in a generally rural area about 40 km (25 mi) southeast of Augusta, Georgia (DOE, 1995g). The Savannah River forms the southwestern border of the Savannah River Site. The Savannah River Site consists primarily of managed upland forest with some wetland areas, and facilities and railways occupy approximately five percent of the Savannah River Site land area. Figure F-13 presents a map of the Savannah River Site with spent nuclear fuel facilities displayed.

1 *Existing facilities in Nevada were not included in the analysis.*

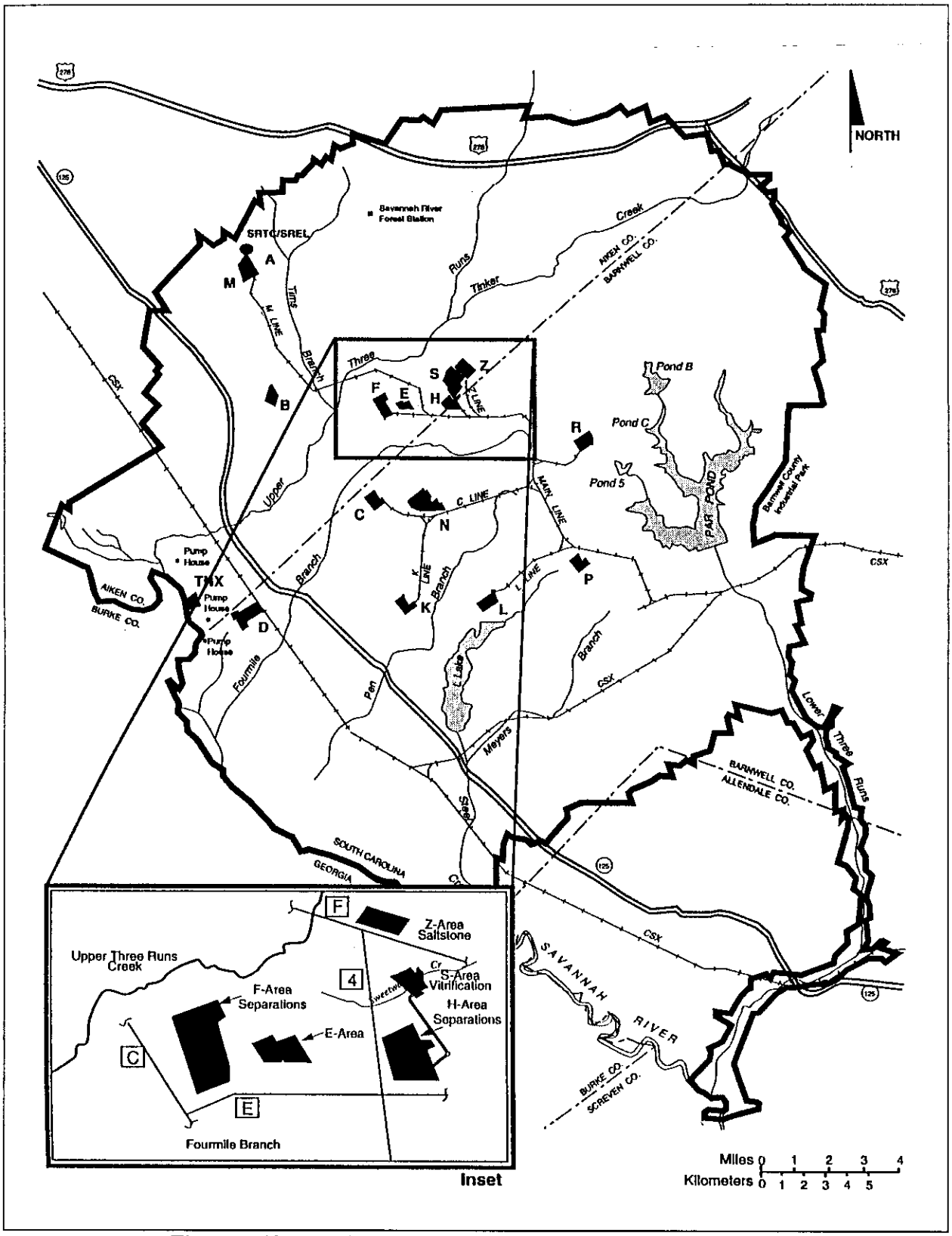


Figure F-13 Location of Principal Savannah River Site Facilities

The primary Savannah River Site facilities were used for the production of nuclear materials. Currently, the production reactor facilities are not operating and are in either shutdown or standby mode. Several large waste management projects are now underway at the site, including the Defense Waste Processing Facility for the vitrification of high-level waste.

F.1.3.1.1 Spent Nuclear Fuel Activities at the Savannah River Site

The Savannah River Site currently stores approximately 201 MTHM of spent nuclear fuel (DOE, 1995g), or approximately 7 percent of the DOE total, including the following:

- 184.4 MTHM of aluminum-based spent nuclear fuel, including plutonium target material,
- 4.6 MTHM of commercial spent nuclear fuel (zircaloy-clad),
- 11.9 MTHM of test and experimental reactor, zircaloy-clad fuel, and
- 5.4 MTHM of test and experimental reactor, stainless steel-clad fuel.

This fuel is stored in several basins onsite. The F- and H-Area Canyons are the processing and separations facilities at the Savannah River Site, and each has a small associated wet storage basin. Three reactor disassembly basins (K, L, and P) contain the reactor fuel and target materials. A fourth reactor disassembly basin (C) currently is the only basin without security upgrades necessary for any storage activities. These basins consist of unlined concrete with inadequate water purification equipment for extended storage of aluminum-clad spent nuclear fuels. These reactor basins were built in the 1950s and were not intended for the long-term storage ("years") of radioactive materials. Furthermore, poor water chemistry has corroded some of the spent nuclear fuel in the K- and L-Reactor disassembly basins, resulting in the release of fissile materials to the pool water. Also, these reactor basins are not seismically qualified and lack modern earthquake resistant features. Ongoing facility upgrades of the L-Reactor disassembly basin are intended to correct the conditions of the basin. Deionization of the basin has lowered the conductivity to acceptable levels for corrosion control. Lower conductivity would greatly reduce the probability of new corrosion and reduce the rate of progression of existing corrosion. The control of the conductivity after the completion of the deionization would be accomplished using the Disassembly Basin Upgrade Project which was initiated to address near term activities and vulnerabilities associated with storing fuel in the L-Reactor disassembly basin. With the upgrades to be completed by mid-1996 (Miller et al., 1995), the L-Reactor basin can be expected to safely store spent nuclear fuel for as long as 10 to 20 years. These upgrades include the following:

- A continuous on-line deionization system to improve water chemistry. The continuous deionization system will lower and control the conductivity levels of the basin thereby minimizing corrosion. The continuous deionizer system also removes ionic radionuclide concentrations, specifically Cesium-137.
- A makeup water deionizer to improve the quality of makeup water supplied to the basin. This action will mitigate any additional load on the continuous deionization system.
- New equipment and systems for alternative packaging and removal of waste.

A Basis for Interim Operation document for the L-Reactor in cold standby conditions was prepared by the Westinghouse Savannah River Company (WSRC, 1995b). The Basis for Interim Operation addressed the effects of process events on the facility worker and the effects of process and natural phenomena hazards events on the public and the environment. The Basis for Interim Operation document concluded that the

facility could continue to operate within the safety envelope, identified in the Basis for Interim Operation, without undue risk to the public or the environment.

The RBOF is the other major facility for spent nuclear fuel storage. The RBOF is more suitable than the reactor basins because it is lined (epoxy sides, stainless steel bottom) and has a water purification system. The BNFP, after refurbishing, would be suitable for foreign research reactor spent nuclear fuel storage because it is fully lined with stainless steel, has water purification systems, and has active heat removal systems. Major spent nuclear fuel storage facilities are summarized in Table F-10.

Table F-10 Major Savannah River Site Spent Nuclear Fuel Storage Facilities

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel Elements*	*Access*
105-K Disassembly Basin	Basin Dimensions: 46.9 x 65.8 x 5.2m (154'W x 216'L x ~17'D) Basin Water: 13.2 million l (3.5 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-L Disassembly Basin	Basin Dimensions: 46.9 x 65.8 x 5.2m (154'W x 216'L x ~17'D) Basin Water: 13.2 million l (3.5 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-C Disassembly Basin	Basin Dimensions: 39.6 x 58.2 x 5.2m (130'W x 191'L x ~17'D) Basin Water: 13.6 million l (3.6 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-P Disassembly Basin	Basin Dimensions: 55.5 x 68.2 x 5.2m (182'W x 223'L x ~17'D) Basin Water: 18.2 million l (4.8 million gal)	None initially 20,000 after upgrades	Truck/Rail
RBOF (244-H)	Basin 1: 8.2 x 12.1 x 6.7m depth over two-thirds of floor space 8.8m depth over one-third of area Basin 2: 8.2 x 3.9 x 8.8m depth Basin Water: 1.7 million l (450,000 gal)	~1000 initially, plus 1,425 after rearranging[b]	Truck/Rail
BNFP[a]	Several Pools: Main Pool: 14.6 x 14.6 x 9.8m (48'L x 48' x 32'D) Basin Water: 2.1 million l (550,000 gal)	None initially 25,000 after acquisition and reactivation[b]	Truck/Rail

[a] *Discussed in more detail in Section F.1.3.1.3; rail spur not currently active but would be included in reactivation.*

[b] *Difference in capacity between RBOF and BNFP is due to greater pool depth of BNFP and different fuel packing density assumptions for the two facilities.*

F.1.3.1.2 Spent Nuclear Fuel Storage Facilities Available for Foreign Research Reactor Spent Nuclear Fuel at the Savannah River Site

The RBOF is the principal facility applicable for foreign research reactor spent nuclear fuel. This basin has been operating and receiving spent nuclear fuel, including foreign research reactor spent nuclear fuel, since 1964, and is located in H-Area, near the center of the Savannah River Site. The 1,393 m^2 (15,000 ft^2) facility consists of an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin. The basins and their interconnecting canals hold approximately 1,893,000 l (500,000 gal) of water. Spent nuclear fuel elements arrive in lead-lined casks weighing from 22 to 64 metric tons (24 to 70 tons), which a crane lifts from a railroad car or a truck trailer and places in the unloading basin. About 30 percent of the fuels in the RBOF consist of uranium clad in stainless steel or zircaloy, which the Savannah River Site facilities cannot process without modifications. The RBOF is discussed in more detail in Section F.3.

In March 1995, the Savannah River Site estimated that the RBOF has the capacity for approximately an additional 1,000 spent nuclear fuel elements (O'Rear, 1995). However, the Savannah River Site has

determined that 1,425 additional spaces can be made available by rearranging fuel in the pools and moving spent nuclear fuel to other storage areas, such as one of the reactor disassembly basins. If empty, the total RBOF capacity would be 6,500 foreign research reactor spent nuclear fuel elements.

F.1.3.1.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Savannah River Site for Foreign Research Reactor Spent Nuclear Fuel

The Savannah River Site is evaluating the use of several new planned or potential facilities for foreign research reactor spent nuclear fuel management. These include:

- a modular dry vault storage building,
- dry cask storage, or
- wet pool storage.

These technologies may require additional support facilities for such functions as: spent nuclear fuel examination, spent nuclear fuel characterization, cask loading and unloading, spent nuclear fuel repackaging, and cask maintenance. The Savannah River Site is also evaluating the use of one or more of the reactor disassembly basins for near-term wet storage of foreign research reactor spent nuclear fuel. These facilities are discussed in more detail in Section F.3.

The Savannah River Site is also evaluating the potential storage of spent nuclear fuel at the BNFP facility. Allied General Nuclear Services constructed a large reprocessing facility for commercial spent nuclear fuel in Barnwell, South Carolina, adjacent to the Savannah River Site (Fields, 1994; Matthews, 1994 and 1991; Taylor et al., 1994; Williams, 1994; WSRC, 1992a-d). This plant was never operated due to a change in Government policy, and was mothballed in the 1980's. The BNFP includes a wet fuel storage basin that is approximately twice the area and potentially has over four times the spent nuclear fuel capacity of the RBOF facility at the Savannah River Site. The wet storage basin is fully lined and seismically qualified and would be capable of storing all of the currently identified foreign research reactor spent nuclear fuel (Jackson, 1994). Facility acquisition, replacement of removed equipment, reactivation, installation of suitable storage racks, and checkout at the facility would be required prior to its use.

Figure F-14 displays the location of the BNFP in relation to the Savannah River Site. This land was originally part of the Savannah River Site. The BNFP site consists of approximately 680 hectares (ha) (1,680 acres).

Allied General Nuclear Services designates the fuel pool area of the plant as the "Fuel Receiving and Storage Stations." Considerable documentation exists for the facility, including the engineering designs, the Environmental Impact Statement (EIS), and the Final Safety Analysis Report submitted to the NRC. The pools and attendant cranes are fully seismically qualified structures. The pool section includes ion exchange systems for pool water purification and a separate radwaste system (solidification may need to be added). The section incorporates capabilities for receipt of either truck or railcarried casks. The main crane is rated at 122 metric tons (135 tons).

The Fuel Receiving and Storage Station facility is shown in Figure F-15 and was designed and constructed to receive, store, and handle spent (irradiated) light water reactor fuel. Spent nuclear fuel assemblies are received in shielded casks by either truck or rail. The assemblies are unloaded underwater and stored underwater to provide cooling and shielding. Stored fuel can be remotely transferred to the adjacent Remote Process Cell and Remote Maintenance and Scrap Cell for mechanical processing. After fuel assemblies are unloaded from the shielded casks, the empty casks are prepared for return shipment.

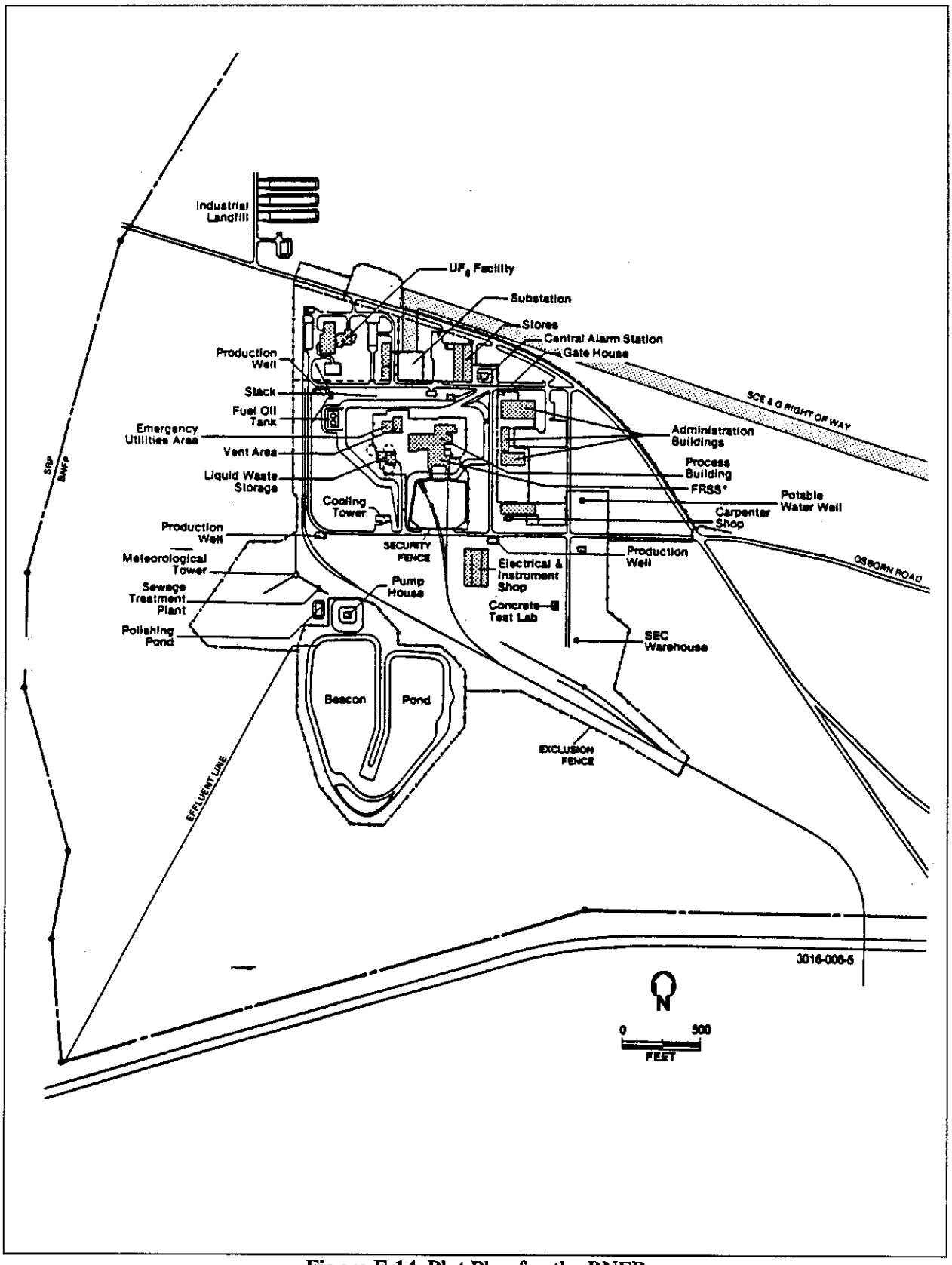


Figure F-14 Plot Plan for the BNFP

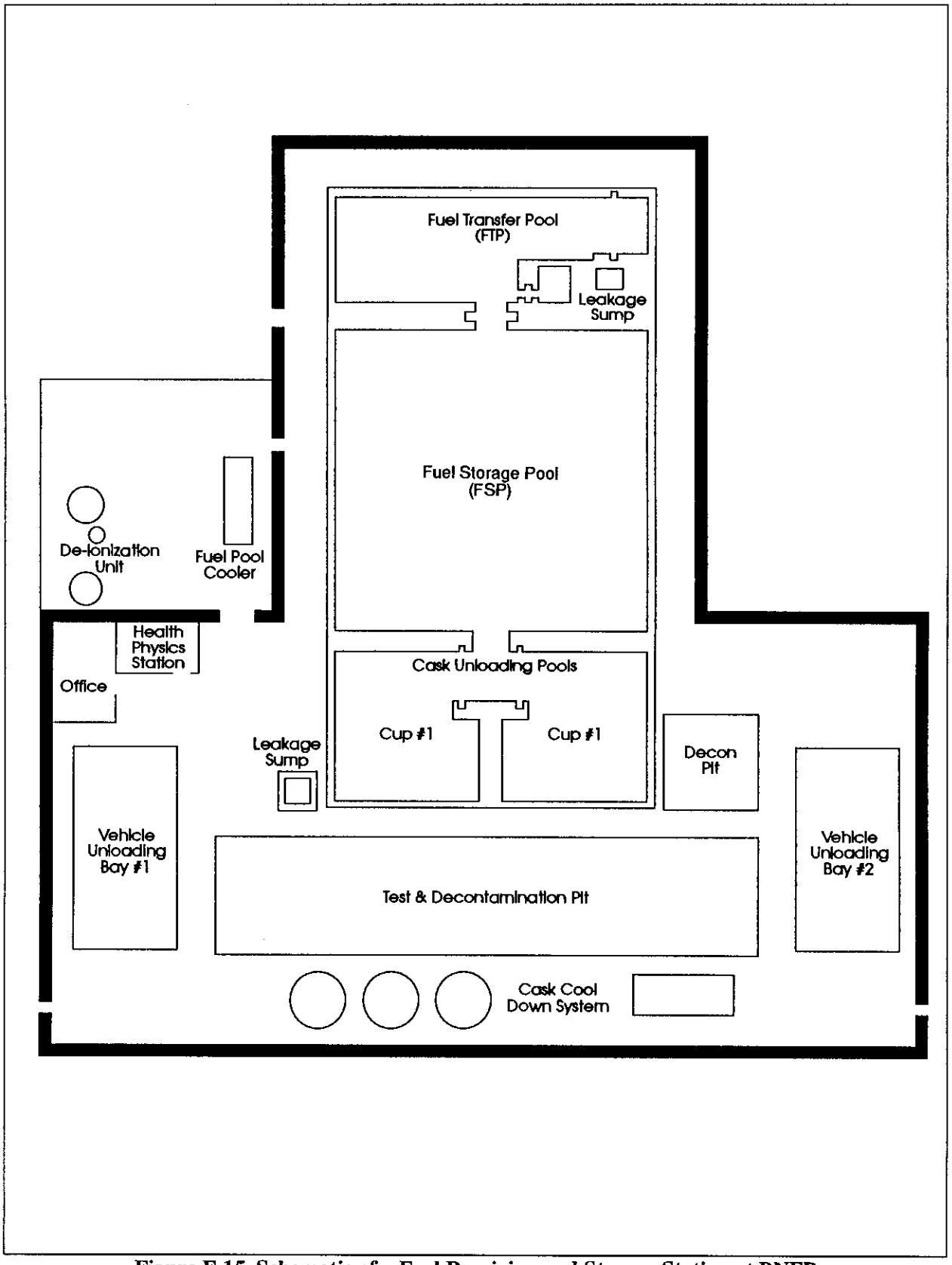


Figure F-15 Schematic of a Fuel Receiving and Storage Station at BNFP

The following areas of the Fuel Receiving and Storage Station are safety class structures:

- pool concrete structure,
- pool and crane column foundations,
- embedments for the fuel storage racks,
- crane rails, rail supports, and restrainer bars which retain the cranes on their rails and prevent their falling into the pools,
- cask barrier beams and embedments,
- energy absorbing pads in the Cask Unloading Pools,
- emergency water supply line, and
- Fuel Receiving and Storage Station walls to the 7.6 m (25 ft) level above grade. Clean spent nuclear fuel casks are moved to the Fuel Receiving and Storage Station water pool area. This area is divided into six pools consisting of two Cask Unloading Pools, one Fuel Storage Pool, one Failed Fuel Pool, one Fuel Transfer Pool, and an examination cell/pool.

Water shielding of 3.7 m (12 ft) is provided in the Fuel Receiving and Storage Station pools. This limits surface dose rates to a calculated 0.08 mrem/hr, assuming design basis Light Water Reactor fuel, and permitting at least 40 hours per week working time for an operator. Handling systems are designed with special limit switches and mechanical stops to prevent raising fuel higher than the design depth of the shielding water.

The water in the five pools of the Fuel Receiving and Storage Station is channeled and treated to promote maximum clarity, to control temperature, and to minimize corrosion and radioactivity. This is accomplished by continuous filtration through 95 percent efficient 5 micron pore size filter elements, cooling in heat exchangers to hold the pool water temperature below 41°C (105°F), and demineralization.

Demineralizing water treatment is designed to maintain radioactivity levels below 0.0005 μCi/ml. Pool water is pumped from the Fuel Storage Pool at 7,570 l/min (2,000 gal/min), directed through the heat exchangers, and returned to the Fuel Storage Pool. A second stream is pumped at 1,135 l/min (300 gal/min) from a pool and is filtered. After filtration, one-half of this stream is treated by ion exchange. The combined filtered and purified solution is then returned to the Fuel Storage Pool. The pool piping system is arranged so that the cleanup stream can be removed from or returned to any of the pool areas, permitting cleanup of contaminated water.

The cooling system is designed to remove heat at a rate of 4,000 kilowatts (14 million Btu/hr). The cooling capacity can be increased by expanding the capacity of the heat exchanger system in the Fuel Receiving and Storage Station. The estimated life for the structure is 50 years (Fields, 1994; Matthews, 1994; Taylor et al., 1994).

The Fuel Receiving and Storage Station has the following six pools:

- two cask pools, each 18.3 m (60 ft) deep,
- failed fuel pool (for degraded fuel),
- fuel transfer pool, 18.3 m (60 ft) deep,

- examination cell/pool, and
- main pool, 14.6 m x 14.6 m x 9.8 m deep (48 ft by 48 ft by 32 ft deep).

All of the pools are lined with stainless steel and are designed to maintain a minimum of 3.7 m (12 ft) of water above the fuel for shielding. The pools include detectors and flow channels for managing potential leaks. The original capacity of the main pool was 400 MTHM. Various analyses have been performed to increase this capacity to the 1,200 to 2,000 MTHM range with reracking and other arrangements. It has been estimated that maximum wet storage corresponds to approximately 5,200 Pressurized Water Reactor assemblies (Taylor et al., 1994). For foreign research reactor spent nuclear fuel, this would correspond to over 25,000 elements; and, thus, as noted previously, the BNFP could accommodate all of the fuel.

The environmental impacts of spent nuclear fuel storage at the BNFP have also been analyzed for between 360 and 5,000 MTHM of commercial fuel (Taylor et al., 1994). The results were:

- Dose commitments to the 80 km (50 mi) population were estimated to be 0.067 person-rem and 0.071 person-rem for 15- and 25-year storage periods, respectively.
- The worst accident would result in a dose commitment of 1 mrem total body, 6 mrem thyroid, and 100 mrem skin to an exposed individual located at the eastern boundary of the site.

These analyses were based upon commercial spent nuclear fuel, but should bound the consequences of foreign research reactor spent nuclear fuel storage at the BNFP. Potential impacts are discussed in more detail in Section F.4.

The BNFP site consists of some 680 ha (1,680 acres), bounded on three sides by the Savannah River Site. Preliminary walkthroughs and analyses by the Savannah River Site indicate the facility is in good condition, and principally needs a main transformer for power supply. The Savannah River Site has estimated a cost of $50 million (Matthews, 1991; WSRC, 1992a-d). Actual acquisition and reactivation costs are claimed to be as low as $25 million (Matthews, 1994; WSRC, 1992a-d). This facility, however, would not be available immediately to receive the foreign research reactor spent nuclear fuel.

Figure F-16 displays the foreign research reactor spent nuclear fuel storage capacity versus time for the Savannah River Site. Clearly, the Savannah River Site can accommodate foreign research reactor spent nuclear fuel at existing facilities supplemented by dry storage, modified reactor disassembly basins, or the potential use of the BNFP. The reactor basins could be used to store the non-aluminum-based spent nuclear fuel currently in the RBOF because the poorer water quality in the basins would not cause additional corrosion for this other fuel that is not aluminum based. Recent improvements in reactor basin water chemistry control have resulted in a substantial decrease in the potential for corrosion of aluminum-clad spent nuclear fuels.

F.1.3.2 Idaho National Engineering Laboratory

The Idaho National Engineering Laboratory has several reactors and critical assemblies operating and also possesses several reactors that are either in standby or shutdown and awaiting decommissioning. From 1953 until 1992, the Idaho National Engineering Laboratory was responsible for processing and recovering highly-enriched uranium (HEU) from naval reactors. The Idaho National Engineering Laboratory discontinued processing spent nuclear fuel in 1992. Consequently, the Idaho National Engineering Laboratory has spent nuclear fuel facilities, spent nuclear fuel in storage, and spent nuclear fuel from

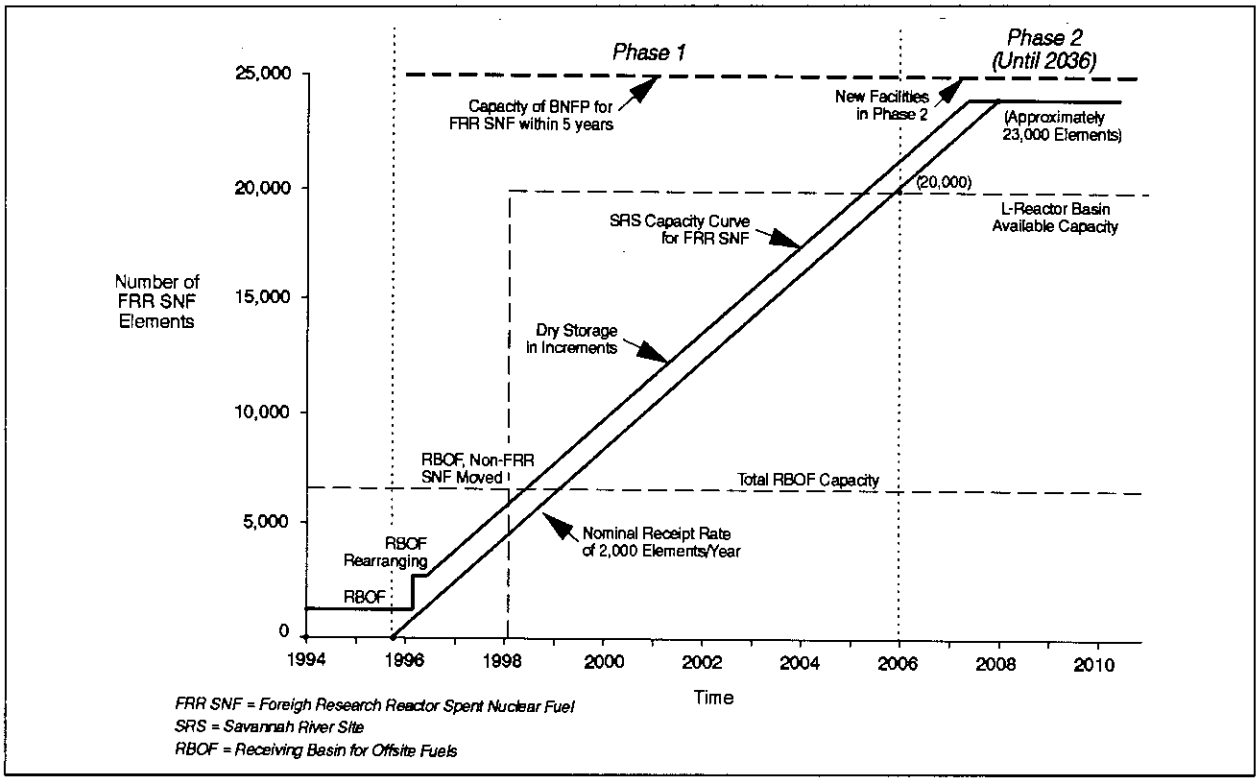


Figure F-16 Foreign Research Reactor Spent Nuclear Fuel Storage at the Savannah River Site

current operations. The Idaho National Engineering Laboratory site map with spent nuclear fuel facilities is shown in Figure F-17.

F.1.3.2.1 Spent Nuclear Fuel Activities at the Idaho National Engineering Laboratory

Six major facility areas at the Idaho National Engineering Laboratory store spent nuclear fuel:

- ICPP,
- Test Area North,
- Power Burst Facility,
- Test Reactor Area,
- Argonne National Laboratory-West, and
- Naval Reactors Facility.

A description of each major facility area and its spent nuclear fuel storage activities is presented below.

F.1.3.2.2 Spent Nuclear Fuel Storage Facilities at the Idaho National Engineering Laboratory

Spent nuclear fuel at the Idaho National Engineering Laboratory is sorted in a variety of dry and wet configurations. The total amount of spent nuclear fuel at the Idaho National Engineering Laboratory

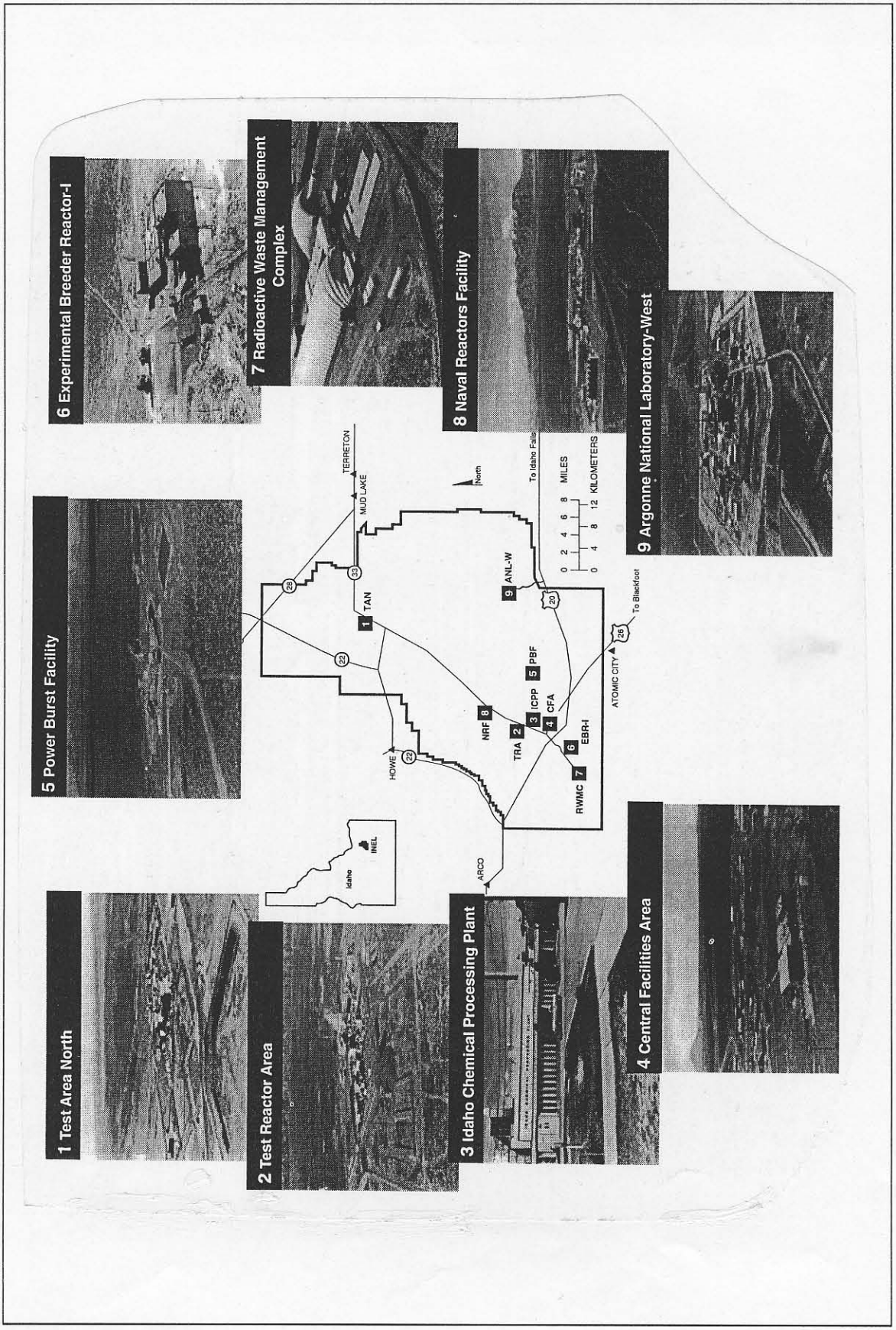


Figure F-17 Location of Spent Nuclear Fuel Storage and Handling Facilities at the Idaho National Engineering Laboratory

accounts for about 10 percent (by weight of heavy metal) of the spent nuclear fuel in the DOE complex (DOE, 1995g).

Table F-11 lists the primary spent nuclear fuel storage facilities, including the type of storage configuration, capacity for foreign research reactor spent nuclear fuel receipts, and accessibility. The number, variety, and location of the wet and dry configurations currently in use at the Idaho National Engineering Laboratory are largely the result of the different purposes for the facilities (e.g., at-reactor storage, storage research and development, reprocessing, and fuel research and development). The condition of the spent nuclear fuel in storage is generally good, with the notable exception of minor amounts of fuel in the Underwater Fuel Storage Facility at the ICPP-603.

The ICPP has received spent nuclear fuel from many onsite and offsite reactors (including foreign research reactor spent nuclear fuel) for reprocessing. Reprocessing for recovery of HEU materials was ceased in 1992. The ICPP now has the mission of managing its current spent nuclear fuel inventory and assigned new spent nuclear fuel receipts, development of technologies in support of dispositioning the spent nuclear fuel, and eventually packaging the material for shipment to a repository. The ICPP stores virtually all types of spent nuclear fuel except production reactor fuel (i.e., fuel from the Hanford Site and the Savannah River Site production reactors). It stores nonproduction reactor aluminum, stainless steel, zirconium, and graphite-clad spent nuclear fuel and uses both wet and dry storage configurations. The ICPP facilities have experience and some capacity for foreign research reactor spent nuclear fuel storage. These are discussed in more detail in Section F.3.

The Test Area North has been a reactor testing facility and has received significant amounts of spent nuclear fuel for examination and testing purposes. This includes the commercial dry storage cask demonstration program and the Three Mile Island debris examination program. It has a very large hot cell and an adjacent underwater storage pool to support the testing programs. It also has a large hot shop where large pieces of equipment, such as transportation casks, have been reconfigured or maintained. At the current time, the Test Area North hot cell and pool have no future mission, but may be used by the U.S. Navy. If Test Area North is not used by the Navy, then the Test Area North hot cell and pool may have significant capacity for receipt of foreign research reactor spent nuclear fuel and for placing it into temporary underwater storage or dry storage casks.

Other storage areas such as the Power Burst Facility reactor canal and the MTR storage pool have limited storage capacities for receipt or storage of foreign research reactor spent nuclear fuel.

The Argonne National Laboratory-West facilities supported the Experimental Breeder Reactor program and also contain the Transient Reactor Test Facility, the Zero Power Physics Reactor, and the Neutron Radiography Reactor. Spent nuclear fuel storage facilities include an at-reactor molten sodium storage pool, in-process lag storage in the Hot Fuel Examination Facility and dry underground SILOs for spent fuel and wastes pending disposition. The Hot Fuel Examination Facility would be suitable for foreign research reactor spent nuclear fuel examination activities.

The Naval Reactors Facility is also located at the Idaho National Engineering Laboratory, but is not included in Table F-11 because of its sole purpose to support the Naval ship propulsion program. The Naval Reactors Facility includes the Expended Core Facility, which receives and examines Naval spent nuclear fuel to support fuel development and performance analyses. In addition, the Expended Core Facility removes structural support material from the Naval spent nuclear fuel before transfer of the fuel portion to the ICPP for reprocessing or interim storage.

Table F-11 Description of Existing Spent Nuclear Fuel Facilities at the Idaho National Engineering Laboratory

Facility	*Description*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel*	*Access*
ICPP-666 Underwater Fuel Storage Area	Water Storage Facility with 6 lined storage basins 9.4 m x 14.2 m by 9.4 m or 12.5 m deep (31 ft x 46.5 ft x 31 ft or 41 ft deep)	Temporary storage after reracking for 8,400 elements	Shipment by truck. Rail shipments to a site receiving area 8 km (5 mi) away.
ICPP-603 Underwater Fuel Storage Area	Water Storage Facility with three hasins of varying sizes, no sealant or liner	Not Available - facility is being shut down	Shipment by truck.
ICPP-603 Irradiated Fuel Storage Facilities	Dry Storage Facility with remote unloading area and vault storage with 636 0.5 x 3.4 m L (18 in x 11 ft long) containers	200 containers available for storage of 9,000 foreign research reactor elements	Shipment by truck. Rail shipments to a site receiving area 8 km (5 mi) away.
ICPP-749 Underground Fuel Storage Area	Dry Storage Facility with 218 underground SILOs	Approximately 60 SILOs available following renovation of first generation SILOs; capacity for 3,600 elements after fiscal year 1998	Requires receipt into ICPP-666 or ICPP-603 IFSF and packaging and conditioning for dry storage.
Test Area North-607 Pool and Hot Cell	Water Storage Facility with adjacent remote hot cell	Approximately 56 m^2 (600 ft^2) of basin 7.3 m (24 ft) deep. Capacity for 4,000 elements after new rack installation.	Additional storage space available in hot cell. Shipment by truck, cask unloading in hot cell.
Test Reactor Area-620 Power Burst Facility	Small water storage pool adjacent to Power Burst Facility reactor	Minimal space available	Shipment by truck. Crane capacity inadequate for foreign research reactor casks.
Test Area North-607 Cask Storage Pad	Five commercial fuel storage casks on concrete pad	Easily expandable for more cask storage as necessary for foreign research reactor spent nuclear fuel shipments	Shipment by truck to hot cell where foreign research reactor spent nuclear fuel can be transferred to storage casks and moved to storage pad.
Test Reactor Area-603 MTR Pool	Water Storage Pool in basement of the MTR	Minimal space available	Shipment by truck. Crane capacity and access inadequate for foreign research reactor casks.
Test Reactor Area-660 ARMF and CFRMF Reactors and Canal	Swimming pool reactors with connecting canal	Minimal space available	Shipment by truck. Crane capacity and access inadequate for foreign research reactor casks.
Argonne National Laboratory-West Hot Fuel Examination Facility	Large two room hot cell facility for fuel examinations with argon atmosphere	Minimal space available without extensive removal of examination equipment	Shipment by truck access to hot cell limited to special designed small transfer cask.
Argonne National Laboratory-West Radioactive Scrap and Waste Facility	1,200 vertical steel-lined underground dry storage wells	About 500 wells are not used	Access requires transfer through Hot Fuel Examination Facility.

F.1.3.2.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Idaho National Engineering Laboratory for Foreign Research Reactor Spent Nuclear Fuel

The main focus of near-term activities is the accurate quantification and characterization of DOE-owned spent nuclear fuel, identification of spent nuclear fuel management facilities and their conditions, identification of safe interim storage for existing and new spent nuclear fuel, and identification of

technologies and requirements to place DOE spent nuclear fuel in safe interim storage. Long-term activities include the development of final waste acceptance criteria requirements and stabilization technologies for alternate fuel disposition, construction of facilities to stabilize fuel to meet waste disposal requirements, processing of the fuel to a final waste form, and transportation of the waste form for disposition (discussed in more detail in Section F.3). As shown in Figure F-18, the Idaho National Engineering Laboratory has sufficient capacity for foreign research reactor spent nuclear fuel if the existing facilities are supplemented by dry casks.

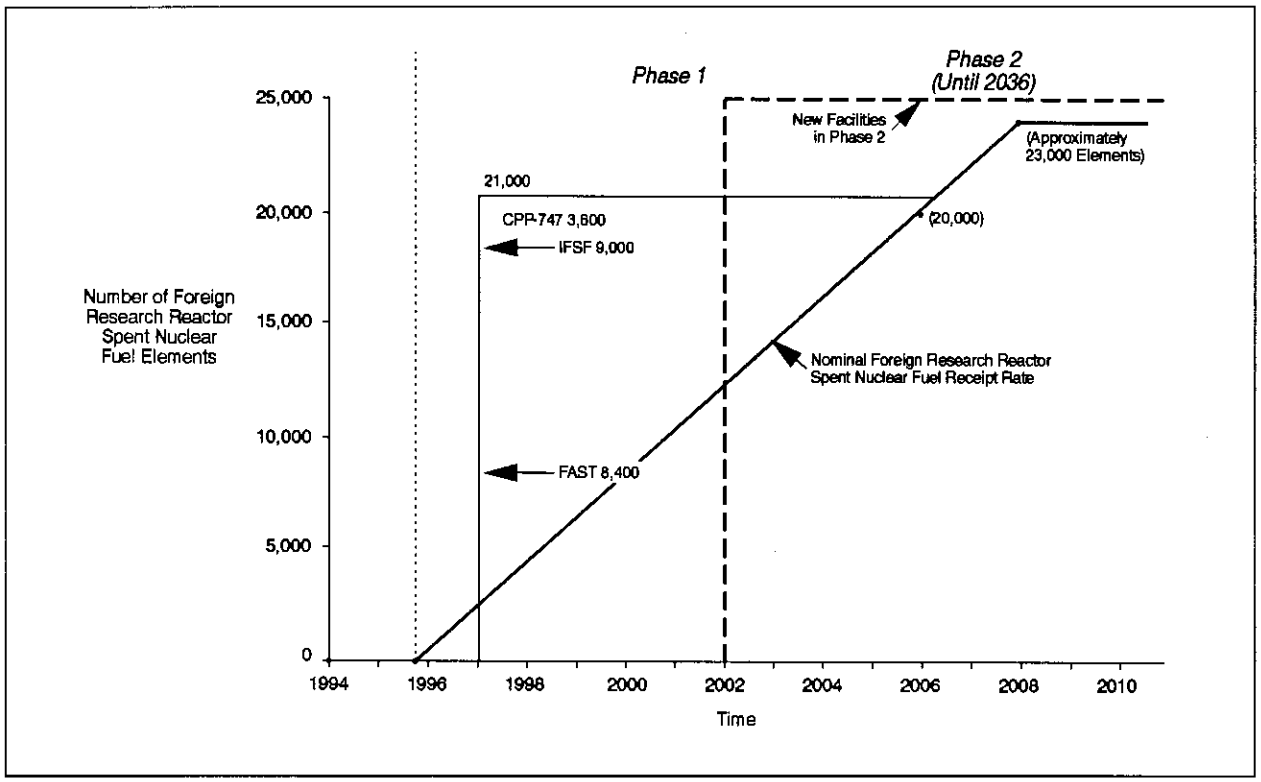


Figure F-18 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Idaho National Engineering Laboratory

F.1.3.3 Hanford Site

The Hanford Site lies within the semi-arid Pasco Basin of the Columbia Plateau in southeastern Washington State (DOE, 1995g). The Hanford Site occupies an area of around 1,450 km^2 (560 mi^2) north of the confluence of the Yakima and Columbia Rivers. Only about six percent of the site has been disturbed in the process of special nuclear materials production for national defense reprocessing and used for DOE purposes, such as nuclear materials production, processing, research and development, and waste management. The Hanford Site facilities include nine shutdown production reactors and several smaller research reactors. Several processing and product finishing facilities are located on the site, but are not currently operating and will not likely operate in the future. Currently, the principal mission of the site is environmental management and includes:

- decontamination and decommissioning of surplus facilities,
- environmental restoration of over 1,500 waste management units and 4 groundwater contamination plumes,

- waste management, including new processing facilities and retrievable disposal, and
- research and development into energy, environmental, and waste management technologies.

A Tri-Party Agreement between DOE, the U.S. Environmental Protection Agency, and the State of Washington provides milestones and guidance for these activities at the Hanford Site. Current schedules use a 2030 date for the completion of most of the restoration activities at the site. A map of the Hanford Site that shows spent nuclear fuel facilities is presented in Figure F-19. Existing spent nuclear fuel facilities are listed in Table F-12.

F.1.3.3.1 Spent Nuclear Fuel Activities at the Hanford Site

The following spent nuclear fuel types and their associated facilities are at the Hanford Site:

- *N Reactor Spent Nuclear Fuel:* This is zircaloy-clad, metallic uranium fuel stored in water in the 105-KE and 105-KW Basins (1,146 and 954 MTHM, respectively), and exposed to air in the plutonium-uranium extraction dissolver cells A, B, and C (0.3 MTHM).
- *Single-Pass Reactor Spent Nuclear Fuel:* This is aluminum-clad, metallic uranium fuel stored in water in the 105-KE and 105-KW Basins (0.4 and 0.1 MTHM, respectively), and stored in water in the plutonium-uranium extraction basin (approximately 2.9 MTHM).
- *Fast Flux Test Facility Spent Nuclear Fuel:* This consists of stainless steel-clad fuel stored in liquid sodium at the Fast Flux Test Facility, comprised mainly of a uranium/plutonium oxide fuel, but with some carbide, metallic, and nitride fuel elements (in all, fuel from 329 assemblies of spent nuclear fuel).
- *Shippingport Core II Spent Nuclear Fuel:* These assemblies are zircaloy-clad uranium dioxide fuel, and are stored in the T-Plant Canyon, Pool Cell 4.
- *Miscellaneous Commercial and Experimental Spent Nuclear Fuel:* This includes primarily zircaloy-clad uranium dioxide fuel stored in air, but does include some Test, Research, Isotope, General Atomic (TRIGA) reactor hydride spent nuclear fuel stored in water and aluminum-clad, uranium-aluminum metallic fuel stored in air. These are principally stored in the 300-Area at Hanford Site.

F.1.3.3.2 Spent Nuclear Fuel Storage Facilities at the Hanford Site

The Hanford Site spent nuclear fuel storage facilities are principally based upon wet methods. Table F-12 provides a brief summary of these facilities. The age, condition, available capacity of these facilities, and the Tri-Party Agreement milestones generally prevent the use of the existing facilities for storage of foreign research reactor spent nuclear fuel. It is extremely unlikely that significant processing activities on spent nuclear fuel will occur in the near future, and thus, new facilities would be required for foreign research reactor spent nuclear fuel management at the Hanford Site.

Two spent nuclear fuel EIS documents address the environmental impacts from spent nuclear fuel management at the Hanford Site. The first is the DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final EIS (SNF&INEL Final EIS) (DOE, 1995g), the Record of Decision of which was issued on May 30, 1995, that in general specifies spent nuclear fuel management throughout DOE; and in particular

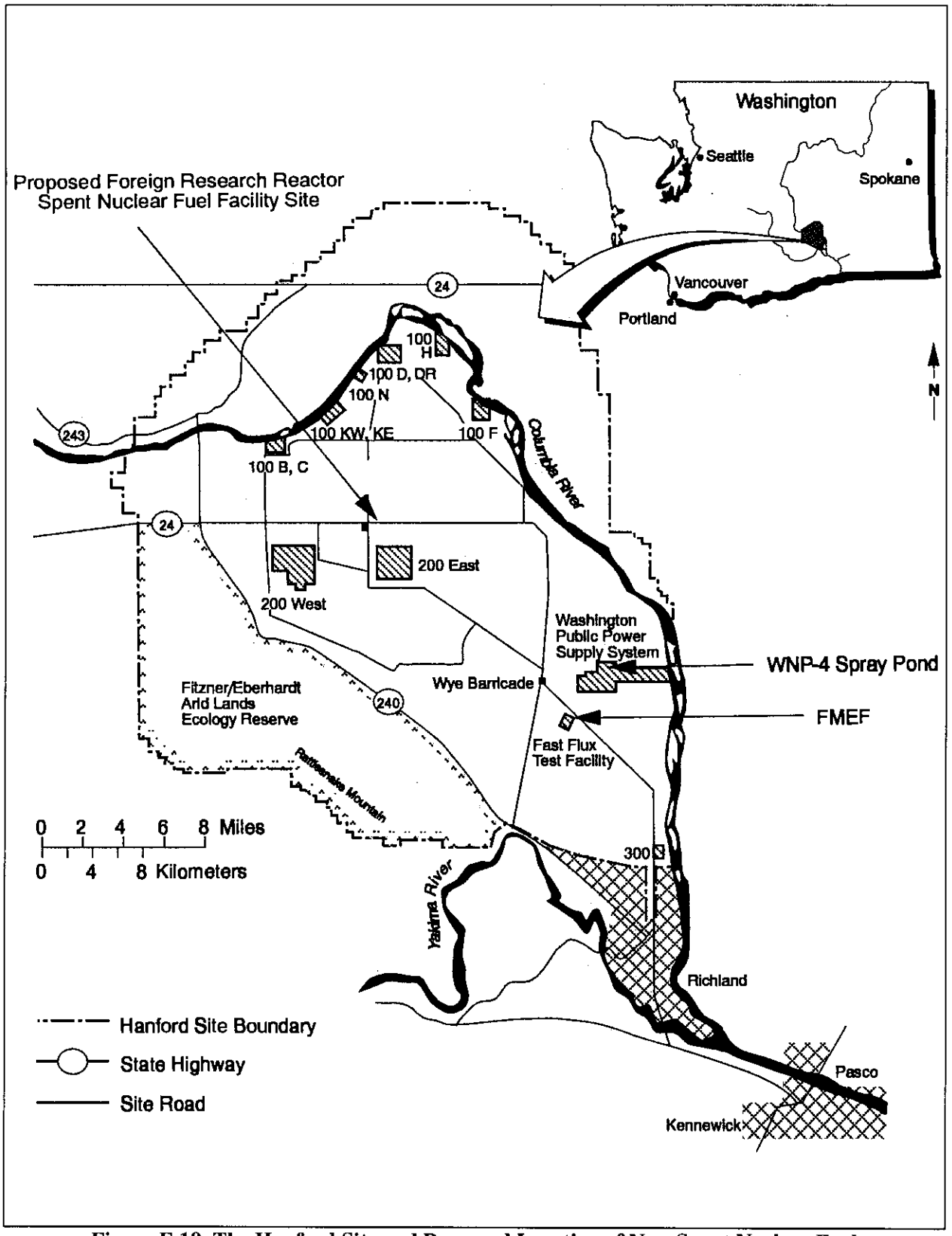


Figure F-19 The Hanford Site and Proposed Location of New Spent Nuclear Fuel Storage Facility

Table F-12 Description of Existing Spent Nuclear Fuel Facilities at Hanford Site

Facility	*Description*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel*	*Access*
105-KE Basin[a]	Water storage pool; 38 m x 20 m x 6 m deep, concrete walls and floor, no sealant or liner	75% - 100% full	By rail, 27 metric tons crane, fairly restrictive
105-KW Basin[a]	Water storage pool, 38 m x 20 m x 6 m deep, concrete walls and floor, epoxy sealant, no liner	75% full; no space for foreign research reactor spent nuclear fuel	By rail, 27 metric tons crane, fairly restrictive
T Plant: Cell 4	Water storage pool, 4 m x 8.4 m x 5.8 m deep	50% full; no space for foreign research reactor spent nuclear fuel	By rail or truck All fuel handling remote
PUREX Plant: East end of 202A Bldg, plus Dissolver Cells A, B, and C[b]	Water storage pool, 9.5 m x 6.1 m x 5.2 m deep, Dissolver Cell sizes vary	No additional capacity	Shipment by rail 36 metric tons crane
Plutonium Finishing Plant: 2736-ZB Bldg.	Dry storage in 208 L[c]	No additional capacity	Shipment by truck
FFTF: Reactor in-vessel storage, interim decay storage, and fuel storage facility locations[b]	Liquid sodium pool storage (fuel storage facility is separate from reactor containment building, with limit of kilowatts/assembly)	More than 75% full; no space for foreign research reactor spent nuclear fuel	By truck 91 metric tons Crane
200 Area LL Burial Grounds: 218-W-4C Trenches 1 and 7; and 218-W-3A Trenches 8 and S6	Dry, retrievable storage, 13 lead-lined, concrete-filled 208 liter drums, soil covered, 22 concrete casks (1.66 m x 1.66 m x 1.22 m or 1.92 m high), soil covered, 39 EBR II casks (1.5 m high x 0.4 m diameter), soil covered; 1 Zircaloy Hull Container (152 cm long x 76 cm diameter)	Large additional capacity; not suitable for foreign research reactor spent nuclear fuel	By truck
308 Building Annex: Neutron Radiography Facility[b]	Built in late 1970s water storage pool, 2.8 m diameter x 6 m deep	Small additional capacity	Truck shipments 4.5 metric tons crane
324 Building: B and D Cells	Dry storage in air, B Cell: 6.7 m x 7.6 m x 9.3 m high (spent nuclear fuel uses 10% of floor space). D Cell: 4 m x 6.4 m x 5.2 m high (small part for fuel), thick concrete walls and floors with steel liners	Small additional capacity	Truck shipments only B Cell - 2.7 and 5.4 metric tons cranes; Airlock - 27 metric tons crane
325 Building: A and B Cells in 325A Radiochemical Facility; 325B Shielded Analytical Laboratory	Dry Storage in air 325A - 1.8 m x 2.1 m x 4.6 m high (typical cell) 325B - 1.7 m x 1.7 m floor area (typical cell)	Small additional capacity	Truck shipments only 325A - 27 metric tons crane 325B - 2.7 metric tons crane
327 Building: A-F and I Cells; Upper and Lower SERF; Dry Storage vault, EBR II cask, Large Basin	Dry storage in air, except for water in small basin; variety of cell sizes, but storage only for fuel research	Small additional capacity	No direct rail Truck shipments 13.5 and 18 metric tons cranes

FFTF = Fast Flux Test Facility; EBR = Experimental Breeder Reactor; PUREX = Plutonium Uranium Extraction

[a] *If 105-KE Basin fuel is consolidated with 105-KW Basin fuel, 105-KE Basin would be shut down. The storage capacity of 105-KW Basin would be increased by replacing all of the storage racks to allow multi-tiered stacking of fuel canisters and by making minor facility modifications.*

[b] *Facility is being shut down.*

[c] *One 55 gal drum.*

specifies that Hanford generated spent nuclear fuel will remain in storage at Hanford pending decisions on ultimate disposition. The second, is the Management of Spent Nuclear Fuel from the K Basins at the Hanford Site Draft EIS (DOE, 1995d) which was issued for comment in October 1995. This EIS addresses the location and method of managing Hanford spent nuclear fuel for up to 40 years or until decisions on ultimate disposition are made.

New facilities would be required for storage of foreign research reactor spent nuclear fuel. However, there may be some economies of scale achieved from overlap with the other spent nuclear fuel activities at the Hanford Site.

F.1.3.3.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Hanford Site for Foreign Research Reactor Spent Nuclear Fuel

The Hanford Site has concluded that there are no existing facilities available and ready for accepting foreign research reactor spent nuclear fuel (Bergsman et al., 1994). Consequently, the Hanford Site proposes the following strategies:

- construction of a "generic" modular dry vault or dry cask storage facility,
- construction of a "generic" wet storage and handling facility,
- modification and completion of the Fuel Maintenance and Examination Facility (FMEF) (located at the Fast Flux Test Facility) as a modular dry vault storage facility, and
- acquisition, modification, and completion of the Washington Nuclear Plant-4 Spray Cooling Pond (at the Washington Public Power Supply System) as a wet storage facility.

These facilities and their potential applications to foreign research reactor spent nuclear fuel storage are discussed in detail in Section F.3.

Figure F-20 displays the Hanford Site capacity for foreign research reactor spent nuclear fuel storage. The Hanford Site is not considered capable of immediately accepting foreign research reactor spent nuclear fuel because of the required construction of new facilities. The Hanford Site would have sufficient capacity for foreign research reactor spent nuclear fuel storage after new facility construction.

F.1.3.4 Oak Ridge Reservation

The Oak Ridge Reservation is located on approximately 140 km^2 (54 mi^2) of federally owned land near Knoxville, TN (DOE, 1995g). There are three primary plant complexes within the Oak Ridge Reservation:

- *Y-12 Plant:* produces various materials used for national defense purposes,
- *K-25 Site (formerly called the Oak Ridge Gaseous Diffusion Plant):* originally used for uranium enrichment and now an environmental management site, and
- *Oak Ridge National Laboratory (also known as X-10):* research and development into nuclear energy and other energy technologies.

The Oak Ridge National Laboratory has operated several small reactors for research and isotope production and, of the Oak Ridge Reservation sites, is the most familiar with spent nuclear fuel

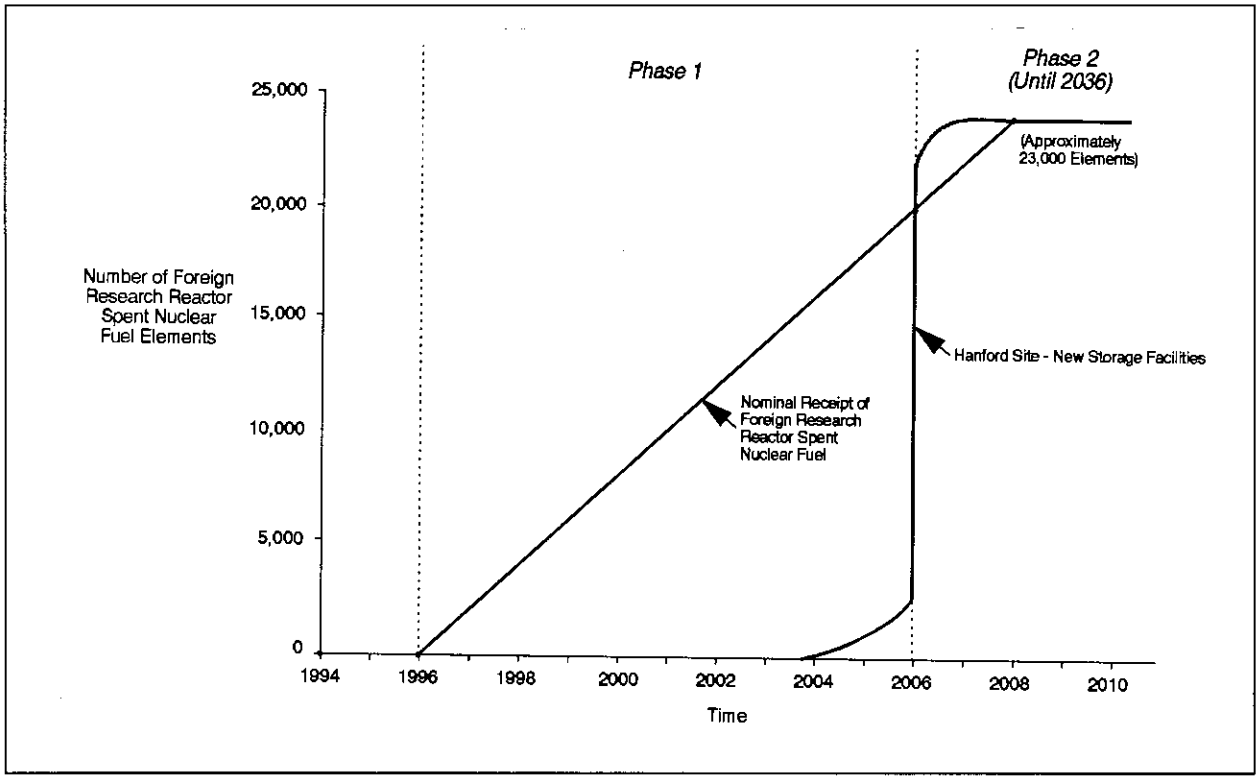


Figure F-20 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Hanford Site

requirements. A map of the Oak Ridge Reservation and its candidate sites for foreign research reactor spent nuclear fuel storage is presented in Figure F-21.

F.1.3.4.1 Spent Nuclear Fuel Activities at the Oak Ridge Reservation

Most Oak Ridge Reservation spent nuclear fuel activities occur at the Oak Ridge National Laboratory. The Oak Ridge National Laboratory has operated several small research reactors, all of which generate (or have generated) spent nuclear fuel. These reactors all have small fuel preparation and handling facilities associated with them ranging up to the single digit MTHM capacity. The spent nuclear fuel storage space is small, and most is either full or committed, with little excess capacity. The Oak Ridge National Laboratory also has hot cell and irradiated fuel examination facilities. Currently, only the High Flux Isotope Reactor is operating and generating spent nuclear fuel. More spent nuclear fuel facilities at Oak Ridge Reservation are presented in Table F-13.

F.1.3.4.2 Spent Nuclear Fuel Storage Facilities at the Oak Ridge Reservation

The Oak Ridge Reservation stores spent nuclear fuel in several small facilities. Most of these facilities are old and are unlikely to meet modern building code and seismic standards. The spent nuclear fuel facilities include the following structures:

- *Building 3525 - Irradiated Fuels Examination Laboratory:* This two-story brick structure was constructed in 1963. It houses hot cells and contains small quantities of irradiated research reactor fuel in the form of samples and targets.

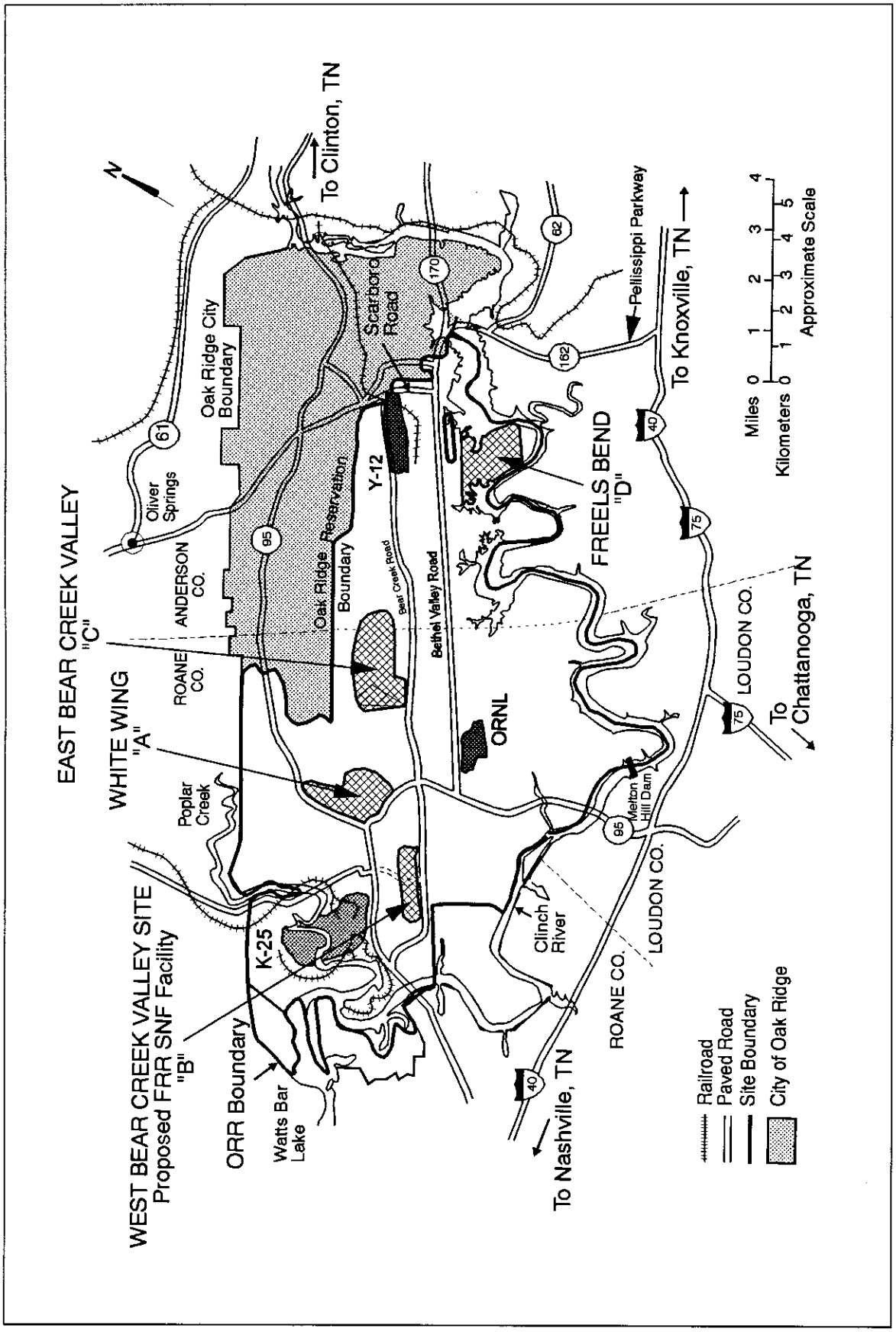


Figure F-21 Candidate Sites at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel Storage

Table F-13 Major Spent Nuclear Fuel Facilities at the Oak Ridge Reservation

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel*	*Access*
Building 3525	Hot Cells	No, too small	Truck
Building 4501	Hot Cells	No, too small	Truck
Building 7827	Drywells	No space	Truck
Building 7920	Hot Cells	No, too small	Truck
Building 9720-5 (Y-12)	Warehouse	No, unirradiated fuel only	Truck
Other	Research Reactors	No, storage space near capacity	Truck

- *Building 4501 - High-Level Radiochemical Facility:* This facility dates from 1951 and contains hot cells for examining irradiated materials. This facility contains small quantities (several kg) of sectioned commercial fuel.
- *Building 7920 - Radiochemical Engineering Development Center:* This is a multi-purpose, hot cell facility for (relatively) large quantities of irradiated spent nuclear fuel. This facility supports target preparation and processing for the High Flux Isotope Reactor and contains samples and targets of research reactor spent nuclear fuel in dry storage.
- *Building 9720-5 (Y-12):* This is a large warehouse for storing and safeguarding unirradiated or low burnup HEU fuel. It currently contains around 0.2 MTHM.
- *Research Reactors:* There are five existing and one planned research reactor at the Oak Ridge Reservation. All of these reactors have small spent nuclear fuel storage basins nearby, and this capacity is essentially full. Only the High Flux Isotope Reactor is currently operating.
- *The Oak Ridge Reservation* also has several drywells such as Building 7827 and drum storage areas for irradiated fuel. Spent nuclear fuel would be relocated in accordance with actions of the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g).

None of these locations have any significant capacity for the potential quantities of foreign research reactor spent nuclear fuel.

F.1.3.4.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel

The Oak Ridge Reservation has plans for dry storage of spent nuclear fuel. This would be accomplished via a modular route at the High Flux Isotope Reactor location. This dry storage area could be extended almost indefinitely to accommodate the Oak Ridge Reservation's needs.

DOE is evaluating a spent nuclear fuel management complex for handling DOE spent nuclear fuel from other sites as an alternative in the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel management complex would include the following:

- Spent Nuclear Fuel Receiving and Canning Facility,
- Technology Development Facility,
- Interim Dry Storage Facility, and

- Expended Core Facility for Naval-type fuel similar to the one at Idaho National Engineering Laboratory.

The receiving and canning facility would receive spent nuclear fuel cask shipments from offsite and prepare the spent nuclear fuel for dry storage. The facility incorporates a pool (wet) storage facility for cooling spent nuclear fuel (tentatively identified as a 5-year period) prior to placement into dry storage, as necessary. The technology development facility would investigate the applicability of dry storage technologies and pilot scale technology development for disposal for various types of spent nuclear fuel. The interim dry storage area would consist of passive storage modules to safely store the spent nuclear fuel for 40 years. Naval fuel would be examined at the Expended Core Facility prior to interim storage. The total land required for the facility, including a buffer zone, is approximately 36 ha (90 acres).

The proposed site for the spent nuclear fuel facilities is located in the West Bear Creek Valley Area, in the western portion of the Oak Ridge Reservation site. This area of the Oak Ridge Reservation is currently in the Natural Areas land use category and is designated for future Waste Management land use. Land uses bordering on the Oak Ridge Reservation in this area are primarily agricultural farmland and commercial forest, with sparsely located residences (i.e., low population density).

Environmental, safety, and health consequences are calculated to be negligible from the spent nuclear fuel facilities, although a preliminary design and/or layout is not provided. Releases of krypton-85, chlorine, and hydrogen fluoride are included in the analysis for incident-free operations, but the source of these emissions is not reported. Facility budgetary requirements are not delineated.

Foreign research reactor spent nuclear fuel represents less than one percent of the DOE spent nuclear fuel quantities in terms of mass and, thus, its effect would be minimal as compared to the other fuels. The foreign research reactor spent nuclear fuel contribution to the operational consequences and its costs are not delineated. Figure F-22 summarizes foreign research reactor spent nuclear fuel capacity at the Oak Ridge Reservation. New facility construction would be required for foreign research reactor spent nuclear storage.

F.1.3.5 Nevada Test Site

The Nevada Test Site is located in the southeastern part of the State of Nevada, and is used as the on-continent site for nuclear weapons testing (DOE, 1995g). The Nevada Test Site encompasses approximately 3,500 km^2 (1,350 mi^2) of desert land, with flats, mesas, and mountain ridges (Figure F-23). Essentially no permanent surface waters exist, and the depth to groundwater routinely exceeds 330 m (1,000 ft). The Nellis Air Force Base Range surrounds the Nevada Test Site to the north, east, and west; and, with the Tonapah Test Range, provides a 24 to 104 km (15 to 65 mi) buffer zone between the Nevada Test Site and public lands. The Bureau of Land Management owns land on the southern and southwestern borders of the Nevada Test Site. Principal access to the site is via the town of Mercury, on the southeastern corner. Las Vegas is approximately 104 km (65 mi) from this corner of the Site.

Activities at the site have included nuclear weapons testing, nuclear reactor tests, nuclear rocket engine development, and waste management. Current activities include nuclear weapons-related activities (e.g., emergency search teams, arms control/verification, etc.), low-level waste/low-level mixed waste disposal, and site characterization for commercial spent nuclear fuel disposal.

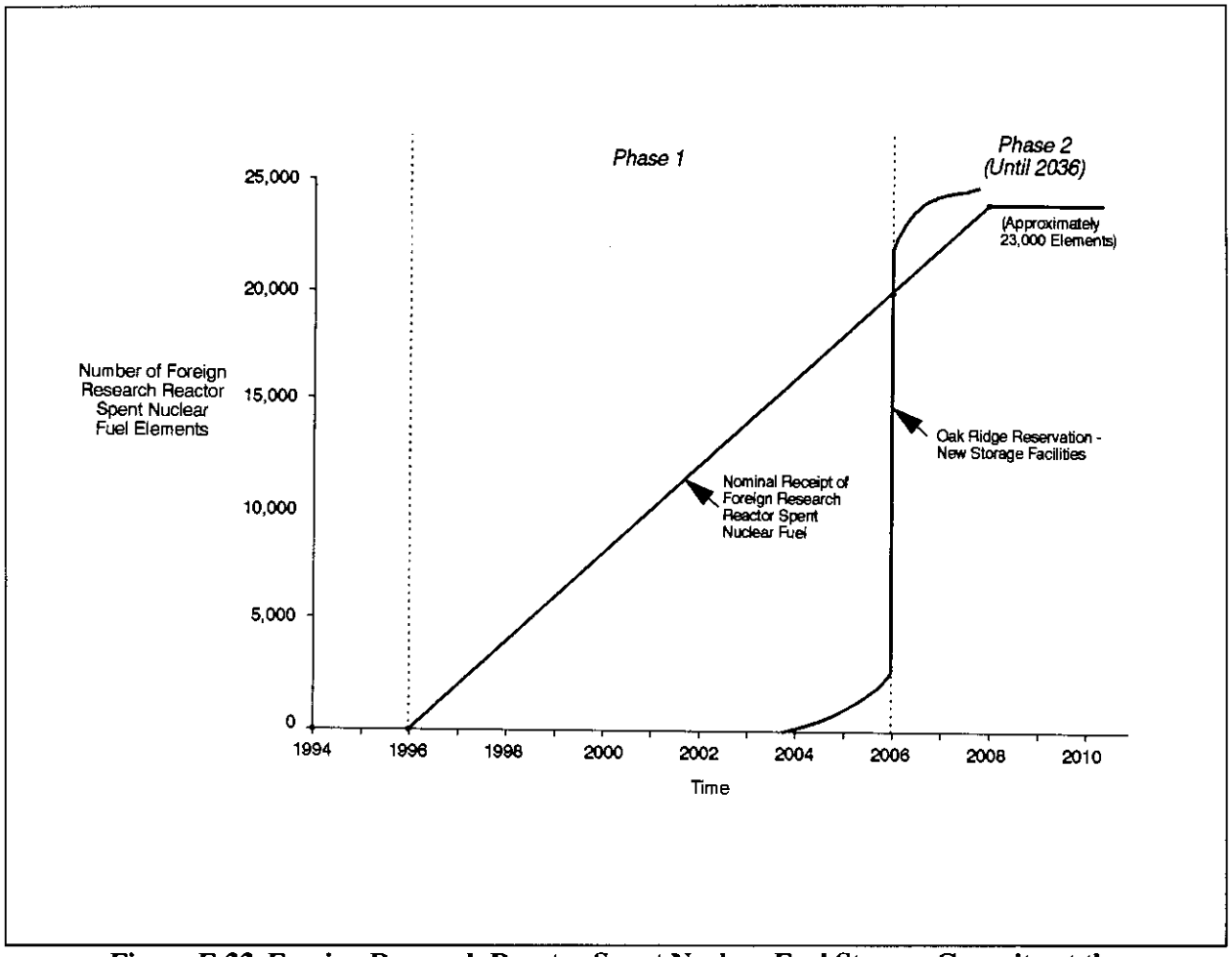


Figure F-22 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Oak Ridge Reservation

F.1.3.5.1 Spent Nuclear Fuel Activities at the Nevada Test Site

The Nevada Test Site has several existing facilities that could be useful for spent nuclear fuel management. These facilities were principally used for nuclear rocket engine development and are located at Jackass Flats, in a southern portion of the Nevada Test Site called the Nevada Research & Development Area (Cosimi, 1994; Chandler et al., 1992; Gertz, 1994; Hynes, 1994; Reed, 1994). The facilities include several large hot cell and fuel examination "shops," with large cranes and manipulators. At least two of these facilities appear to be ideally suited for handling and storing foreign research reactor spent nuclear fuel after relatively minor upgrades and refurbishments. Table F-14 summarizes the capabilities of these facilities for foreign research reactor spent nuclear fuel.

The Engine Maintenance and Disassembly (E-MAD) facility was originally constructed for the assembly and preparation of nuclear rocket engines for testing, refurbishment of activated engines, and disassembly and inspection of tested engines and components. The facility is designed for remote handling and examination of highly radioactive components. The building is a T-shaped, multi-storied structure, with overall dimensions of 85 x 107 m (280 ft x 350 ft) (Figure F-24). Numerous hot cells exist, with remote handling and transfer equipment, and the largest hot cell is 20 m wide x 45 m long x 23.5 m high (66 ft wide by 146 ft long and 77 ft high). Typically, 1.5 m (5 ft) thick concrete walls provide the shielding. Material transfer capabilities include several 36 metric tons (40 ton) cranes and a cask handling system of

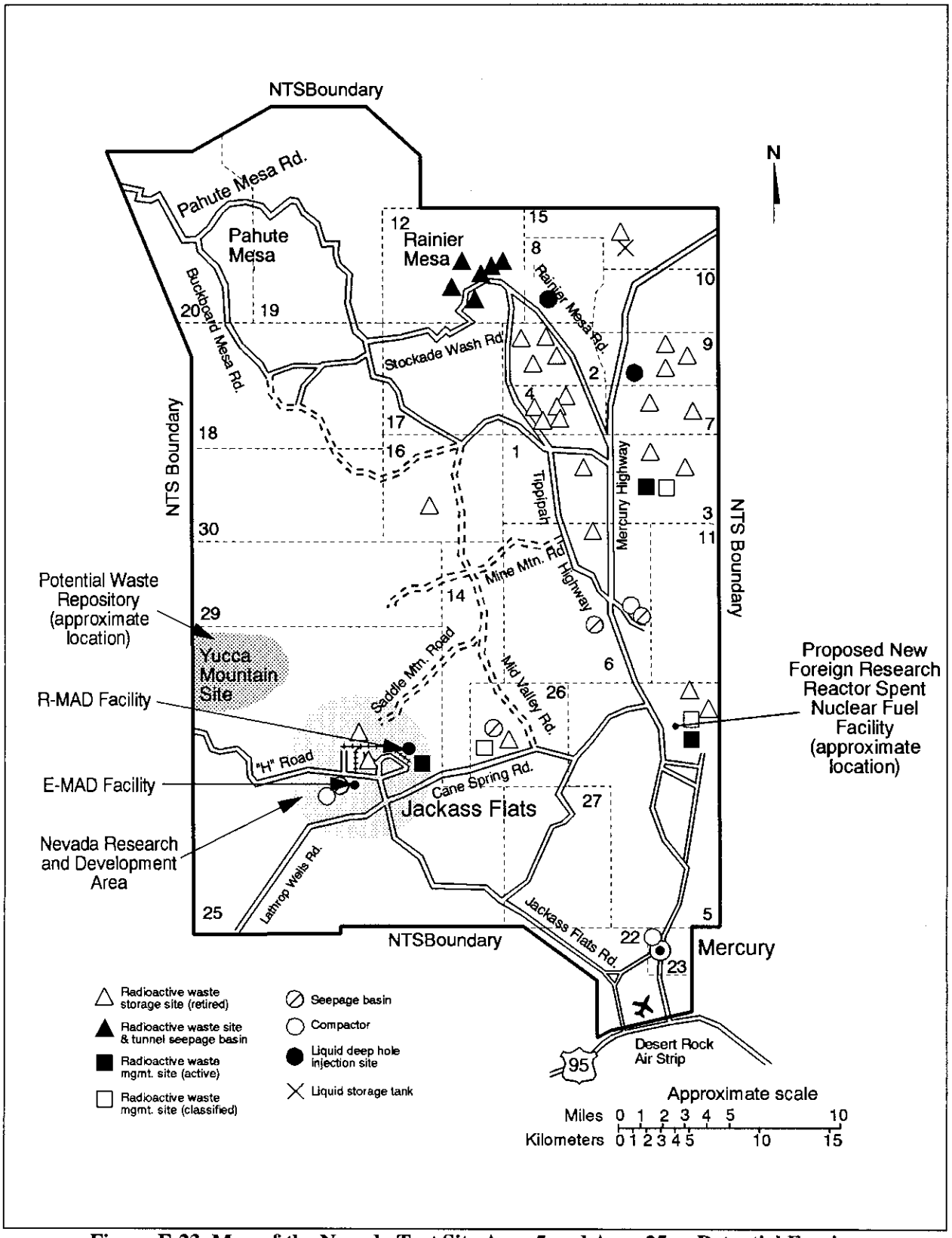


Figure F-23 Map of the Nevada Test Site Area 5 and Area 25 as Potential Foreign Research Reactor Spent Nuclear Fuel Storage Areas

Table F-14 Major Spent Nuclear Fuel-Capable Facilities at the Nevada Test Site

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel*	*Access*
E-MAD	Large hot cell, with smaller hot cells. Main hot cell area is 895m^2 (9,600 ft^2)	Yes, as either a vault or a staging facility for dry casks, after 1-2 years refurbishments, 25,000 elements	Truck
R-MAD	Large hot cell, with smaller hot cells. Main hot cell area is 223m^2 (2,400 ft^2)	Yes, as a staging facility for dry casks or a small vault, after 1-3 years refurbishment, 25,000 elements	Truck

approximately 91 metric tons (100 tons) capacity. The heating, ventilation, and air conditioning systems for the hot cell areas maintain negative pressure and exhaust through High Efficiency Particulate Air filters.

The E-MAD facility is currently unused and last saw service during the 1980s for commercial spent nuclear fuel storage experiments (e.g., Climax Mine Project) (Gertz, 1994). Thirteen commercial spent nuclear fuel assemblies were tested in casks and drywells. The E-MAD facility was subsequently used to load transportation casks for shipment of the spent nuclear fuel to Idaho. Several of these spent nuclear fuel storage casks remain at the site (Hynes, 1994). The Los Alamos National Laboratory assessment (Chandler et al., 1992) considers the facility to require only minor upgrades and routine maintenance.

The Reactor Maintenance and Disassembly facility is located a short distance from the E-MAD facility. This facility contains two (contact) assembly bays and one remotely operated hot disassembly bay. The hot bay dimensions are 18 x 12 x 18 m (60 by 40 by 60 ft) high, with 1.8 m (6 ft) thick walls for shielding. A transfer system connects six hot cells to the hot disassembly bay. The Los Alamos National Laboratory assessment (Chandler et al., 1992) found the Reactor Maintenance and Disassembly facility to require a minor upgrade.

F.1.3.5.2 Spent Nuclear Fuel Storage Facilities at the Nevada Test Site

At the present time, the Nevada Test Site is not storing spent nuclear fuel. As noted, facilities in the Jackass Flats area have handled spent nuclear fuel in the past and could be adapted to accommodate foreign research reactor spent nuclear fuel and serve as the nucleus of a spent nuclear fuel storage facility. The E-MAD and Reactor Maintenance and Disassembly facilities appear to have sufficient size and design for accommodating all of the foreign research reactor spent nuclear fuel in a dry storage mode, either vault or cask, and for accomplishing any required transfer, examination, and canning operations.

F.1.3.5.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Nevada Test Site for Foreign Research Reactor Spent Nuclear Fuel

Besides the Area-25 facilities, DOE evaluated an elaborate spent nuclear fuel handling system as an alternative in the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel management complex would be located in Test Area 5, near the eastern border of the site, and in the general proximity of the low-level waste/low-level mixed waste disposal areas. The spent nuclear fuel complex would include:

- Spent Nuclear Fuel Receiving and Canning Facility,
- Technology Development Facility,
- Interim Dry Storage Area, and
- Expended Core Facility, similar to the one at Idaho National Engineering Laboratory.

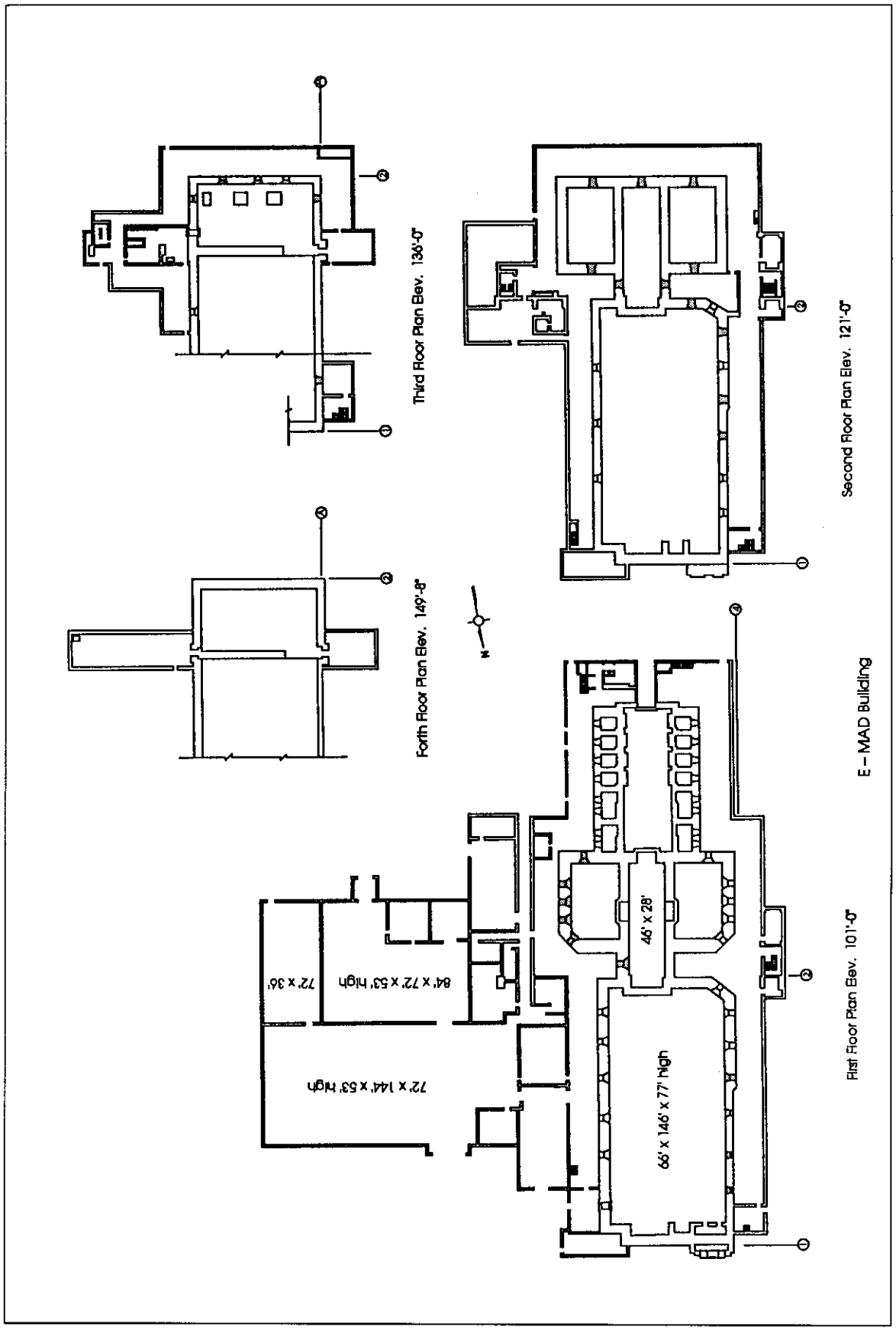


Figure F-24 Schematic of the Engine Maintenance and Disassembly Facility at the Nevada Test Site

The receiving and canning facility would receive spent nuclear fuel cask shipments from offsite and prepare the spent nuclear fuel for dry storage. The facility incorporates a pool (wet) storage facility for cooling spent nuclear fuel (tentatively identified as a 5-year period) prior to placement into dry storage, as necessary. The technology development facility would investigate the applicability of dry storage technologies and pilot scale technology development for disposal for various types of spent nuclear fuel. The interim dry storage area would consist of passive storage modules to safely store the spent nuclear fuel for 40 years. Naval fuel would be examined at the Expended Core Facility prior to interim storage. The total land required for the facility, including a buffer zone, is approximately 36 ha (90 acres).

Environmental, safety, and health consequences are calculated to be negligible from the spent nuclear fuel facilities, although a preliminary design and/or layout is not provided. Releases of krypton-85, chlorine, and hydrogen fluoride are included in the analysis for incident-free operations, but the source of these emissions is not reported. Facility budgetary requirements are not reported.

Foreign research reactor spent nuclear fuel represents less than 1 percent of the DOE spent nuclear fuel quantities in terms of mass (i.e., potential source term), and about 10 percent in terms of volume. Thus, its effect would be minimal as compared to the other fuels. The foreign research reactor spent nuclear fuel contribution to the operational consequences and its costs are not delineated.

Figure F-25 summarizes the Nevada Test Site storage capabilities for foreign research reactor spent nuclear fuel. The Area-25 facilities could all receive and provide for dry storage of foreign research reactor spent nuclear fuel close to the proposed Yucca Mountain repository. It should be noted that these facilities have comparable shielded floor areas and volumes as compared to the generic modular dry vault

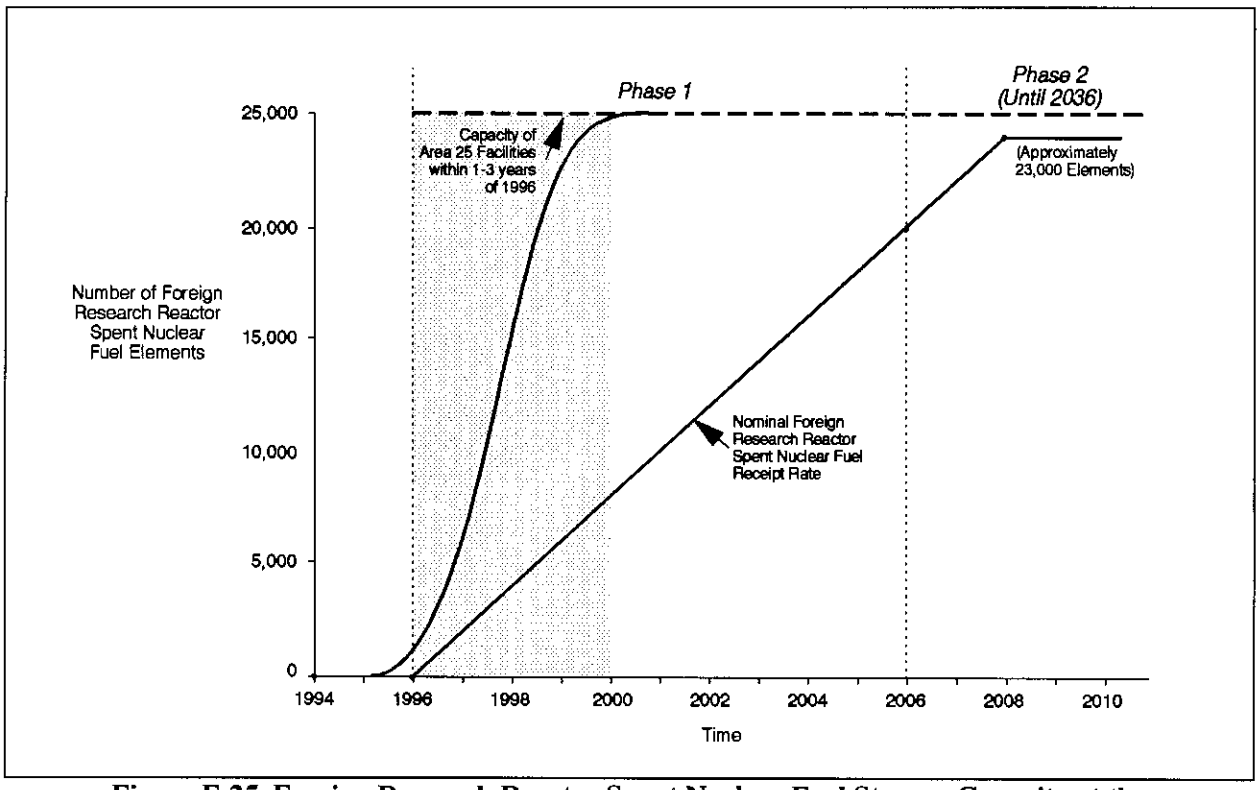


Figure F-25 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Nevada Test Site

for foreign research reactor spent nuclear fuel discussed in Section F.3. Alternatively, new facilities could be built, but these would require a longer transportation path to the proposed Yucca Mountain repository.

F.1.3.6 Storage at Overseas Facilities

Currently, foreign research reactor spent nuclear fuel is being stored in wet pools at foreign research reactor sites. These pools are approaching the levels of their capacity, which is why the foreign research reactor operators would like the United States to accept their spent nuclear fuel. An alternative being considered by DOE is foreign research reactor spent nuclear fuel storage at overseas facilities. Several facilities exist in Europe for contractual storage of both commercial and research reactor spent nuclear fuel for a fee, including:

- British facilities at Dounreay, Scotland and Sellafield, England. The former has several small pools for research reactor fuels, while the latter has several large pools with a capacity of 3,000 MTHM for commercial spent nuclear fuel (Bonser, 1994).
- French facilities at La Hague, with several large pools having a total capacity of 14,000 MTHM for commercial spent nuclear fuel (Nuclear Fuel, 1993); facilities at Marcoule, for research and metallic spent nuclear fuel.

Electricite De France has also announced its intention of constructing a commercial spent nuclear fuel wet storage facility with a capacity of 12,000 MTHM (Nuclear Fuel, 1994b). Dry storage of spent nuclear fuel is also being considered.

These facilities are predominantly stainless-steel lined wet storage pools that meet modern seismicity and confinement standards and maintain good water chemistry. Wet storage pools designed for commercial spent nuclear fuel could, after license modification and new rack installation, store foreign research reactor spent nuclear fuel. These overseas wet storage pools are similar in design and layout to the generic wet storage facility discussed in Section F.3.

F.1.4 Vitrified Waste Storage Facilities

If foreign research reactor spent nuclear fuel is processed, the resulting high-level waste would be vitrified and placed into stainless steel canisters. The Savannah River Site is the only domestic site that currently has a storage facility designed and built for storing vitrified high-level waste from the processing of spent nuclear fuel. This facility is termed the Glass Waste Storage Building, and it is located immediately adjacent to the Savannah River Site vitrification facility (the Defense Waste Processing Facility), in the S-Area of the site near the H-Area processing facilities (DOE, 1994g). Figure F-26 provides a general overview of the facilities in the S-Area. Figure F-27 displays a general layout of the building. The Glass Waste Storage Building is designed to accommodate the standard Defense Waste Processing Facility vitrified waste canister (Figure F-28). The existing building has space for 2,286 of these canisters. A second, almost identical building, is planned for construction starting in 2007. Additional buildings may be built, up to a total interim storage capacity of 10,000 canisters if delays in the Federal Repository Program are encountered (DOE, 1994g). The Defense Waste Processing Facility/Glass Waste Storage Building area does not currently include a cask receiving/shipping facility, but one is planned for future construction.

The facility is relatively simple in design and operation. It consists of a structure enclosing a concrete floor that functions as the charging face to the vault beneath it. Shield plugs are removed from the floor to provide access to storage tubes in the vaults that would contain the canisters. Each storage tube contains

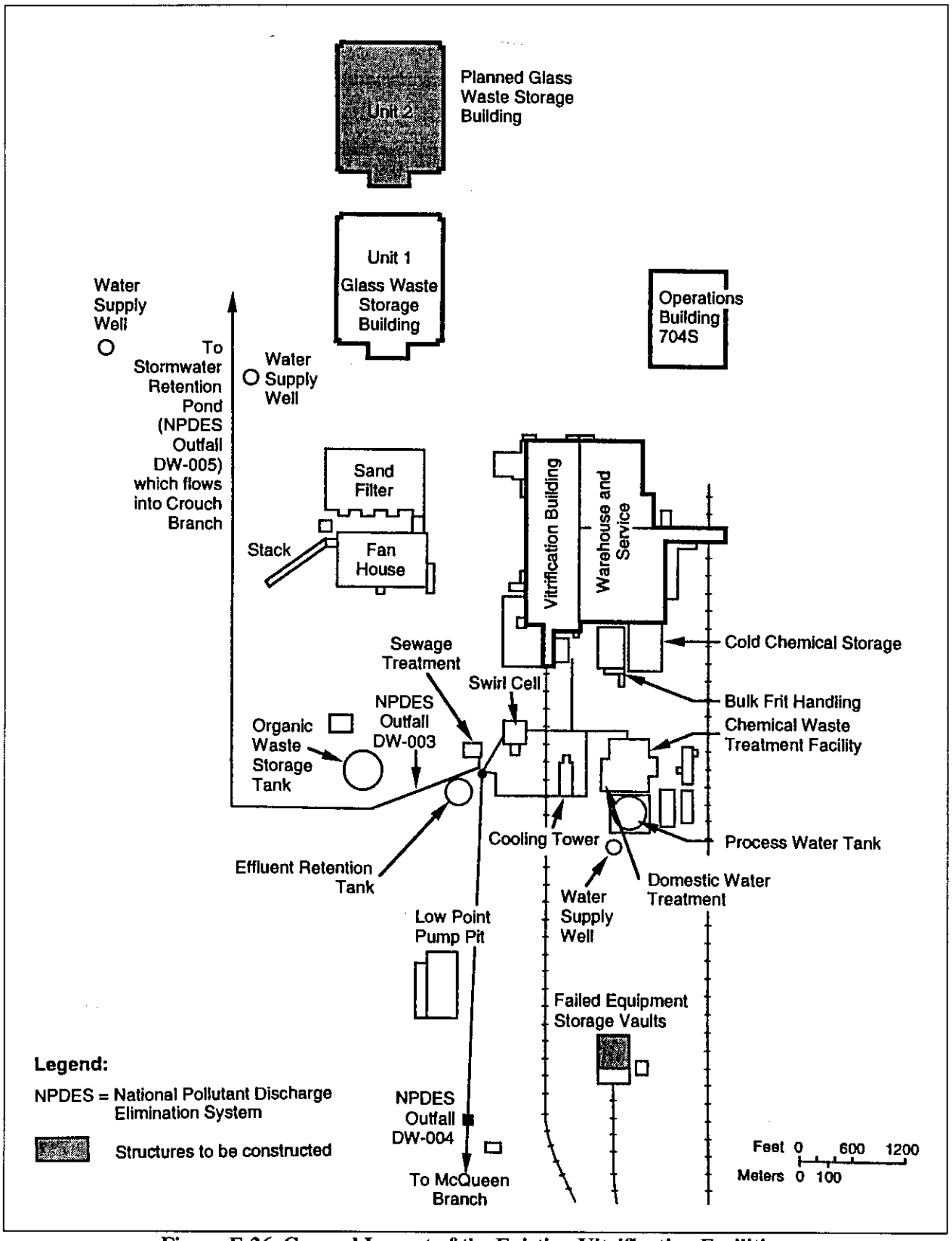


Figure F-26 General Layout of the Existing Vitrification Facilities at the Savannah River Site

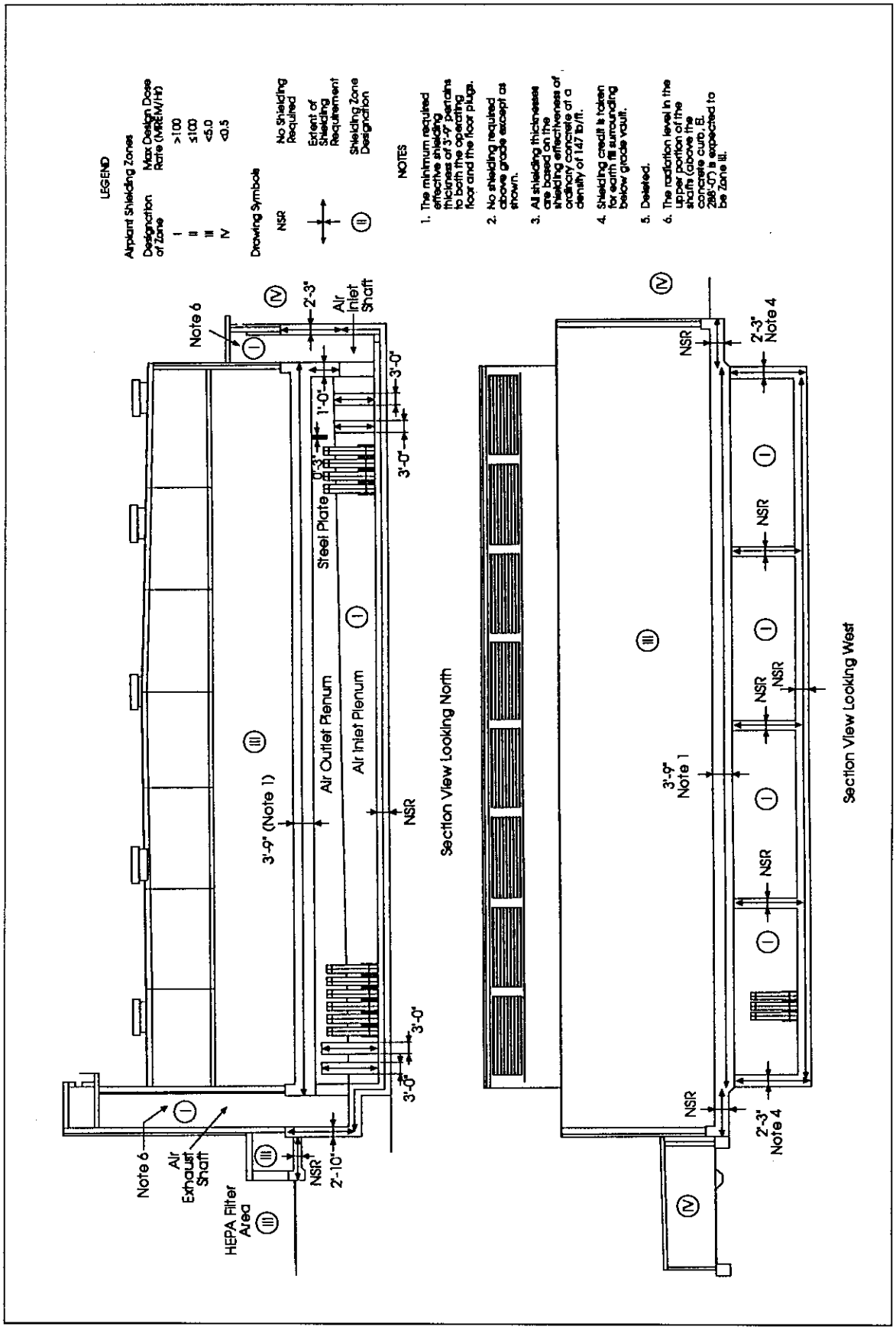
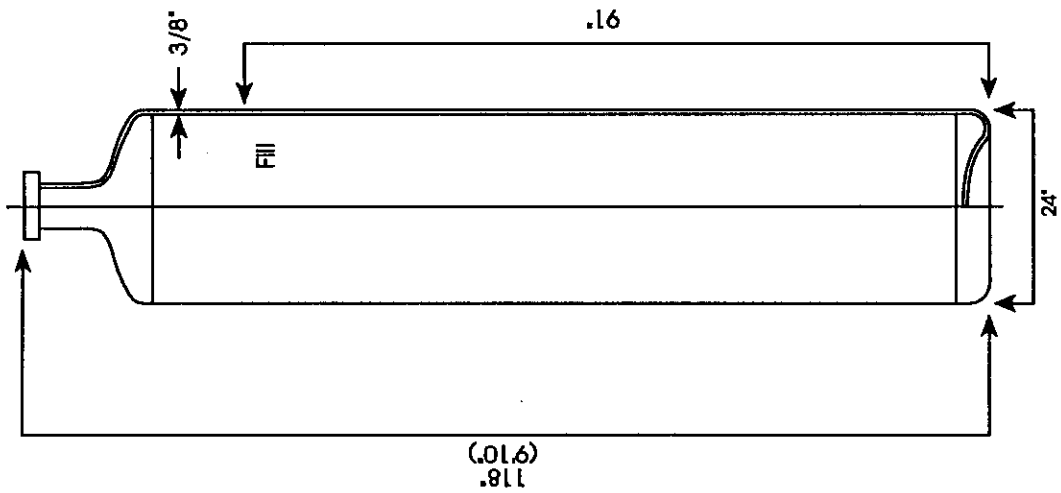


Figure F-27 Layout of the Glass Waste Storage Building



Material:	304L Stainless Steel
Empty Weight:	1,100 lb
Net Weight:	3,540 lb (Glass)
Weight of Radionuclide:	41 kg (90 lb)
Activity:	8.3×10^{15} Bq (223,000 Ci)
Decay Heat:	660 W
Radiation Field (at Surface):	6,300 rad/hr
Surface Contamination:	Less than 10^{-4} µCi/cm^2

Figure F-28 Typical Defense Waste Processing Facility Glass Waste Canister

two canisters, stacked vertically. The vault area consists primarily of steel-reinforced concrete and is designed to resist all earthquake and severe weather incidents.

Radioactive decay heat from the canisters is removed by the Glass Waste Storage Building's forced air fan exhaust system. The exhaust air is drawn around the canisters and then exhausted through the building's High Efficiency Particulate Air filtered ventilation system and discharged to the atmosphere via a stack. No condensate is expected to form, although the building does include a sump for exhaust air condensate. No radioactivity is expected in the exhaust air or in any condensates that might form.

During operation, a special dedicated transporter vehicle moves the canister from the Defense Waste Processing Facility vitrification building to the Glass Waste Storage Building in a shielded transporter. The transporter's cask is placed over the appropriate vault borehole, the shield plug is removed, and the canister is lowered via a crane mechanism into the borehole. The shield plug is replaced, and the transporter returns to the vitrification plant for the next shipment.

Several overseas facilities also exist for vitrified waste storage at Marcoule (France), La Hague (France), and Sellafield (England) (COGEMA, 1994a and 1994b; BNFL, 1994a and 1994b). These facilities are designed as natural circulation vaults and, thus, do not require fans for storage cooling. These vaults use a smaller canister, with several thousand currently in storage.

F.2 Storage Technology Evaluation Methodology

The selection of a spent nuclear fuel storage technology for foreign research reactor spent nuclear fuel requires a multi-disciplinary approach including the evaluation of, at the minimum, the environmental impact of alternatives and the following key design and performance areas:

- chemical compatibility,
- subcriticality assurance,
- shielding effectiveness,
- structural integrity (i.e., containment),
- thermal performance,
- ease of use,
- cost, and
- regulatory basis and licensing.

Other factors that may affect the decision process are whether the design has been previously licensed and actually used to store spent nuclear fuel, and its perceived ability to meet applicable regulations and standards if it has not yet been licensed.

Two principal types of spent nuclear fuel storage can be used for foreign research reactor spent nuclear fuel, wet and dry. Wet storage denotes the immersion of fuel in a pool of water, which performs the dual functions of shielding/leaking radionuclide removal and decay heat removal, but which relies on active systems. Dry storage encompasses a wide spectrum of structures that house the fuel in a dry inert gas environment, with an emphasis on passive system design and operation.

F.2.1 Chemical Compatibility

The most important criterion in assessing foreign research reactor spent nuclear fuel storage technologies is the compatibility of the spent nuclear fuel with the fuel storage technology environment. The research reactor fuel cladding is either aluminum or stainless steel. Aluminum cladding fuel is the predominant type in the mix of foreign research reactor spent nuclear fuel being considered for acceptance in this EIS. The selected method of storage for any foreign research reactor spent nuclear fuel that may be accepted must provide a benign and noncorrosive environment for the fuel.

In reviewing the corrosive potential of aluminum, acidic, alkaline, and even many neutral chemical solutions have been found to be significantly corrosive. Therefore, the use of wet storage technology for the majority of the foreign research reactor spent nuclear fuel that contains aluminum cladding would require the maintenance of high water purity throughout the storage life of the pool, which may equal or exceed 40 years.

Unlike wet storage, most of the dry storage technologies utilize a dry inert gas atmosphere for the fuel, which is a noncorrosive noble gas that also enhances conduction heat transfer from the fuel to the encapsulating container. Some dry storage technologies use dry nitrogen instead of inert gas. Even in the event of a loss of inert gas atmosphere, the air atmosphere would be less corrosive than a less-than-high-purity water pool. Finally, previous experience at many DOE wet storage facilities has shown that poor water quality dramatically deteriorates the integrity of aluminum fuel. Thus, the chemical compatibility criterion indicates that dry storage is a more appropriate technology than wet storage for extended storage of foreign research reactor spent nuclear fuel.

F.2.2 Subcriticality Assurance

Uranium and plutonium are the principal elements that have the unique ability to split or fission after absorbing a neutron, and release energy and several new neutrons from this fission process. The particular forms or isotopes of uranium that are effective in the fission process are called fissile materials and are ^{233}U, ^{235}U, and ^{239}Pu. Of these three isotopes, only ^{235}U exists naturally, while the other two isotopes can be produced artificially. Under the right conditions, the fission process can be self-sustaining or even grow by a chain reaction. This chain reaction produces as many or more neutrons than are absorbed in an assembly of fissile materials.

In nuclear engineering terminology, the numerical measure of a mass of fissile material to achieve and maintain a self-sustaining fission chain reaction is termed K-effective. K-effective is the net ratio of neutrons produced per neutron absorbed in the fissile material mass. When K-effective equals 1.0, the mass is said to be critical because it can maintain the fission process. When K-effective is less than 1.0, the mass is considered to be subcritical.

Subcriticality can be ensured by a number of factors, including:

- diluted concentration of fissile materials,
- adequate separation distance between masses of fissile materials such as nuclear fuel rods or assemblies,
- presence of materials (such as boron) mixed with the fissile material that absorbs neutrons before they can be captured by the fissile material,

- exclusion of substances such as water that can encourage the absorption of neutrons by the fissile isotopes or reflect neutrons leaving the mass of fissile materials back into the mass, and
- restricting the mass of fissile material below the minimum that nature requires to initiate, maintain, and/or sustain a fission chain reaction.

In nuclear criticality safety, the principle of double contingency is used to protect against criticality. Double contingency requires that the design of any system containing fissile material use two of the aforementioned factors to prevent the onset of criticality. Criticality analyses would be required to confirm spacing and the effects of optimum moderation, as well as the different structural materials in foreign research reactor spent nuclear fuel as compared to commercial fuel (e.g., aluminum, stainless steel, hydride, inconel in foreign research reactor spent nuclear fuel as compared to zircaloy in commercial fuel).

Another important factor is that most of the products of the fission reaction are radioactive fission products that are not capable of sustaining a fission reaction. Like most elements in nature, these fission products can absorb neutrons, but do not produce any neutrons or energy during this absorption. Mixed with these fission products is a small amount of fissile ^{239}Pu, which was also created during the fission process. However, the sum of the remaining ^{235}U and the created ^{239}Pu is much smaller than the original quantity and concentration of ^{235}U in the new fuel. The relatively low (in comparison to its initial value) ^{235}U enrichment and ^{239}Pu present in foreign research reactor spent nuclear fuel, coupled with the presence of neutron-absorbing fission products, greatly reduces the physical ability of foreign research reactor spent nuclear fuel to become critical.

Another important aspect in selecting an appropriate storage technology is the maintenance of subcriticality. For wet (pool) storage, subcriticality is ensured by fuel spacing, and in some cases, the use of spacing plates between adjacent fuel assemblies that contain boron. In addition, control of the maximum allowable concentration of fissile isotopes (i.e., ^{235}U enrichment) is another method used to control subcriticality. Since all dry storage technologies use a storage canister for fuel, the subcriticality design relies on controlling the fissile material inventory, fuel spacing, and if necessary, the use of neutron-absorbing materials. The subcriticality control design of all fuel storage technologies is acceptable and does not provide any discriminating factors for selecting one technology over another.

F.2.3 Shielding Effectiveness

Shielding effectiveness design impacts both onsite worker and public dose rates during the loading and subsequent storage of spent nuclear fuel. Both neutron and gamma ray shielding must be provided and ensured throughout the life of the storage facility. Wet storage technology uses pool water as a shield, which effectively reduces both neutron and gamma doses to acceptable levels. The only weakness of this shielding design is any event in which the water could be lost due to a leak in the pool wall or, if the design includes piping at a low enough level, a pipe break in a line connected to the pool for the water purification or decay heat removal systems. Precluding piping or wall leakage, spent nuclear fuel pool water is an inexpensive shield medium that offers the advantage of providing visual inspection of the stored fuel as long as water purity and clarity are maintained.

Dry storage technology relies on a number of solid shielding materials, sometimes in combination, to reduce gamma and neutron dose rates. The most common materials are different forms of concrete (i.e., low-density, high-density, hydrogenated), cast iron, carbon or stainless steel, lead, borated resin, and polyethylene (for neutrons). As with water, these materials have been widely used in the nuclear industry

for shielding and their properties are well known. They are more costly than pool water and prevent visual inspection of the spent nuclear fuel, but are not prone to material loss like pool water.

In comparing shielding designs, it is important to note that most of the shielding materials have inherent limiting temperatures (i.e., maximum allowable temperature) with the exception of the steels and cast iron. These metals' temperature limits are much greater than the aluminum-based fuel cladding temperature limit. Shielding material thermal limits include both absolute values of temperature and, in the case of concrete, temperature gradients that create thermal stresses. Wet storage pool water also has a thermal limit that is the prevention of local or bulk boiling in the pool. Operation of the spent nuclear fuel pool heat removal system prevents pool water boiling, but a postulated accident in which this system is disabled requires calculation of the time before the inception of bulk pool boiling. Adequate natural convection between adjacent fuel assemblies and within storage racks prevents local nucleate boiling in any fuel flow channel.

Shielding geometry plays an important role in the determination of a dose rate profile around the storage facility. A continuous and constant thickness of shielding completely surrounding the fuel provides a relatively constant dose rate at all locations. A shielding design that is asymmetric and contains air gaps and/or varying material thickness results in hot spots and a relatively larger variation in surface dose rates. The wet pool design offers a continuous shield of water with resulting low constant dose rates throughout the pool surface. The water and pool wall, usually steel-lined concrete, also maintain a low continuous dose rate profile outside the walls.

The dry storage concrete building and concrete cask technologies rely on concrete walls for shielding with some steel internal to the walls. The need for internal airflow passages in the concrete introduces gaps in these walls. These gaps, which are labyrinths, require complex shielding analyses and typically allow a relatively larger dose rate at the air inlets and/or outlets than at the bulk concrete wall. This effect is more significant for concrete casks than for concrete buildings because the casks are more limited in the concrete thickness that is used in their shield wall. In the case of metal casks, since there are no internal air passages in the metal shield, the dose rate is relatively uniform around the surface. Different axial shielding and neutron-gamma source terms will result in different axial dose rates for the metal and concrete casks. Inground storage systems use a relatively small amount of concrete radially coupled with the surrounding earth for shielding and employ thick steel plugs for axial shielding. The hybrid metal-concrete cask design uses shielding principles similar to the concrete cask.

In comparing spent nuclear fuel storage shielding designs, the four basic technologies can be characterized as water, lead, metal, and concrete. Water and metal provide the most uniform dose rate reduction because they do not require the inclusion of labyrinth airflow passages for decay heat removal necessary for concrete. Water is most susceptible to a sudden rapid loss of shielding effectiveness because it is a liquid requiring confinement. It should be noted, however, that pool storage of spent nuclear fuel has been effectively used in the nuclear industry for over 40 years. Both concrete and water are susceptible to degradation of their shielding effectiveness if temperature limits are exceeded. These thermal limits can be accommodated by proper design of the spent nuclear fuel pool cooling system for water shields and conservative design along with airflow passage surveillance for the concrete shield. The shielding properties of all three shields are well known and therefore not subject to significant design uncertainties.

Shielding design is dictated by the regulatory dose limits, maximum bounding radiological neutron and gamma ray source terms of the fuel to be stored, cost, weight (in some cases), and thermal limits of some shielding materials. In general, the commercial nuclear power spent nuclear fuel storage technologies discussed in this appendix were designed to provide adequate shielding for fuel assemblies containing several hundred thousand curies (Ci) of fission products per assembly. The foreign research reactor spent

nuclear fuel being considered for acceptance in the United States will contain fission product inventories of from 1,000 to 100,000 (maximum) Ci per assembly. Therefore, the radiation source term for shielding design purposes, assuming the same number of fuel assemblies in each storage technology unit, may be significantly smaller for foreign research reactor spent nuclear fuel than for commercial fuel. The cost savings associated with a reduction in shielding thickness are expected to be more significant for the metal cask and concrete building designs because of their relatively higher costs. At the present time, it appears appropriate to use available designs from the vendors.

Based on the aforementioned vulnerabilities, the best shield would be the metal cask. Concrete shields are judged second best after metal based on their lack of dependence on any active systems. The water shield requires active systems for decay heat removal to prevent heatup and makeup to compensate for long term evaporation. It is also vulnerable to leaks from connected piping and its enclosing structure. Although water appears to be the least expensive shield material, its requirements for several active systems and qualified walls and floor actually make it one of the more expensive shields.

F.2.4 Structural Integrity

All of the spent nuclear fuel storage technologies are required to meet the same standards for structural integrity in accordance with appropriate codes. Structural integrity ensures that the confinement boundary around the spent nuclear fuel is maintained under all operational and accident conditions.

For incident-free operation, the dry storage designs are analyzed in terms of peak stresses on their canister and enclosing structure (i.e., metal cask, concrete cask, or vault). In wet storage designs, the fuel racks and pool structure are analyzed for operating loads. The source of these loads, in accordance with appropriate American Society of Mechanical Engineers codes, include such factors as deadweight, pressure, fill gas pressure, and thermal gradients.

For accident cases, additional loads are imposed upon the structures. These additional loads include seismic acceleration, high (or low) ambient temperature and solar heat flux, component drop or tip over, airflow passage blockage, external fire, tornado missile, flooding, etc. As with incident-free operation, specific prescribed margins of safety between the peak calculated stresses and the maximum allowable stress for a given component, location, and material must be maintained to substantiate structural integrity.

The principal structural-related differences between foreign research reactor spent nuclear fuel and commercial fuel for storage technology design purposes are:

- a typical foreign research reactor spent nuclear fuel element is much lighter [5 kg (11 lbs) as compared to 800 kg (1,760 lbs)] and shorter than a commercial fuel assembly (a stack of 5 typical foreign research reactor spent nuclear fuel elements is approximately equal in length to I commercial fuel assembly), and

- the strength of foreign research reactor spent nuclear fuel, in particular the predominant aluminum-clad design, is expected to be less than the commercial fuel assembly.

The much lower foreign research reactor spent nuclear fuel weight will reduce the total weight and load on the storage technology unit by about 19 metric tons (21 tons) for a 24-commercial fuel assembly design. For metal and concrete casks, this is a significant fraction of the total cask's weight, and can only improve the structural strength of the cask. The lower weight of the foreign research reactor spent nuclear fuel will increase the structural margins in the design and possibly allow for the use of less material in the structure compared with the commercial cask design. Any design changes to take advantage of the lower fuel weight would require detailed re-analysis, and are probably unnecessary.

The lower strength of the foreign research reactor spent nuclear fuel would require analyses to demonstrate that operational and postulated accident events do not result in structural failure of the fuel. However, since the principal means of confinement is the canister surrounding the fuel, its structural integrity is expected to be maintained, as it has already been qualified for the heavier commercial fuel under the same conditions and accidents.

Assuming that the same structural design limits apply for foreign research reactor spent nuclear fuel storage as for commercial fuel storage, the lower weight and strength of the foreign research reactor spent nuclear fuel would be expected to increase the original stress design margins.

The basket of any currently licensed cask would require redesign to accommodate the foreign research reactor spent nuclear fuel. Furthermore, it could be anticipated that permanently installed neutron poisons may be required in the basket to prevent criticality for the highly enriched fuels (initially 90 to 93 percent enrichment).

Each of the spent nuclear fuel storage technology designs that have been licensed by the NRC have undergone rigorous structural analyses and have been shown to meet all applicable standards and codes. Designs which have not yet been licensed would be required to present detailed structural analyses for review and confirmation to ensure structural integrity. No design has specific structural vulnerabilities that make it unsuitable for the storage of foreign research reactor spent nuclear fuel. It should be noted that any changes in existing NRC-approved storage designs that are deemed to impact stresses (i.e., reducing shielding wall thickness) would require extensive re-analysis and technical review for structural integrity. Thus, use of existing designs is favored.

F.2.5 Thermal Performance

Adequate decay heat removal is vital to preventing degradation of the fuel cladding barrier to fission product releases. The wet and dry storage technologies rely on a combination of conduction, convection (natural or forced), and radiation heat transfer mechanisms to ensure fuel cladding temperatures below appropriate long term storage limits.

In wet pool designs, fuel decay heat is transferred to the pool water by conduction and natural convection, which is induced by the axial enthalpy rise of the water as it passes over the active region of the fuel. An active cooling system consisting of redundant pumps, heat exchangers, and piping connected to the pool removes the heat in the bulk pool water. Careful thermal design of the spent nuclear fuel storage racks allows for sufficient natural convection flow over each fuel assembly to prevent any local nucleate boiling on the cladding surface throughout the pool. Therefore, the thermal performance of the pool technology relies on storage rack design for local thermal effects and an active external system for global heat removal. As previously discussed, this design has a long-established history of satisfactory performance. Wet storage can accommodate fuel of any power level.

The metal cask, dry storage design relies on a totally passive system for heat removal. The fuel decay heat, in an encapsulating inert gas atmosphere canister, is transferred to the canister's walls by a combination of radiation and conduction heat transfer. The canister walls, in contact with the metal (or sometimes metal sandwiched with a neutron-absorbing material) cask wall transfers this heat by conduction through the metal wall. At the outside of the metal cask, the heat is removed by conduction and natural convection to the environment. Some designs incorporate cooling metal fins on the exterior of the cask to enhance heat transmission to the air. The four metals used in spent nuclear fuel storage cask designs are ductile cast iron, carbon steel, lead, and stainless steel. In terms of their heat conduction properties, cast iron, lead, and carbon steel are superior to stainless steel because they have a thermal

conductivity which is about three times that of stainless steel. The metal cask heat transfer system is not susceptible to thermal limits, since these metals have a higher temperature limit than fuel cladding. The only possible degradation of heat transfer could occur if the fuel canister seal was broken and the inert gas atmosphere lost. The sealing system is designed to withstand all postulated accidents and maintain integrity over the lifetime of the cask, because it constitutes part of the radioisotope confinement boundary.

As with metal casks, concrete casks use a passive heat removal system, but the concrete cask system has one inherent vulnerability. To remove fuel decay heat and stay below both the fuel cladding and concrete temperature limits, concrete casks must include a labyrinth airflow passage design that allows natural convection-driven air to enter the cavity enclosing the canister inside the concrete. The air then exits through higher elevation paths through the concrete to the environment. Concrete thermal conductivity is a factor of 10 to 40 lower than that of the previously discussed cask metals. The need for these airflow passages and their associated inlets and outlets introduces the possibility of an accident in which the inlet and/or outlets could be blocked by debris, snow, or even nests or hives. Therefore, concrete casks require surveillance of their air inlet and outlet flow passages. Typically, measurement of the air temperature rise between the inlet and outlet is also used to validate the thermal design and as an operating specification. The elevation difference between the air inlets and outlets is an important design factor in the effectiveness of natural convection-driven airflow through the cask. Larger elevation differences induce a greater airflow rate, which improves heat removal. The above discussion would not apply to a solid concrete design (such as a SILO), because it does not use internal airflow passages for decay heat removal; it uses only concrete conduction.

In a concrete cask, heat transfer within the canister is identical to that of the metal cask. The canister transfers heat by conduction, natural convection, and radiation heat transfer to the airflow around it and the concrete walls surrounding it. That portion of the heat transferred to the concrete walls is then conducted through the concrete to the outside air. The concrete building technology uses a heat transfer system identical to that of the concrete cask. However, its relatively larger size, translating to a larger flow area for air inlets and outlets, makes it less susceptible to flow passage inlet and/or outlet blockage. Also, the concrete building designs typically incorporate a much larger elevation difference between air inlets and outlets than the concrete cask, which further enhances natural convection flow driven heat transfer.

For commercial nuclear fuel, the long-term storage temperature limits are well known (Levy et al., 1987; Johnson and Gilbert, 1983; Einziger and Cook, 1985; and Kohli et al., 1985), and typically about 350°C (662°F). The shielding material limits usually apply to concrete or concrete-like materials, but may also apply to resins or polyethylene. Shielding material temperature limits are not affected by the use of foreign research reactor spent nuclear fuel instead of commercial nuclear fuel in the storage technology. However, the thermal limits of the TRIGA and MTR foreign research reactor spent nuclear fuel could affect the thermal design.

Currently, there is limited well-documented information available on long-term foreign research reactor spent nuclear fuel storage temperature limits for fuel cladding. However, the aluminum cladding of the MTR-type foreign research reactor spent nuclear fuel has a much lower melting point than commercial nuclear fuel zircaloy cladding [649°C (~1,200°F) for aluminum versus 1,832°C (~3,330°F) for zircaloy]. Thus, the maximum long-term storage temperature limit for aluminum-clad foreign research reactor spent nuclear fuel would be expected to be considerably lower than that for zircaloy-clad commercial nuclear power fuel. Aluminum also undergoes a phase change at around 250°C (482°F), which results in a reduction of its tensile properties. An offsetting physical property of aluminum that may partially compensate for its lower melting point is that aluminum has a thermal conductivity more than 10 times greater than zircaloy. This would tend to reduce the temperature difference across the aluminum cladding as compared to zircaloy cladding. TRIGA foreign research reactor spent nuclear fuel cladding is

composed of stainless steel or inconel, which have similar thermal conductivities to zircaloy, but a melting temperature of about 1,371°C (2,500°F). TRIGA fuel storage temperature limits are expected to be greater than for aluminum-clad fuel. The Savannah River Site is conducting a research and development project, initiated in FY 1994, to examine the applicability of aluminum-clad spent nuclear fuel dry storage.

At a minimum, a new thermal analysis would need to be performed for existing designs of spent nuclear fuel storage technologies. This analysis would use the parameters associated with foreign research reactor spent nuclear fuel instead of commercial spent nuclear fuel. The important changes in thermal performance parameters for the foreign research reactor spent nuclear fuel are:

- lower individual fuel assembly decay heat power,
- lower and/or different fuel temperature limits for aluminum-clad and stainless steel-inconel-clad fuels, and
- higher clad and fuel thermal conductivity for aluminum foreign research reactor spent nuclear fuel.

A temperature limit of 175°C (347°F) has been tentatively identified to avoid damage to the cladding of aluminum-clad spent nuclear fuel (Shedrow, 1994a and 1994b; Taylor et al., 1994). The results of this revised thermal analysis could impact the thermal design of the spent nuclear fuel storage technology. If the existing design results in unacceptable fuel and/or shielding temperatures, redesign could reduce the maximum heat load of each module or cask or increase the airflow passage area or height for concrete casks that rely on natural convection heat transfer. The new thermal analysis must take into account design restrictions that are imposed by criticality limits (i.e., the maximum allowable number of foreign research reactor spent nuclear fuel assemblies), and possible changes in shielding thickness due to lower gamma and neutron source terms that would improve the storage technology's thermal performance. Again, existing designs should be used to the greatest extent possible.

Commercial spent nuclear fuel dry storage systems require a minimum cooldown period of 5 years. For aluminum-clad foreign research reactor spent nuclear fuel, the preliminary cladding temperature limit of 175°C (347°F) becomes the determining criteria for dry storage loading above an average spent nuclear fuel element power level of 40 Watts each. Foreign research reactor spent nuclear fuel averages more than 40 Watts per element after a single year's discharge from the reactor and, if immediately placed into dry storage, would result in oversized facilities within several years as the radionuclides decay. Consequently, for the size of a foreign research reactor spent nuclear fuel dry storage facility to be minimized, an average foreign research reactor spent nuclear fuel power level below 40 Watts per element is necessary. On average, a 3-year cooldown period would be required. This results in the element's volume being the constraining criteria, and corresponds to maximum density of spent nuclear fuel (hence, minimum size of the facility) in the dry storage method. Consequently, the storage approach uses a minimum wet storage period of 3 years prior to emplacement into dry storage.

A comparative evaluation of the thermal performance of each fuel storage technology points to the metal cask and the solid concrete SILO as the simplest, effective, and least susceptible to any degradation. However, another design which has many merits is the concrete building. Although concrete buildings require open airflow passages to remove decay heat, size and a large elevation difference are factors which compensate for this weakness and make them good candidates. The concrete cask, with adequate design margins and surveillance is an acceptable thermal system. Finally, the wet pool system is a proven technology, but is dependent on an active system to remove heat. The inground concrete system in

Denmark (RISO National Laboratory) relies on a forced air active system and is characterized similar to the wet system in terms of its heat removal capabilities.

F.2.6 Ease of Use

For spent nuclear fuel storage, ease of use is defined as the lack of complexity involved in the process of loading spent nuclear fuel, and operating and maintaining the storage technology. For all storage designs, the spent nuclear fuel must be removed from the transportation cask to be placed into the storage facility, unless the design is a dual-purpose cask.

The technology that requires the fewest steps and lowest complexity for transferring spent nuclear fuel from the transportation cask to its storage location is wet pool storage. At a pool, the transportation cask is simply immersed under the water, opened underwater, and the fuel moved underwater to its final location in a storage rack in the pool. Pool water provides shielding, heat removal, and viewing of the fuel. The dry storage technologies all require additional intermediate steps, which include the insertion of fuel into a canister that must be subsequently drained of all water and air, seal welded, tested for leakage, and backfilled with inert gas. The canister is then placed into its dry storage structure (i.e., vault, concrete, or metal cask). The vault provides for this entire process within a shielded enclosing building, whereas the casks require transport by some vehicle between the transportation cask fuel transfer location and the cask site. Thus, for spent nuclear fuel transfer and loading, the wet storage design is easiest to use, followed by the dry vault.

After loading, operation of the storage facility is another important factor in determining ease of use. For operation, the individual metal or concrete casks are easiest, since they are designed as totally passive systems requiring only periodic visual inspection from a distance. The vault is slightly more complex than the casks because it includes a number of active systems (i.e., crane, power supplies, fuel handling machine) that may require some operational support. The wet storage is the most complex from an operational viewpoint because it includes a number of vital safety-related systems that must be monitored and controlled (e.g., heat removal system, water purification system, makeup water system, ventilation exhaust system).

Maintenance ease of use is closely related to operational ease of use since designs with more operational complexity require greater maintenance. Thus, the cask systems can be considered easiest, followed by the vault, and the wet technology.

In ranking the relative importance of the three aforementioned factors of ease of use (fuel loading, operation, and maintenance) the fraction of time spent during the life of the facility for fuel loading is expected to be much smaller than for operation and maintenance. Fuel loading will be a sporadic event over a long period of time, whereas operation and maintenance are considered continuous over this same period of time. Thus, operation and maintenance ease of use is given greater importance than fuel loading ease of use. With this ranking, the cask (both metal and concrete) technology is judged to have the greatest ease of use, followed by the vault system. The wet storage technology is judged to have the lowest ease of use principally because of its safety-related active systems.

F.2.7 Cost

Information on the cost of different spent nuclear fuel storage technologies is limited because of its proprietary nature, but several comparative statements apply to the different designs. Operation and maintenance costs are expected to be highest for those technologies that rely on active systems for safety. Thus, the wet pool and inground forced air technologies have higher operations and maintenance costs

than dry metal casks, dry concrete casks, and dry concrete buildings. The dry concrete vault/building technology would be expected to have slightly higher operations and maintenance costs than the individual metal or concrete casks, since these buildings use active nonsafety systems such as lighting, cranes, and fuel drying dedicated to the vault facility.

For the construction of a new fuel storage facility for the purpose of storing foreign research reactor spent nuclear fuel on the order of approximately 23,000 assemblies, elements and/or rods, it is assumed that 5 "trimmed" foreign research reactor spent nuclear fuel assemblies would occupy the same approximate space as one commercial nuclear power plant fuel assembly (Boiling Water Reactor-type). "Trimmed" means that the non-essential portions (i.e., ends) of the spent nuclear fuel element have been removed, as detailed in Appendix B. Therefore, storage of the total amount of foreign research reactor spent nuclear fuel under consideration in this EIS would be the equivalent of about 5,000 commercial Boiling Water Reactor spent nuclear fuel assemblies. Since typical concrete, inground, or metal casks can store 52 power fuel assemblies, this foreign research reactor spent nuclear fuel inventory would require around 100 casks. A suitably sized single pool or concrete building could accommodate this inventory of spent nuclear fuel. Spent nuclear fuel storage manufacturers have indicated that metal casks typically cost about twice as much as concrete casks for the same quantity of fuel storage due to the higher costs of metal as compared to concrete. Based on its design, the least expensive concrete cask is expected to be the simple concrete SILO, since it does not have steel-lined internal air passages. The number of foreign research reactor spent nuclear fuel assemblies under consideration in this EIS may be amenable to the economic advantages that a single building or pool offers over a large number of individual casks. Another potential cost advantage of the pool or concrete building is that these are self-contained, not requiring access to any other facilities for the transfer of the fuel from the transport cask. Presented below is a brief summary of commercial cost experience with storage.

F.2.7.1 Costs for Dry Storage Designs

The cost for different spent nuclear fuel storage technologies varies significantly between designs. Some information on cost has been obtained from manufacturers and openly available literature. Relative order of magnitude cost data was obtained for the horizontal concrete NUHOMS module, vertical concrete Ventilated Storage Cask design, vertical concrete SILO, metal CASTOR vertical cask, and the modular dry vault concrete building design.

The principal elements of cost that should be considered for the storage of foreign research reactor spent nuclear fuel are: (1) engineering for redesign and licensing, (2) capital for the construction of the facility, and (3) operations and maintenance. In the interest of minimizing cost and schedule for the completion of any storage facility for foreign research reactor spent nuclear fuel, the licensing basis of 10 CFR 72 used by the NRC for commercial nuclear power plant spent nuclear fuel should be adopted for the foreign research reactor spent nuclear fuel. This regulation provides all the requirements for licensing foreign research reactor spent nuclear fuel storage and has been successfully applied to numerous dry spent nuclear fuel storage installations in the United States.

Redesign engineering should be limited to changes in the design of the basket that encapsulates the fuel, since foreign research reactor spent nuclear fuel has different dimensions, would probably be stacked, and could require different spacing and/or the incorporation of neutron absorbing plates to maintain subcriticality safety margins. Outside the basket, all remaining components should be identical to those already licensed for commercial nuclear fuel by the NRC, thereby significantly reducing engineering analysis and license review time and costs as well as drawing and specification changes. This could result in some overdesign in the shielding and heat removal of the system, but would have the benefit of greatly reduced engineering, licensing, and schedule costs. If thermal analyses show that unacceptable foreign

research reactor spent nuclear fuel temperatures would occur in the storage facility, then more extensive redesign would be required (e.g., reduce excess concrete wall thickness not needed for shielding, which then improves the conduction heat transfer) or fewer fuel assemblies could be stored in each unit.

Information obtained on the unit capital costs for different storage designs shows a significant variation. The least expensive unit is the SILO due to its simple concrete-canister design and lack of internal air passage labyrinth. The most expensive unit cost, excluding the modular dry vault (which stores a larger number of fuel assemblies than the other storage designs), is the CASTOR metal cask, due to its use of a thick metal wall instead of concrete. The Ventilated Storage Cask and NUHOMS designs' costs fall between the SILO and CASTOR. If one were to rank, in decreasing order, the unit cost of the four cask designs, they would be: CASTOR, NUHOMS, Ventilated Storage Cask, and SILO. There is more than a factor of 10 difference between the SILO and the CASTOR.

An estimate of the capital costs for storing approximately 23,000 foreign research reactor spent nuclear fuel elements can be made with the following two assumptions. First, the average spent nuclear fuel assembly decay heat is between 10 and 40 Watts, which is reasonable and conservative based on the status of foreign research reactor spent nuclear fuel under consideration in this EIS. Second, five trimmed foreign research reactor spent nuclear fuel assemblies can be stacked to fit into the same approximate space as one commercial nuclear power plant spent nuclear fuel assembly (Boiling Water Reactor-type). Using these assumptions, 23,000 foreign research reactor spent nuclear fuel elements would require 375 SILOs, 100 VSC-24s, 100 NUHOMS-24Ps, or 150 CASTOR V21s. One sufficiently sized and designed modular dry vault would also accommodate the foreign research reactor spent nuclear fuel.

The operations and maintenance costs for all these designs are expected to be small based on utility experience in operating dry spent nuclear fuel storage at numerous sites throughout the United States. The passive nature of these designs eliminates the need for any control room or continuous monitoring. Once completed and loaded with fuel, the storage facility would not require any onsite staff. Remote security surveillance cameras, fences, and thermoluminescent dosimeters for radiation monitoring would be utilized. Some designs (Ventilated Storage Cask and NUHOMS) require a periodic visual inspection of the labyrinth airflow inlets and outlets to ensure that there is no blockage. Periodic security fence thermoluminescent dosimeter retrieval and analysis would also be an expected operating requirement. No onsite utility consumption would be necessary except for that used in security lighting and cameras. Under incident-free conditions, no significant maintenance costs would be anticipated for most of the designs, with the exception of the modular dry vault, where equipment used in the movement and encapsulation of fuel would require some periodic maintenance.

Table F-15 provides a summary of the dry storage costs. Full-Time Equivalent estimates assume full-time assignment, whereas the utility experience indicates only part-time assignment would be necessary. Thus, Table F-15 costs are extremely conservative.

F.2.7.2 Costs for Wet Storage Designs

Based on previous utility experience, the costs associated with the design, licensing, construction, operation, and maintenance of a new spent nuclear fuel pool are expected to be higher than most of the dry storage designs. This is due to the structural, equipment, and active system requirements of a pool that must also be enclosed in a properly qualified structure. The need for active operating safety-related systems at a wet facility increases the operations and maintenance costs. These important systems include water purification and chemistry, water heat removal, water level, and heating, ventilation, and air conditioning for the building. Table F-16 provides a summary of the wet storage costs. Thus, it can be stated that the overall costs associated with the selection of a new wet storage facility for foreign research

Table F-15 Summary of Dry Storage Facility Costs Based Upon Utility Experience[a]

	Approximate Unit Capital Cost Range, $	*Approximate # of Canisters/Sleeves for Foreign Research Reactor Spent Nuclear Fuel*	*Total Capital Cost $M*	*Full-Time Equivalents, Loading/ Inspection[b]*	*Full-Time Equivalents Monitoring[c]*	*Other Annual Costs, $M*	*Total Annual Operating Cost $M[d,e]*
Metal Cask	800,000-1.1M	150 (max)	165	15 (max)	3 (max)	1	3.7
Horizontal Dry Storage Cask	400,000-500,000	100 (max)	50	15 (max)	3 (max)	1	3.7
Vertical Concrete Storage Cask	350,000	100 (max)	35	15 (max)	3 (max)	1	3.7
Modular Dry Vault	13,000/tube	5 foreign research reactor/vaul t tube	65	15 (max)	3 (max)	1	3.7
SILO	100,000	375 (max)	37.5	15 (max)	3 (max)	1	3.7

Reference for costs: (EPRI, 1993)

[a] *Intermediate wet pool required for dry storage facility not included because utilities already possess an on-site pool*

[b] *One shift operation*

[c] *Monitoring based on one Full-Time Equivalent per shift*

[d] *Average Full-Time Equivalent cost of $150,000 per year*

[e] *Estimated absolute maximum from utility experience; around $1 million per year appears to be the average*

Table F-16 Summary of Wet Storage Costs

Approximate Facility Capital Cost	$80-100 million
Full-Time Equivalents Loading/Inspection	30
Full-Time Equivalents Operations/Monitoring	(in above)
Other Annual Costs	around $1million
Total Annual Operating Cost	$6-12 million

(Nuclear Fuel, 1994a)

reactor spent nuclear fuel could be significantly larger than for any comparably sized dry storage design using concrete.

F.2.8 Design, Construction, and Operational Requirements

The DOE has orders dealing indirectly with the storage of spent nuclear fuel. A search was also made to determine if other Federal requirements exist that specifically deal with spent nuclear fuel.

DOE Order 5400.1 (DOE, 1988), entitled "General Environmental Protection Program," establishes environmental protection program requirements. It applies to all Departmental elements and contractors performing work for DOE. Although the order makes no direct reference to spent nuclear fuel, Chapter IV deals with environmental monitoring requirements, which could be useful in establishing a legal framework for storing spent nuclear fuel.

DOE Order 5400.5 (DOE, 1990), entitled "Radiation Protection of the Public and the Environment," establishes standards and requirements for operations of the DOE and DOE contractors with respect to

protection of the public from undue radiological risk. The provisions apply to all Departmental elements. This order references the storage of spent nuclear fuel. It also references instances where some DOE facilities are subject to provisions of 10 CFR 72. It was not made clear in the order which DOE facilities are subject to 10 CFR 72, which deals directly with all aspects of interim storage of spent nuclear fuel.

DOE Order 5480.22 (DOE, 1992b), entitled "Technical Safety Requirements," requires that DOE nuclear facilities delineate criteria, content, scope, documents, etc. The scope includes DOE elements, but excludes facilities exempt from NRC licensing and Naval Propulsion Program facilities. Although this order does not reference spent nuclear fuel, there are useful discussions of limiting conditions for operation of nonreactor nuclear facilities.

DOE Order 5633.3A (DOE, 1994e), entitled "Control and Accountability of Nuclear Materials," prescribes minimum requirements and procedures for control and accountability of nuclear materials at DOE facilities, which are exempt from NRC licensing requirements. By DOE definition, "nuclear materials" includes spent nuclear fuel. Storage of nuclear material is mentioned with respect to repositories.

DOE Order 6430.1A, (DOE, 1989a), entitled "General Design Criteria," has a section dealing with irradiated fissile material storage facilities (Section 1320). General criteria for nuclear criticality, confinement systems, effluent control and monitoring, and decontamination and decommissioning are discussed. Reference is made that, "the design professional shall consider the criteria provided in 10 CFR 72," (NRC, 1994) as well as NRC Regulatory Guides 3.49 (NRC, 1981) and 3.54 (NRC, 1984) for applicability to irradiated fissile material storage facilities. Other important standards for dry storage are ANSI/ANS-57.9 (ANSI, 1984a) and NRC Regulatory Guide 1.13 (NRC, 1975).

F.2.9 Aluminum-Clad Research Reactor Spent Nuclear Fuel Dry Storage Experience

F.2.9.1 Australia

Australia has successfully operated an underground dry storage facility for High Flux Australian Reactor MTR-type aluminum-clad research reactor spent nuclear fuel for 31 years at the Lucas Heights Facility (Australia, 1993; Ridal, 1994; Silver, 1993). The facility consists of a building enclosing a concrete floor with 50 steel plugs that are bolted to a steel collar set into the concrete.

Each plug covers a stainless steel-lined 0.64 cm (0.25 in) thick, 14 cm (5.5 in) inner diameter, 15.2 m (50 ft) deep borehole tube that is sealed at the bottom. The rock around these 50 borehole tubes is sandstone with a variable clay matrix and bands of enriched siderite. The actual boreholes in the sandstone are 16.5 cm (6.5 in) in diameter, 16.8 m (55 ft) deep, and spaced 1.14 m (45 in) center-to-center apart.

Each borehole liner is filled with 11 stainless steel canisters that hold 2 stacked fuel assemblies each. The borehole liner is evacuated and backfilled with dry nitrogen. The borehole liner plug is designed with its own plug to allow for atmosphere purging, backfilling, and annual monitoring of any fission product gases that would indicate canister breach.

The stored High Flux Australian Reactor spent nuclear fuel is uranium-aluminum alloy with aluminum cladding in the shape of four concentric tubes. Each fuel assembly has an outer diameter of 10 cm (3.93 in) and a length of 66 cm (26 in). The ^{235}U content for each fresh fuel assembly was 170 g (0.37 lbs), and the ^{235}U was enriched to 60 percent. Fuel has been stored at this facility for 8 to 31 years with no radioactivity releases or evidence of corrosion over this time period. No nuclear poisons for

criticality safety or heat transfer analyses were deemed necessary because of the relatively low ^{235}U content, large borehole spacing, and low fuel assembly decay heat. The storage criteria for each fuel assembly is a maximum decay heat of less than 4.5 Watts, which after 20 years, drops to 1.5 Watts per fuel assembly. Fuel examined in a hot cell after 10 and 25 years of storage at this facility showed no visible signs of corrosion. Figure F-29 illustrates the facility design.

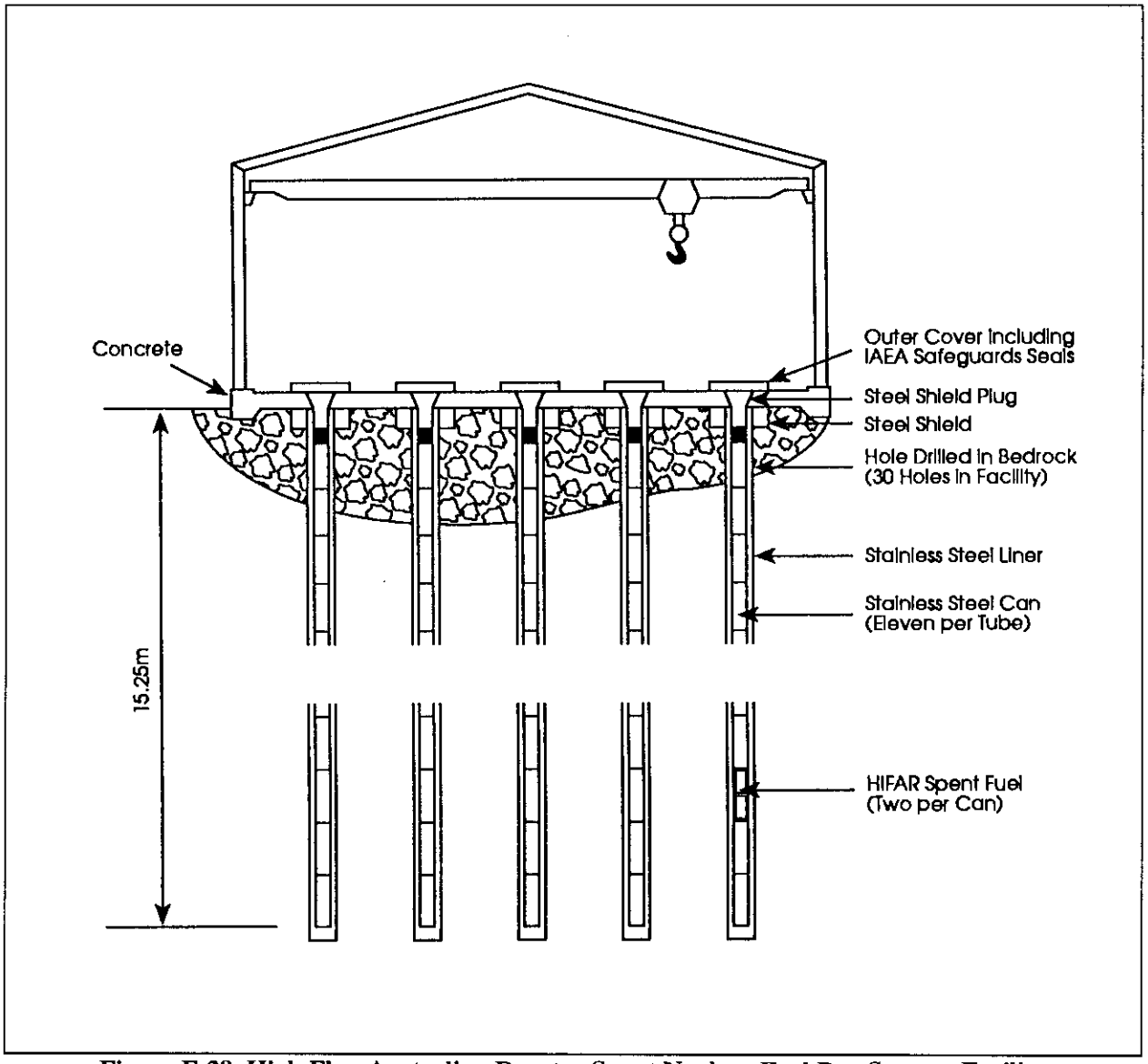


Figure F-29 High Flux Australian Reactor Spent Nuclear Fuel Dry Storage Facility

F.2.9.2 Japan

In 1982, The Japan Atomic Energy Research Institute completed construction of a dry spent nuclear fuel storage facility at Tokai, Japan for the storage of JRR-3 research reactor spent nuclear fuel (Shirai et al., 1991). The facility consists of a building enclosing several support areas (cask receipt, loading, cask maintenance, and control room) and the drywell storage structure (Figure F-30). The storage structure is 12 m (39.4 ft) long, 13 m (42.7 ft) wide, 5 m (16.4 ft) high concrete box that encapsulates a 10 x 10 lattice array of drywells (Figure F-31). Each drywell storage canister (Figure F-32) comprises a

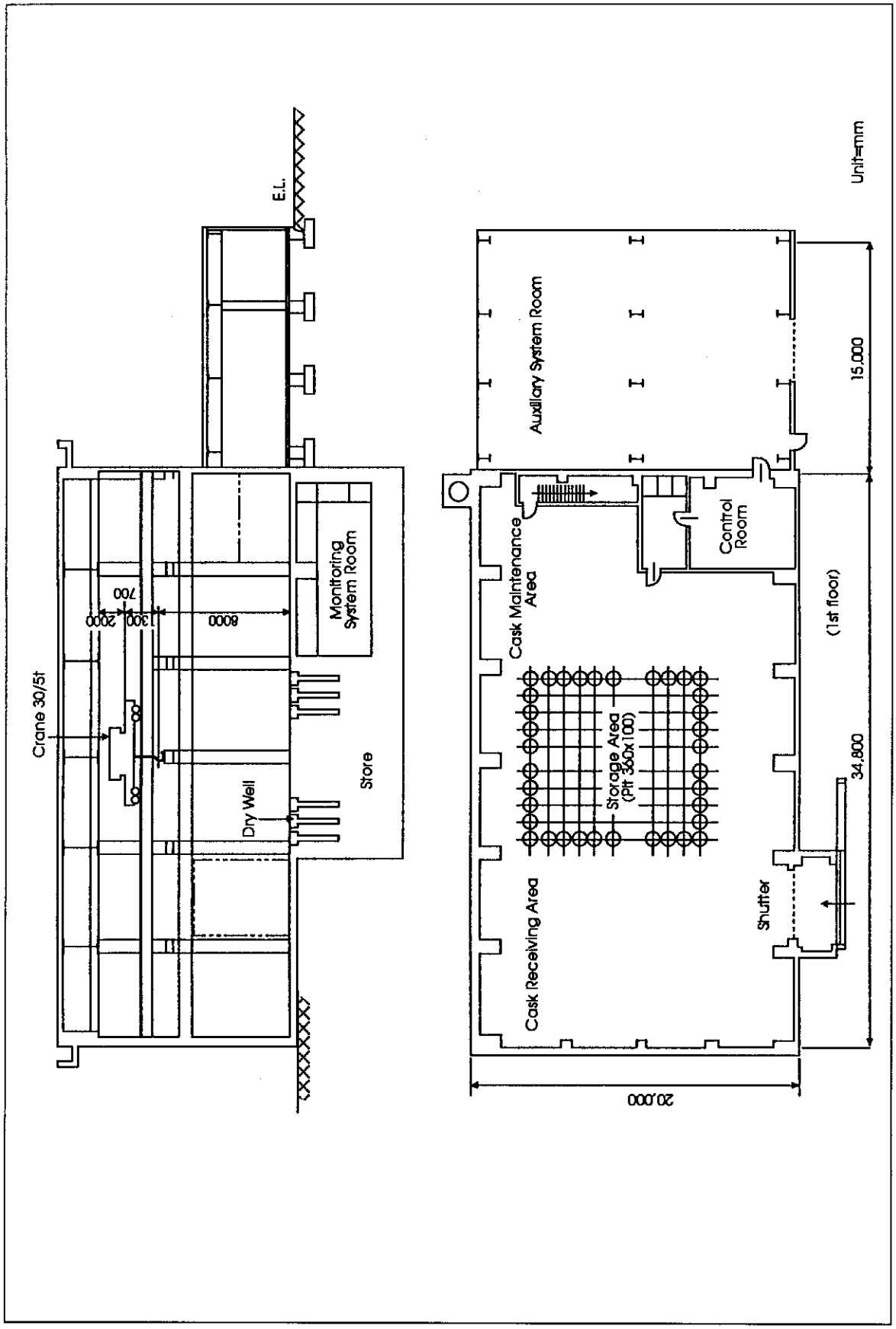


Figure F-30 General Arrangement of Dry Storage Facility at Tokai, Japan

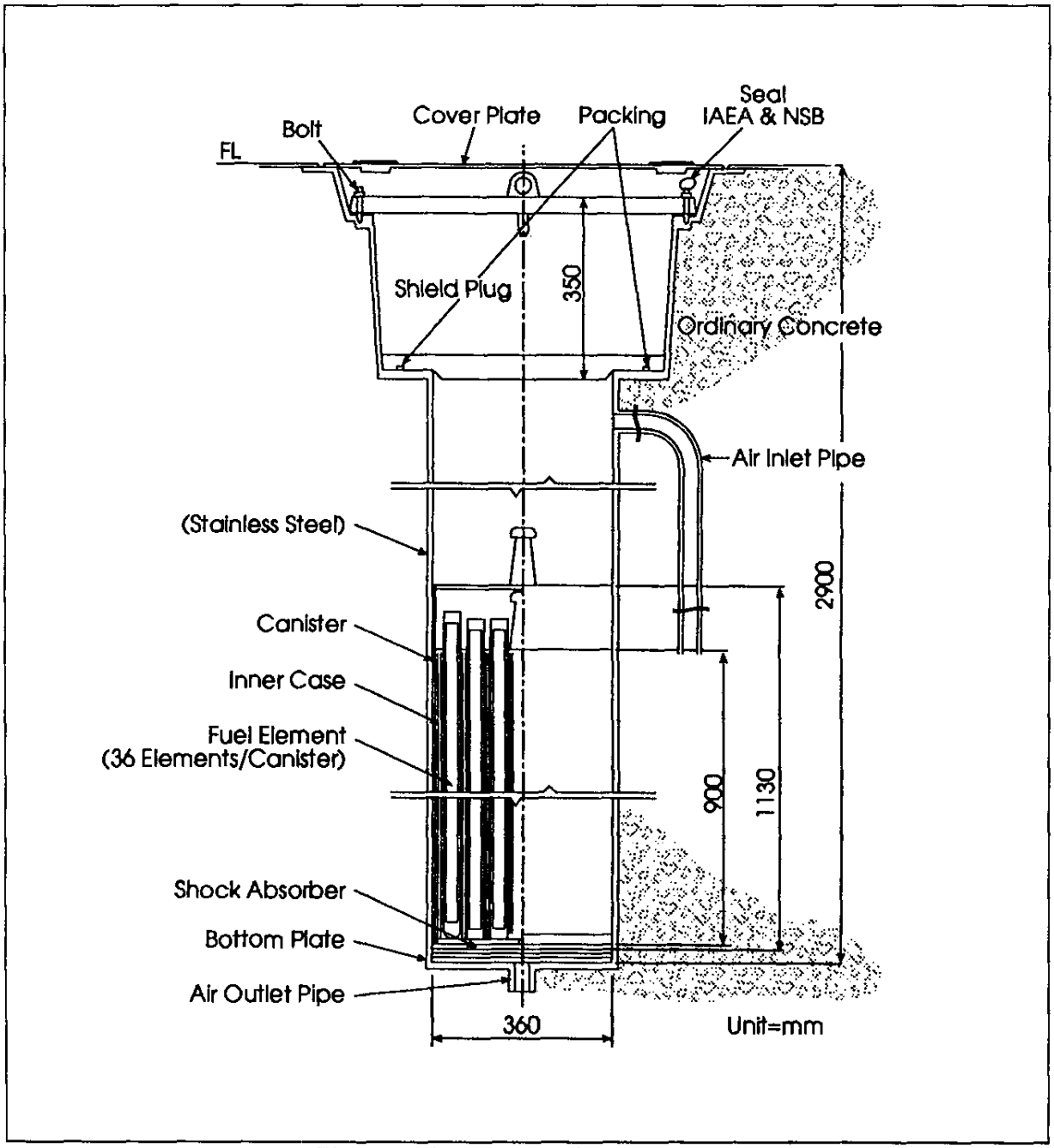


Figure F-31 JRR-3 Dry Storage Facility

0.8 cm (0.3 in) thick stainless steel liner 2.5 m (8.2 ft) deep and has a 36 cm (14.2 in) inner diameter. Each drywell has a labyrinth air inlet and outlet pipe for radiation monitoring and decay heat removal, and is covered with a 35 cm (13.8 in) thick carbon steel shield plug. The plug is bolted to the concrete and has a cover plate above it. Each drywell has a minimum of 1.5 m (4.9 ft) of concrete shielding around it.

A cylindrical stainless steel canister (Figure F-32) is placed in each drywell. The canister has 0.5 cm (0.2 in) thick walls, a 35 cm (13.8 in) outer diameter, and a height of 1.25 m (4.1 ft). Each canister holds 36 fuel elements and is fusion welded after being loaded with spent nuclear fuel, evacuated, and filled with inert gas. Each element is a natural metallic or 1.5 percent ^{235}U-enriched uranium oxide cylinder encased

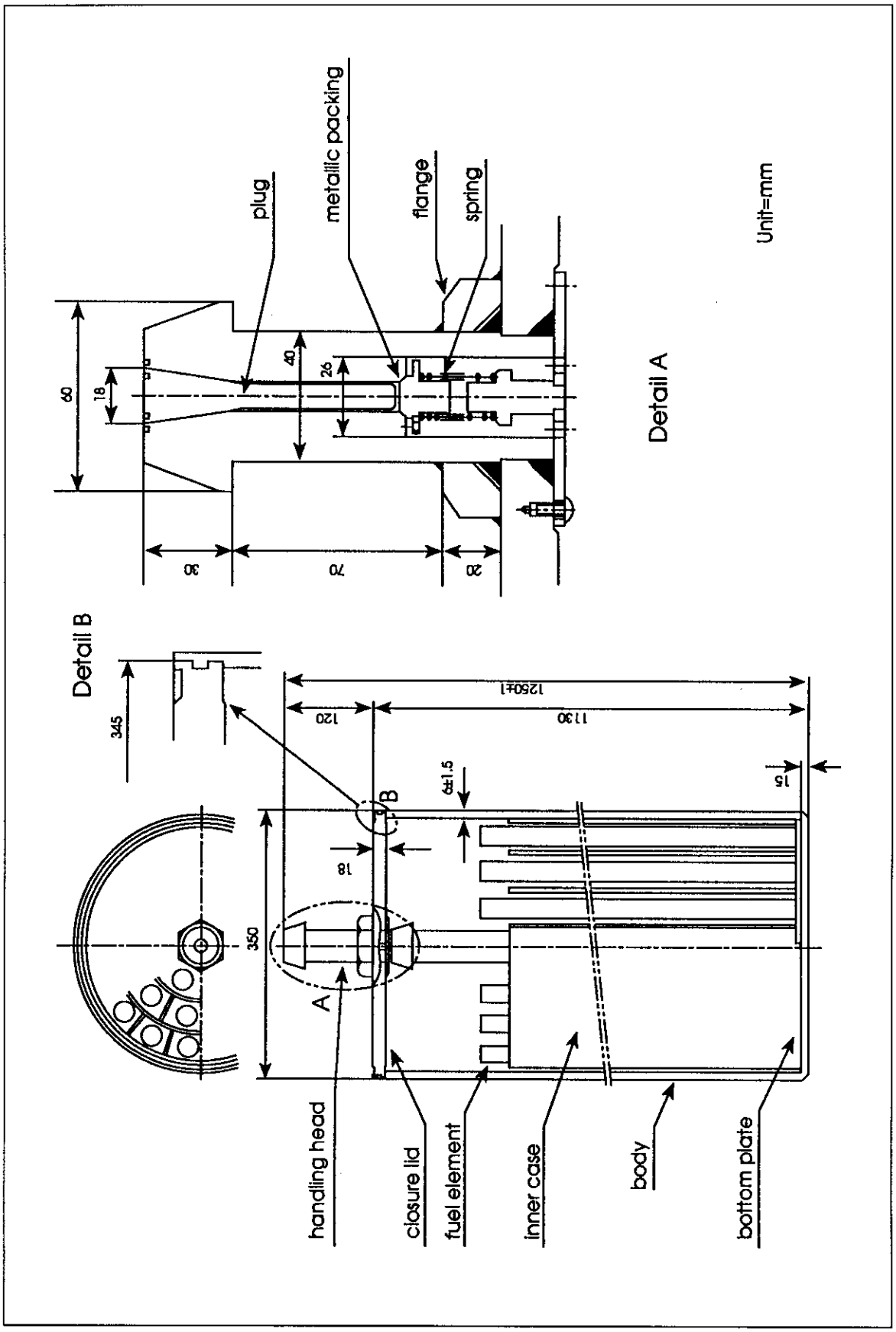


Figure F-32 Storage Canister

in aluminum cladding. The element is about 95 cm (37.4 in) long, with a 2.5 cm (1 in) outer diameter. Before storage, each element must meet the specifications of:

- maximum burnup = 800 Megawatt Days per metric ton,
- minimum cooling time = 2,500 days (6.8 years),
- maximum fission product activity = 110 Ci, and
- maximum decay heat = 0.5 Watt.

Although the low decay heat for the fuel elements eliminates the need for any cooling system (i.e., natural convection and conduction are sufficient), a system of blowers, filters, dehumidifiers, and monitors is provided for the facility. This system is used to provide subatmospheric pressure in the drywell so that any leakage would always be into the structure and its associated filters. This system ensures low humidity for minimizing any corrosion, and is also part of the radiation monitoring system designed to sample airflow for any escaping Krypton-85, a long-lived fission product that would be indicative of canister and fuel degradation. Maximum fuel storage temperature is maintained below 45°C (113°F) in this storage facility.

As of 1991, 1,800 fuel elements [14 metric tons (15 tons) of uranium] have been stored at this facility without any incidents. In 1987, after 5 years of storage, 2 canisters and their 72 fuel elements were removed and examined. None of the fuel elements or canisters exhibited any signs of corrosion, cracking, degradation, or failure.

F.2.10 Summary

In summary, the preceeding discussion indicates the following:

- Dry storage of spent nuclear fuel is a mature technology and requires the least maintenance.
- Dry storage of foreign research reactor spent nuclear fuel appears to be practical using existing designs from commercial utility experience and has been demonstrated in operating storage facilities for foreign research reactor spent nuclear fuel in Australia and Japan.
- Wet storage technology is the most common method for spent nuclear fuel.

F.3 Selection of Storage Technologies for Further Evaluation

The preceding discussions have identified several storage technologies suitable for foreign research reactor spent nuclear fuel. Three basic categories encompass these technologies:

- Dry Vault Storage,
- Dry Cask Storage, and
- Wet Pool Storage.

The three technologies are discussed in this section, including site-specific modifications, while Section F.4 describes the potential impacts and ramifications at the five candidate management sites. All

three approaches are estimated to require less than 4.5 ha (11 acres) of site land for receipt of all of the foreign research reactor spent nuclear fuel under consideration in this EIS.

F.3.1 Dry Storage Facility Designs

F.3.1.1 Spent Nuclear Fuel Storage Using a Dry Vault (Modular Dry Vault Storage)

As noted previously, the dry vault facility is an aboveground, self-contained concrete structure that includes dry fuel loading and unloading (Fort St. Vrain, 1992; Shedrow, 1994a and 1994b; Taylor et al., 1994; Claxton et al., 1993). The vault approach design consists of four major components: a receiving/loading area, fuel storage canisters, a shielded container handling machine, and a modular array for storing the fuel storage canisters (Figure F-33). The receiving area uses a small wet pool for unloading the transportation casks and for short-term (1 to 3 year) storage of foreign research reactor spent nuclear fuel exceeding 40 Watts per element. Table F-17 summarizes some typical modular dry vault storage parameters. The vault consists of several array units, and each unit provides storage for hundreds of fuel elements. The vault itself consists of a charge/discharge bay with a fuel handling machine above a floor containing steel tubes that house the (removable) fuel canisters. Shielding above the spent nuclear fuel is provided by the thick concrete floor and shield plugs inserted into the top of the steel storage tubes. The steel tubes serve as secondary containment for the foreign research reactor spent nuclear fuel and descend into an open storage area. Large labyrinth air supply ducts and discharge chimneys permit natural convection cooling of the steel spent nuclear fuel storage tubes, while the perimeter concrete walls provide for shielding. The design allows for expansion by adding additional units of arrays to the end of the vault, or by construction of another module. The vault facility also includes a receiving and loading bay that allows handling of the shielded transportation casks and unloading of the foreign research reactor spent nuclear fuel into the short-term wet storage pool. The receiving bay provides for spent nuclear fuel inspection, canning as required, and could be used for spent nuclear fuel characterization with additional equipment and modifications.

In operation, the transportation cask is lifted by a crane and placed in the unloading area of the small wet pool. The fuel elements are removed under water, examined, and, if the heat generation rate is below 40 Watts per element, the spent nuclear fuel is placed within the transfer canister. The transfer canister is subsequently drained, dried, and seal-welded. The handling machine then transports the loaded canister to the storage tubes. The handling machine includes radiation shielding. Heat dissipation is accomplished by natural convection from the surfaces of the handling machine and canister. The handling machine transfers the spent nuclear fuel canister from the receiving area to the vault and places the canister vertically into the storage tubes. The shield plug is placed on top of the loaded storage tube. Decay heat is dissipated by natural convection; air enters through inlet ducts at the bottom of the vault module, passes around the outside of the steel storage tubes containing the spent nuclear fuel canisters, and exits through outlet ducts at the top of the module. The vault facility stores spent nuclear fuel in canisters that are approximately 40.6 cm (16 in) in diameter by some 4.6 m (15 ft) long. As currently envisioned, foreign research reactor spent nuclear fuel would be stored in 5 levels of fuel with 4 elements per level, for a total of 20 fuel elements per spent nuclear fuel canister (MTR aluminum-clad type design). The vault design allows for 36 to 44 canisters per array unit, depending upon the decay heat of the spent nuclear fuel and a cladding temperature limit [175°C (347°F) for aluminum cladding with an air inlet temperature of 49°C (120°F)] (Shedrow, 1994a; Taylor et al., 1994).

Thus, the number of vault units/arrays required is as follows:

- spent nuclear fuel decay heat exceeding 80 Watts per element: 35 vault units;

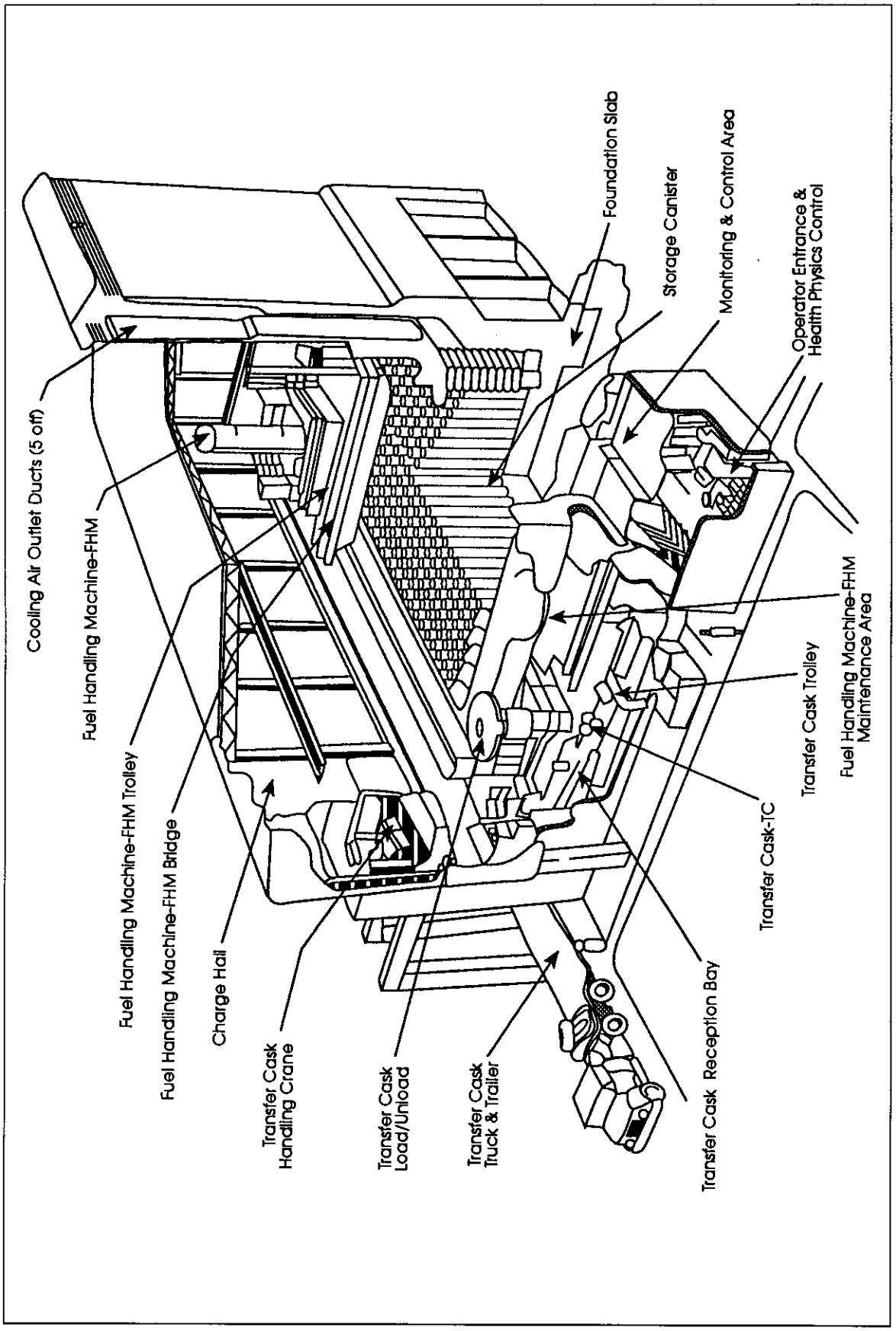


Figure F-33 Vault Elevation View

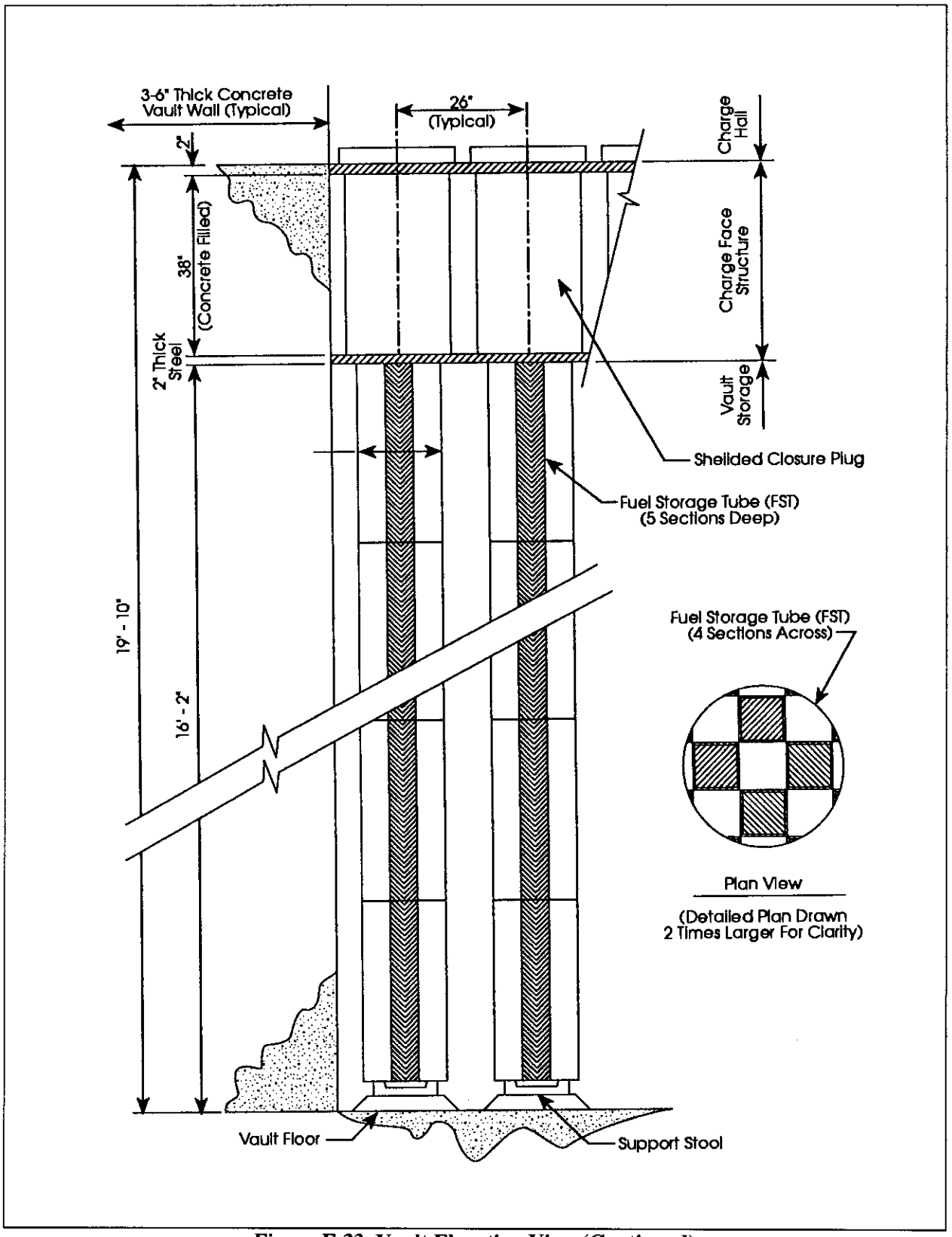


Figure F-33 Vault Elevation View (Continued)

Table F-17 Summary of Modular Dry Vault Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3.7 ha (9 acres)
Facility: Size (Area) Concrete Steel	 5,000 m^2 (54,000 ft^2) 21,800 m^3 (28,500 yd^3) 5,200 mt (5,750 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	835,000 l (221,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	190/yr average, 234/yr peak
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$370 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) for first 10 years, 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams	
Solid Low Level Waste	22 m^3/yr (780 ft^3/yr) during receipt, 1 m^3/yr (35 ft^3/yr) thereafter
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt, 8 thereafter
Annual Operating Cost	$15.6 million during handling, $0.6 million during storage[b]

[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke et al., 1994).*

[b] *Cost estimates are in $1993 (EG&G, 1993)*

- spent nuclear fuel decay heat between 40 and 80 Watts per element: 32 vault units; and
- spent nuclear fuel decay heat between 10 and 40 Watts per element: 28 vault units.

For "cold" fuel (10 Watts per element), potentially more than 44 spent nuclear fuel canisters could be placed per vault unit. This would require a customized design. Figure F-34 displays the 10 to 40 Watts and 80 Watts per element cases.

Higher decay heat foreign research reactor spent nuclear fuel would have to be temporarily stored in the small wet storage pool. The storage period is not expected to exceed 3 years.

Criticality concerns are addressed by fuel geometry within the canister and by the use of nuclear poisons (e.g., borated steel in the baskets, etc.). Vault geometry is used to maintain a minimum spacing between adjacent fuel elements or groups of fuel elements to prevent criticality. Nuclear poisons absorb neutrons, thus preventing criticality.

The vault/canyon design has been licensed by the NRC for a specific site. It represents a complete stand-alone facility that can be dedicated to foreign research reactor spent nuclear fuel without requiring the utilization of any other facilities at the host site. Cask handling, maintenance, spent nuclear fuel loading, spent nuclear fuel inspection, spent nuclear fuel storage, and (potentially) characterization can all be accomplished within the same facility.

The cost to construct a modular dry vault storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the vault storage

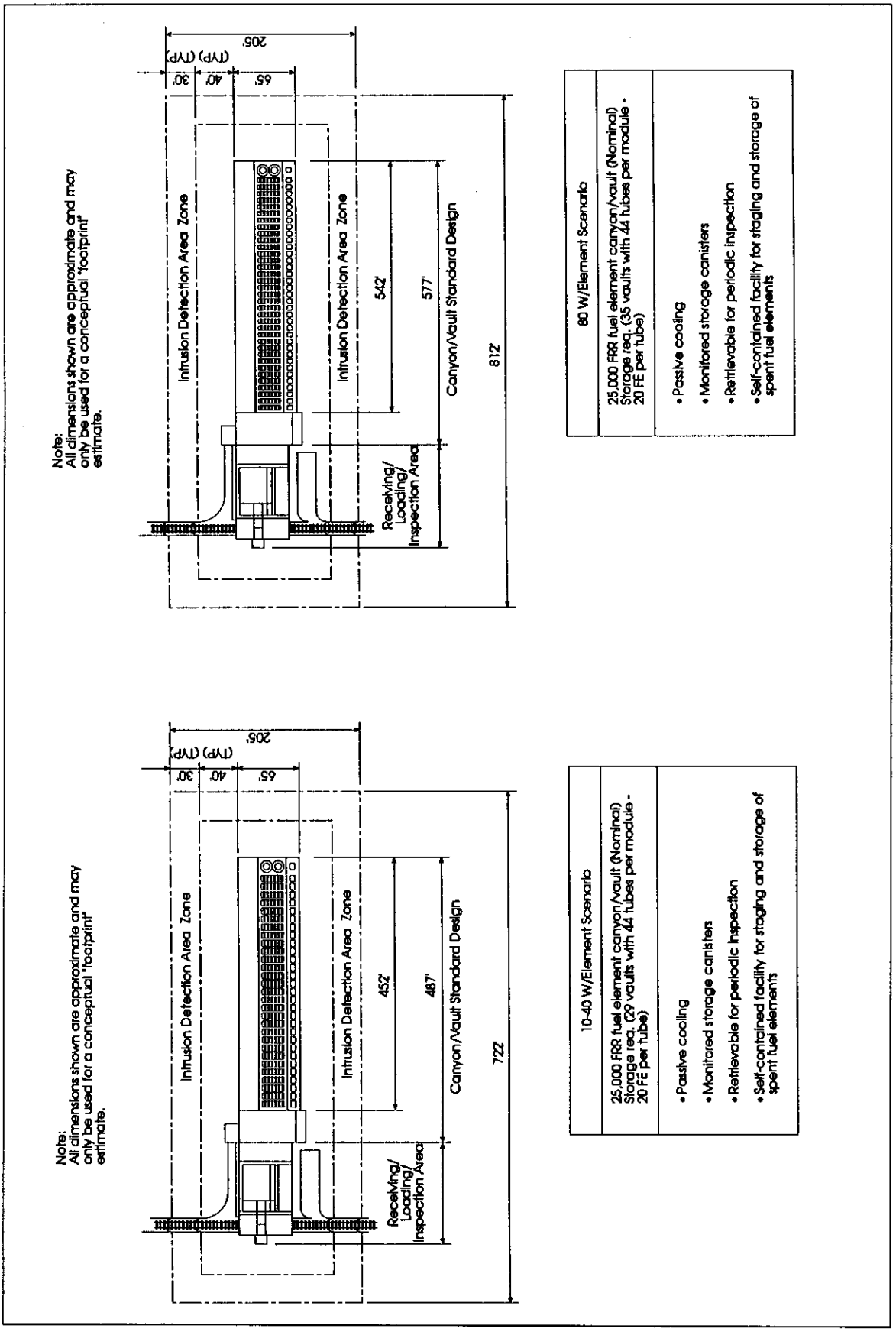


Figure F-34 Canyon/Vault Standard Design (10 to 40 Watt and 80 Watt Scenarios)

area is estimated to be $370 million. The annual operating cost for this facility is estimated to be $15.6 million during the period of handling and transfers of the spent nuclear fuel and $0.6 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

F.3.1.2 Spent Nuclear Fuel Storage Using Dry Casks

The dry cask storage approach consists of the following components (BG&E, 1989; Duke Power Company, 1988; Shedrow, 1994a and 1994b; Taylor et al., 1994; Claxton et al., 1993):

- a staging facility for cask receipt and unloading and for loading foreign research reactor spent nuclear fuel into the dry storage casks [a wet pool is used for this purpose, and for short-term (1 to 3 years) storage of foreign research reactor spent nuclear fuel with a heat load exceeding 40 Watts per element],
- an inspection/characterization facility, for examining fuel integrity and canning degraded spent nuclear fuel as required (this may be incorporated into the staging facility as an inspection cell or be immediately adjacent to it),
- a dry storage cask (usually concrete) [this provides for the shielding and the structural stability of the spent nuclear fuel storage],
- a transfer mechanism, such as a dedicated truck/trailer combination with a ram for horizontal modules, or a crane for vertical modules, and
- a separate fuel canister which may or may not be used [if used, it is typically around 4.6 m (15 ft) long and 1.7 m (5.5 ft) in diameter and weighs around 32 metric tons (36 tons)].

The dry cask approach requires the staging facility to receive and inspect the spent nuclear fuel shipment. The transportation cask would be unloaded in a small wet pool within the facility. Subsequently, the spent nuclear fuel is loaded into the dry cask (or spent nuclear fuel canister for the horizontal cask), and the cask is placed upon an outside concrete slab. The horizontal approach uses a dry spent nuclear fuel transfer canister for containing the spent nuclear fuel. This is placed within a shielded transfer cask and moved to the outside modular storage facility. A hydraulic ram inserts the transfer canister inside the horizontal storage module, followed by sealing with a shield plug.

The dry storage modules are designed to withstand normal loads and design basis accident effects, such as earthquakes, tornadoes, and floods. The concrete provides radiation shielding for gamma rays and neutrons. Natural air circulation dissipates the heat as air enters through inlet vents near the bottom of the cask, passes around the spent nuclear fuel canister, and exits near the top. Screens and grills keep birds and other animals out of the cooling duct area. Some of the candidate sites have facilities which may be used for cask receipt and unloading and spent nuclear fuel inspection and transfer to storage.

The application of dry cask storage technology to foreign research reactor spent nuclear fuel depends upon the heat load. Horizontal casks are anticipated to be slightly more restrictive than the vertical casks with respect to the heat load, and are thus the focus of the discussion. The standard design for a horizontal fuel canister provides for 24 or 52 sleeves (i.e., Pressurized Water Reactor or Boiling Water Reactor spent nuclear fuel, respectively), each about 4.6 m (15 ft) long. As with the vault approach, it is conservatively assumed that each sleeve contains five foreign research reactor spent nuclear fuel elements (i.e., in layers), within a basket or can arrangement for maintaining spacings and retrievability. As with the vault

approach, the number of dry storage casks depends upon the decay heat of the spent nuclear fuel and a cladding temperature limit [175°C (347°F) for aluminum cladding, with an air inlet temperature of 49°C (120°F)]. The 24-sleeve design allows for a maximum of 120 elements of foreign research reactor spent nuclear fuel with 40 to 80 Watts per element of decay heat, while the 52-sleeve design provides for a maximum of 260 elements per dry storage cask. Thus, the number of casks required is as follows:

- decay heat between 40 and 80 Watts per element: 205 casks, and
- decay heat between 10 and 40 Watts per element: 94 casks.

Note that these values are very conservative and correspond to a maximum of around 40 percent of the NRC-licensed heat loads per cask. Initially, foreign research reactor spent nuclear fuel with higher heat loads could be unsuitable for the dry storage cask pending detailed heat transfer analysis and a determination of limiting fuel storage temperature for aluminum-clad and TRIGA spent nuclear fuel. However, such relatively high decay heat fuel represents a small percentage of the currently identified foreign research reactor spent nuclear fuel so that its impact would be small; and after 1 to 5 years of wet storage, it would all be below a heat duty of 80 Watts per elements. The storage approach assures a minimum spent nuclear fuel wet storage time of 3 years after discharge prior to dry storage. This would essentially ensure that all foreign research reactor spent nuclear fuel is below a heat output of 40 Watts per assembly.

Figure F-35 displays approximate layouts for the dry cask storage facility predicated upon a horizontal cask design. Table F-18 summarizes some general parameters of dry cask storage.

The dry storage cask technology requires a separate staging facility for foreign research reactor spent nuclear fuel unloading, canning, and storage cask loading and transportation cask maintenance. This facility has the following operational areas:

- *Transportation Cask Handling*: This incorporates cask maintenance, truck/railcar unloading, decontamination/washdown, radioactive material control, and cask sampling/flushing/degassing.
- A small wet pool for fuel transfer and short-term storage.
- *Spent Nuclear Fuel Unit Handling*: Fuel removal, decontamination, fuel drying, fuel canning, inerting, and thermal measurements.
- *Spent Nuclear Fuel Unit Transfer*: This constitutes placement of the spent nuclear fuel into the cask or canister, followed by sealing.
- *Radwaste Treatment*: This includes collection, treatment, and preparation for disposal of contaminated effluents and radwaste treatment and solidification.
- *Heating, Ventilation, and Air Conditioning*: This represents heating, ventilation, and air conditioning of the facility so that contamination of the workers and the environment is avoided.

The inspection/characterization facility includes a shielded dry hot cell for spent nuclear fuel analysis and examination, and canning of degraded spent nuclear fuel. All equipment and instrumentation within the cells is remotely operated to provide chemical, physical, and radiological properties, as needed. The facility is maintained under negative pressure with exhaust through High Efficiency Particulate Air filters

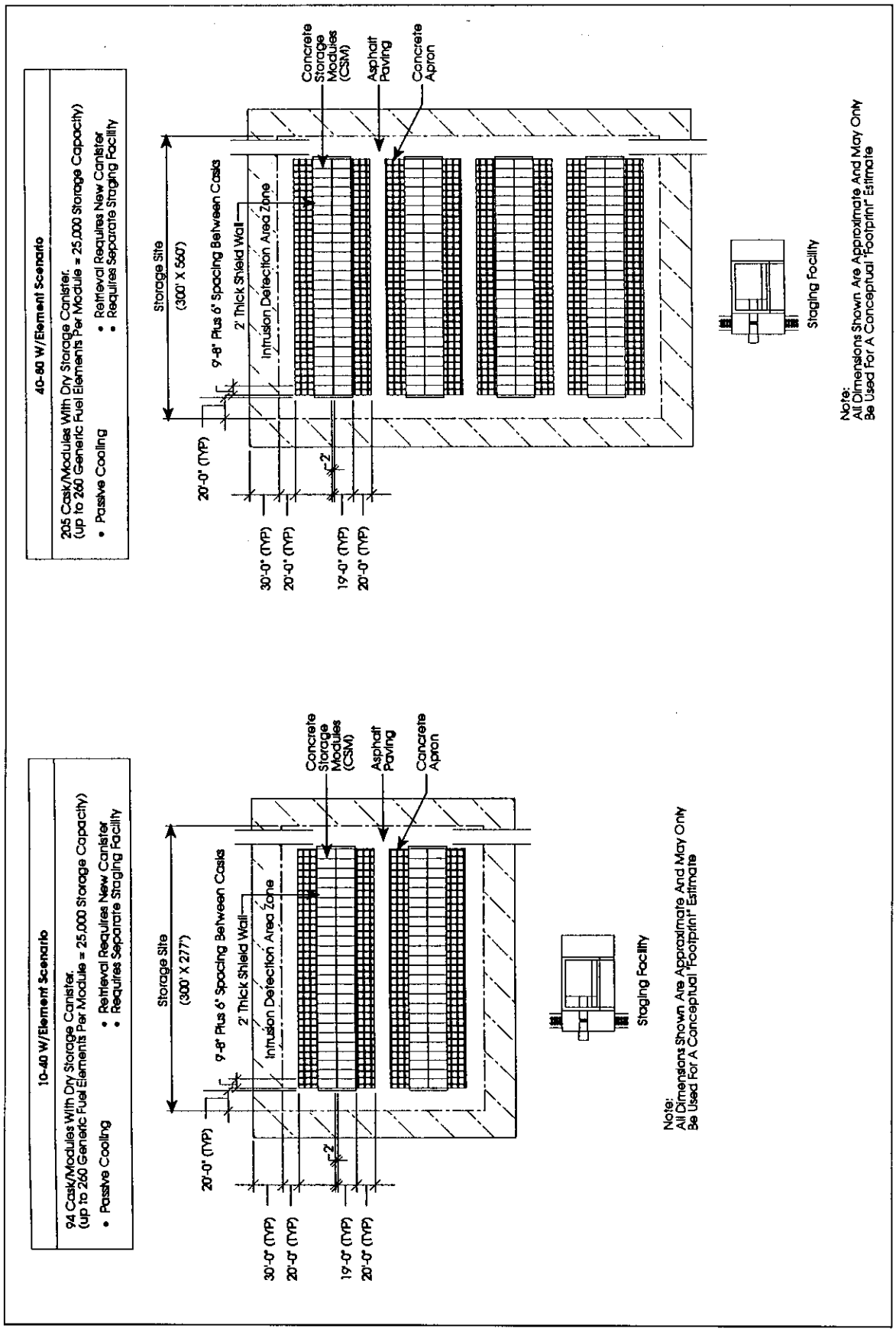


Figure F-35 Schematic of Expanded Dry Cask Storage Facility (10 to 40 Watts Scenario and 40 to 80 Watts Scenario)

Table F-18 Summary of Dry Cask Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3 ha (7.7 acres)
Facility: Size (Area) Concrete Steel	 2,200 m^2 (24,000 ft^2) 17,500 m^3 (22,900 yd^3) 4,500 metric tons (5,000 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	810,000 l (214,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	50/yr for staging facility, 50 per 25 cask array, 1 array/yr
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$366 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/yr) during receipt, 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams	
Solid Low Level Waste	16 m^3/year (565 ft^3/year) during receipt, 1 m^3/yr (35 ft^3/yr) thereafter
Waste Water	1.58 million l/yr (412,000 gal/year) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt, 8 thereafter
Annual Operating Cost	$17.3 million during handling, $0.3 million during storage[b]

[a] *Staging facility parameters based upon the Regionalized, Small Wet Pool (Dahlke et al., 1994)*

[b] *Cost estimates are in $1993 (EG&G, 1993)*

to mitigate the environmental effects of any radionuclide releases. This facility is normally located immediately adjacent to, or within, the staging facility.

Dry cask storage is unique among the three storage technologies because of its ability to be operationally integrated with existing facilities, which allows for faster implementation as compared to the other two storage technologies. Several DOE sites have facilities with spent nuclear fuel handling capabilities similar to the requirements of the staging facility. Potential examples include the RBOF at the Savannah River Site and the ICPP-666 storage pool area. For dry cask storage, the spent nuclear fuel would be shipped to the existing facility and unloaded from the transportation cask. The spent nuclear fuel would be inspected, canned if identified as a degraded element, and placed inside the storage canister. Spent nuclear fuel with heat loads exceeding 40 Watts per element would be stored in the existing facility to allow cooldown prior to cask storage. After filling, the canister would be sealed and placed inside the storage cask. The only new construction required would be the concrete storage pad (for vertical casks) or the concrete storage modules (for horizontal casks). For the foreign research reactor spent nuclear fuel receipt rate of approximately 2,000 elements per year considered in the analyses in this EIS, approximately 8 storage casks would be needed annually.

The cost to construct a dry cask storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the cask storage area is estimated to be $366 million. The annual operating cost for this facility is estimated to be $17.3 million during the period of handling and transfers of the spent nuclear fuel and $0.3 million during the period of storage.

The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

F.3.2 Wet Storage Facility

Three generic wet storage facility options have been proposed for foreign research reactor spent nuclear fuel. They are denoted Centralized-Underwater Fuel Storage Facility, Regionalized Large-Underwater Fuel Storage Facility, and Regionalized Small-Underwater Fuel Storage Facility (Dahlke et al., 1994). The difference between these 3 options is that Centralized-Underwater Fuel Storage Facility is sized to store 100 percent of the foreign research reactor spent nuclear fuel under consideration in this EIS (Figure F-36), Regionalized Large-Underwater Fuel Storage Facility is designed for the storage of 75 percent of the foreign research reactor spent nuclear fuel, and Regionalized Small-Underwater Fuel Storage Facility will accommodate 25 percent of the foreign research reactor spent nuclear fuel. These three options were selected to encompass any conceivable decision regarding centralization or regionalization (by geography or fuel type) for the foreign research reactor spent nuclear fuel storage sites. The design features of all three wet storage facility options are identical with the exception that building and pool sizes and, in the case of the Regionalized Small-Underwater Fuel Storage Facility, the number of storage pools and receiving bays is smaller for the Regionalized Large-Underwater Fuel Storage Facility and Regionalized Small-Underwater Fuel Storage Facility. Table F-19 presents the difference in design between these three facilities. Because the design and environmental impacts of the larger Centralized-Underwater Fuel Storage Facility would bound the two smaller facility designs, the balance of the presentation in this section addresses the specific design of the Centralized-Underwater Fuel Storage Facility for storage of 100 percent of the foreign research reactor spent nuclear fuel.

The proposed new wet storage facilities consist of a fuel storage area and support areas (Dahlke et al., 1994). The Fuel Storage Area provides for the receipt of cask transportation vehicles, cask unloading and decontamination, fuel handling, transfer, and storage. Support areas provide for the equipment necessary to maintain and operate the storage area (e.g., heating, ventilation, air conditioning, water treatment, and waste management). The wet storage facility would be constructed as a structure that meets all current nuclear regulations for withstanding natural events such as seismic, tornado, and flood, as well as aircraft impact loads. All systems supporting the operation of the fuel storage facility would also meet these safety requirements. The facility is equipped with a 118-metric ton (130-ton) overhead cask handling crane, and a 9-metric ton (10 ton) fuel handling crane. Each cask transportation vehicle would enter the facility through one of two bays, where it would be monitored and washed from transportation dust. When the external surfaces are cleaned, the cask would be placed into a decontamination room where the cask would be prepared as needed to facilitate underwater unloading. The cask would then be placed in an unloading pool. Transportation casks would be monitored and, if clean of radioactive contamination, placed in an unloading pool. The cask receiving area can accept two simultaneous shipments on 3 m (10 ft) by 24.4 m (80 ft) trucks or railcars and casks weighing up to 114 metric tons (126 tons), each with a total individual cask and transport vehicle weight of 177 metric tons (195 tons).

There are two stainless steel-lined unloading pools, one measuring 6.4 m (21 ft) long, 5.8 m (19 ft) wide by 13.4 m (44 ft) deep, and the other measuring 6.1 m (20 ft) long, 6.1 m (20 ft) wide, and 11 m (36 ft) deep. There are two decontamination hot cells. Each unloading pool has a cask washdown system. Prior to being placed in one of the two storage pools, each fuel element would be checked to ensure that it is properly configured for direct transfer to the fuel storage pool buckets. If not, it would be transferred to the fuel cutting/canning pool, which is 10.4 m (34 ft) long, 5.8 m (19 ft) wide, and 9.4 m (31 ft) deep. Here it would be prepared for transfer to the storage pool buckets.

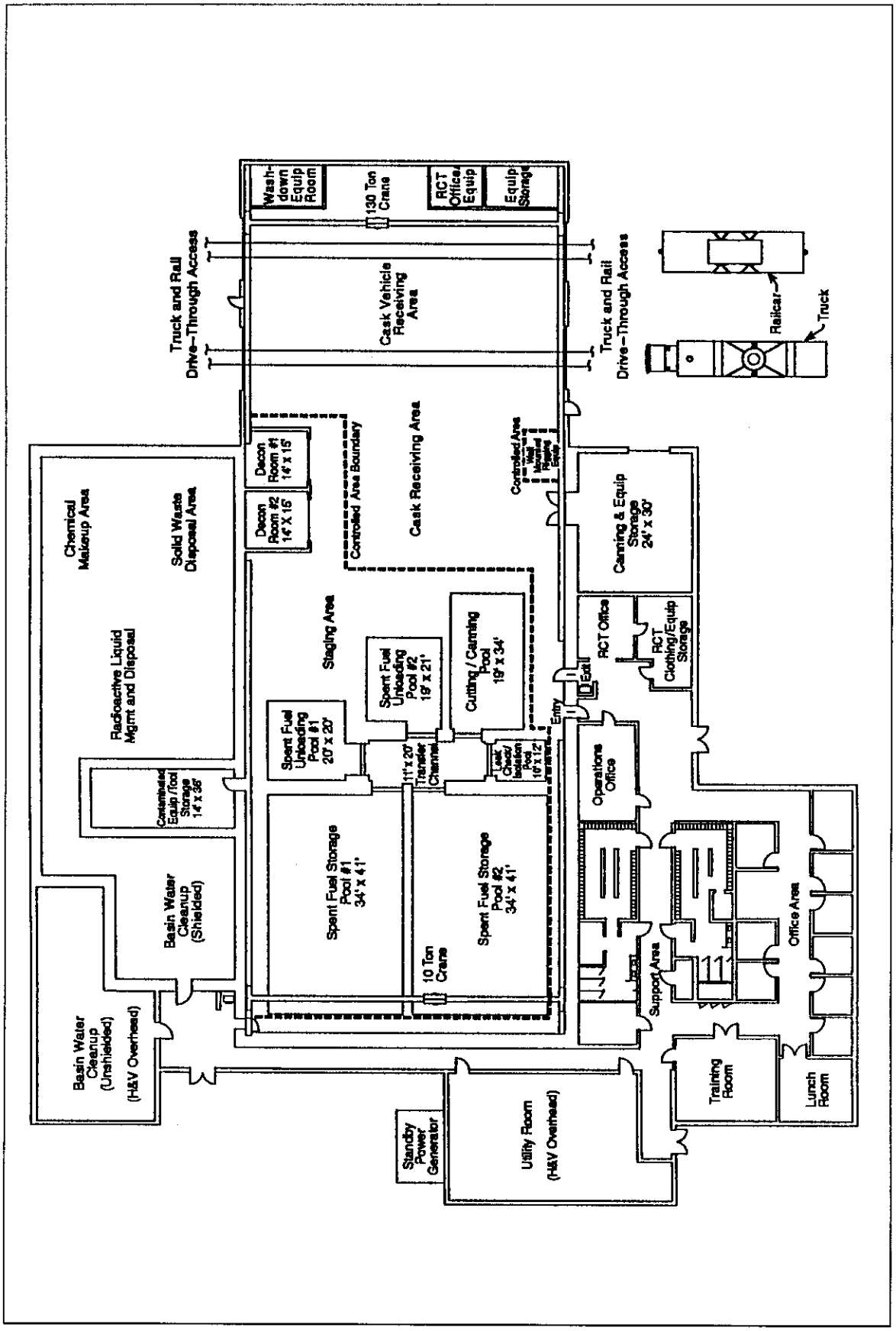


Figure F-36 Generic Wet Storage Facility for All of the Foreign Research Reactor Spent Nuclear Fuel

Table F-19 Design Difference Between 100 Percent, 75 Percent, and 25 Percent Generic Wet Storage Facilities

Design Parameter	*Wet Storage Capacity (Amount of Foreign Research Reactor Spent Nuclear Fuel)*		
	100%	*75%*	*25%*
Number of Storage Pools	2	2	1
Storage Pool Length and Width, m (ft)	16.5 x 10.4 (54 x 34)	12.5 x 10.4 (41 x 34)	10.4 x 8.2 (34 x 27)
Transfer Channel Length and Width, m (ft)	6.1 x 3.4 (20 x 11)	6.1 x 3.4 (20 x 11)	6.1 x 3 (20 x 10)
Fuel Unloading Pool Length and Width, m (ft)	6.4 x 5.8 (21 x 19)	6.4 x 5.8 (21 x 19)	6.1 x 6.1 (20 x 20)
Number of Receiving Bays	2	2	1

If cask measurements indicate that fuel is degraded, the fuel would be transferred to the isolation pool which is 3.7 m (12 ft) long, 3 m (10 ft) wide, and 9.4 m (31 ft) deep. This pool is equipped so that wet sipping, dry sipping, or vacuum sipping of the suspect fuel element could be performed. Sipping is a method of measuring radioisotope leakage from spent nuclear fuel. An identified degraded fuel element would then be transferred to the cutting/canning pool, where it would be canned before transfer to the storage pool. If it is not found to be degraded, it would be transferred directly to the storage pool.

All six pools in this facility (two unloading; two storage, cutting/canning; and two leak check/isolation) are hydraulically connected by a stainless steel-lined transfer channel/pool which is 6.1 m (20 ft) long, 3.3 m (11 ft) wide, and 9.4 m (31 ft) deep. Gates between this transfer channel and each pool allow for hydraulic watertight isolation of the other pools to control contamination and allow for individual pool water pump-out. All pools and channels are constructed of concrete with stainless steel floors and liners. Pool water leak detection and collection systems are provided in accordance with NRC Regulatory Guide 1.13 (NRC, 1975) and American National Standards Institute Standard N305-1975 (ANSI, 1975b).

Each of the two stainless steel-lined, interconnected storage pools is 16.5 m (54 ft) long, 10.4 m (34 ft) wide, and 9.4 m (31 ft) deep. Each contains stainless steel storage racks which hold 1,000 fuel storage holes, with a 0.2 m (8 in) spacing maintained between adjacent storage holes (Figure F-37). Each storage hole can hold three stacked stainless steel fuel storage buckets (Figure F-38), which can each contain up to four fuel elements. A loading fixture is used during spent nuclear fuel emplacement. Thus, each pool has the capacity for 12,000 fuel elements (or 24,000 for both pools). The 0.2-m (8-in) space provides neutron isolation between adjacent storage holes and, therefore, ensures criticality safety. Each rack is 2 m (6.7 ft) square and 3.2 m (10.4 ft) high, and consists of a 5 x 5 array of 25 fuel positions. A hinged lid is above each of these fuel positions. Each pool can hold 40 of these racks. Fuel elements are stored in the racks so that at least 25.4 cm (10 in) of rack protrudes above the top of the fuel.

The heating, ventilation, and air conditioning system for the wet storage facility would include a room for air supply equipment and a room for air exhaust equipment with separate filtering and monitoring rooms for the different areas within the facility. There are two heating, ventilation and air conditioning equipment rooms. Although a total of six different rooms are used for heating, ventilation, and air conditioning, all exhaust air is directed through pre-filters, High Efficiency Particulate Air filters, radiation monitors, filter fire protection components, and heat recovery coils before it exhausts to the atmosphere.

The wet storage facility's water treatment system consists of redundant pumps, piping, filters, deionizers, and ultraviolet microorganism control systems. A heat removal system is sized to maintain the bulk water temperature at or below 43°C (109°F). The system's filters and deionizers include anion and cation exchangers that maintain water chemistry and remove radionuclides from the pool water.

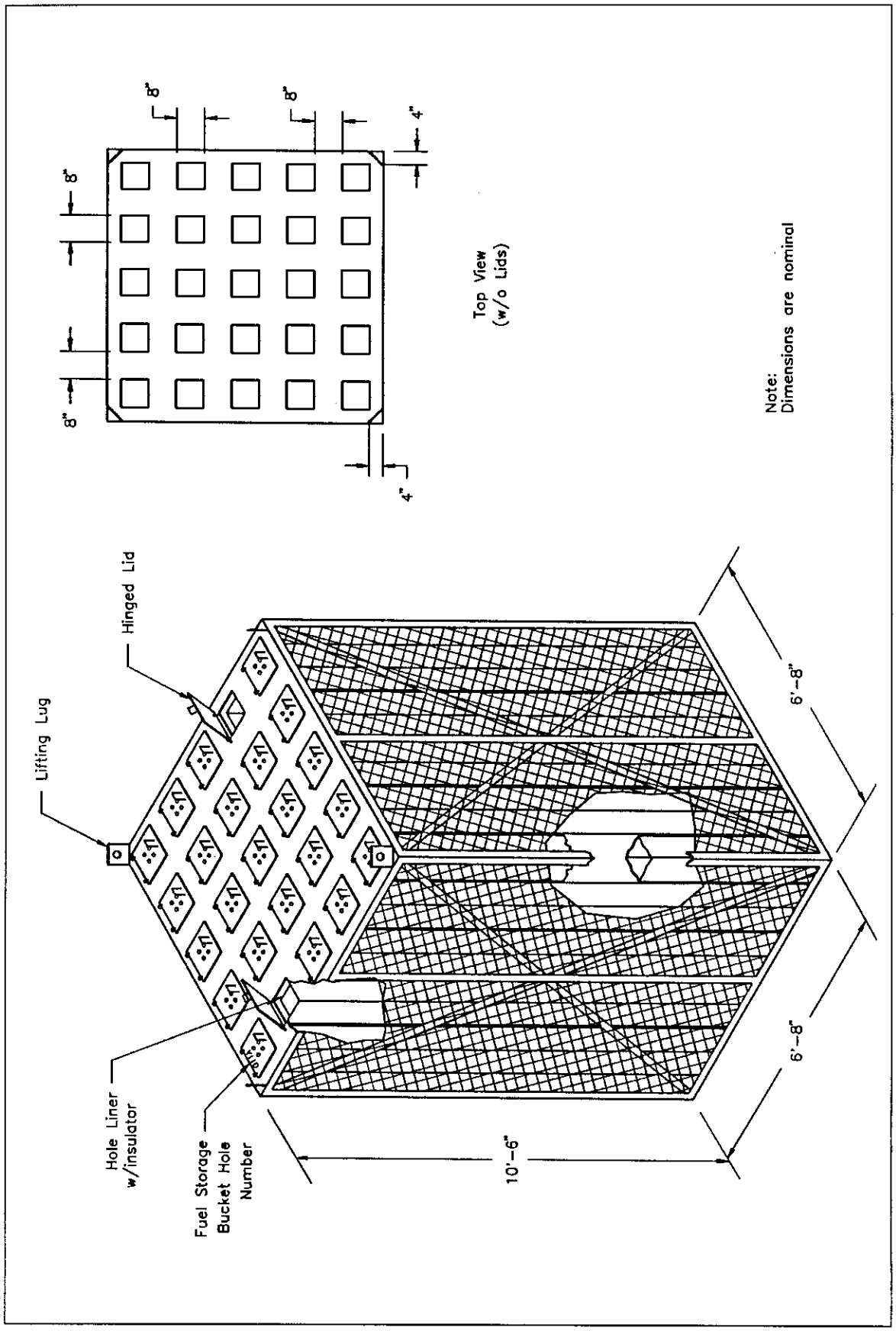


Figure F-37 Storage Racks

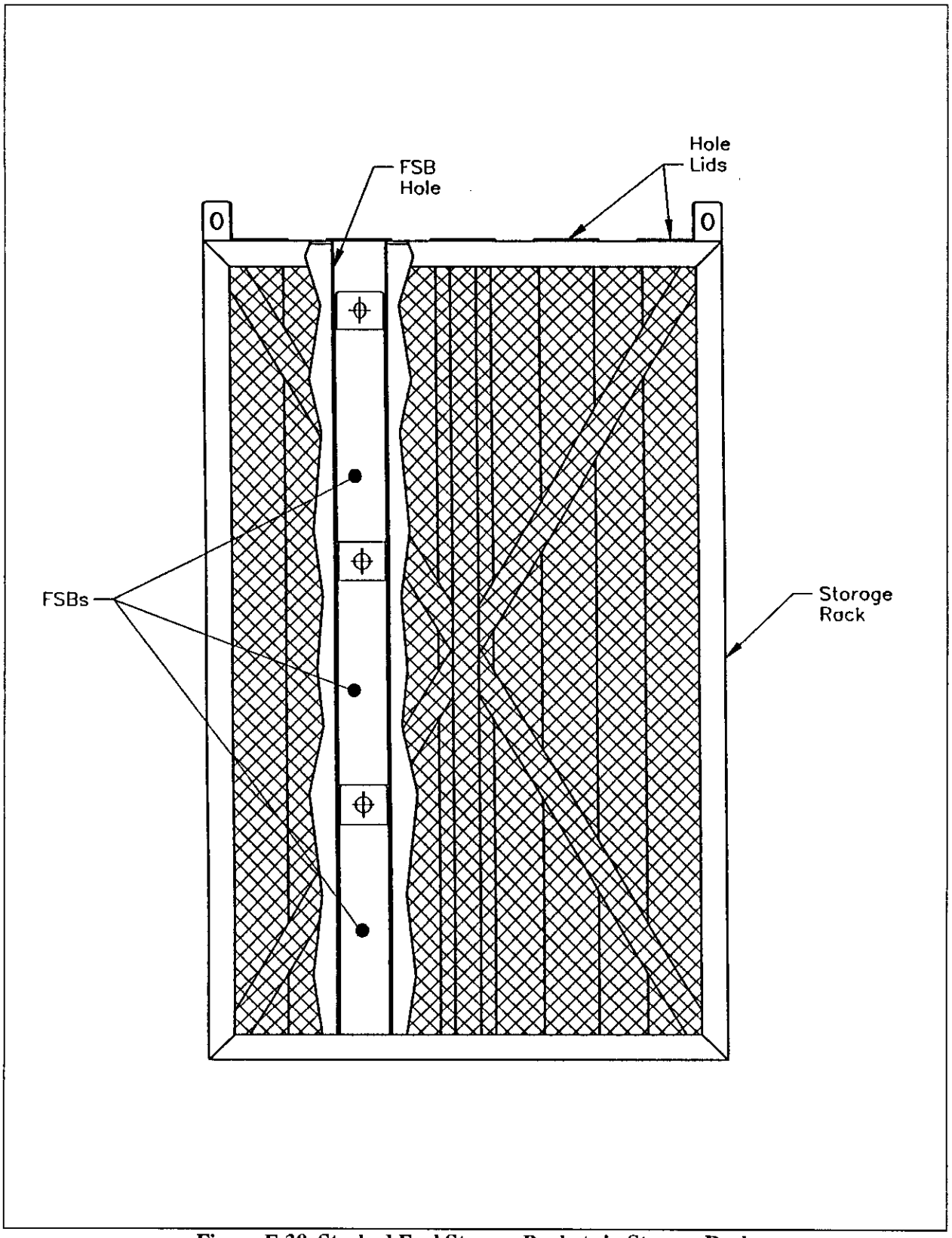


Figure F-38 Stacked Fuel Storage Buckets in Storage Rack

The staff required to operate the wet storage facility is estimated to be a maximum of 30 when 24-hour-a-day fuel loading is being performed, with only occasional maintenance visits by administrative personnel for operation. Potential radiological consequences are extrapolated from other operating wet storage facilities and are discussed in Section F.4.

No high-level radioactive waste is expected to be generated by the wet storage facility. Low-level solid radioactive waste generated over the 40-year life of the facility is expected to be about 488 m^3 (17,200 ft^3). Nonradioactive solid waste generated over the facility's life is expected to be about 300 m^3 (10,594 ft^3). No nonradioactive air emissions are expected to be generated by this facility. Table F-20 summarizes the parameters for the facility.

The cost to construct a wet storage facility with a staging area sufficient to unload, characterize, can, and transfer the spent nuclear fuel to the storage area is estimated to be $449 million. This cost may include some duplicate facilities and equipment present in both the staging facility and the rest of the wet storage facility which were costed separately. The annual operating cost for this facility is estimated to be $23.3 million during the period of handling the spent nuclear fuel and $3.5 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993).

F.3.3 Site-Specific Facilities Proposed for Foreign Research Reactor Spent Nuclear Fuel Storage and Management

F.3.3.1 Savannah River Site

RBOF

The Savannah River Site has proposed the use of its RBOF, which is also designated as Building 244-H (DuPont, 1983a and 1983b; Shedrow, 1994a and 1994b; WSRC, 1994a; Claxton et al., 1993; DOE 1993c). The RBOF is a 30-year-old steel and concrete block structure that contains several water pools that have been used for the storage of spent nuclear fuel including foreign research reactor spent nuclear fuel since approximately 1964.

The RBOF facility is located in H-Area on 0.8 ha (2 acres) of land about 397 m (1,300 ft) west of the 221-H Canyon building. A railroad track terminates within the facility, and a roadway surrounds it for access by trucks. The RBOF Building (244-H) is about 42 m (139 ft) wide and 45 m (148 ft) long and contains water-filled basins. The basin area extends below grade to a maximum depth of 13.7 m (45 ft), the roof over the 91 metric ton (100 ton) crane bay is about 13.7 m (45 ft) above grade, and most of the remainder of the roof is at an elevation of 4.6 m (15 ft).

The building consists of seven main sections separated by partition and shielding walls. A ventilation system is provided to exhaust any airborne particulate contamination through filters. The basins, cubicles, and shielding walls are made of reinforced concrete. Most of the above-grade structure consists of standard structural steel shapes with an exterior wall of Transite™ (registered trademark of Johns-Manville Co.). The walls are insulated with Fiberglas™ (registered trademark of Owens-Corning Corp.).

The basin, or working area, of the building has an inner wall of Transite™ to prevent water damage to the insulation from condensation. The disassembly and inspection basins are separated by an inner concrete block wall, and the repackaging basins are enclosed by concrete block walls.

Table F-20 Summary of Wet Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel

Construction Phase:	
Disturbed Land Area	2.8 ha (7 acres)
Facility: Size (Area) Concrete Steel	 3,800 m^2 (41,000 ft^2) 12,400 m^3 (16,260 yd^3) 3,100 metric tons (3,443 tons)
Soil Moved	18,000 m^3 (24,000 yd^3)
Equipment Fuel	600,000 l (159,000 gal)
Construction Debris/Waste	2,600 m^3 (10,300 yd^3)
Work Force	157/yr average, 184 peak
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$449 million[a,b]
Operation Phase:	
Electricity	1,000 - 1,500 MW-hr/yr
Water	2.7 million l/yr (720,000 gal/yr) during receipt, 1.5 million l/yr (409,000 gal/yr) thereafter
Wastestreams	
High Level Waste	none
Transuranic Waste (TRU)	none
Solid Low Level Waste	16 m^3/yr (580 ft^3/yr)
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30
Annual Operating Cost	$23.3 million during handling $3.5 million during storage[a]

Source: (Dahlke et al., 1994)

[a] *Cost estimates are in $1993 (EG&G, 1993)*

[b] *The cost may include duplicate equipment costed in both the staging facility and the wet storage facility*

The two storage pools are between 6.7 and 8.8 m (22 and 29 ft) deep, and approximately 70 percent of their capacity is filled with a variety of fuel types, including aluminum-clad fuel with a ^{235}U enrichment up to 93.91 percent. Subcriticality is maintained by appropriate fuel spacing [center-to-center fuel spacing in these racks currently varies between 23 and 65 cm (9 and 25.5 in) depending on the specific rack] in the storage racks, since no neutron absorption material is used in the pool water. Rack height is 3.4 m (11.17 ft), but some fuel protrudes above the top of the racks. Some of this fuel has been stored at the RBOF for as long as 15 years without any significant degradation. These aluminum storage racks have been present in the pools for 30 years without degradation. Figure F-39 shows the floor plan, and Figure F-40 displays an elevation view.

The RBOF includes specific design, operating, and maintenance procedures for the receipt of a wide variety of fuel types and casks, including damaged fuel elements. The RBOF has the facilities and experience in all aspects of spent nuclear fuel receipt including cask wash, fuel unloading, fuel transfer, fuel storage, fuel inspection, fuel disassembly, and fuel repackaging.

The RBOF pools have a stainless steel bottom and epoxy-coated walls. Pool walls are made of reinforced concrete that varies in thickness from 0.9 m (3 ft) at the top of the pool to 2 m (6.5 ft) at the lower elevations, and the pool floor is a stainless steel liner over a 91-cm (3-ft) thick reinforced concrete slab. Most of the pools have a 6.4-mm (0.25-in) thick stainless steel liner on the floor. The disassembly, inspection, and repackaging basins also have a 3.2-mm (0.125-in) thick stainless steel liner on the walls. Pools or basins are connected by transfer canals with underwater doors.

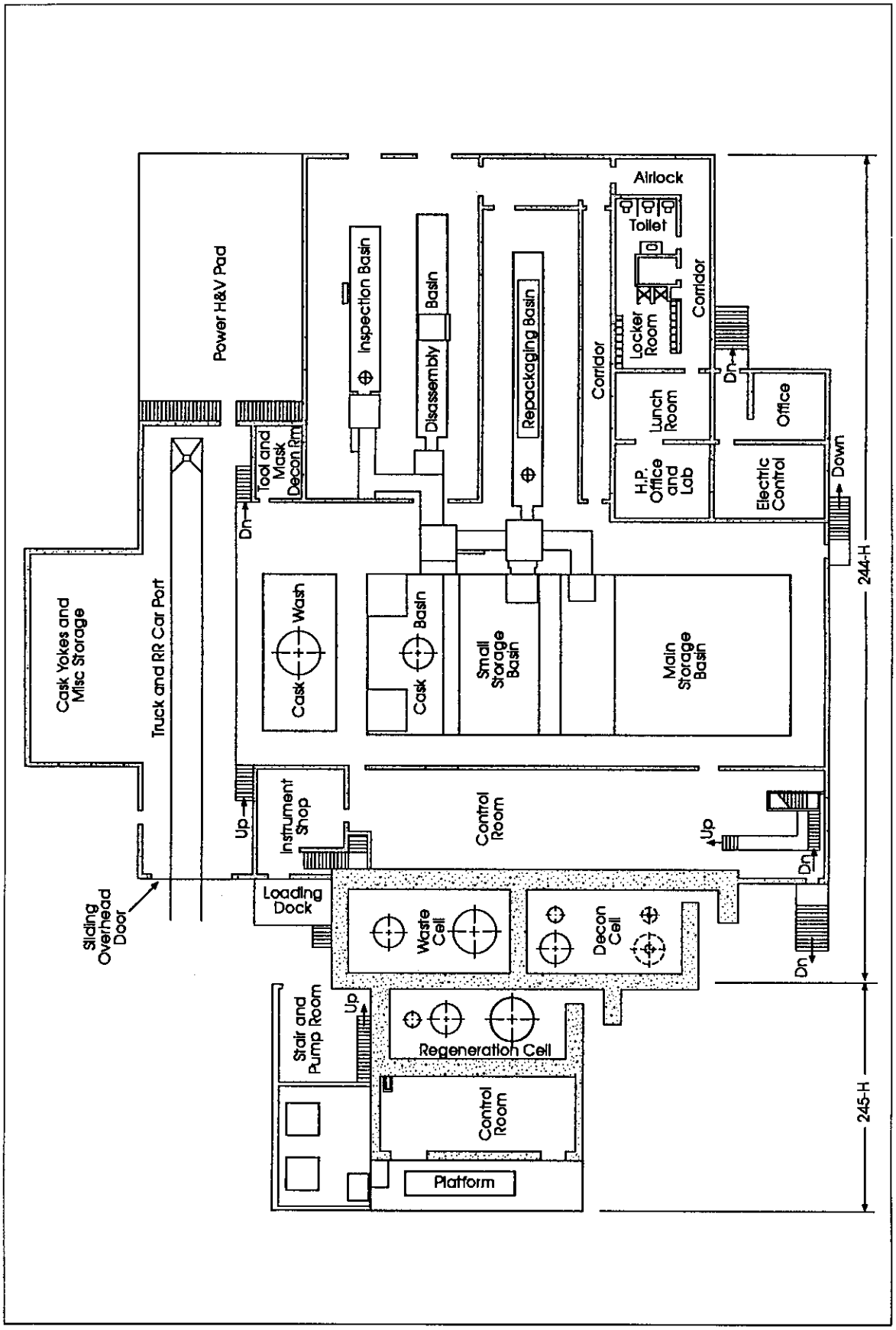


Figure F-39 Plan View of the RBOF

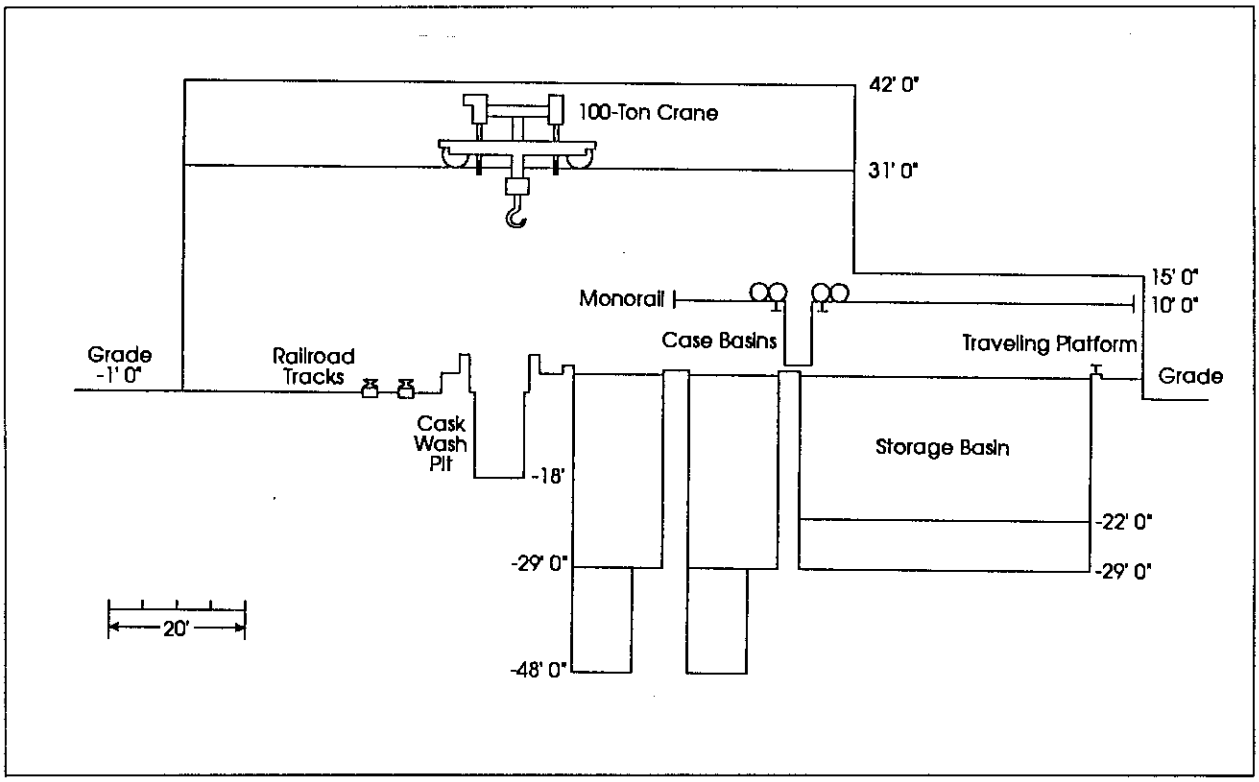


Figure F-40 Elevation Schematic of the RBOF (Facing East)

Water from any basin can be pumped through a filter-deionizer and then returned to the basin as purified water with a conductivity in the range of 0.5 to 1.5 μmhos/cm. In addition, the activity level of the water, which is typically in the range of 0.5 to 1.5 x 10^{-4} μCi/ml, is reduced to less than 0.05 x 10^{-4} μCi/ml by this process. The normal inventory of activity in the approximately 1,700,000 l (450,000 gal) of total basin water is thus 0.1 to 0.3 Ci. For typical flow rates of 454 l/min (120 gal/min), the deionizer processes approximately 1.1 x 10^8 l (3 x 10^7 gal) during 6 months of service and may contain about 20 Ci of radioactivity (mostly as cesium) when regeneration is required. The deionizer is typically regenerated every 4 to 5 months. The "Porostone" (aluminum oxide) filter that precedes the deionizer is normally backflushed and recoated with filter-aid whenever a significant pressure drop occurs, which, in practice, is about three to four times per month. When the filter tubes become plugged, they are chemically treated with oxalic acid and sodium hydroxide to open the pores of the filter. This occurs once or twice per year. The purification system maintains excellent chemistry, with mercury and copper kept below two parts per billion, and iron and aluminum maintained below 2 parts per million (ppm). Chloride is maintained below 10 parts per billion.

In the event of an interruption of normal power to the RBOF, critical equipment essential for maintaining personnel safety and containing radioactivity are automatically supplied with emergency power from a 12.5-kilovolt amps, 10-kilowatts, 460-volts gasoline-driven generator.

The building ventilation system serves to minimize airborne radioactivity both inside and outside the facility because of the "once-through" airflow system and the use of High Efficiency Particulate Air filters on the building exhaust. Because of the once-through airflow, activity levels do not tend to increase with time within the building, and the High Efficiency Particulate Air filters serve to effectively remove particulate radioactivity that would otherwise be released to the atmosphere. In addition, the facility is maintained at less than atmospheric pressure so that all building air will be filtered before release. The

basin areas are also kept at a lower pressure than the rest of the building, and a separate ventilation system supplies air to the control room, offices, and change room. All air, after filtration, is discharged through a 1.5-m (5-ft) diameter by 16.2-m (53-ft) high stack. The exhaust system for the process vessels in the Waste Cell and the Decontamination Cell is similar to the building system, but is separate and employs acid resistant components. This exhaust is discharged through a 25-cm (10-in) diameter by 16.2-m (53-ft) high pipe.

The 91-metric ton (100-ton) capacity bridge crane travels on a 27.4-m (90-ft) long runway located 9.4-m (31-ft) above grade which permits access to the carport, the cask wash pit, and the cask basin. It is used to handle transportation casks, cask lids, cask basin shims, and a semi-remote impact wrench.

The twin hook crane consists of two 45-metric ton (50-ton) capacity hoist trolleys, which can be arranged for independent travel, or which can be electrically locked to provide for operation as a single unit. Load clearance above the 1.1-m (3-ft 6-in) high cask basin railing is 8.1-m (26-ft 6-in). The bridge crane is pendant-operated from a walkway on the west side of the basins.

A 2.7-metric ton (3-ton) hoist, suspended from a monorail on the south girder of the bridge, is used in the handling of a semi-remotely operated impact wrench. The other 2.7-metric ton (3-ton) hoist is used primarily for the handling of yokes and other ancillary equipment in the yoke storage area adjoining the carport.

Brakes on the cranes and hoist are applied automatically in the event of a power outage.

Two small bridge cranes, one motorized and one manually operated, are employed over the repackaging basin. Both have a load capacity of 2.7 metric tons (3 tons).

The RBOF includes a High Efficiency Particulate Air heating, ventilation, and air conditioning system and maintains subatmospheric pressure within the building to minimize environmental releases of radionuclides. Automatic atmospheric isolation is actuated by activity level monitors inside the RBOF. The RBOF is also equipped with groundwater monitoring for detecting leakage from the pool confinement boundary.

Analysis of the RBOF was performed and included an evaluation of the reliability of process equipment and controls, administrative controls, and engineered safety features. The evaluation identified potential scenarios and radiological consequences. Risks were calculated in terms of 50-year population dose commitment per year (person-rem per year) to the onsite staff and to an individual at the plant boundary. Risk is defined as the product of the expected frequency of a release and the consequences of the release. Consequences are expressed in terms of dose commitment to onsite and offsite populations surrounding the release point.

An evaluation of the RBOF as a potential storage site for foreign research reactor spent nuclear fuel indicates a number of problem areas. The current cask handling capacity of the RBOF is approximately one cask per week. This capacity is based upon facility operations at two shifts per day, 5 days per week. The cask handling capacity could be increased, perhaps to as much as 84 casks per year, if facility operations were expanded to around-the-clock (3 shifts per day), 7 days per week. However, considering that shipments out of the RBOF also require cask handling, the net receipt capacity of the RBOF is practically limited to four casks per month. This capacity would not be sufficient for the potential foreign research reactor spent nuclear fuel cask receipt rate of ~60 casks per year. If the RBOF were used for the receipt and loading of dry storage canisters, its receipt rate could be reduced by half. Only ~1,000 fuel storage spaces are available at the RBOF. Consolidation of the spent nuclear fuel might open an additional 1,425 spaces, but this is much less than that required for the number of foreign research reactor spent

nuclear fuel elements under consideration in this EIS. The Savannah River Site has proposed movement of other spent nuclear fuel to the reactor storage basins, and use of dry storage for foreign research reactor spent nuclear fuel.

The DOE Spent Fuel Working Group Report has identified a number of vulnerabilities at the RBOF, including insufficient training, inadequate tornado missile protection, no seismic qualification, lack of water leak detection system, and no up-to-date and approved Safety Analysis Report (DOE, 1993b). It should be noted that a system description and a Safety Analysis Report for the RBOF do exist and were published in 1983. Current recommendations are to address and correct these problems by FY 1996 (DOE, 1993b; Taylor et al., 1994). The 30-year age of these pools may also require analyses to determine the remaining safe lifetime without significant replacement or design modifications.

Reactor Disassembly Basins

Savannah River Site has also proposed the use of one or more of its reactor disassembly basins for Phase 1 storage of foreign research reactor spent nuclear fuel (Shedrow, 1994a and 1994b; Taylor et al., 1994). All of these basins were constructed in the early 1950's and became operational in the mid-1950's. The disassembly basins are similar to each other and are briefly described in the sections that follow, using the L-Reactor disassembly basin as an example.

The L-Reactor performed the basic function of irradiating elements in a heavy water moderated and cooled reactor for the purpose of supplying special nuclear materials for national defense, medical, and research applications. The Savannah River Site production reactors are not currently operating. The disassembly area of the Savannah River Site Production Reactors was designed to serve as a processing area for reactor target and fuel assemblies. This processing included removal of decay heat, disassembly of components, short term storage of fissile product material, and cask loading operations. Total residence time from reactor discharge to shipment to the separation areas was typically 12 to 18 months.

The disassembly basin is arranged into three major sections: the machine basin, the vertical tube storage basin, and the transfer area (Figure F-41). The machine basin and vertical tube storage basins are divided into the following interconnected basins:

Vertical Tube Storage Basin	*Machine Basin*
Deposit and Exit Canal*	Machine Area
Vertical Tube Storage	Horizontal tube storage Target bucket storage Dry cave

* *The Deposit and Exit canal is a water-filled canal that connects the disassembly area and the area that houses the reactor tank top or process room. This canal also acts as a water seal to allow access by an underwater conveyer, but no airflow.*

The disassembly basin contains 12,776,000 l (3,375,000 gal) of water. Vertical tube storage holds 3,730,000 l (985,000 gal), while the machine basin holds 9,050,000 l (2,390,000 gal) (transfer area included). The depth of the disassembly basin ranges from 5.2 to 9.2 m (17 to 30 ft), but most of the basin is around 5.2 m (17 ft) deep. The approximate overall dimensions are 47 m (154 ft) wide by 66 m (216 ft) long. There is a small 0.9 m (3 ft) diameter circular section of the machine basin that is 15.6 m (50 ft) deep.

Figure F-42 shows a basic block diagram of the disassembly process. The reactor assemblies were discharged from the reactor tank to the Deposit and Exit canal, placed in the Deposit and Exit conveyer,

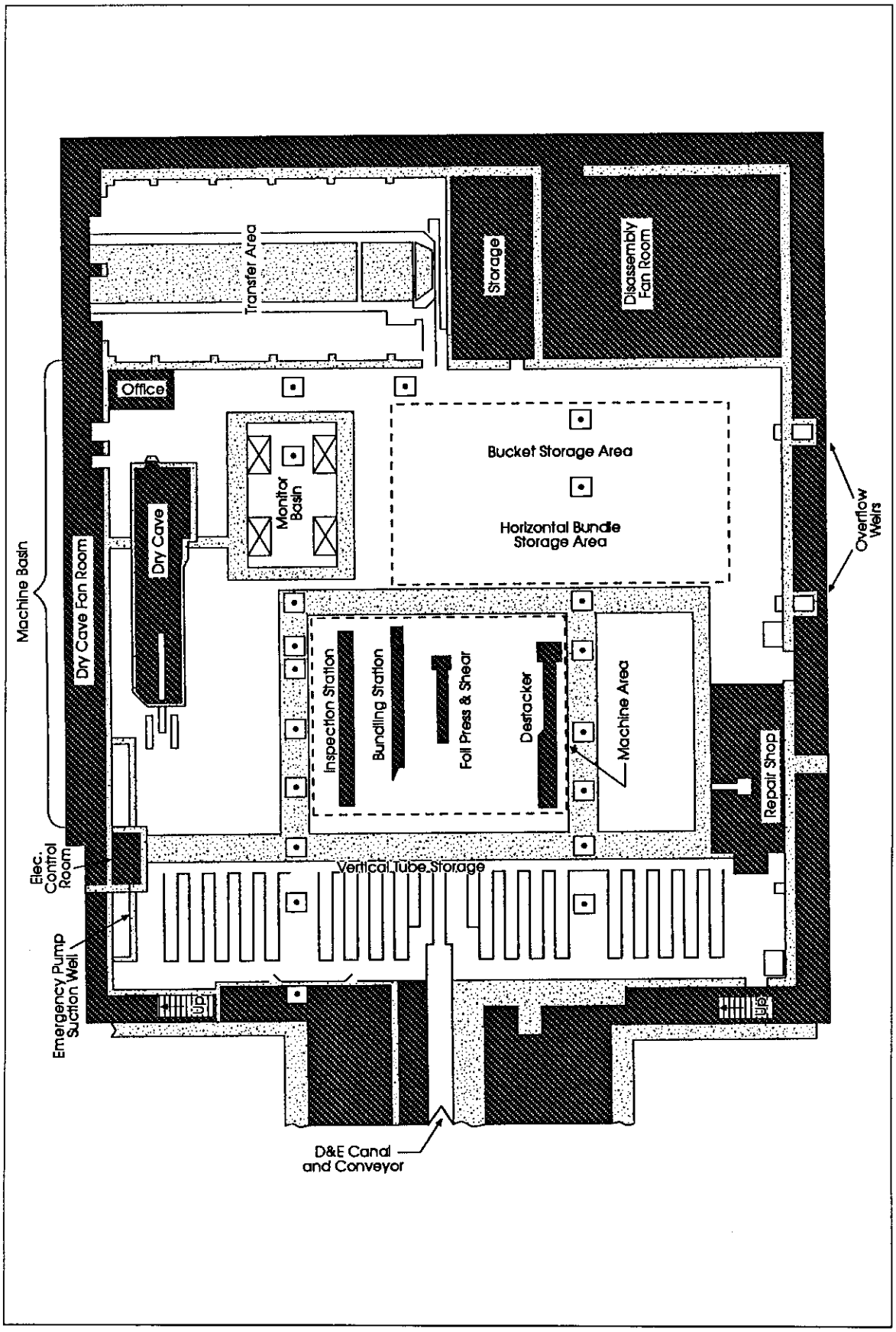


Figure F-41 Typical Reactor Disassembly Basin Area Layout

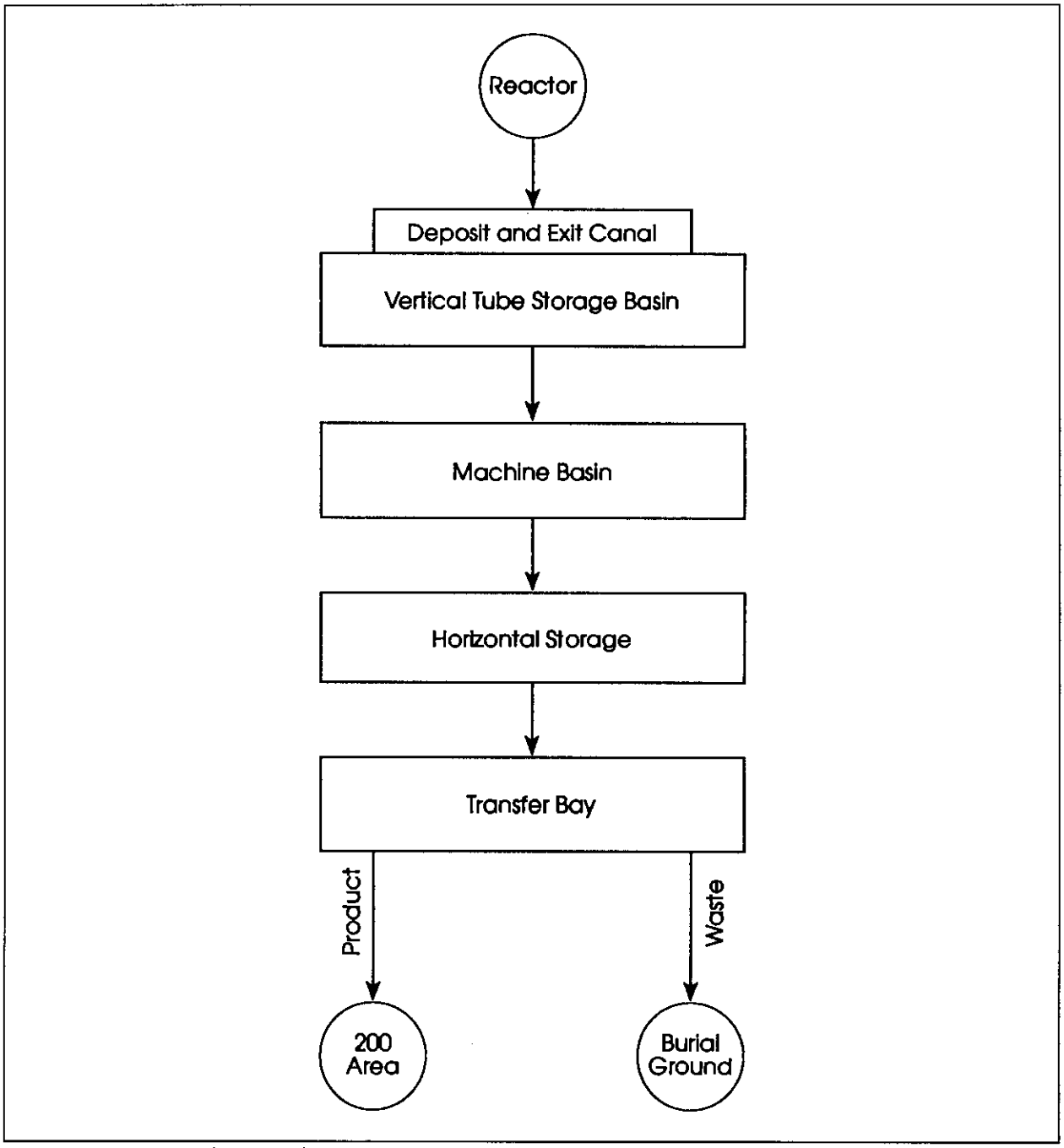


Figure F-42 Reactor Disassembly Basin Process Block Diagram

and transferred to the disassembly area side of the Deposit and Exit canal. The assemblies were transferred to hangers suspended from overhead monorails and were initially stored in the vertical tube storage area for 3 to 8 months. Fuel and target assemblies were moved to the machine basin area where they were disassembled. Target material was placed in stainless steel buckets and then stored in the bucket storage area. Fuel was bundled in aluminum bundles and stored in horizontal storage racks. The components were then allowed to cool for up to another 8 months. The fuel and target material was expected to be in the basin no longer than 18 months. The components, once sufficiently cooled, were moved to the transfer area to be shipped by cask to the processing facility.

Use of a disassembly basin for foreign research reactor spent nuclear fuel would require continuous demineralizer treatment for water quality, and new storage racks. Existing heat exchanger systems can remove upwards of 6,800 kilowatts (24 x 10^6 BTU/hr), which should far exceed the 240-1,000 kilowatts heat generation rate of foreign research reactor spent nuclear fuel (i.e., 10 to 40 Watts per element). These changes would allow each basin to accommodate approximately 20,000 elements.

The transfer area provides an area for shipping or receiving material and equipment. This area consists of two water-filled basins which are designated the scrap pit and the transfer pit. Irradiated material ready for shipping is transferred from horizontal storage to the transfer bay. Transportation casks are moved to and from the transfer pit and irradiated material placed in the cask using hoists mounted on the monorail system. Transportation casks can be transported to and from the reactor areas by tractor trailer or railroad. Trailers or railcars are positioned inside the transfer pit and casks are lifted and transported into/out of the basin using an 85/30 ton overhead crane.

Over the course of the site's history, at least 10 different casks have been used for various applications, many of which are still available for use pending proper inspection and maintenance. Two types are now used for most, if not all, disassembly work. EP-85 is a 63.5-metric ton (70-ton) fuel and target transport cask and EP-383 is a 13.6 metric ton (15-ton) cask used to move scrap to the burial ground.

The transfer area cranes would have to be modified to accommodate the different casks used for offsite shipments. These changes would allow a disassembly basin to receive up to seven casks per month in addition to the projected Savannah River Site shipping requirements.

A monorail system is mounted to the ceiling throughout the disassembly area. This system is used to transport and store all types of spent nuclear fuel and reactor components in the disassembly basin area. Most of the disassembly monorail system is designed for a working load of 907 kg (2,000 lbs) per foot of rail, which would be adequate for moving foreign research reactor spent nuclear fuel.

H-Canyon

The H-Area facilities occupy approximately 160 ha (395 acres). The H-Area Canyon processed irradiated fuel elements via modifications of the plutonium-uranium extraction process (Figure F-43) and is oriented toward HEU recovery. Primary operations also include the dissolution of fuel tubes, chemical and physical separations, and purification of materials. DOE stores the high-level waste from the operations in large tanks (nominally, 3.8 million l or 1 million gal each) for future stabilization and disposal via the Defense Waste Processing Facility.

The facility arrangement of the H-Canyon is identical to the F-Canyon. The main facility in the H-Area is the 221 H-Canyon, where most of the separations of irradiated materials were accomplished. The H-Canyon is a Class 1 reinforced concrete structure with exterior dimensions of 254.4 m (835 ft) by 37.2 m (122 ft) by 20.1 m (66 ft) high. The facility houses two parallel process canyons, each 9.1 m (30 ft) wide. The two process canyons handle high activity and lower activity materials, and are designated as the hot and warm canyons, respectively. Each of these process canyons is divided into 14 process cells, 13.1 m (43 ft) long by 4.6 m (15 ft) wide by 13.7 m (45 ft) high. Each process canyon is serviced by an overhead crane that operates the entire length of the canyon. The cranes perform remote operations for the processes such as equipment replacement, piping and electrical changes, leak repair, and inspections. Recently, new cranes were installed in both canyons. These cranes are remotely operated and include a Closed Circuit Television system for better monitoring capabilities. The warm process canyon crane has a lifting capacity of 27.2 metric tons (30 tons) and the hot process canyon's crane capacity is 45.4 metric

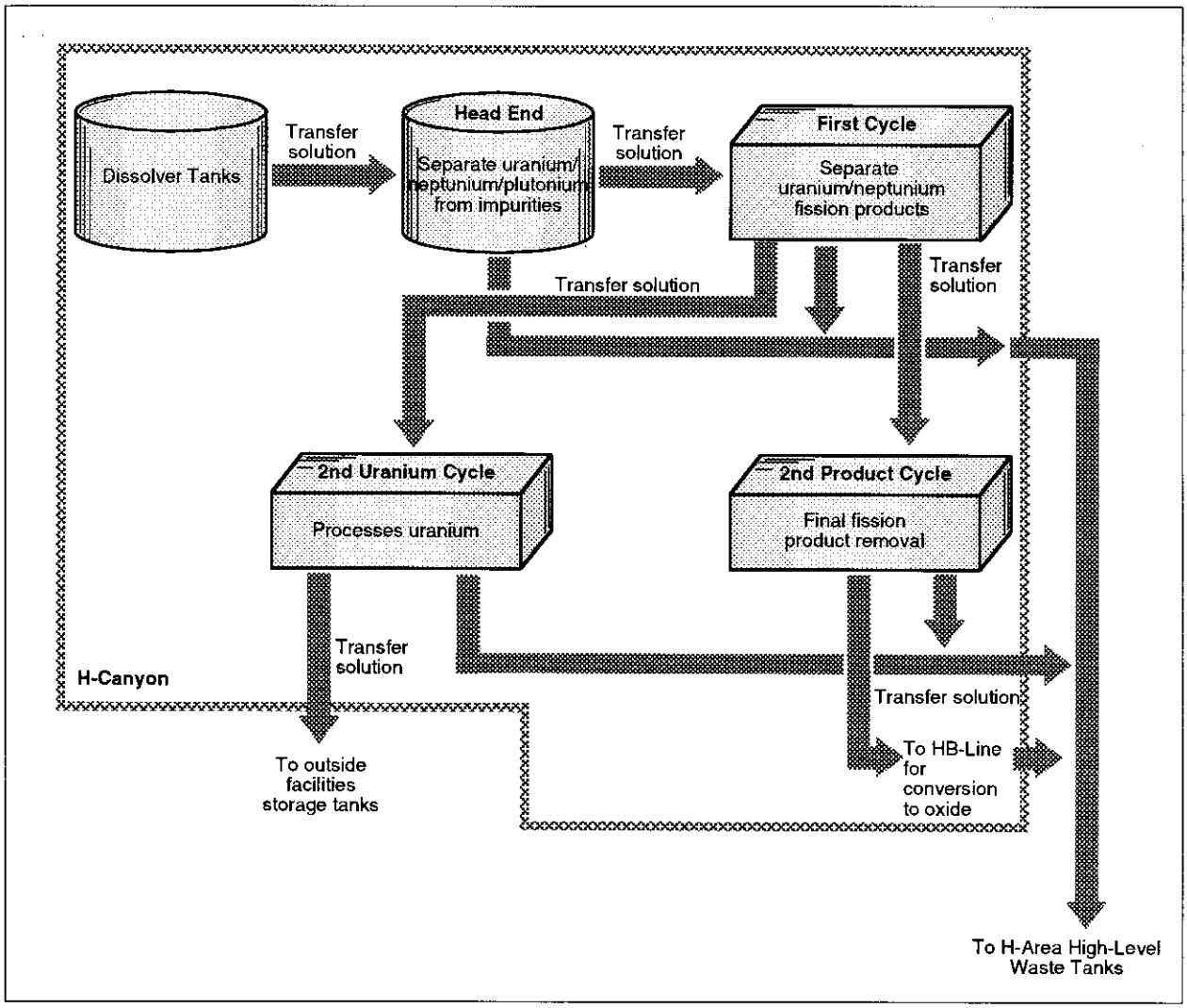


Figure F-43 H-Canyon Modified Plutonium-Uranium Extraction Process Flow

tons (50 tons). A separate maintenance area for each canyon provides for crane maintenance and repair. The original canyon cranes are still in place and can be utilized as backups to the new ones.

Spent nuclear fuel is delivered by a water-filled railway mounted cask into a shielded tunnel. The hot canyon overhead crane unloads spent nuclear fuel either into a storage basin in Section 3 or into the dissolver. The H-Canyon basin was never designed for storage of spent nuclear fuel, but only as a staging area prior to reprocessing. As such, it does not have the capability to establish, circulate, or maintain water chemistry. Basin water is sampled and analyzed every 3 months and domestic water is added manually every 3 months by opening an appropriate valve to the basin to make up for evaporation.

The H-Canyon storage basin is constructed of concrete and is 5.4 m (17.75 ft) long, 2.38 m (7.83 ft) wide, and 7.6 m (25 ft) deep. The floor is covered by a stainless steel liner that also extends 4.3 m (14 ft) up each wall. A stainless steel storage rack sits in the bottom of the basin and provides adequate spent nuclear fuel separation. The basin is filled with water to a level of 3.8 m (12.5 ft). The fuel currently being stored in the basin contains approximately 40 kg (88 lbs) of ^{235}U. The H-Canyon is currently in a status ready for restarting. An EIS was prepared to cover the canyon's restart for processing of the liquids

currently stored in tanks within the facility. Future missions for the facility are still being analyzed. For foreign research reactor spent nuclear fuel, H-Canyon could be used for chemical separation and blending down of HEU to LEU material.

F.3.3.2 Idaho National Engineering Laboratory

On October 17, 1995, litigation with the State of Idaho was settled by stipulation of the parties and entry of a consent order. This settlement would provide for the transportation of up to 61 shipments of foreign research reactor spent nuclear fuel to the Idaho National Engineering Laboratory prior to the year 2000, if DOE and the Department of State choose to adopt a policy of accepting such spent nuclear fuel. After the year 2000, additional shipments of such spent nuclear fuel could be made to the Idaho National Engineering Laboratory under the stipulated settlement and consent order. However, the following discussion is to provide a full understanding of the existing capabilities at the Idaho National Engineering Laboratory in light of the fact that this site has been considered as a reasonable alternative to manage the foreign research reactor spent nuclear fuel as in the preparation of the Draft EIS.

Irradiated Fuel Storage Facility

The ICPP-603 includes an underwater fuel storage basin area and the IFSF, which is a remotely operated, dry-vault facility specifically constructed for the storage of graphite fuel from the Fort St. Vrain and Peach Bottom reactors. It was built in 1974 as an addition to the underwater Fuel Storage Facility and contains 636 storage positions. This facility can handle casks weighing up to 55 metric tons (60 tons). Spent nuclear fuel currently stored here is from two commercial high-temperature, gas-cooled reactors (Fort St. Vrain and Peach Bottom), some for the ROVER Nuclear Rocket Program, and some Tory 2C and BER II TRIGA fuel. The IFSF is a good candidate for spent nuclear fuel requiring frequent monitoring because of the ease of visual fuel inspections.

Since the facility can accommodate fuels up to 3.0 m (130 in) in length, all types of foreign research reactor spent nuclear fuel under consideration in this EIS could be handled. Transfer cart modifications would be needed for the proposed foreign research reactor spent nuclear fuel transportation casks, since the existing transfer cart only has the capability to handle the Rover cask, the Fort St. Vrain cask, and the Peach Bottom cask. New fuel handling tools, such as a new can grapple, would be needed. New cell preparations and work stations would also be needed. Sipping, unloading, canning, sealing, and leak checking equipment would need to be added to the cell. A 14 metric ton (15 ton) crane is present in the vault room for fuel handling. Visual inspection and gamma spectroscopy could readily be performed in the existing vault room.

The vault room is 7 m by 7.1 m by 6.6 m (22 ft 10 in by 23 ft 3 in by 21 ft 6 in) high, and the storage room is 437 m^2 (4,700 ft^2). Loaded fuel cans can be transferred from the vault room to the storage area by a shuttle bin. The vault room is being reanalyzed structurally to validate its capability to meet the current seismic requirements of 10 CFR 72. Recent reliable data regarding the effectiveness of the filtering and ventilation systems must be obtained in order to assess the amount of radionuclides that may be vented into the outside air. The cost to add the required capabilities to the IFSF for storage of foreign research reactor spent nuclear fuel is approximately $5 million.

It should be mentioned that, although this facility was originally constructed to accommodate the Fort St. Vrain High Temperature Gas-Cooled Reactor graphite fuel, it will not be used for this purpose because of the October 16, 1995 Settlement Agreement with the State of Idaho that declared that the Fort St. Vrain fuel will not be brought to the Idaho National Engineering Laboratory for interim storage. Public Service of Colorado has obtained a 10 CFR 72 license to store all of the fuel in the Foster Wheeler

modular dry vault facility built adjacent to the reactor site. That modular dry vault is currently completely loaded with Fort St. Vrain fuel, and the reactor is being decommissioned. Availability of the IFSF for foreign research reactor spent nuclear fuel will be dependent on decisions to consolidate spent nuclear fuel from other Idaho National Engineering Laboratory facilities at this facility.

Approximately 300 positions are available in the IFSF dry storage facility for foreign research reactor spent nuclear fuel. Preparations to receive the foreign research reactor spent nuclear fuel could be completed as soon as calendar year 1997. However, many activities are already scheduled for this facility. A new canning station for spent nuclear fuel from the ICPP-603 basins is being constructed in the handling cave, and the canning will then be accomplished. Other spent nuclear fuel management activities being considered in this facility include ROVER reactor fuel shipments, Experimental Breeder Reactor II and FERMI movements fuel from ICPP-666. Naval fuel inspection sample receipts from the Expended Core Facility are scheduled. The Peach Bottom fuel in the ICPP-749 facility and Rover fuel ash transfers from a shutdown fuel processing facility are also being considered for repackaging in the IFSF handling cell, and the Idaho National Engineering Laboratory spent fuel consolidation activities are scheduled to begin within 2 years. Detailed facility usage schedules have been drafted to demonstrate how the foreign research reactor spent nuclear fuels could be accommodated in this workload.

With this schedule as background, foreign research reactor spent nuclear fuel preparations could take place during early 1996. Fuel could be received beginning in 1997 (61 shipments through FY2000) and up to 101 shipments in the years beyond this timeframe.

The 300 positions could store up to approximately 9,000 foreign research reactor spent nuclear fuel elements. Approximately 60 dry storage positions in the ICPP-749 drywells could also be utilized to store foreign research reactor spent nuclear fuel. Some refurbishment would have to take place to receive spent nuclear fuel. This could be completed in 1997, and the facility would be ready to receive fuel at the beginning of 1998. With 60 fuel elements per position, up to 3,600 fuel elements could be stored in the ICPP-749. Any fuel that is stored in the ICPP-749 would have to go through the ICPP-603-IFSF to be placed in sealed containers and transferred to an interfacility transfer cask.

Idaho Chemical Processing Plant-666 Fuel Storage Area

This facility (Figure F-44) is the modern Idaho National Engineering Laboratory underwater storage facility. Receipt and storage of foreign research reactor spent nuclear fuel has been accomplished in the past as one of its many missions. It has the capability of receiving and unloading spent nuclear fuel casks at a rate of approximately five per week. Storage capability for up to 8,400 foreign research reactor spent nuclear fuel elements can be provided for an approximate 10-year period by using the increased capacity fuel storage racks that will be installed in Pool 1 via a reracking project planned following the Programmatic SNF&INEL Final EIS, and by installing additional fuel storage racks in the cutting pool. The increased capacity being provided in Pool 1 will be required for Naval spent nuclear fuel receipts beginning in about FY 2005. The racks being removed from Pool 1 as part of the reracking project could be placed in the cutting pool for the foreign research reactor spent nuclear fuel.

The capability of the ICPP-666 facility to receive foreign research reactor spent nuclear fuel in the near term is limited, however, due to the number of activities scheduled through 1998. These activities include reracking in Pools 1, 6, and 5. Fuel receipts from the Navy and the Advanced Test Reactor, and fuel transfers from CPP-603 will continue. The CPP-603 fuel transfers are required to meet a court order requirement. These activities utilize nearly all resources, such as cranes, manpower, and health physics personnel in the near term of 1995 through 1997. A limited number of shipments (one per month) are possible for the 1996 and 1997 schedules. In 1998, the schedule relaxes enough that up to 30 shipments

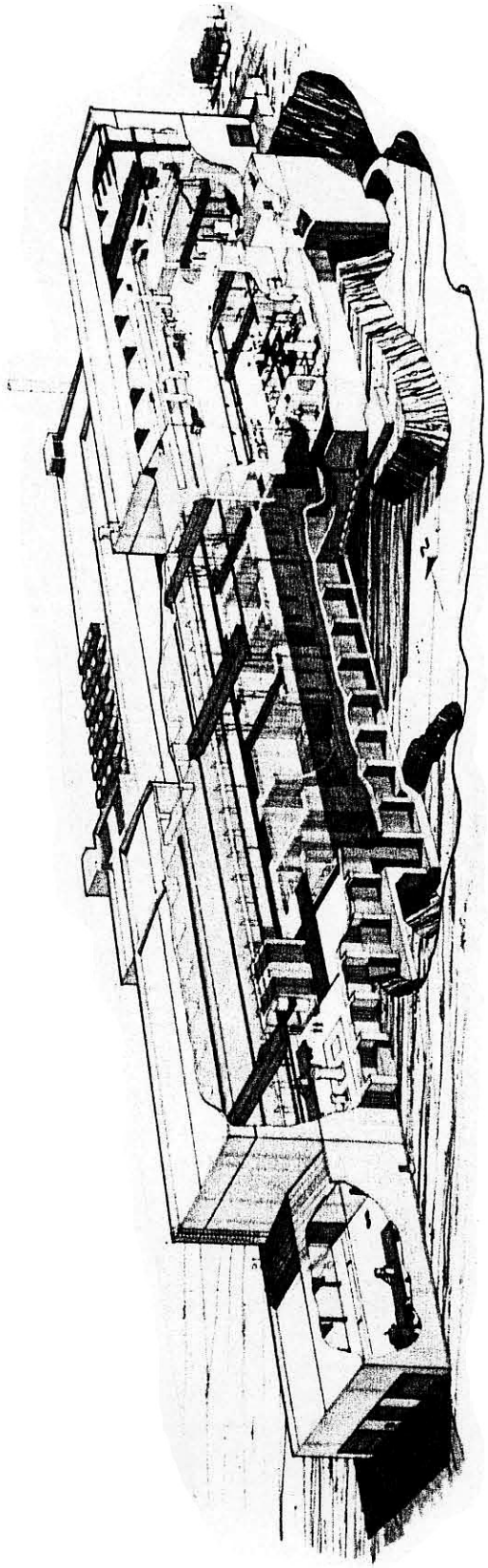


Figure F-44 Pictorial of the Idaho Chemical Processing Plant-666

could be received because most of the ICPP-603 fuel transfers to Fluorinel Dissolution and Fuel Storage (FAST) will have been completed. By the end of 1999, a total of 3,600 fuel elements could be received under this scenario. This schedule is predicated on the assumption that resolution of Idaho National Engineering Laboratory facility vulnerabilities and Naval and Advanced Test Reactor fuel receipts have a higher priority than foreign fuel receipts.

Idaho Chemical Processing Plant-666 Fluorinel Dissolution Process

The conversion of the ICPP-666 Flourinel Dissolution Process cell for canning of spent nuclear fuel without removal of any existing equipment or decontamination of the cell is proposed as a low-cost option to prepare foreign research reactor spent nuclear fuel for dry storage. Only minor modifications would be made to the cell, so as to preserve the current dissolution capability for possible future use. This option would utilize the Flourinel Dissolution Process cell, which currently has remote fuel handling, sampling, and waste load-out capabilities, as well as a connection to the ICPP-666 Fuel Storage Area, for fuel inspection, stabilization, and packaging for interim dry storage.

A potential disadvantage is noted. The equipment inside the Flourinel Dissolution Process cell is contaminated, and radiation fields are too high for manned entry. Retaining the dissolution cell equipment will make it impossible to adequately clean the cell to allow personnel entry. For this reason, equipment requiring installation within the Flourinel Dissolution Process cell must be assembled outside the cell and installed remotely with the in-cell crane and master-slave manipulators. Preventative and corrective maintenance of the equipment inside the cell would be done remotely. The modular design of the components would facilitate removal and replacement.

No general Flourinel Dissolution Process utility upgrades, such as electrical power or ventilation, would be required. Piping services could be added to support the vacuum drying and inert gas backfilling functions envisioned to meet dry storage requirements. All of the Reduced Enrichment for Research and Test Reactors (RERTR) program spent nuclear fuel could be accommodated by the existing transfer tunnel and transfer cart. Modifications of the existing fuel shear tools, or new ones, could be acquired to shear larger objects such as cans and lids.

Fuel Processing Restoration

Another structure that may represent an option for fuel storage is the Fuel Processing Restoration building (Figure F-45) that was constructed to house the Fuel Processing Restoration process. It is approximately 56.4 m (185 ft) long and contains shielded, below-grade process cells. These cells vary in dimension, with the nine main process cells measuring 5 to 6 m (16.5 to 20 ft) wide, 10.4 m (34 ft) long, and 12.2 m (40 ft) deep. Fuel racks could be designed to accommodate cans of the type proposed for dry storage in arrays that could contain as many as 17 cans along the 10.4 m (34 ft) axis by 8 cans along the 4.9 m (16 ft) axis, and be 3 cans deep. Airflow through each of these cells could be controlled by dampers in the cell ductwork. Construction of this facility was interrupted prior to completion, and it currently does not include cell ventilation, fire safety equipment, instrumentation, or lighting. These additions would cost approximately $15 million. Several other processes are being considered for use of this facility, and there is no assurance that it would be used for fuel storage. When completed, the building will be seismically qualified, and could physically accommodate approximately 540 canisters per main process cell. Special remote handling tools and techniques would be developed to allow the fuel cans to be inserted into the storage cell. The total estimate to make all necessary conversions is $65 million.

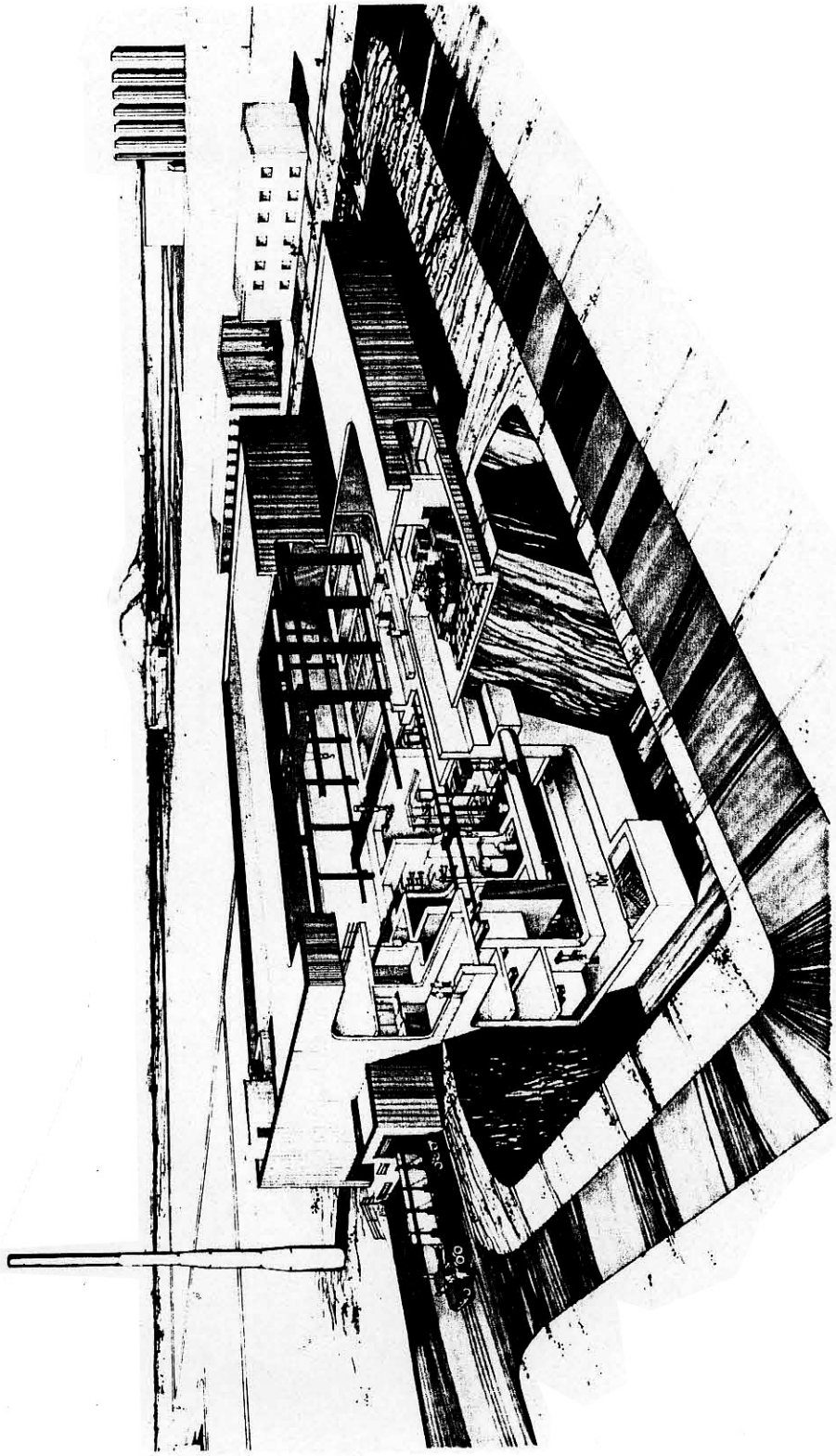


Figure F-45 Fuel Processing Restoration Facility (Unfinished)

Hot Fuel Examination Facility

The Hot Fuel Examination Facility is a facility used for examining and storing irradiated fuels from the EBR-II breeder reactor at Idaho National Engineering Laboratory. Figure F-46 presents the layout of the main cell of the Hot Fuel Examination Facility. Although it was not designed for storage, the Hot Fuel Examination Facility could be used to receive, inspect, examine, and transfer foreign research reactor spent nuclear fuel to dry cask storage if it is fitted with an appropriate spent nuclear fuel examination station. The cost of these modifications and the purchase and installation of the dry casks and their equipment would be the principal costs involved.

Test Area North-607 Pool, Hot Cell, and Cask Storage Pad

The utilization of the Test Area North-607 facilities is a potential option for receipt and storage of the foreign research reactor spent nuclear fuel. This could be accomplished without significant modification to the hot cell. The hot cell has significant lag capacity for interim storage of foreign research reactor spent nuclear fuel and has most of the equipment necessary for placement of the fuel into dry interim storage casks. There is adequate space for installation of the characterization and conditioning equipment needed for dry storage. There are significant vulnerabilities associated with the underwater storage pool which would need to be corrected if underwater interim storage were desired. The cask storage pad could be easily expanded to accept additional dry storage casks. At the current time, the entire Test Area North area is being planned for shutdown in approximately ten years due to reduced mission needs. If Test Area North had adequate new missions and it was determined to be economical, the Test Area North hot cell and cask storage area would have significant capacity for receipt and temporary storage of foreign research reactor spent nuclear fuel.

F.3.3.3 Hanford Site

In addition to the generic dry and wet storage facilities, two existing facilities at Hanford Site have been identified as potential candidates for the storage of foreign research reactor spent nuclear fuel: the FMEF and the Washington Nuclear Plant-4 Spray Pond. They will be discussed in greater detail in the following sections.

F.3.3.3.1 Fuel Maintenance and Examination Facility (FMEF)

The FMEF, built during the late 1970s and early 1980s (but never completed), consists of a 82.3 m (270 ft) long, 53.3 m (175 ft) wide, 29.9 m (98 ft) high Process Building with attached mechanical equipment and entry wings. The FMEF was intended to receive and extensively examine irradiated breeder reactor test fuels. The seismically-qualified FMEF Process Building, which extends 10.7 m (35 ft) below the surface, consists of 6 operating floors or levels and encloses a total of 17,466 m^2 (188,000 ft^2) of operations space. The FMEF has a 68 metric ton (75 ton) overhead crane, 18 metric ton (20 ton) hoist, and 9 metric ton (10 ton) hoist. Three areas within the FMEF, the Shipping and Receiving area, Decon Cell, and the Entry Tunnel would be used for the storage of foreign research reactor spent nuclear fuel. Figure F-47 presents a ground floor plan, and Figure F-48 shows vertical storage of spent nuclear fuel canisters.

The Shipping and Receiving Area, also known as Room 300, is the access area to the FMEF for truck or rail shipments of spent nuclear fuel. This area provides a 24.4 m (80 ft) long working area, washdown and decontamination of casks and shipping vehicles, and crane interface access to other areas within the FMEF. The 68 metric ton (75 ton) crane and hoists can transport casks to the Entry Tunnel from this area. The principal modification needed for this area would be the construction of a 0.8 km (0.5 mi) rail extension to the existing Hanford Site rail system.

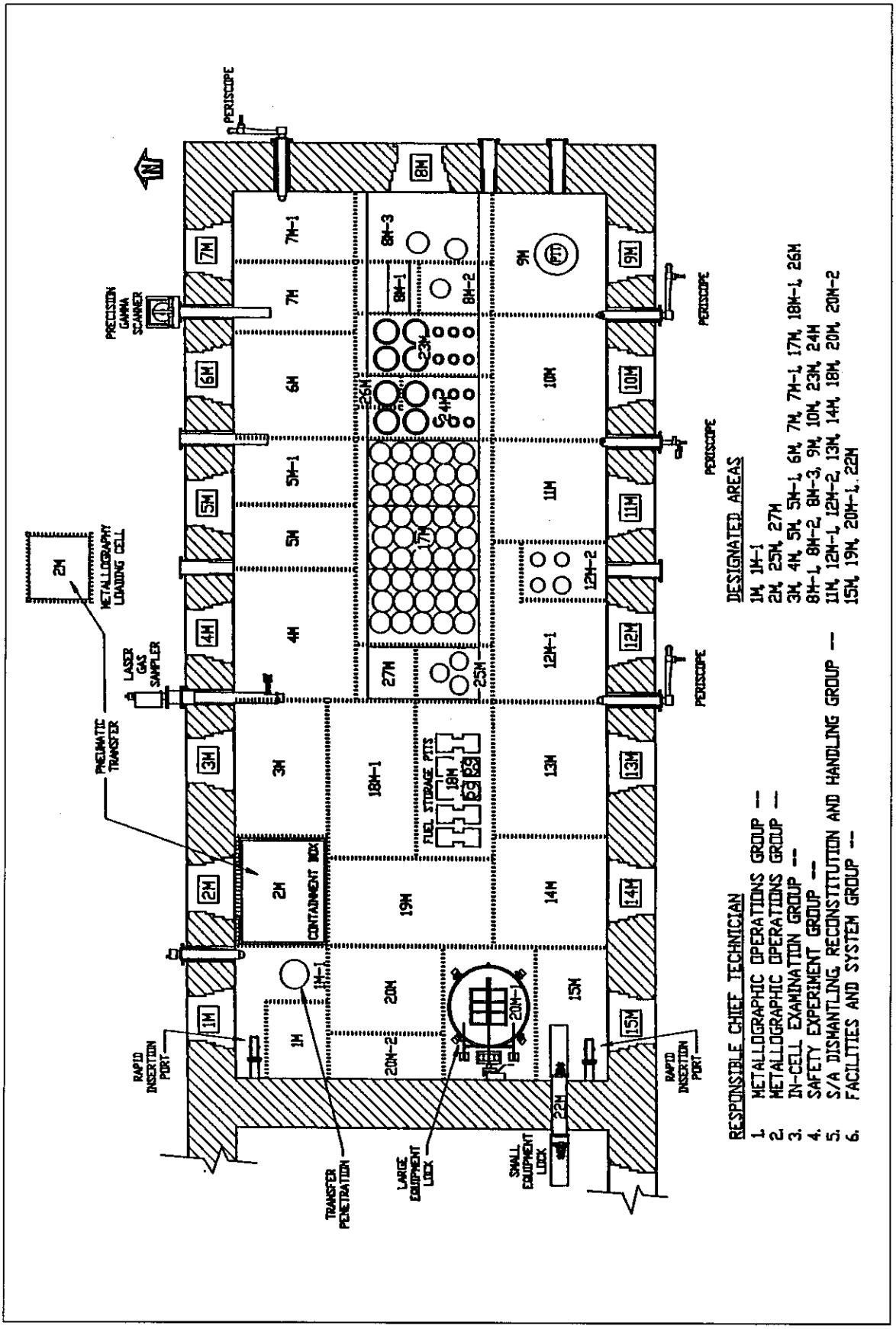


Figure F-46 Hot Fuel Examination Facility Main Cell Layout

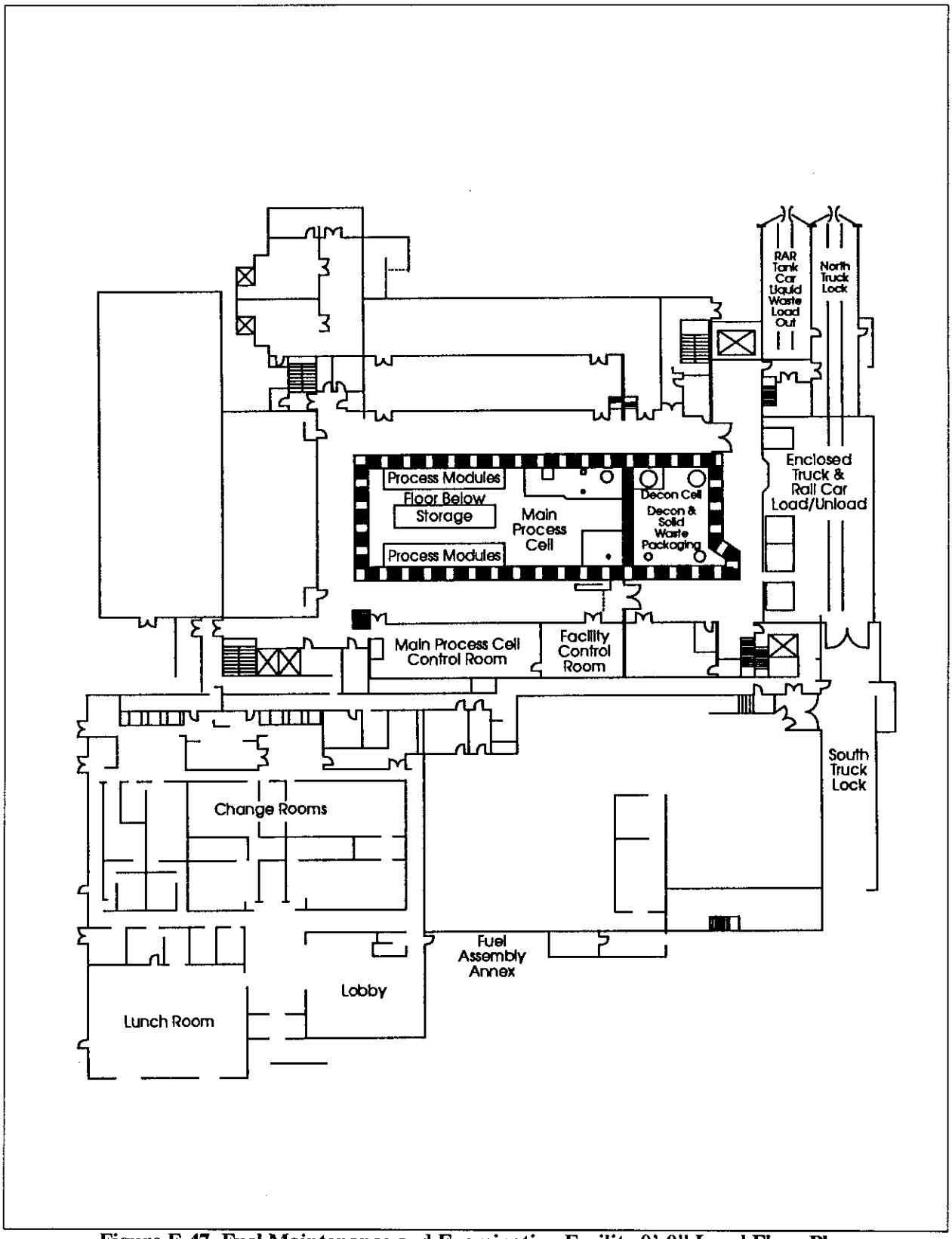


Figure F-47 Fuel Maintenance and Examination Facility 0'-0" Level Floor Plan

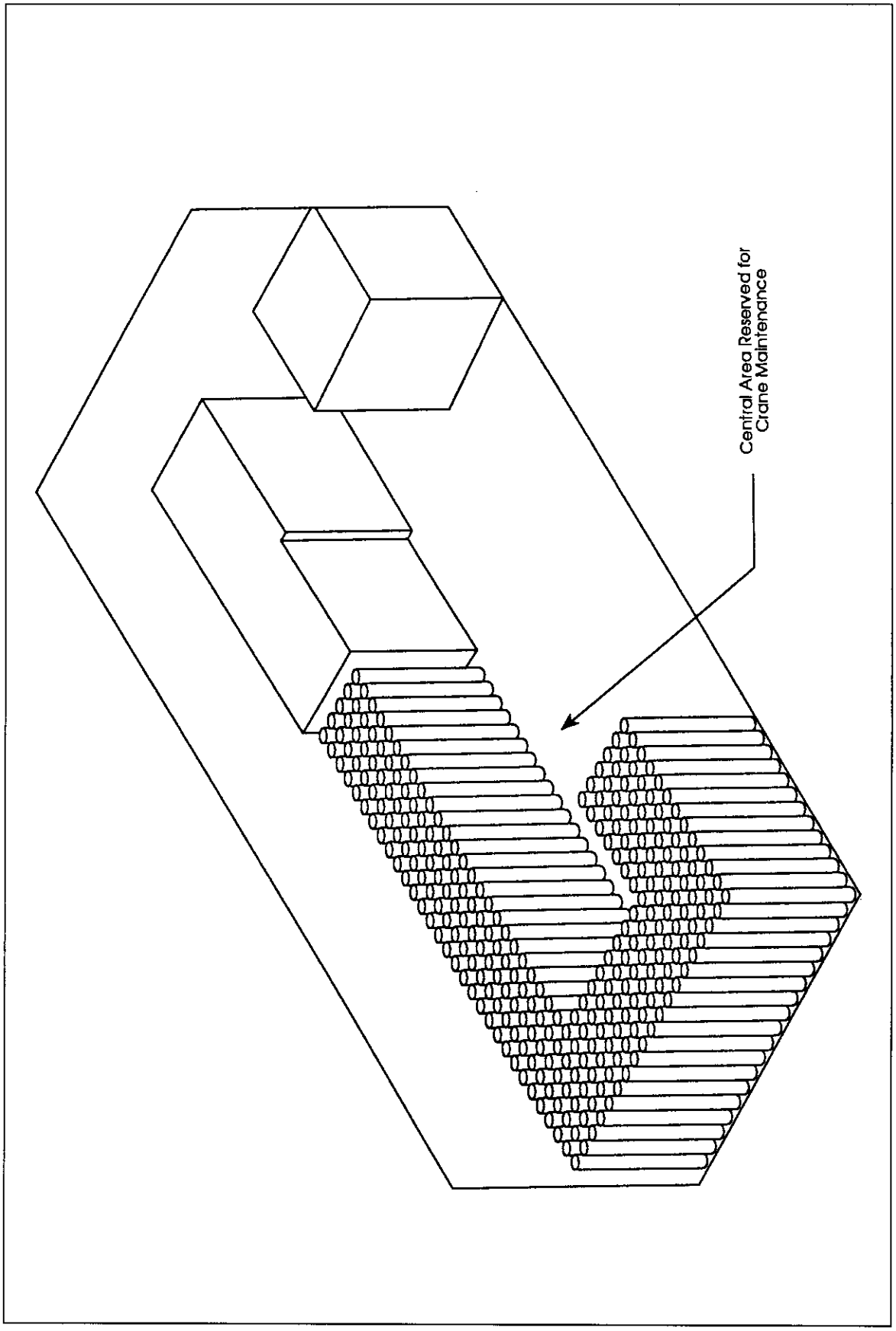


Figure F-48 Potential Use of the Fuel Maintenance and Examination Facility for Fuel Storage
Partial Plan View

The Entry Tunnel is a 9.9 m (32.5 ft) high tunnel below the Shipping and Receiving area floor designed to transfer transportation casks to the Decon and Main Process Cell areas of the FMEF. It includes a 68 metric ton (75 ton) overhead bridge crane. To transport heavier multi-purpose casks when the foreign research reactor spent nuclear fuel would be shipped out of the FMEF in the future, the tunnel would be extended and modified to accommodate an above-grade 114 metric ton (125 ton) crane. The entry tunnel would be extended to connect to the new adjacent storage facility.

The Decon Cell is a room 12.2 m (40 ft) long, 9.1 m (30 ft) wide, and 11.6 m (38 ft) high, with thick concrete shielding. Nine work stations with remote manipulators and viewing windows are part of the design of this cell, although it should be noted that the viewing windows and equipment have not been installed. Access is available through two 2.1 m (84 in) diameter hatches, a 0.8 m (30 in) diameter port, and a 0.3 m (12 in) diameter opening. The Decon Cell includes material handling capability by cranes, manipulators, and hoists ranging from 1.4 to 6.8 metric tons (1.5 tons to 7.5 tons). The Decon Cell would be used to inspect foreign research reactor spent nuclear fuel that has been unloaded from transportation casks and to subsequently load this spent nuclear fuel into storage baskets. Each basket holds three foreign research reactor spent nuclear fuel elements. After basket loading, four baskets would be stacked into a 4.6 m (15 ft) high stainless steel canister. The canisters would be moved to the adjacent storage facility using a Transfer Tunnel, which is equipped with a cart.

The Main Process Cell represents a potential storage location for foreign research reactor spent nuclear fuel at the FMEF (Figure F-47). This room is 30.5 m (100 ft) long, 12.2 m (40 ft) wide, and 11.6 m (38 ft) high with concrete walls either 1.2 or 1.5 m (4 or 5 ft) thick, depending on the concrete's density. The Main Process Cell design includes two 4.5 metric ton (5 ton) bridge cranes and two 1.4 metric ton (1.5 ton) electro-mechanical manipulators.

A zoned heating, ventilation, and air conditioning system with negative differential pressure, redundant cooling systems, and staged multiple High Efficiency Particulate Air filters provides decay heat removal and protection from environmental releases of radioisotopes. This system provides for flow, by negative air pressure differential, from the least contaminated zones to the most contaminated zones, thereby maintaining individual zone relative contamination potential. Supply air is drawn from tornado-hardened and seismically qualified intake shafts and dampers. All heating, ventilation, and air conditioning equipment required to supply high contamination zones is designed as Seismic Category 1. After multiple High Efficiency Particulate Air filtration and monitoring for radioactivity, heating, ventilation, and air conditioning exhaust air is released from a seismically qualified reinforced concrete 35.7 m (117 ft) tall stack.

The FMEF is provided normal power by two separate 115 kV electric power supply lines from the Bonneville Power Administration. Emergency power is provided by two 100 percent redundant 900-kilowatt gas turbines which, along with their seismically qualified support and fuel oil systems, are capable of 24 hours of continuous operation. An Uninterruptible Power Supply, consisting of two 150-kVA lead calcium batteries, can provide full load for 30 minutes. Emergency generators require 2 minutes to start up and produce rated power.

A number of modifications would be required for the FMEF to be used as a storage facility for foreign research reactor spent nuclear fuel. They can be categorized as: addition of a 114 metric ton (125 ton) crane, railroad tunnel extension, and storage rack canisters. Even with these modifications, the FMEF does not have sufficient space to store 23,000 foreign research reactor spent nuclear fuel elements, but it could be used as an unloading and support facility for an adjacent dry vault storage facility. Costs for the necessary modifications to the FMEF have been estimated to be approximately $32 million. The adjacent dry storage facility is estimated to cost an additional $100 million. It should be noted that the FMEF is

being considered for other spent nuclear fuel storage which could eliminate it for use with foreign research reactor spent nuclear fuel.

F.3.3.3.2 Washington Nuclear Plant-4 Spray Pond Wet Storage

The Washington Nuclear Plant-4 Spray Pond is a nuclear safety-related structure that was originally designed for decay heat removal following a Loss of Coolant Accident at the Washington Nuclear Plant-4 commercial nuclear power plant. The Washington Nuclear Plant-4 was canceled, but the spray pond structure is essentially complete. This pond is 91.4 m (300 ft) long, 76.2 m (250 ft) wide, and 8.2 m (27 ft) deep, and was designed and built to 10 CFR 50 Appendix B quality assurance standards as a seismic and safety class structure. It should be noted that the size of this spray pond is much greater than that needed to store the foreign research reactor spent nuclear fuel. Figure F-49 presents a schematic of the Washington Nuclear Plant-4 Spray Ponds and the necessary modifications required to store foreign research reactor spent nuclear fuel.

In order for this pond to be used for wet storage of foreign research reactor spent nuclear fuel, several modifications would have to be made to duplicate the features of the Generic Pool Facility. These modifications include: an enclosure with a qualified building superstructure, inclusion of a shipping-receiving-handling facility in this structure; a deeper loading pool; a 114 metric ton (125 ton) loading crane; a 274 m (900 ft) long railroad line extension; installation of heating, ventilation, and air conditioning, and pond water cooling and water chemistry/cleaning systems; canister support racks into the pond; installation of a stainless steel liner in the pond; installation of partition walls to isolate the fuel storage and handling areas from the unused portion of the spray pond; and a leak detection system. The supporting pumphouse is a 12 m (40 ft) by 30 m (100 ft) concrete building designed to the same standards as the spray pond and would be used to house some of the fuel handling and storage support equipment. Total cost for these modifications was estimated to be approximately $113 million (Bergsman et al., 1994).

F.3.3.4 Oak Ridge Reservation

No existing facilities at the Oak Ridge Reservation are being used for the storage of foreign research reactor spent nuclear fuel. However, either the generic wet (pool) or dry storage facilities, as described in Sections F.3.1 and F.3.2, could be constructed and operated at Oak Ridge Reservation.

F.3.3.5 Nevada Test Site

No existing facilities at the Nevada Test Site are being used for storage of foreign research reactor spent nuclear fuel, although the Area 25 facilities (E-MAD/Reactor Maintenance and Disassembly) have been used in the past and might be suitable in 1 to 3 years. These Area-25 facilities appear capable of accommodating the required cask receipt rate and dry storing all of the foreign research reactor spent nuclear fuel under consideration in this EIS. However, in addition to Area 25, either the generic wet (pool) or dry storage facilities, as described in Sections F.3.1 and F.3.2, could be constructed and operated at Nevada Test Site in Area 5.

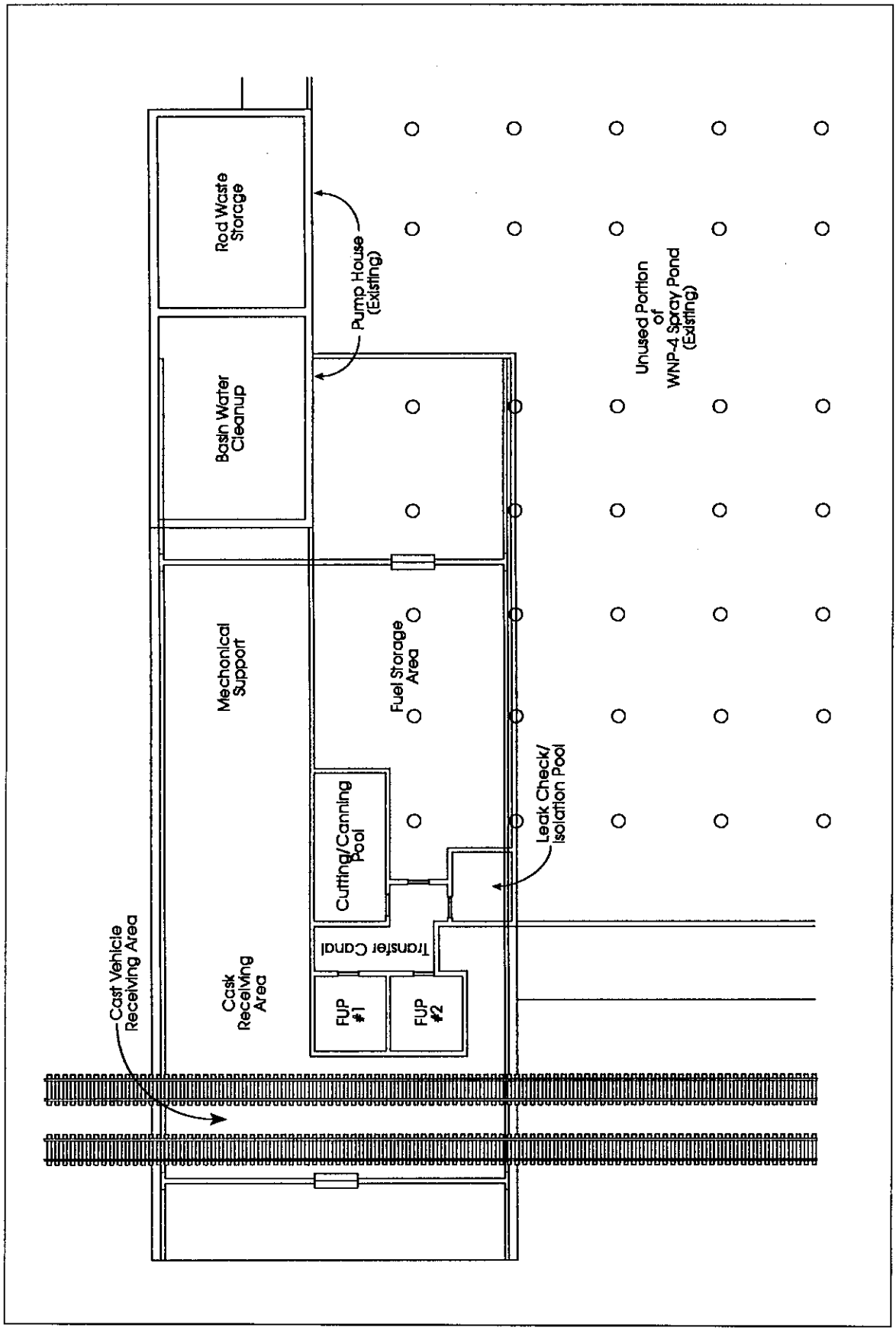


Figure F-49 Washington Nuclear Plant-4 Spray Pond (with Modifications) Schematic

F.4 Environmental Impacts at Foreign Research Reactor Spent Nuclear Fuel Management Sites

This section analyzes the environmental impacts associated with the storage of foreign research reactor spent nuclear fuel at the five potential management sites considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g), namely: the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

The Record of Decision for the Programmatic SNF&INEL Final EIS was issued on May 30, 1995. In accordance with this Record of Decision, all of the aluminum-based foreign research reactor spent nuclear fuel accepted by DOE would be managed at the Savannah River Site in South Carolina, and any other foreign research reactor spent nuclear fuel to be accepted by DOE would be managed at the Idaho National Engineering Laboratory. Nevertheless, all five of the spent nuclear fuel management sites originally considered in this EIS and the spent nuclear fuel distribution alternatives have been kept in the final to maintain maximum consistency with the analyses provided in the Programmatic SNF&INEL EIS (DOE, 1994h and 1995g).

The environmental impacts analyzed pertain to management of foreign research reactor spent nuclear fuel under the basic implementation, and implementation alternatives of Management Alternative 1, the storage of vitrified waste that may be accepted by the United States under Management Alternative 2, and the management of foreign research reactor spent nuclear fuel under Management Alternative 3, the Hybrid Alternative. Chemical separation, which is an implementation alternative to storage, is analyzed in Section 4.3 of this EIS.

Since foreign research reactor spent nuclear fuel is part of the DOE's overall management of spent nuclear fuel, the management options in this EIS must be consistent with the site management alternatives considered in the Programmatic SNF&INEL Final EIS. The alternatives considered in the Programmatic SNF&INEL Final EIS are: Decentralization and 1992/1993 Planning Basis (even distribution of foreign research reactor spent nuclear fuel between the Idaho National Engineering Laboratory and the Savannah River Site), Regionalization by Geography, Regionalization by Fuel Type, and Centralization (all foreign research reactor spent nuclear fuel eligible under the policy).

The foreign research reactor spent nuclear fuel management options also depend on the availability of the sites to implement the policy immediately. Of the five sites, only the Savannah River Site and the Idaho National Engineering Laboratory would be available in late 1995. The other three sites could become available at a later date when appropriate facilities would be completed (either constructed or refurbished). This constraint has necessitated a two-phased approach to foreign research reactor spent nuclear fuel management in the United States in which foreign research reactor spent nuclear fuel is received and managed first at an available management site and is shipped to another site later. For the purpose of this analysis, the implementation of the policy was divided into two functional periods — the period during which receipt and storage of foreign research reactor spent nuclear fuel would be accomplished by using existing facilities (Phase 1), and the period during which new or refurbished facilities could be used (Phase 2). The first phase would be characterized by operational activities only, while the second involves impacts from construction in addition to operational activities.

The environmental impacts from the basic implementation of each Management Alternative, as they relate to storage of the foreign research reactor spent nuclear fuel in the United States, are analyzed in Sections F.4.1 through F.4.5. Elements of this analysis are combined and summarized in Section 4 of this EIS to present the impacts of all Implementation Alternatives under the proposed action.

F.4.1 Savannah River Site

If the Savannah River Site is the site to manage all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed at the site until ultimate disposition. If the Savannah River Site is not the site to manage DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel could be received and managed at the Savannah River Site until the selected site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years. Modifications to existing facilities for the same purpose could take less time. This period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2. The amount of spent nuclear fuel that could be received and managed at the Savannah River Site under Management Alternative 1, as discussed in Section 2.2.2, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, the Savannah River Site could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, the aluminum-based foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, the foreign research reactor spent nuclear fuel from eastern ports under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As discussed in Section 2.6.4.1, the split of foreign research reactor spent nuclear fuel evenly between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and 1992/1993 Planning Basis alternatives in the Programmatic SNF&INEL Final EIS was not considered to have a practical basis, and was therefore not evaluated in detail.

As a potential Phase 1 site under Management Alternative 1, the Savannah River Site would receive and manage foreign research reactor spent nuclear fuel at existing wet storage facilities: RBOF and the L-Reactor disassembly basin. Descriptions of RBOF and the L-Reactor disassembly basin are provided in Section F.3. RBOF is located at the H-Area. It is a facility with provisions for the receipt and storage of irradiated nuclear fuel elements. Since 1963, irradiated spent nuclear fuel elements have been received from offsite reactors and from the Savannah River Site reactors. RBOF provides the capability for underwater unloading of the transportation casks and the handling and storage of the foreign research reactor spent nuclear fuel. The foreign research reactor spent nuclear fuel would be stored in RBOF until its storage capacity is exhausted. Currently, RBOF has space for approximately 1,170 foreign research reactor spent nuclear fuel elements. This capacity could be increased to a total of 2,425 elements by rearrangement and consolidation of existing inventory (O'Rear, 1995).

The L-Reactor disassembly basin is not currently configured for storage of aluminum-based foreign research reactor spent nuclear fuel; however, minor modifications which would provide new storage racks, new handling equipment, safety documentation, etc., along with upgrades in progress to address vulnerabilities associated with water chemistry control, would permit receipt and management of foreign research reactor spent nuclear fuel. Installation of racks equivalent to those in RBOF would provide storage for approximately 20,000 foreign research reactor spent nuclear fuel elements. The modifications to RBOF and L-Reactor disassembly basin are part of the ongoing programs at the site to be performed independent of the proposed action in this EIS.

Between the RBOF and the L-Reactor disassembly basin, there would be sufficient storage capacity and handling capability to accommodate the receipt and management of foreign research reactor spent nuclear fuel during the estimated 10-year period for Phase 1.

An additional option to enhance storage capacity during Phase 1 would be to use RBOF and/or L-Reactor disassembly basin to unload the transportation casks and provide storage capacity in dry storage casks

which would be placed near the existing facility. Descriptions of the dry storage casks are provided in Section F.3.

As a Phase 2 site under the basic implementation of Management Alternative 1, the Savannah River Site would continue to receive foreign research reactor spent nuclear fuel beyond Phase 1 in a new dry storage facility that would be constructed at the H-Area. The H-Area is the preferred site among several considered for the construction of new foreign research spent nuclear fuel storage facilities, and is the location assumed for the environmental impacts calculations. An alternative site, equally qualified for construction of new storage facilities is located on a ridge between the P-Reactor and the Pen Branch watershed as indicated in Section 2, Figure 2-14 of this EIS (Shedrow, 1994a). Foreign research reactor spent nuclear fuel stored during Phase 1 would be transferred to the new facility and would be stored there for an additional 30 years until ultimate disposition. The dry storage would encompass a number of designs, examples of which were provided in Section 2.6.5.1.1 and in Section F.3.

The analysis of environmental impacts from the management of foreign research reactor spent nuclear fuel at the Savannah River Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

1A. The Savannah River Site would receive foreign research reactor spent nuclear fuel during Phase 1 and store it at the RBOF and/or the L-Reactor disassembly basin. For the purpose of this analysis, the amount of fuel to be stored is all foreign research reactor spent nuclear fuel that would be received during Phase 1 (approximately 17,500 elements). The spent nuclear fuel would be shipped offsite at the end of Phase 1.

1B. Foreign research reactor spent nuclear fuel stored under analysis option 1A would be transferred to a newly constructed dry storage facility, where it would be stored until ultimate disposition. Foreign research reactor spent nuclear fuel arriving in the United States after Phase 1 concludes would be received and stored at the new dry storage facility. For the purpose of this analysis, the amount of spent nuclear fuel that would be stored would be all the foreign research reactor spent nuclear fuel eligible under the policy (22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, as discussed in Section 2.2.2, introduce additional analysis options that would be considered for the Savannah River Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Savannah River Site would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 1A (above). Impacts of construction and operation of the dry storage facility considered in analysis option 1B would bound those of the facility required to accommodate this amount of fuel. The spent nuclear fuel would either be shipped offsite after Phase 1, or it would be managed along with the rest of the spent nuclear fuel at the Savannah River Site.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Savannah River Site would receive only HEU from the foreign research reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts

from the management of this amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A and 1B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Savannah River Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which in uranium content represents approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 1A or 1B (above) by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years, and therefore the amount of spent nuclear fuel available for acceptance would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A or 1B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and stored would remain constant. The impacts would be the same as in analysis options 1A or 1B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States because the foreign research reactors would consider their own alternatives as to whether or not to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel, in this case, cannot be quantified. The upper limit, however, as considered under analysis options 1A and 1B (above), would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the impacts at the Savannah River Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider wet storage technology for new construction. DOE would implement the policy by constructing a new wet storage facility at the H-Area or by using the BNFP, owned by Allied General Nuclear Services. DOE would have to acquire the facility which could be ready for use in approximately 5 years. Therefore, if the Savannah River Site is selected under either the Regionalization or Centralization Alternatives of the Programmatic SNF&INEL Final EIS, Phase 2 at the Savannah River Site could start as early as 5 years from the start of implementation period by using BNFP. The new wet storage facility is described in Section 2.6.5.1.2. BNFP is described in Section F.1. For this implementation alternative, an analysis option 1C is considered, which is similar to 1B, as follows:

1C. The spent nuclear fuel managed under analysis option 1A would be transferred to a newly constructed wet storage facility or the BNFP where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 would be received and managed at these facilities. For the purpose of this analysis, the amount of spent nuclear

fuel that would be managed in these facilities would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in Section 2.3.6, the Savannah River Site is currently limited to chemical separation of aluminum-based foreign research reactor spent nuclear fuel.

Under Management Alternative 2, as discussed in Section 2.3, DOE and the Department of State would assess the management of foreign research reactor spent nuclear fuel in a foreign location which would include an evaluation of foreign reprocessing with acceptance by the United States of the vitrified high-level waste resulting from reprocessing. The waste would be received and managed at the Defense Waste Process Facility at the Savannah River Site. DOE estimates that the total volume of the vitrified high-level waste would be about 2.4 m^3 (8.5 ft^3) and it would fill about 16 European-size canisters. A European-size canister is about four times smaller than the canister used in the Defense Waste Process Facility at the Savannah River Site.

Under Management Alternative 3 (Hybrid Alternative), as discussed in Section 2.4, the Savannah River Site would receive the aluminum-based fuel which would not be reprocessed overseas. This spent nuclear fuel would be processed at the Savannah River Site chemical separation facilities in the same manner as in Implementation Alternative 6 above. The amount of foreign research reactor aluminum-based spent nuclear fuel to be chemically separated would be approximately 12,200 elements, 12.9 MTHM, 79 m^3 (2,600 ft^3).

F.4.1.1 Existing Facilities (Phase 1)

Analysis option 1A utilizes existing facilities that would be ready to receive and store foreign research reactor spent nuclear fuel by late 1995. The environmental impacts from this analysis option include only those related to operations, specifically: socioeconomics; occupational and public health and safety; materials, utilities, and energy; air quality; and waste management. For this analysis, it was assumed that the amount of foreign research reactor spent nuclear fuel to be received at the management site is the maximum, and the receipt rate is uniform at approximately 1,800 elements per year.

F.4.1.1.1 Socioeconomics

Potential socioeconomic impacts associated with analysis option 1A would be attributable to the staffing requirements for existing facilities. Currently, these facilities are being used to store spent nuclear fuel, so any incremental staffing requirements related to foreign research reactor spent nuclear fuel storage would be small. All personnel required for the operation and support of the existing facilities could be acquired from the current work force at the Savannah River Site. Use of the current work force would not result in any net socioeconomic impact relative to baseline employment data. In fact, using the current work force may partially compensate for the decline in employment expected from changes in site mission from 20,000 persons in 1995 to approximately 15,800 persons in 2004 (DOE, 1995g).

F.4.1.1.2 Occupational and Public Health and Safety

Radiological exposures could affect occupational and public health and safety. Possible sources of radiological exposure from the receipt and storage of foreign research reactor spent nuclear fuel include: (1) airborne emissions from incident-free operations; (2) incident-free handling activities; and (3) airborne emissions from accident conditions. Radiological exposures are presented in individual subsections for

emissions-related impacts and handling-related impacts. Accident-related impacts are presented in Section F.4.1.3.

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to airborne emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage would take place to receive any dose from direct exposure. Doses were calculated for the maximally exposed individual (MEI), defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the existing storage facility (RBOF and/or L-Reactor disassembly basin) during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-21 summarizes the annual emission-related doses to the public and the associated risks for the MEI and the population at the Savannah River Site during Phase 1 operations.

Table F-21 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage in Existing Facilities at the Savannah River Site (Phase 1)

	MEI Dose (mrem/yr)	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• RBOF (wet)	0.00011	5.5×10^{-11}	0.0057	0.0000028
• L-Reactor Basin (wet)	0.000073	3.7×10^{-11}	0.0046	0.0000023
Storage at:				
• RBOF (wet)	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
• L-Reactor Basin (wet)[a]	0.00036	1.8×10^{-10}	0.022	0.000011

[a] *L-Reactor basin doses are due to existing conditions. The foreign research reactor spent nuclear fuel contribution would be six orders of magnitude lower.*

Handling-Related Impacts: Management site workers would receive radiation doses during handling operations, such as receiving and unloading the transportation casks, transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1A involves the receipt of 644 shipments of spent nuclear fuel into the existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, and the preparation of 161 transportation casks for offsite shipment at the end of Phase 1. It was assumed that at the end of a 10-year period (i.e., Phase 1), the spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodology used to calculate the doses to a working crew associated with the handling activities of the spent nuclear fuel are described in Section F.5 of this appendix.

The collective doses that would be received by the members of the working crew and the associated risk were calculated for Phase 1 operations. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculation. However, the upper bound for such a dose would be equal to administrative or regulatory limits at the management site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and

if any worker's dose approached this limit, he or she would be rotated into a different job. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If one worker received the full 5,000 mrem per year for the full 13 years of potential spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. The associated risk of incurring a latent cancer fatality (LCF) would be 2.6 percent. The collective dose to the workers would be 250 person-rem with an associated LCF risk of 0.10.

F.4.1.1.3 Material, Utility, and Energy Requirements

The estimated annual consumption of materials, utilities, and energy from the use of existing storage facilities is shown in Table F-22.

Table F-22 Annual Utility and Energy Requirements for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Commodity	*Baseline Site Usage*	*RBOF*	*L-Reactor Basin*	*Percent Increase*
Electricity (MW-hr/yr)	659,000	1,430	784	0 percent
Fuel (l/yr)	28,400,000	6,570	15,000	0.05 percent
Water (l/yr)	88,200,000,000	35,100,000	2,900,000	0.04 percent

The material, utility, and energy requirements for analysis option 1A would represent a small percentage of current requirements. No new generation or treatment facilities would be necessary. Increases in fuel consumption at the Savannah River Site would be minimal because overall onsite activity would not increase due to changes in the Savannah River Site mission and the general reduction in employment levels. The Programmatic SNF&INEL Final EIS concluded that the existing capacities and distribution systems for electricity, steam, water, and domestic wastewater treatment are adequate to support any of the five alternatives considered for spent nuclear fuel management at the Savannah River Site. This conclusion would also be valid for analysis option 1A because it is bounded by the alternatives considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g).

F.4.1.1.4 Waste Management

The estimated annual waste generation for foreign research reactor spent nuclear fuel at the RBOF and L-Reactor disassembly basin is shown in Table F-23. These quantities represent a very small percent increase above current levels at the Savannah River Site. Existing waste management storage and disposal activities at the Savannah River Site could accommodate the waste generated by foreign research reactor spent nuclear fuel storage at the RBOF and L-Reactor disassembly basin. Therefore, the impact of this waste on existing Savannah River Site waste management activities would be minimal.

Table F-23 Annual Waste Generation for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Waste Form	*Baseline Site Generation*	*RBOF*	*L-Reactor Basin*	*Percent Increase*
High-Level Waste (m^3/yr)	127,400[a]	0	0	0 percent
Transuranic Waste (m^3/yr)	760	0	0	0 percent
Solid Low-Level Waste (m^3/yr)	19,750	161	510	2.6 percent
Wastewater (l/yr)	690,000,000	2,650,000	35,000,000	5.1 percent

[a] *Total inventory (m^3) at the Savannah River Site.*

F.4.1.1.5 Air Quality

Nonradiological Emissions: Impact assessments for nonradiological air emissions associated with implementation of the respective spent nuclear fuel management alternatives (excluding construction-related activities) are based primarily on analyses performed for the Programmatic SNF&INEL Final EIS (DOE, 1995g). These analyses were based on the following assumptions and qualifications:

- Air emissions data for wet storage are based upon releases from the RBOF
- All air pollutant sources, except standby diesel generators, are assumed to operate continuously. The standby diesel generators are assumed to operate daily for 1 hour; whereas their actual operation consists of a single monthly test.

Table F-24 lists the annual potential maximum emissions of criteria and toxic air pollutants (in tons per year) attributable to existing facilities. These data indicate little or no difference in pollutant loading between the baseline and the regionalization alternative considered in the Programmatic SNF&INEL Final EIS. The greatest pollutant contribution for criteria air pollutants would be for nitrogen oxides (7.7 metric tons per year, or 8.5 tons per year) and carbon monoxide (1.8 metric tons per year, or 2.0 tons per year). Assuming that the foreign research reactor spent nuclear fuel comprises approximately nine percent of the total spent nuclear fuel managed under the Centralization Alternative of the Programmatic SNF&INEL Final EIS, the incremental and cumulative nonradiological air quality impacts attributable to the storage of foreign research reactor spent nuclear fuel in existing facilities (RBOF, L-Reactor disassembly basin) would be small.

Table F-24 Annual Maximum Emissions of Criteria Air Pollutants Attributable to Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Pollutant	*Baseline Wet (tons/yr)*[a,b,c]	*RBOF (tons/yr)*[a,b]	*Percent Increase*
Sulfur Oxides	0.4	0.005	1.3 percent
Nitrogen Oxides	6.0	0.77	12.8 percent
Total Suspended Particulates	0.4	0.006	1.5 percent
Carbon Monoxide	1.5	0.18	12 percent
Total Volatile Organic Compounds	0.6	0.077	12.8 percent
Gaseous Fluorides	none	none	none

[a] *Source: Hunter and Stewart, 1994*

[b] *To convert tons to metric tons, multiply by 0.907*

[c] *Decentralization based on management of the existing Savannah River Site inventory of spent nuclear fuel*

Radiological Emissions: The ventilation system serving the RBOF is designed to minimize airborne radioactivity levels both inside and outside of the facility. This ventilation system is based on a "once through" multiple-air-zone concept in which air flows from areas with low potential for contamination to areas with higher potential. The airflow is passed through High-Efficiency Particulate Air filters mounted on the building exhaust. A review of 1993 emissions data from the RBOF indicates emissions of approximately 2.7×10^{-7} Ci per year of ^{137}Cs (DOE, 1995g).

The RBOF and L-Reactor disassembly basin are currently being utilized to wet store spent nuclear fuel, and their emissions are reflected within baseline environmental conditions.

F.4.1.1.6 Water Resources

The use of RBOF and the L-Reactor disassembly basin for the interim storage of foreign research reactor spent nuclear fuel would not change the current levels of water usage at these facilities. Nor would it change thermal discharges from cooling water or the quantity or quality of radioactive and nonradioactive wastewater effluents.

Viable accidents during this interim storage period could be a release of pool water onto the ground surface or a breach of the liner of the wet storage basins in which the spent nuclear fuel would be stored. These type of accidents have been analyzed for both the RBOF and the L-Reactor disassembly basin in the safety analysis documentation (Dupont, 1983a and 1983b; WSRC, 1995b and 1995c) and the Programmatic SNF&INEL Final EIS (DOE, 1995g). As discussed in the Programmatic SNF&INEL Final EIS, radionuclides in the released water would enter the water table aquifer but would not reach any surface-water or any drinking water aquifer on or off the Savannah River Site. Basin water contains no toxic or hazardous chemicals, therefore, accidental releases from the basins would have minimal impacts on surface- and groundwater resources.

F.4.1.2 New Facilities (Phase 2)

Analysis options 1B and 1C involve the use of new facilities for the storage of foreign research reactor spent nuclear fuel at the Savannah River Site. The environmental impacts analyzed relate to the construction and operation of these new facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities and energy; and waste management.

F.4.1.2.1 Dry Storage

Analysis option 1B is associated with the construction and operation of new dry storage facilities. The dry storage option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 of this EIS and earlier in this appendix. None of the environmental impact parameters discriminate between the two designs. For the purpose of this analysis, the impacts from the larger dry vault design are presented.

F.4.1.2.1.1 Land Use

A new dry storage facility would be located in one of two 60-plus ha (150-plus acres) undeveloped areas near the H- and P-areas, respectively. Predominant land use at both areas is managed timber land. Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land. This represents about 6 percent of the available space at either area. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at either area would significantly impact land use patterns on the Savannah River Site.

F.4.1.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is

evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents about 7.7 percent of the estimated FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual construction expenditures on the region of influence would be small but positive. The annual operations costs from a new dry storage facility are estimated to be $15.6 million for receipt and handling and $.6 million for storage. These costs represent about 1.3 percent and 0.05 percent of FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Savannah River Site of approximately 20,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to eight persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Savannah River Site.

F.4.1.2.1.3 Cultural Resources

There are no known cultural or historic resources located within the two proposed construction locations for a new dry storage facility. Both locations are within an area of low archaeological site density. Activities within this zone would have a low probability of encountering archaeological sites and virtually no chance of impacting large sites with more than three prehistoric components. Neither location has been specifically surveyed for archaeological resources, but this would occur prior to initiation of any construction-related activities.

Three Native American groups have expressed concerns relating to the possible existence on the Savannah River Site of several plant species traditionally used in Tribal ceremonies. These plant species are known to occur on the Savannah River Site, typically in wet, sandy areas such as evergreen shrub bogs and savannas. However, these plants are not likely to be found in the two proposed construction locations because of a lack of suitable habitat.

F.4.1.2.1.4 Aesthetics and Scenic Resources

Construction and operation of a new dry storage facility would not adversely impact aesthetic or scenic resources. A new dry storage facility would not be visible from any onsite or offsite public access roads. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the new dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility.

F.4.1.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Savannah River Site that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils and would necessitate the clearing and grading of 3.7 ha (9 acres).

Site preparation, land shaping and grading activities associated with construction would present a slight to moderate erosion hazard, which would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the Savannah River Site.

F.4.1.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction-related activities include the generation of fugitive dust (particulate matter), smoke from earth moving and clearing operations, and emissions from the construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Cleared vegetation would be burned at the construction site rather than hauled to a landfill. The open burning of this material is not expected to adversely impact ambient air quality at the Savannah River Site. As shown in Table F-25, air quality impacts associated with construction-related activities would be minimal and compliance with Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-25 Estimated Maximum Concentrations of Criteria Pollutants at the Savannah River Site Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[a]	*Baseline Concentration*[b]	*Construction Activities*
Savannah River Site Boundary ($\mu g/m^3$):				
• Total Suspended Particulate (TSP)	Annual	75	11	0.002 - 0.003
• Particulate Matter (PM_{10}), Daily	24-hr	150	56	0.1
• Particulate Matter (PM_{10}), Daily	Annual	50	2.7	0.003

[a] *Source: (DOE, 1995g)*

[b] *Baseline values due to actual emissions from all the Savannah River Site sources during 1990 plus sources permitted through 1992*

No nonradiological air emissions would be expected during operation of a new dry storage facility. Any emissions associated with new dry storage would be directly attributable to front-end wet storage activities only.

Radiological Emissions: No radiological emissions would be produced during construction of a dry new storage facility.

Based on fuel drying and storage operations conducted at the Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and

generally consists of radionuclides, such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively high contamination) would help provide contamination control. High-Efficiency Particulate Air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depend on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask; and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a vapor state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

Radiological emissions from the operation of a new dry storage facility were calculated based on the methodology and assumptions discussed in Section F.6. The radiological consequences of air emissions are discussed in Section F.4.1.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.1.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 88,200 million l (23,300 million gal) for the Savannah River Site, these amounts represent no more than a 0.002 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Savannah River Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Savannah River Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Savannah River Site could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Savannah River Site would still be well within the design capacities of treatment systems at the Savannah River Site. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.1.2.1.8 Ecology

Terrestrial Resources: The two proposed locations for new spent nuclear fuel management facilities encompass approximately 60-plus ha (150-plus acres) of undeveloped forest land. Surface vegetation consists primarily of upland pine stands. Loblolly and slash pine dominate, but small pockets of hardwoods (oaks, hickory, sweetgum, and yellow poplar) are also evident. The locations possess suitable habitat for white-tailed deer and feral hogs, as well as other faunal species common to the mixed pine/hardwood forests of South Carolina. The locations contain no suitable habitat for the various threatened and endangered species found on the Savannah River Site. The construction of a new dry storage facility would necessitate the clearing of 3.7 ha (9 acres) and is therefore not expected to significantly affect the terrestrial ecology of the area.

Wetlands: Dry storage of foreign research reactor spent nuclear fuel would not adversely impact wetlands. Although two small wetland areas are located along the southeastern perimeters of the potential storage locations, there is sufficient land area available within these locations to avoid these critical habitats. The implementation of best-management practices to control surface runoff and sedimentation would ensure the protection of wetlands and the aquatic ecosystem during construction activities.

Threatened and Endangered Species: The potential locations contain no suitable habitat for threatened, endangered, or candidate species known to occur on or near the Savannah River Site (DOE, 1995g). The southern bald eagle and wood stork feed and nest near wetlands, streams, and reservoirs, and thus would not be attracted to the highly industrialized foreign research reactor spent nuclear fuel management sites. Red-cockaded woodpeckers prefer open pine forests with mature trees greater than 70 years old for nesting and 30 years old for foraging. It is not believed that this species utilizes the relatively young pine stands (5 to 40 years of age) present within the potential storage locations. The nearest red-cockaded woodpecker colony is located across Upper Three Runs Creek, approximately 3.2 km (2 mi) north of H-Area. DOE has begun consultations with the U.S. Fish and Wildlife Service to determine the potential for endangered species to be affected, as required by the Endangered Species Act. Impacts to threatened and endangered species are not anticipated.

F.4.1.2.1.9 Noise

Noise generated onsite by construction or operation of a new dry storage facility should not adversely affect the public or the Savannah River Site environment. Noise generated by construction would be site specific and short lived. A small number of new construction jobs would be generated, but the resultant temporary increase in worker and materials traffic is expected to be insignificant compared to existing baseline traffic loads. Noise generated by operation would not significantly impact the environment because the facility would be located adjacent to previously developed, industrialized areas. The number of daily freight trains in the region and through the site (approximately 13) are not expected to change as a result of dry storage. There may be a slight increase in truck traffic to and from the potential storage locations, but this is not expected to result in a perceptible increase in traffic noise or any change in community reaction to noise along the major access routes to the Savannah River Site.

F.4.1.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction

workers, commercial truck traffic, and in traffic associated with the daily operations of the Savannah River Site. Again, traffic congestion would not be a significant problem. As long as commercial trucks are complying with the Federal and State loading and speed regulations, truck traffic would not significantly damage the roadbed.

Traffic due to operations of a new dry storage facility would not increase site levels because the required workers would be drawn from the existing the Savannah River Site labor force.

F.4.1.2.1.11 Occupational and Public Health and Safety

Emissions Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-26 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Savannah River Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-26 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at Savannah River Site (New Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility	0.00018	9.0×10^{-11}	0.0086	0.0000043
Storage at:				
• New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1B involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into an existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, the preparation of 161 transportation casks for shipment to a dry storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel directly from the ports to the new dry storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. Doses were calculated for the dry vault and dry cask designs. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-27 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Savannah River Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-27 Handling-Related Impacts to Workers at the Savannah River Site (New Dry Storage)

	Worker Population Dose (person-rem)			*Worker Population Risk (LCF)*		
	RBOF/L-Reactor	*New Dry Storage Cask*	*New Dry Storage Vault*	*RBOF/L-Reactor*	*New Dry Storage Cask*	*New Dry Storage Vault*
Phase 1	250	NA	NA	0.10	NA	NA
Phases 1 and 2	NA	416	363	NA	0.17	0.15

NA = Not Applicable

F.4.1.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Savannah River Site would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel; and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-28. These requirements represent a small percent of current requirements for the Savannah River Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Savannah River Site is expected to decrease due to changes in site mission and a general reduction in employment.

Table F-28 Annual Utility and Energy Requirements for New Dry Storage at the Savannah River Site

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	659,000	800 - 1,000	0.15 percent
Fuel (l/yr)	28,400,000	0	0 percent
Water (l/yr)	88,200,000,000	1,590,000[a] 400,000[b]	0.002 percent[a] 0.00046 percent[b]

[a] *During receipt and handling.*

[b] *During storage.*

F.4.1.2.1.13 Waste Management

Construction of a new dry storage facility at the Savannah River Site would generate approximately 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-29. These quantities represent a very small percent increase above current levels at the Savannah River Site. Existing waste management storage and disposal activities at the Savannah River Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on the existing Savannah River Site waste management capacities would be minimal.

Table F-29 Annual Waste Generated from New Dry Storage at the Savannah River Site

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level Waste (m^3/yr)	127,400[a]	none	0 percent
Transuranic Waste (m^3/yr)	760	none	0 percent
Solid Low-Level Waste (m^3/yr)	19,750	22[b] 1[c]	0.11 percent[b] 0.005 percent[c]
Wastewater (l/yr)	690,000,000	1,590,000[b] 400,000[c]	0.21 percent[b] 0.06 percent[c]

[a] *Total inventory (m^3) at the Savannah River Site*

[b] *During receipt and handling*

[c] *During storage*

F.4.1.2.2 Wet Storage

Analysis option 1C is associated with the construction and operation of a new wet storage facility or the modification and operation of BNFP at the Savannah River Site (Implementation Alternative 5 to Management Alternative 1). The environmental impacts from the modification of the BNFP would be bounded by the impacts associated with the construction of a new wet storage facility.

F.4.1.2.2.1 Land Use

A new wet storage facility would be located in one of two 60-plus ha (150-plus acres) undeveloped areas near the H- and P-areas, respectively. Predominant land use at both areas is managed timber land. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. This represents less than 5 percent of the available space at either area. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at either area would significantly impact land use patterns on the Savannah River Site.

F.4.1.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 9.4 percent of the estimated FY 1995 total expenditures for the Savannah River Site (1,198 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be small but positive. The annual operations costs of a new wet storage facility

are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 1.9 percent and 0.3 percent of FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Savannah River Site of approximately 20,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Savannah River Site.

F.4.1.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.1.2.1.3).

F.4.1.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.1.2.1.4).

F.1.4.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.1.2.1.5).

F.4.1.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 2.8 ha (7 acres) of land. In comparison, approximately 3.7 ha (9 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with new dry storage construction, as presented in Table F-25.

Operations-related impacts associated with wet storage would be similar to those discussed under existing facilities.

Radiological Emissions: Incident-free airborne releases from a new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3-micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. The truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b).

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE Order 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

F.4.1.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 88,200 million l (23,300 million gal) for the Savannah River Site, these amounts represent an increase of less than 0.01 percent for both. Therefore, a new wet storage facility would have minimal impact on water resources at the Savannah River Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Savannah River Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Savannah River Site could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Savannah River Site would still be well within the design capacities of treatment systems at the Savannah River Site. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.1.2.2.8 Ecology

Impacts to the ecology would be the same as for new dry storage (Section F.4.1.2.1.8).

F.4.1.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.1.2.1.9).

F.4.1.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.1.2.1.10).

F.4.1.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-30 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Savannah River Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-30 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• BNFP	0.00065	3.3×10^{-10}	0.0045	0.0000023
• New Wet Storage Facility	0.00011	5.5×10^{-11}	0.0057	0.0000028
Storage at:				
• BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
• New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1C involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into an existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, the preparation of 161 transportation casks for shipment to a wet storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel directly from the ports into the new wet storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-31 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Savannah River Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this

limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-31 Handling-Related Impacts to Workers at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*Worker Population Dose (person-rem)*	*Worker Population Risk (LCF)*
Phase 1: RBOF/L-Reactor Basin Phase 1 and Phase 2: New Wet Storage Facility	250 360	0.10 0.14
Phase 1: RBOF/L-Reactor Basin Phase 1 and Phase 2: BNFP	250 360	0.10 0.14
Phase 1: RBOF/L-Reactor Basin Phase 1 and Phase 2: BNFP[a]	250 310	0.10 0.12

[a] *Assumes that BNFP would be ready in 5 years instead of 10 years.*

F.4.1.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Savannah River Site would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-32. These requirements represent a small percent of current requirements for the Savannah River Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Savannah River Site is expected to decrease due to changes in site mission and a general reduction in employment.

Table F-32 Annual Utility and Energy Requirements for New Wet Storage at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Commodity	*Baseline Site Usage*	*Wet Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	659,000	1,000 - 1,500	0.23 percent
Fuel (l/yr)	28,400,000	0	0 percent
Water (l/yr)	88,200,000,000	2,700,000[a] 1,500,000[b]	0.003 percent 0.001 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.1.2.2.13 Waste Management

Construction of a new wet storage facility would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-33. These quantities represent a very small percent increase in current levels at the Savannah River Site. Existing waste management storage and disposal activities at the Savannah River Site could accommodate the waste generated by a new wet

Table F-33 Annual Waste Generated from New Wet Storage at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Waste Form	*Baseline Site Generation*	*Wet Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	127,400[a]	none	0 percent
Transuranic (m^3/yr)	760	none	0 percent
Solid Low-Level (m^3/yr)	19,750	16[b] 1[c]	0.08 percent 0.005 percent
Wastewater (l/yr)	690,000,000	1,590,000[b] 400,000[c]	0.23 percent 0.06 percent

[a] *Total inventory (m^3) at the Savannah River Site.*

[b] *During receipt and handling*

[c] *During storage*

storage facility. Therefore, the impact of this waste on the existing Savannah River Site waste management capacities would be minimal.

F.4.1.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Savannah River Site is presented here, based on the methodology presented in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated.

The radiation doses to the following individuals and the general population are calculated for accident conditions at the spent nuclear fuel storage facility:

- ***Worker*:** An individual located 100 m (330 ft) from the radioactive material release point. For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (330 ft) for elevated release. The direction to the worker was chosen as the direction to the maximally exposed sector. The dose to the worker is calculated for the 50th-percentile meteorological condition (DOE, 1992a).

- ***Maximally Exposed Offsite Individual (MEI):*** A theoretical individual living at the storage site boundary receiving the maximum exposure. The individual is assumed to be located in a direction downwind from the release point. The dose to the MEI is shown for the 95th-percentile meteorological condition.

- ***Nearest Public Access Individual (NPAI).*** An individual stranded on a highway or public access road near to the facility at the time of an accident. The distance to the NPAI was chosen as the distance to the nearest public access point; the direction was chosen as the direction to that point. The dose to the NPAI is shown for the 95th-percentile meteorological condition.

- ***General Population Within an 80 km (50 mi) Radius of the Facility:*** The dose calculations are performed for the direction downwind from the release point that results in highest dose to the public. The dose to the population is shown for the 95th-percentile meteorological condition.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using external (direct exposure), inhalation, and ingestion pathways. Dispersion in air from point of release was estimated with both 50th-percentile and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition. The 95th-percentile condition is defined as that condition which is not exceeded more than 5 percent of the time, and is more conservative than the 50th-percentile condition.

The ingestion dose is calculated by considering that the individual and the public would consume contaminated food produced in the vicinity [up to 80 km (50 mi)] of the accident. This is conservative, and it is expected that continued consumption of contaminated food products by the public would be suspended after a protective action guideline is reached. In 1991, the U.S. Environmental Protection Agency recommended protective action guidelines in the range of one to five rem whole-body exposure (EPA, 1991). To ensure a consistent analytical basis, no reduction of exposure due to a protective action guideline was used in this analysis.

Accidents considered for detailed analysis are similar to those analyzed in the Programmatic SNF&INEL Final EIS. The selection of accidents was based on the following considerations:

- Accidents in the Programmatic SNF&INEL Final EIS were reviewed to select reasonably foreseeable accidents. They are: (1) criticality caused by human error during operation, equipment failure, or earthquake; (2) mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a spent nuclear fuel element); and (3) accident involving an impact by either an internal or an external initiator with and without an ensuing fire.

Six accident scenarios were evaluated at each storage location using identical source terms (estimated amounts of radioactive material released during postulated accidents). The wet pool accidents are assumed to be cutting into the fuel region or mechanical damage due to operator error, an accidental criticality, and an aircraft crash into the water pool facility. The dry storage accidents are assumed to be cutting into the fuel region or mechanical damage during examination work and handling in a dry cell, dropping of a fuel cask, and an aircraft crash with an ensuing fire.

Table F-34 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Multiplying the frequency of each accident times its consequences at the Savannah River Site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Savannah River Site. These annual risks are multiplied by the maximum duration of the policy at each site to obtain conservative estimates of risks for the entire program at the Savannah River Site. These risk estimates are presented in Table F-35.

Table F-36 presents the frequency and consequences of the accidents analyzed for the Savannah River Site for wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences at the Savannah River Site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Savannah River Site. These annual risks are multiplied by the maximum duration of this implementation alternative at the Savannah

Table F-34 Frequency and Consequences of Accidents at the Savannah River Site

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Dry Storage Accidents - New					
• Spent Nuclear Fuel Assembly Breach	0.16	0.24	0.068	9.2	28
• Dropped Fuel Cask	0.0001	0.018	0.00034	0.55	0.28
• Aircraft Crash w\Fire	1×10^{-6}	40	0.29	1300	120
Wet Storage Accidents at RBOF					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	0.00039	0.23	0.14
• Accidental Criticality	0.0031	130	44	4,800	16,000
• Aircraft Crash	1×10^{-6}	4.1	0.98	150	400
Wet Storage Accidents at L-Reactor Basin					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0093	0.00097	0.14	0.11
• Accidental Criticality	0.0031	170	120	3,000	14,000
• Aircraft Crash	1×10^{-6}	4.2	2.6	93	70

Table F-35 Annual Risks of Accidents at the Savannah River Site

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents - New				
• Spent Nuclear Fuel Assembly Breach	1.9×10^{-8}	5.5×10^{-9}	0.00075	0.0000018
• Dropped Fuel Cask	9.0×10^{-13}	1.7×10^{-14}	2.8×10^{-8}	1.1×10^{-11}
• Aircraft Crash w\Fire	2.0×10^{-11}	1.5×10^{-13}	6.5×10^{-7}	4.8×10^{-11}
Wet Storage Accidents at RBOF				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.0×10^{-7}	7.0×10^{-8}	0.0074	0.000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
Wet Storage Accidents at L-Reactor Basin				
• Spent Nuclear Fuel Assembly Breach	7.4×10^{-10}	8.0×10^{-11}	0.000011	7.1×10^{-9}
• Accidental Criticality	2.6×10^{-7}	1.9×10^{-7}	0.0047	0.000017
• Aircraft Crash	2.1×10^{-12}	1.3×10^{-12}	4.7×10^{-8}	2.8×10^{-11}

Table F-36 Frequency and Consequences of Accidents at Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Wet Storage Facility - New					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	0.00039	0.23	0.14
• Accidental Criticality	0.0031	17	9.5	370	1,600
• Aircraft Crash	1×10^{-6}	4.1	0.98	150	400
BNFP					
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.018	0.00099	0.028	0.0008
• Accidental Criticality[a]	0.0031	80	75	44	75
• Aircraft Crash	1×10^{-6}	92	31	23	70

[a] *Emissions would be released through a tall stack.*

River Site to obtain conservative estimates of risks at the Savannah River Site. Table F-37 presents the risk estimates from this implementation alternative.

Table F-37 Annual Risks of Accidents at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Wet Storage Facility - New				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.7×10^{-8}	1.5×10^{-8}	0.00060	0.0000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
BNFP				
• Spent Nuclear Fuel Assembly Breach[a]	2.8×10^{-9}	8.0×10^{-11}	0.0000023	5.2×10^{-11}
• Accidental Criticality[a]	1.3×10^{-7}	1.2×10^{-7}	0.000070	9.2×10^{-8}
• Aircraft Crash[a]	4.6×10^{-10}	1.6×10^{-11}	1.2×10^{-8}	2.8×10^{-10}

[a] *Emissions would be released through a tall stack.*

F.4.1.3.1 Secondary Impact of Radiological Accidents at the Savannah River Site

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the results of the accident calculations presented in Section F.4.1.3. Radiological accidents that would result in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Savannah River Site, the radiological accident with the highest MEI dose is the accidental criticality at a wet storage facility (Table F-34). For this accident, the MEI dose would be 170 mrem. For the air immersion and ground surface pathways only, the dose would be 50 mrem, (Table F-115A) which is lower than the 100 mrem limit used in this analysis. Local contamination would be likely around the dry storage facility, but is expected to be contained entirely within the boundaries of the Savannah River Site. Cleanup activities should be small and any impacts to land uses, cultural resources, water quality, and ecology would be reversible. No impacts to national defense or local economies would be expected.

F.4.1.4 Cumulative Impacts at the Savannah River Site

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions, and other offsite (non-DOE) facility impacts at the Savannah River Site. A major portion of the presentation is based on information included in the Interim Management of Nuclear Materials Final EIS for the Savannah River Site, issued in October 1995 (DOE, 1995b). The cumulative impacts include those associated with the handling and dry storage of foreign research reactor spent

nuclear fuel at the Savannah River Site and the following existing or major foreseeable activities proposed for the site:

- The operation of the Vogtle Electric Generating Plant located approximately 16 km (10 mi) south west of the center of the Savannah River Site.
- The implementation of the preferred scenario in the Interim Management of Nuclear Materials EIS (DOE, 1995b).
- Shipment of aluminum-based spent nuclear fuel to the Savannah River Site for storage and disposal discussed in Appendix C of the Programmatic SNF&INEL Final EIS (DOE, 1995g).
- Completion of the construction and operation of the Defense Waste Processing Facility (DOE, 1994g).
- Processing of F-Canyon plutonium solutions to metal (DOE, 1994a).
- Treatment and minimization of radioactive and hazardous wastes at the site as identified in the Savannah River Site Waste Management Final EIS (DOE, 1995f).
- Construction of an accelerator for tritium production at the Savannah River Site, along with associated support facilities (DOE, 1995a).
- Disposition of Surplus Highly Enriched Uranium at the site (DOE, 1995e).
- Storage and Disposition of Weapons-Usable Fissile Materials.
- Stockpile Stewardship and Management Program.
- Current Savannah River Site projects (based on 1993 data).

Any other foreseeable activities would have minimal impacts compared to the activities considered above.

Table F-38 summarizes the cumulative impacts for land use, socioeconomics, nonradiological air quality, occupational and public health and safety, waste generation, and energy and water consumption. As shown in Table F-38, the contribution of foreign research reactor spent nuclear fuel to the cumulative impacts at the Savannah River Site would be minimal.

F.4.1.5 Unavoidable Adverse Environmental Impacts

The construction and operation of facilities for the receipt and management of foreign research reactor spent nuclear fuel at the Savannah River Site would result in some adverse impacts to the environment. Changes in designs and other methods of mitigation could eliminate, avoid, or reduce most impacts to minimal levels. The following paragraphs identify adverse impacts that mitigation could not reduce to minimal levels or avoid altogether.

The generation of some fugitive dust during construction would be unavoidable, but could be controlled by water and dust suppressants. Similarly, construction activities would result in some minor, yet unavoidable, noise impacts from heavy equipment, generators, and vehicles.

The maximum loss of habitat would result from conversion of approximately 4 ha (10 acres) of managed pine forest to industrial land use.

Table F-38 Cumulative Impacts at the Savannah River Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Current Activities*[a]	*Other Activities*[b]	*Cumulative Impact*
Land Use (acres)	9	9,075[c]	3,975	13,059
Socioeconomics (persons)	190[d]/30[e]	(f)	11,000[d]/6,200[e]	11,190[d]/6,230[e]
Air Quality (nonradiological)	See Table F-38A	See Table F-38A	See Table F-38A	See Table F-38A
Occupational and Public Health and Safety:				
• MEI Dose (rem/yr)	3.6×10^{-7}	0.00025	0.0041	0.0043
LCF (per year)	1.8×10^{-10}	1.25×10^{-7}	0.000002	0.000002
• Population Dose (person-rem/yr)	0.022	9.1	295	304
LCF (per year)	0.000011	0.0045	0.15	0.154
• Worker Collective Dose (person-rem/yr)	10[g]	263	1,418	1,691
LCF (per year)	0.004	0.10	0.57	0.67
Energy and Water Consumption[j]				
• Electricity (MW-hr/yr)	1,000	659,000	4,104,106[h]	4,764,106
• Fuel (million l/yr)	0	28.4	3.06	31.47
• Steam (million kg/yr)	0	1,700	1,550	3,250
• Coal (tons/yr)	0	210,000	20,440	230,440
• Water (million l/yr)	2.2	88,200	6,796	94,996
Waste Generation				
• High-Level (m^3/yr)	0	(i)	6,330	6,330
• Low-Level (m^3/yr)	22	(i)	35,600	35,622
• Saltstone (m^3/yr)	0	(i)	60,000	60,000
• Transuranic (m^3/yr)	0	(i)	1,038	1,038
• Mixed/Hazardous (m^3/yr)	0	(i)	2,561	2,561

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Based on 1993 site data*

[b] *Other activities include: interim management of nuclear materials, spent nuclear fuel management, Vogtle plant operation, defense waste processing facility operations, stabilization of Pu-solutions, site-wide waste management activities, tritium accelerator facility, disposition of surplus HEU, storage and disposition of weapons-usable fissile materials, and the stockpile stewardship and management program activities*

[c] *Five percent of the total SRS site area of 181,500 acres*

[d] *Increase over baseline during construction activities*

[e] *Increase over baseline during operation activities*

[f] *Baseline working force approximately 20,600 persons*

[g] *The dose is due to the handling of the FRR SNF during receipt and transfer between facilities averaged over 40 years*

[h] *Major portion is the requirement for electricity by the tritium production accelerator facility (3,740,000 MW-hr/yr)*

[i] *Included in "other activities"*

[j] *During operation activities*

The amount of radioactivity that incident-free operation of the spent nuclear fuel facilities would release is a small fraction of the cumulative operational releases at the Savannah River Site and would be well below applicable regulatory standards (see Tables F-38 and F-38A).

Table F-38A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Savannah River Site Boundary[a]

Pollutant	*Averaging Time*	*Regulatory Standard, (μg/m³)*	*Cumulative Concentration (μg/m³)[b]*
Carbon Monoxide	1-hour	40,000	331.7 (0.83%)
	8-hour	10,000	52.3 (0.52%)
Nitrogen Oxides	Annual	100	19.5 (19.5%)
Sulfur Dioxide	3-hour	1,300	1,159 (89.1%)
	24-hour	365	248 (68.2%)
	Annual	80	17 (21.3%)
Gaseous Fluorides	12-hour	3.7	1.38 (37.3%)
	24-hour	2.9	0.58 (20%)
	1 week	1.6	0.56 (34.8%)
	1 month	0.8	0.066 (8.2%)
Nitric Acid	24-hour	125	9.8 (7.8%)

[a] *Concentrations represent: foreign research reactor spent nuclear fuel management, other DOE-owned spent nuclear fuel management, defense waste processing facility operations, consolidated incineration facility operation, stabilization of Pu-solutions, waste management activities, tritium supply and recycling, disposition of surplus highly enriched uranium, storage and disposition of weapons-usable fissile materials, and stockpile and stewardship management program activities*

[b] *Numbers in parentheses indicate the percentage of the regulatory standard*

F.4.1.6 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of facilities for foreign research reactor spent nuclear fuel facilities at the Savannah River Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery. Construction and operation of facilities for foreign research reactor spent nuclear fuel management would also require the withdrawal of water from surface- and groundwater sources, but most of this water would return to onsite streams or the Savannah River after use and treatment.

F.4.1.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Savannah River Site could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Savannah River Site would be minimal in most environmental media.

Pollution Prevention: DOE is committed to comply with Executive Order 12856, "Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements;" Executive Order 12780, "Federal Acquisition, Recycling and Waste Prevention;" and applicable DOE orders and guidance documents in planning and implementing pollution prevention at the Savannah River Site. The pollution prevention program at the Savannah River Site was initiated in 1990 as a waste minimization program. Currently, the program consists of four major initiatives: solid waste minimization, source reduction and recycling of

wastewater discharges, source reduction of air emissions, and potential procurement of products manufactured from recycled materials. Since 1991, waste (all types) generated at the Savannah River Site has decreased, with the greatest reductions in hazardous and mixed wastes. These reductions are attributable primarily to material substitutions (DOE, 1995g).

All foreign research reactor spent nuclear fuel activities at the Savannah River Site would be subject to a pollution prevention program. Implementation of the program plan would minimize waste generated by these activities (DOE, 1995g).

Cultural Resources: A Programmatic Memorandum of Understanding, ratified on August 24, 1990, between the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Office, and the Advisory Council on Historic Preservation is the instrument for the management of cultural resources at the Savannah River Site. DOE uses this memorandum to identify cultural resources and develop mitigation plans for affected resources in consultation with the State Historic Preservation Office.

DOE would comply with the terms of the memorandum in support of foreign research reactor spent nuclear fuel activities at the Savannah River Site. For example, DOE would survey sites prior to disturbance and reduce impacts to any potentially significant resources discovered through avoidance or removal. Any artifacts encountered would be protected from further disturbance and the elements until removed (DOE, 1995g).

DOE conducted an investigation of Native American concerns over religious rights in the Central Savannah River Valley in conjunction with studies in 1991 related to a New Production Reactor. During this study, three Native American groups expressed concern over sites and items of religious significance on the Savannah River Site. DOE has included these organizations on its environmental mailing list, solicits their comments on NEPA actions on the Savannah River Site, and sends them documents about the Savannah River Site environmental activities, including those related to foreign research reactor spent nuclear fuel management considerations. These Native American groups would be consulted on any actions that may follow subsequent site-specific environmental reviews (DOE, 1995g).

Geology: DOE expects that there would be no impacts to geologic resources at the Savannah River Site under any storage option evaluated. Potential soil erosion in areas of ground disturbance would be minimized through sound engineering practices such as implementing controls for storm water runoff (e.g., sediment barriers), slope stability (e.g., rip-rap placement), and wind erosion (e.g., covering soil stockpiles). Relandscaping would minimize soil loss after construction was completed. These measures would be included in a site-specific Storm Water Pollution Prevention Plan that the Savannah River Site would prepare prior to initiating any construction (DOE, 1995g).

Air Resources: DOE would meet applicable standards and permit limits for all radiological and nonradiological releases to the atmosphere. In addition, the Savannah River Site would follow the DOE policy of maintaining radiological emissions to levels "as low as reasonably achievable" (ALARA). ALARA is an approach to radiation protection to control or manage exposures (both individual and collective) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit, but rather a process that has as its objective the attainment of dose levels as far below the applicable limits as practicable (DOE, 1995g).

Water Resources: DOE would minimize the potential for adverse impacts on surface water during construction through the implementation of a storm water pollution prevention plan that details controls

for erosion and sedimentation. The plan would also establish measures for prevention of spills of fuel and chemicals and for rapid containment and cleanup (DOE, 1994g).

DOE could minimize water usage during both construction and operation of facilities by instituting water conservation measures such as instructing workers in water conservation (e.g., turn off hoses when not in use), installing flow restrictors, and using self-closing hose nozzles (DOE, 1995g).

Ecological Resources: DOE does not anticipate any impact on wetlands on the Savannah River Site as a result of the spent nuclear fuel program. In any case, it is DOE and the Savannah River Site policy to achieve "no net loss" of wetlands. Pursuant to this goal, DOE has issued a guidance document, "Information for Mitigation Impacts at the Savannah River Site," for project planners that puts forth a practical approach to wetlands protection that begins with avoidance of impacts (if possible), moves to minimization of impacts (if avoidance is impossible), and requires compensatory measures (wetlands restoration, creation, or acquisition) in the event that impacts cannot be avoided (DOE, 1995g).

The analysis indicates that there are no threatened or endangered species or sensitive habitats in the areas considered as representative of potential sites for foreign research reactor spent nuclear fuel activities at the Savannah River Site. However, DOE would perform site-specific predevelopment surveys to ensure that development of new facilities would not impact any of these biological resources (DOE, 1995g).

Noise: DOE anticipates that noise impacts both on and off the Savannah River Site would be minimal. DOE does not foresee noise impacts from the management of foreign research reactor spent nuclear fuel that would warrant mitigation measures beyond those consistent with good construction, engineering, operations, and management practices.

Traffic and Transportation: DOE has a system of onsite buses operating at the Savannah River Site. The Savannah River Site would evaluate the need for upgrades or changes in service that might be required for foreign research reactor spent nuclear fuel management activities and would make changes, as necessary.

DOE would manage changes in traffic volume or patterns during construction through such measures as designating routes for construction vehicles, providing workers with safety reminders, and upgrading onsite police traffic patrols, if necessary.

Occupational and Public Health and Safety: The DOE program for maintaining radiological emissions to levels "as low as reasonably achievable" would minimize any impacts to workers and the public due to atmospheric releases. Likewise, the Site Pollution Prevention Plan and emergency preparedness measures would enhance safety both on and off the Site (DOE, 1995g).

Accidents: The Savannah River Site has in place emergency action plans that would be activated in the case of an accident. These plans contain both onsite provisions (e.g., evacuation plans, response teams, medical and fire response, training and drills, communications equipment) and offsite arrangements (e.g., response plans for medical and fire agencies, coordination with local and State agencies, communication plans). The Savannah River Site plans would be updated to include any new facilities or activities related to spent nuclear fuel management that would involve the Savannah River Site. The execution of the plans in response to an accident would mitigate adverse effects both on the Savannah River Site and in all the surrounding areas (DOE, 1995g).

F.4.2 Idaho National Engineering Laboratory

If the Idaho National Engineering Laboratory is the site to manage all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed at the site until ultimate

disposition. If the Idaho National Engineering Laboratory is not the site to manage DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel could be received and managed at the Idaho National Engineering Laboratory until the selected site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years; this period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2 (approximately 30 years). The amount of spent nuclear fuel that could be received at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, the Idaho National Engineering Laboratory could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, all of the TRIGA-type foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, only the foreign research reactor spent nuclear fuel from Western ports under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As discussed in Section 2.6.4.1, the split of foreign research reactor spent nuclear fuel evenly between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and 1992/1993 Planning Basis alternatives in the Programmatic SNF&INEL Final EIS was not considered to have a practical basis, and was therefore not evaluated in detail.

As a potential Phase 1 site, the Idaho National Engineering Laboratory would receive and manage foreign research reactor spent nuclear fuel at existing dry and wet storage facilities. The existing facilities identified for this purpose would be the FAST facility in CPP-666, the IFSF in CPP-603, and the CPP-749 storage area. Descriptions of these facilities are provided in Appendix F, Section F.3.

The FAST facility is a modern underwater storage facility which has been used in the past for receipt and storage of foreign research reactor spent nuclear fuel. It has the capability to receive and unload spent nuclear fuel casks at a rate of approximately five per week. Storage capacity for up to 8,400 foreign research reactor spent nuclear fuel elements could be provided in a 10-year period by using the spent nuclear fuel storage racks that would be installed. The capability of the FAST facility to receive foreign research reactor spent nuclear fuel in the near term is limited due to the number of activities scheduled through FY 1998. Considering these activities, DOE estimates that 3,600 elements could be received by the end of 1999 at the FAST facility.

The IFSF is a shielded dry storage vault originally constructed for Fort St. Vrain reactor fuel. The storage capacity available is for approximately 9,000 foreign research reactor spent nuclear fuel elements. However, as with the FAST facility, many activities are already scheduled for the facility. Considering these activities, foreign research reactor spent nuclear fuel could not be received until sometime in FY 1997 and could continue at the rate of 50 shipments per year (approximately 1,500 elements) thereafter.

The CPP-749 underground spent nuclear fuel storage area is a dry storage facility with a remote unloading area and vault storage. With some refurbishment it could provide space for 3,600 elements starting in FY 1998 and 7,000 elements after FY 2002. The spent nuclear fuel would go through the IFSF to be placed in baskets and transferred to a compatible storage cask. The refurbishments of existing facilities are part of the ongoing programs at the site to be performed independent of the proposed action in this EIS.

Between these facilities there is sufficient storage space and handling capacity to accommodate the receipt and management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory during the Phase 1 period. The storage capacity available and estimated maximum receipt rate

of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory was shown earlier in Figure F-18.

An additional option to enhance storage capacity during Phase 1 would be to use the existing facilities to unload the transportation casks, and provide storage capacity in dry storage casks which would be placed near the existing facility.

As a Phase 2 site, the Idaho National Engineering Laboratory would continue to receive and manage foreign research reactor spent nuclear fuel at existing facilities until a new dry storage facility becomes operational at the site. Foreign research reactor spent nuclear fuel managed at existing facilities would then be transferred to the new facility where it would remain until ultimate disposition. The new facility would also receive foreign research reactor spent nuclear fuel shipments directly from ports after Phase 1 concluded. Dry storage encompasses both the dry vault design and the dry cask design as described in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options under the basic implementation of Management Alternative 1 are as follows:

2A. The Idaho National Engineering Laboratory would receive foreign research reactor spent nuclear fuel during Phase 1 and manage it at the FAST, the IFSF, and/or the CPP-749 facilities. For the purpose of this analysis, the amount of fuel to be stored is all foreign research reactor spent nuclear fuel that would be received in a 10-year period (17,500 elements). The fuel would be shipped offsite at the end of Phase 1.

2B. Foreign research reactor spent nuclear fuel managed under analysis option 2A would be transferred to a newly constructed dry storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving at the United States after Phase 1 concludes would be received and managed at the new dry storage facility until ultimate disposition. For the purpose of this analysis, the amount of spent nuclear fuel that would be stored in the dry storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, as discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Idaho National Engineering Laboratory as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Idaho National Engineering Laboratory would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 2A above. The dry storage facility considered in analysis option 2B would be sized to accommodate this amount of fuel. The spent nuclear fuel would either be shipped offsite after Phase 1, or it would be managed along with the rest of the spent nuclear fuel that would be managed at the Idaho National Engineering Laboratory.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive only HEU from the reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the storage of this amount of fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A and 2B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 2A or 2B by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and therefore the amount of spent nuclear fuel available for acceptance would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A or 2B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis options 2A or 2B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States because the foreign research reactors would consider their own alternatives about whether to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel in this case cannot be quantified. The upper limit, however, as considered under analysis options 2A or 2B, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the impacts at the Idaho National Engineering Laboratory.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Idaho National Engineering Laboratory for Phase 2 until ultimate disposition. The new wet storage facility is described in Section 2.6.5.1.2. For this implementation alternative, an analysis option 2C, which is similar to option 2B, is considered as follows:

2C. The spent nuclear fuel managed under option 2A would be transferred to a newly constructed wet storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes would be received and managed at the new wet storage facility until ultimate disposition. For the purpose of this analysis, the amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in the discussion in Section 2.3.6, chemical separation of both aluminum-based and TRIGA foreign research reactor spent nuclear fuel is evaluated for the Idaho National Engineering Laboratory.

Under Management Alternative 3 (Hybrid Alternative), as discussed in Section 2.4, the Idaho National Engineering Laboratory would receive the foreign research reactor TRIGA spent nuclear fuel. This spent nuclear fuel would be managed at the Idaho National Engineering Laboratory in existing facilities until ultimate disposition. The amount of TRIGA spent nuclear fuel that would be stored is 4,900 elements, 1.0 MTHM, 19 m^3 (670 ft^3).

F.4.2.1 Existing Facilities (Phase 1)

Analysis option 2A utilizes existing facilities for receipt and storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory. The impacts from this analysis option include only those related to operations, specifically: socioeconomics, occupational and public health and safety, utilities and energy, air quality, and waste management. For this analysis, it was assumed that the amount of foreign research reactor spent nuclear fuel to be received at this management site is the maximum and the receipt rate is uniform at approximately 1,800 elements per year.

F.4.2.1.1 Socioeconomics

Potential socioeconomic impacts associated with analysis option 2A would be attributable to staffing requirements at existing facilities (FAST and IFSF). Currently, these facilities are being used to store spent nuclear fuel, so any incremental staffing requirements related to foreign research reactor spent nuclear fuel storage would be insignificant. All personnel required for the operation and support of the existing facilities could be acquired from the current work force at the Idaho National Engineering Laboratory. Use of the current work force would not result in any net socioeconomic impact relative to baseline environmental conditions. In fact, using the current work force would partially compensate for the decline in employment expected from changes in site mission.

F.4.2.1.2 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage would take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-39 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-39 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage in Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• IFSF/CPP-749 (dry storage)	0.00056	2.8×10^{-10}	0.0045	0.0000023
• FAST (wet storage)	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at:				
• IFSF/CPP-749 (dry storage)	0	0	0	0
• FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 2A involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into existing storage facilities (IFSF/CPP-749 and FAST) during Phase 1, and the preparation of 161 transportation casks for shipment at the end of Phase 1. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of the analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. Calculations were performed for both dry and wet existing storage facilities. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

The collective doses that would be received by the members of the working crew and the associated risk were calculated for Phase 1 operations. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent. The collective dose to the workers handling the transportation casks is 257 person-rem at the dry storage facilities and 250 person-rem at the wet storage facilities. The associated risk of incurring an LCF is 0.10.

F.4.2.1.3 Material, Utility, and Energy Requirements

The material, utility, and energy requirements at the FAST and IFSF are typical of those for wet storage and dry storage, respectively. They are discussed in more detail in Sections F.4.2.2.1.12 and F.4.2.2.2.12. Table F-40 summarizes the estimated annual requirements for these technologies.

Table F-40 Annual Utility and Energy Requirements for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Commodity	*Baseline Site Usage*	*FAST*	*IFSF*	*Percent Increase*
Electricity (MW-hr/year)	208,000	1,490	1,490	0.72 percent
Water (l/year)	6,500,000,000	1.93 million	2.12 million	0.033 percent
Fuel (l/year)	11,123,400	0	0	0 percent

The requirements for all storage options represent a small percentage of current requirements. No new generation or treatment facilities would be necessary. Increases in the Idaho National Engineering Laboratory fuel consumption would be minimal because overall activity would not increase due to changes in the Idaho National Engineering Laboratory mission and the general reduction in employment levels. The overall impacts of any of the storage options at the Idaho National Engineering Laboratory on materials, utilities, and energy resources would be minimal.

The existing capacities and distribution systems at the Idaho National Engineering Laboratory for electricity, steam, water, and domestic wastewater treatment are adequate to support the receipt and storage of foreign research reactor spent nuclear fuel for all storage options.

Some of the electric power at the Idaho National Engineering Laboratory is generated onsite, and the remainder is provided by the Idaho Power Company. The Utah Power and Light Company Antelope Substation, which is located on the Idaho National Engineering Laboratory, connects to the Scoville Substation, from which electricity is distributed to various facilities over a 138-kilovolt loop at the Idaho National Engineering Laboratory.

All water supplies for the Idaho National Engineering Laboratory are obtained from the Snake River Plain aquifer through wells. Pumping totals approximately 7 million m^3 per year (1.8 billion gallons per year). ICPP has a coal-fired steam system. Natural gas is not used at the Idaho National Engineering Laboratory.

F.4.2.1.4 Waste Management

Waste production associated with the operation of the FAST and IFSF facilities is characteristic to wet and dry storage, respectively, and is discussed in detail in Sections F.4.2.2.1.13 and F.4.2.2.2.13.

F.4.2.1.5 Air Quality

Nonradiological Emissions: It is expected that the ambient concentration levels from incident-free operation of existing facilities would not change from baseline concentrations due to the addition of foreign research reactor spent nuclear fuel. The baseline ambient concentrations are given in Table F-41. They are all below applicable standards and guidelines.

Radiological Emissions: Radiological emissions from the receipt and storage of foreign research reactor spent nuclear fuel in the existing facilities at the Idaho National Engineering Laboratory are discussed in Section F.4.2.1.2.

F.4.2.1.6 Water Resources

The use of FAST and ISFS facilities for the interim storage of foreign research reactor spent nuclear fuel would not change the current levels of water and usage of these facilities. Nor would it change thermal discharges from cooling water or the quantity or quality of radioactive and nonradioactive wastewater effluents.

Table F-41 Maximum Impacts to Nonradiological Air Quality from Spent Nuclear Fuel[a,b] at Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Pollutant	*Averaging Time*	*Applicable Standard ($\mu g/m^3$)[c]*	*Maximum Baseline Concentration ($\mu g/m^3$)*	*Baseline plus Foreign Research Reactor Spent Nuclear Fuel ($\mu g/m^3$)*	*Percent of Standard*
Criteria pollutants					
• Carbon Monoxide	1-hr	40,000	1,200	1,200	3.8
• Nitrogen Dioxide	Annual	100	14.1	14.1	14.1
• Lead	Quarterly	1.5	0.002	0.002	0.1
• Particulate Matter (PM_{10})	24-hr	150	112	112	75
	Annual	50	19	19	38
• Sulfur dioxide	3-hr	1,300	534	534	41.1
	24-hr	365	238	238	65.3
	Annual	80	4.2	4.2	5.3
Other pollutants mandated by Idaho					
• Total Suspended Particulates	24-hr	150	120[d]	120	80
	Annual	60	45	45	75
• Fluorides	Monthly	62,168	0	0	0
	Bimonthly	46,626	0	0	0
	Annual	31,084	0	0	0
Hazardous/toxic air pollutants (carcinogens)					
• Ammonia Hydroxide	8-hr	180	0.33	36	20
• Benzene	Annual	12	0.029	0.029	16
• Formaldehyde	Annual	770	0.012	0.012	16
• Hexone	8-hr	2,100	0	0	0
• Hydrofluoric Acid	8-hr	25	0	0	0
• Tributylphosphate	8-hr	25	0	0	0

[a] *Source: (DOE, 1995g).*

[b] *Listed concentrations are the maximum of those calculated at the Idaho National Engineering Laboratory site boundary, public access roads inside the Idaho National Engineering Laboratory site boundary, and the Craters of the Moon National Monument.*

[c] *To convert to $\mu g/ft^3$, multiply by 0.0283.*

[d] *The background concentration for the 24-hour standard is the same as the background for annual average concentration.*

Interim storage of foreign research reactor spent nuclear fuel in existing facilities would not affect the quality of water resources because it would be stored in contained storage pools or above-grade and below-grade dry storage containers isolated from the environment.

With respect to accident conditions, the Programmatic SNF&INEL Final EIS concluded that on the basis of a bounding accident scenario for high-level waste tank failure, accidental leakage would cause negligible impacts to water resources (DOE, 1995g).

F.4.2.2 New Facilities (Phase 2)

Analysis options 2B and 2C involve the use of new facilities. The environmental impacts analyzed relate to the construction and operation of these new facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities and energy; and waste management.

The impacts are presented in terms of storage technologies: dry storage in Sections F.4.2.2.1 and wet storage in Section F.4.2.2.2. Accident analysis, which is associated primarily with the storage technology rather than specific facilities, is presented in Section F.4.2.3.

F.4.2.2.1 Dry Storage

Analysis option 2B is associated with dry storage of foreign research reactor spent nuclear fuel in new facilities. This analysis option would require the construction of a new dry storage facility at the Idaho National Engineering Laboratory. The dry storage option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 of this EIS and earlier in this appendix. There are no environmental impact parameters that would discriminate between the two designs. For the purpose of this analysis, the impacts from the larger dry vault design are presented.

F.4.2.2.1.1 Land Use

A new dry storage facility could be located in one of several developed areas, including the ICPP. These areas, which have already been developed for industrial use, occupy about 4,560 ha (11,400 acres). Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land. This represents about 0.06 percent of the developed space at these areas. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at any of the areas would significantly impact land use patterns on the Idaho National Engineering Laboratory.

F.4.2.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents approximately 15.4 percent of the estimated FY 1995 total expenditures for the Idaho National Engineering Laboratory (600 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be $15.6 million for receipt and handling and $0.6 million for storage. These costs represent approximately 2.6 percent and 0.1 percent of FY 1995 total expenditures for the Idaho National Engineering Laboratory. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from direct and secondary construction employment on the region of influence would be negligible. In addition, when compared to the projected FY 1995 work force at the Idaho National Engineering Laboratory of approximately 11,600 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to 8 persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Idaho National Engineering Laboratory.

F.4.2.2.1.3 Cultural Resources

No direct impacts on any cultural resources would be expected from the construction and operation of a new dry storage facility. Surveys of previously disturbed areas at the Idaho National Engineering Laboratory found no eligible cultural resources. Native American treaty rights that would affect any future land use on the Idaho National Engineering Laboratory would not be impacted (DOE, 1995g). Because activities associated with spent nuclear fuel management would take place within existing facility areas currently engaged in similar activities, DOE does not expect any impacts to important Native American resources from alteration of the visual setting or noise associated with the construction or operation of any new facilities. DOE has developed plans to be in full compliance with cultural resource laws (DOE, 1995g).

F.4.2.2.1.4 Aesthetic and Scenic Resources

Construction and operation of a new dry storage facility would not adversely impact aesthetic or scenic resources. A new dry storage facility would not be visible from any onsite or offsite public access roads. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility (DOE, 1995g).

F.4.2.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Idaho National Engineering Laboratory that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils, and would necessitate the clearing and grading of 3.7 ha (9 acres). Site preparation, land shaping, and grading activities associated with construction would present a slight to moderate erosion hazard, but would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.2.2.1.6 Air Quality

Nonradiological Emissions: Potential impacts from construction activities at the Idaho National Engineering Laboratory would include fugitive dust from construction activities (e.g., clearing of land, grading, road preparation) and vehicle emissions from the heavy equipment utilized during the construction phase of the project. Construction of a new dry fuel storage facility would be located near the center of the Idaho National Engineering Laboratory. The construction of this facility would require disturbance of approximately 3.7 ha (9 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years). As outlined in Table F-42, the ambient air quality impacts associated with construction-related activities would be minimal and the Idaho National Engineering Laboratory compliance with Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-42 Estimated Maximum Concentrations of Criteria Pollutants at the Idaho National Engineering Laboratory Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[a]	*Baseline Concentration*	*Construction Activities*
Idaho National Engineering Laboratory Boundary (μg/m³)[b]				
• Particulate Matter (PM_{10})	24-hr	150	112	0.0274
	Annual	50	19	0.0014
• Carbon Monoxide	1-hr	40,000	1,200	2.42
	8-hr	10,000	340	0.97
• Sulfur Dioxide	3-hr	1,300	534	0.397
	24-hr	365	238	0.085
	Annual	80	4.2	0.004
• Nitrogen Dioxide	Annual	100	14.1	0.068
Public Roads Boundary (μg/m³)				
• Particulate Matter (PM_{10})	24-hr	150	112	0.0050
	Annual	50	19	0.0006
• Carbon Monoxide	1-hr	40,000	1,200	6.69
	8-hr	10,000	340	1.28
• Sulfur Dioxide	3-hr	1,300	534	0.727
	24-hr	365	238	0.117
• Nitrogen Dioxide	Annual	100	14.1	0.211
Craters of the Moon Boundary (μg/m³)				
• Particulate Matter (PM_{10})	24-hr	150	112	0.00037
	Annual	50	19	0.00003
• Carbon Monoxide	1-hr	40,000	1,200	0.61
	8-hr	10,000	340	0.08
• Sulfur Dioxide	3-hr	1,300	534	0.054
	24-hr	365	238	0.009
	Annual	80	4.2	0.0006
• Nitrogen Dioxide	Annual	100	14.1	0.009

[a] *Source: DOE, 1995g.*

[b] *To convert to μg/ft³, multiply by 0.0283.*

No nonradiological air emissions would be expected during operation of a new dry storage facility. Any emissions would be directly attributable to front-end wet storage activities only.

Radiological Emissions: No radiological emissions would be produced during construction of a new dry storage facility.

Based on fuel drying and storage operations conducted at the Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides, such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency

particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects, (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask, and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage, small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility for foreign research reactor spent nuclear fuel were calculated based on the methodology and assumptions described in Section F.6. The radiological consequences of air emissions from dry storage operation at the Idaho National Engineering Laboratory are discussed in Section F.4.2.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.2.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 6,500 million l (1,717 million gal) for the Idaho National Engineering Laboratory, these amounts represent approximately a 0.03 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Idaho National Engineering Laboratory.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Idaho National Engineering Laboratory. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Idaho National Engineering Laboratory could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Idaho National Engineering Laboratory would still be well within the design capacities of treatment systems at the Idaho National Engineering Laboratory. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.2.2.1.8 Ecology

Terrestrial Resources: DOE expects that construction impacts, which would include the loss of some wildlife habitat due to land clearing and facility development, would be greatest under the Regionalization and Centralization Alternatives under the Programmatic SNF&INEL Final EIS at the Idaho National Engineering Laboratory. Construction impacts from foreign research reactor spent nuclear fuel storage would not be significant because the construction activity would take place either within the boundaries of heavily developed areas or adjacent to those areas. However, construction activities could provide opportunities for the spread of exotic plant species, such as cheatgrass and Russian thistle (DOE, 1995g).

Wetlands: There would be no construction impacts to wetlands, which would be excluded from development, and impacts to threatened and endangered species would be unlikely given the location of previously-developed areas and the maximum size of the affected area of 3.7 ha (9 acres). Construction activities at the Idaho National Engineering Laboratory probably would not affect either of the endangered species found onsite (e.g., bald eagle and peregrine falcon). Both of these birds of prey are associated with riparian areas, wetlands, and larger bodies of water (e.g., reservoirs) and inhabit dry upland areas only temporarily when migrating. Disturbance to other sensitive (but not Federally-listed) species (e.g., the burrowing owl, northern goshawk, ferruginous hawk, Swainson's hawk, gyrfalcon, Townsend's western big-eared bat, and pygmy rabbit) would be possible but unlikely given the scale of the planned construction. Any impacts would be negligible and would last only as long as construction activities continue (DOE, 1995g).

Threatened and Endangered Species: Representative impacts from operations would include the disturbance and displacement of animals (such as the pronghorn antelope) caused by the movement and noise of personnel, equipment, and vehicles. Such impacts would be greatest under the Regionalization by Fuel Type and Geography, and Centralization Alternatives under the Programmatic SNF&INEL Final EIS at the Idaho National Engineering Laboratory, which would involve a generally higher level of operational activity; however, these impacts would be minor (DOE, 1995g). DOE has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the potential construction site of foreign research reactor spent nuclear fuel storage facilities at the Idaho National Engineering Laboratory, as required by the Endangered Species Act. Letters regarding consultation under the Endangered Species Act are included in Volume 2, Appendix B of the Programmatic SNF&INEL Final EIS (DOE, 1995g).

F.4.2.2.1.9 Noise

Noise generated onsite by construction or operation of a new dry storage facility should not adversely affect the public or the Idaho National Engineering Laboratory environment. Noise generated by construction would be site-specific and short lived. A limited number of new construction jobs would be generated, but the resulting temporary increase in worker and truck traffic is expected to be insignificant within the context of existing site traffic loads. Noise generated by operation would not significantly impact the environment because the facility would be located adjacent to previously developed, industrialized areas. Rail shipments of foreign research reactor spent nuclear fuel would be a small fraction of the rail traffic on the Blackfoot-to-Arco Branch of the Union Pacific System line that crosses the Idaho National Engineering Laboratory. There may be a slight increase in truck traffic to and from the potential storage site, but it is not expected to result in a perceptible increase in traffic noise or any change in community reaction to noise along the major access routes to the Idaho National Engineering Laboratory (DOE, 1995g).

F.4.2.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with the daily operations of the Idaho National Engineering Laboratory. Again, traffic congestion would not be a significant problem.

Traffic due to operations of a new dry storage facility would not increase site levels because the required workers would be drawn from the existing Idaho National Engineering Laboratory labor force.

F.4.2.2.1.11 Occupational and Public Health and Safety

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-43 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-43 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Idaho National Engineering Laboratory (New Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility[a]	0.00056	2.8×10^{-10}	0.0045	0.0000023
Storage at:				
• New Dry Storage Facility	0	0	0	0

[a] *The doses for this new dry storage facility are assumed to be equal to those for IFSF/CPP-749.*

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 2B involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into the existing dry and wet storage facilities (IFSF/CPP-749 and FAST) during Phase 1, the preparation of 161 transportation casks for shipment to a dry storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel at the new dry storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the

transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. Collective doses were calculated for both dry storage designs, the vault and the dry cask. The assumptions and methodology used to calculate the doses are described in Section F.5 of this appendix.

Table F-44 presents the doses that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Idaho National Engineering Laboratory. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-44 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory (New Dry Storage)

	Worker Population Dose (person-rem)		*Worker Population Risk (LCF)*	
	New Dry Storage Cask	*New Dry Storage Vault*	*New Dry Storage Cask*	*New Dry Storage Vault*
Phases 1 and 2[a]	424	370	0.17	0.15
Phases 1 and 2[b]	416	363	0.17	0.15

[a] *Phase 1 at IFSF/CPP-749*

[b] *Phase 1 at FAST*

F.4.2.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Idaho National Engineering Laboratory would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-45. These requirements represent a small percent of current requirements for the Idaho National Engineering Laboratory. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Idaho National Engineering Laboratory is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-45 Annual Utility and Energy Requirements for New Dry Storage at the Idaho National Engineering Laboratory

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	208,000	800 - 1,000	0.48 percent
Fuel (l/yr)	11,123,400	0	0 percent
Water (l/yr)	6,500,000,000	1,590,000[a] 400,000[b]	0.025 percent[a] 0.006 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.2.2.1.13 Waste Management

Construction of a new dry storage facility at the Idaho National Engineering Laboratory would generate 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-46. These quantities represent a very small percent increase above current levels at the Idaho National Engineering Laboratory. Existing waste management storage and disposal activities at the Idaho National Engineering Laboratory could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Idaho National Engineering Laboratory waste management capacities would be minimal.

Table F-46 Annual Waste Generated for New Dry Storage at the Idaho National Engineering Laboratory

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	750	none	0 percent
Transuranic (m^3/yr)	712	none	0 percent
Solid Low-Level (m^3/yr)	4,795	22[a] 1[b]	0.5 percent[a] 0.02 percent[b]
Wastewater (l/yr)	540,000,000	1,590,000[a] 400,000[b]	0.29 percent[a] 0.074 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.2.2.2 Wet Storage

Analysis option 2C involves long-term wet storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory. This analysis option would require the construction of a new wet storage facility at the site (Implementation Alternative 5 of Management Alternative 1).

F.4.2.2.2.1 Land Use

A new wet storage facility could be located in one of several developed areas, including the ICPP. These areas, which have already been developed for industrial use, occupy about 4,560 ha (11,400 acres). Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. This represents about 0.06 percent of the developed space at these areas. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at any of the areas would significantly impact land use patterns on the Idaho National Engineering Laboratory.

F.4.2.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 18.7 percent of the estimated FY 1995 total expenditures for the Idaho National Engineering Laboratory (600 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage.

These costs represent about 3.8 percent and 0.6 percent of FY 1995 total expenditures for the Idaho National Engineering Laboratory. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Idaho National Engineering Laboratory of approximately 11,600 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Idaho National Engineering Laboratory.

F.4.2.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.2.2.1.3).

F.4.2.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.2.2.1.4).

F.4.2.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.2.2.1.5).

F.4.2.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 3 ha (7 acres) of land. In comparison, approximately 4 ha (10 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction, as presented in Section F.4.2.2.1.6.

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3-micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs.

For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b).

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission level for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE Order 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

No radiological emissions would be produced during construction of a new wet storage facility.

F.4.2.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (0.72 million gal), respectively. With an annual average water usage of approximately 6,500 million l (1,717 million gal) for the Idaho National Engineering Laboratory, these amounts represent an increase of about 0.03 percent and less than 0.04 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Idaho National Engineering Laboratory.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Idaho National Engineering Laboratory. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Idaho National Engineering Laboratory could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Idaho National Engineering Laboratory would still be well within the design capacities of treatment systems at the Idaho National Engineering Laboratory. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.2.2.2.8 Ecology

Impacts to ecology would be the same as for new dry storage (Section F.4.2.2.1.8).

F.4.2.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.2.2.1.9).

F.4.2.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.2.2.1.10).

F.4.2.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities and storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-47 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory for wet storage. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-47 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at • New Wet Storage Facility	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at: • New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 2C involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into the existing facilities (IFSF/CPP-749 and FAST) during Phase 1, the preparation of 161 transportation casks for shipment to a wet storage facility at the end of Phase 1, and the receipt of 193 shipments directly from the ports into the new wet storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-48 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Idaho National Engineering Laboratory. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This

regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-48 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

Facility	*Worker Population Dose (person-rem)*	*Worker Population Risk (LCF)*
Phase 1: IFSF/CPP-749	257	0.10
Phase 1 and Phase 2	367	0.15
Phase 1: FAST	250	0.10
Phase 1 and Phase 2	360	0.14

F.4.2.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Idaho National Engineering Laboratory would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-49. These requirements represent a small percent of current requirements for the Idaho National Engineering Laboratory. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity at the Idaho National Engineering Laboratory is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-49 Annual Utility and Energy Requirements for New Wet Storage at the Idaho National Engineering Laboratory (Implementation Alternative 5 to Management Alternative 1)

Commodity	*Baseline Site Usage*	*Wet Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	208,000	1,000 - 1,500	0.72 percent
Fuel (l/yr)	11,123,400	0	0 percent
Water (l/yr)	6,500,000,000	2,700,000[a] 1,500,000[b]	0.04 percent 0.02 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.2.2.2.13 Waste Management

Construction of a new wet storage facility at the Idaho National Engineering Laboratory would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-50. These quantities represent a very small percentage increase above current levels at the Idaho National Engineering Laboratory. Existing waste management storage and disposal activities at the Idaho National Engineering Laboratory could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on existing Idaho National Engineering Laboratory waste management capacities would be minimal.

Table F-50 Annual Waste Generated for New Wet Storage at the Idaho National Engineering Laboratory (Implementation Alternative 5 to Management Alternative 1)

Waste Form	*Baseline Site Generation*	*Wet Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	750	none	0 percent
Transuranic (m^3/yr)	712	none	0 percent
Solid Low-Level (m^3/yr)	4,795	16[a] 1[b]	0.33 percent 0.02 percent
Wastewater (l/yr)	540,000,000	1,590,000[a] 400,000[b]	0.3 percent 0.07 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.2.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Idaho National Engineering Laboratory is presented here based on the methodology presented in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated. Information concerning radiation doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-51 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-51 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Dry Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	1.3	0.67	15	28
• Dropped Fuel Cask	0.0001	0.074	0.0033	0.83	0.12
• Aircraft Crash w\Fire	1×10^{-6}	180	2.9	2,000	120
Wet Storage Accidents[b]					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	0.0036	0.43	0.14
• Accidental Criticality	0.0031	28	30	140	1800
• Aircraft Crash	1×10^{-6}	22	9.8	250	400

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *New Wet Storage and FAST facility*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Idaho National Engineering Laboratory. These annual risks are multiplied by the maximum duration of this implementation alternative to obtain conservative estimates of risks at the Idaho National Engineering Laboratory presented in Table F-52 .

Table F-52 Annual Risks of Accidents at Idaho National Engineering Laboratory

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	1.1×10^{-7}	5.5×10^{-8}	0.0012	0.0000018
• Dropped Fuel Cask	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	4.8×10^{-12}
• Aircraft Crash w\Fire	9.0×10^{-11}	1.5×10^{-12}	0.0000010	4.8×10^{-11}
Wet Storage Accidents[b]				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *New Wet Storage and FAST Facility*

Table F-53 presents the frequency and consequences of the accidents analyzed for Idaho National Engineering Laboratory for new wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Idaho National Engineering Laboratory. These annual risks are multiplied by the maximum duration of implementation alternative at each site to obtain conservative estimates of risks at the Idaho National Engineering Laboratory. Table F-54 presents the risk estimates from this implementation alternative.

Table F-53 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Wet Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	0.0036	0.43	0.14
• Accidental Criticality	0.0031	28	30	140	1800
• Aircraft Crash	1×10^{-6}	22	9.8	250	400

[a] *New Wet Storage Facility*

Table F-54 Annual Risks of Accidents at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Wet Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *New Wet Storage Facility*

F.4.2.3.1 Secondary Impact of Radiological Accidents at the Idaho National Engineering Laboratory

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.2.3. Radiological accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Idaho National Engineering Laboratory, the radiological accident with the highest MEI dose is the aircraft crash into a dry storage facility with fire (Table F-51). For this accident, the MEI dose would be 180 mrem. For the air immersion and ground surface pathways only, the dose would be 3.1 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage are expected at the Idaho National Engineering Laboratory.

F.4.2.4 Cumulative Impacts at the Idaho National Engineering Laboratory

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions and current activities at the Idaho National Engineering Laboratory. The contemplated DOE actions are the proposed construction and operation of an accelerator facility for tritium production (along with associated support facilities) (DOE, 1995d), the management of DOE-owned spent nuclear fuel discussed in Appendix B of the Programmatic SNF&INEL Final EIS (DOE, 1995g), and the storage and disposition of weapons-usable fissile materials at the Idaho National Engineering Laboratory site.

Tables F-55 and F-55A summarize the cumulative impacts for land use, socioeconomics, nonradiological air quality, occupational and public health and safety, energy and water consumption, and waste generation. As shown in the tables, the contribution of foreign research reactor spent nuclear fuel management to the cumulative impacts at the Idaho National Engineering Laboratory would be minimal.

F.4.2.5 Unavoidable Adverse Environmental Impacts

The construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would result in some adverse impacts to the environment. Changes in designs and other methods of mitigation could eliminate, avoid, or reduce most of these to minimal levels. The following paragraphs identify adverse impacts that mitigation could not reduce to minimal levels or avoid altogether.

The generation of some fugitive dust during construction would be unavoidable, but would be controlled by water and dust suppressants. Similarly, construction activities would result in some minor, yet unavoidable, noise impacts from heavy equipment, generators, and vehicles.

Table F-55 Cumulative Impacts at the Idaho National Engineering Laboratory

Environmental Impact Parameter	*FRR SNF Contribution*	*Current Activities*[a]	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	9	11,400[b]	604	12,013[b]
Socioeconomics (persons)	190[c]/30[d]	(e)	1980[c]/1080[d]	2,170[c]/1,110[d]
Air Quality (nonradiological)	See Table F-55A	See Table F-55A	See Table F-55A	See Table F-55A
Occupational and Public Health and Safety				
• MEI Dose (rem/yr)	5.6×10^{-7}	0.000056	0.0000057	0.000062
LCF (per year)	2.8×10^{-10}	2.8×10^{-8}	2.8×10^{-9}	3.1×10^{-8}
• Population Dose (person-rem/yr)	0.0045	0.34	32	32.3
LCF (per year)	2.25×10^{-6}	0.00017	0.016	0.016
• Worker Collective dose (person-rem/yr)	10[f]	30	344	384
LCF (per year)	0.004	0.012	0.137	0.154
Energy and Water Consumption				
• Electricity (MW-hr/yr)	1,000	208,000	3,897,000[g]	4,106,000
• Fuel (million l/yr)	0	11.1	1.35	12.45
• Coal (tons/yr)	0	12,500	13,660	26,160
• Water (million l/yr)	2.2	6,500	1,314	7,816
Waste Generation				
• High-Level (m^3/yr)	0	750	160	910
• Low-Level (m^3/yr)	22	4,795	2,800	7,617
• Transuranic (m^3/yr)	0	712	46	758
• Mixed (m^3/yr)	0	243	8	251

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a tritium accelerator facility, and the disposition of weapons-usable fissile materials*

[b] *Two percent of the total Idaho National Engineering Laboratory site area of 570,000 acres*

[c] *Increase over baseline during construction activities*

[d] *Increase over baseline during operation activities*

[e] *Baseline working force is approximately 11,600 persons*

[f] *The dose is due to the handling of FRR SNF during receipt and transfer, averaged over 40 years*

[g] *Major portion is the requirement for electricity by the tritium production accelerator facility (3,740,000 MW-hr/yr)*

The maximum loss of habitat would involve the conversion of approximately 4 ha (10 acres) of previously disturbed habitat that is of low quality and limited use to wildlife.

The amount of radioactivity that incident-free operation of the spent nuclear fuel facilities would release is a small fraction of the cumulative operational releases at the Idaho National Engineering Laboratory and would be well below applicable regulatory standards.

F.4.2.6 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of facilities for foreign research reactor spent nuclear fuel facilities at the Idaho National Engineering Laboratory would consume irretrievable amounts of electrical energy,

Table F-55A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Idaho National Engineering Laboratory Boundary[a]

Pollutant	*Averaging Time*	*Regulatory Standard (μg/m³)*	*Cumulative Concentration (μg/m³)[b]*	
Carbon Monoxide	1-hour	40,000	1,245	(3.1%)
	8-hour	10,000	354	(3.5%)
Nitrogen Oxides	Annual	100	15	(15%)
Surfur Dioxide	3-hour	1,300	660	(51%)
	24-hour	365	267	(73%)
	Annual	80	7.5	(9.3%)
Particulate Matter (PM_{10})	24-hour	150	82	(55%)
	Annual	50	5	(10%)

[a] *Concentrations represent: foreign research reactor spent nuclear fuel management, other DOE-owned spent fuel management, construction and operation of a tritium supply facility and recycling activities, storage and disposition of weapons-usable fissile material activities, and current activities*

[b] *Numbers in parentheses indicate the percentage of the regulatory standard*

fuel, concrete, sand, and gravel. Other resources used in construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery.

F.4.2.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Idaho National Engineering Laboratory could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Idaho National Engineering Laboratory would be minimal in most environmental media.

Pollution Prevention: DOE is committed to comply with Executive Order 12856, "Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements;" Executive Order 12873, "Federal Acquisition, Recycling and Waste Prevention;" and applicable DOE orders and guidance documents in planning and implementing pollution prevention at the Idaho National Engineering Laboratory. The DOE views source reduction as the first priority in its pollution prevention program, followed by an increased emphasis on recycling. Waste treatment and disposal are considered only when prevention or recycling is not possible or practical (DOE, 1995g).

Cultural Resources: The lack of detailed specifications associated with the potential construction at the Idaho National Engineering Laboratory under the various storage options prevents the identification of specific project impacts and mitigation measures for particular structures and facilities. Basic compliance under cultural resource law involves five steps that would be essentially the same under all alternatives. These steps are: (a) identification and evaluation of resources in danger of impact, (b) assessment of effects to these resources in consultation with the State Historic Preservation Office and representatives of the Shoshone-Bannock Tribes, (c) development of plans and documents to minimize any adverse effects, (d) consultation with the Advisory Council on Historic Preservation and Tribal representatives as to the appropriateness of mitigation measures, and (e) implementation of mitigation measures. Therefore, if a cultural resource survey has not been performed in an area planned for ground disturbance under one of the storage options, consultation would be initiated with the Idaho State Historic Preservation Office, and

the survey would be conducted prior to any disturbance. If cultural resources were discovered, they would be evaluated according to National Register criteria. Wherever possible, important resources would be left undisturbed. If the impacts are determined to be adverse and it is not feasible to leave the resource undisturbed, then measures would be initiated to reduce impacts. All mitigation plans would be developed in consultation with the State Historic Preservation Office and the Advisory Council on Historic Preservation and would conform to appropriate standards and guidelines established for historic preservation activities by the Secretary of the Interior (DOE, 1995g).

Some actions may affect areas of religious, cultural, or historic value to Native Americans. DOE has implemented a Working Agreement to ensure communication with the Shoshone-Bannock Tribes, especially relating to the treatment of archaeological sites during excavation, as mandated by the Archaeological Resources Protection Act; the protection of human remains, as required under the Native American Graves Protection and Repatriation Act; and the free exercise of religion as protected by the American Indian Religious Freedom Act. In keeping with DOE Native American policy, DOE Order 1230.2, and procedures to be defined in the Final Cultural Resources Management Plan, DOE would conduct Native American consultation during the planning and implementation of the policy. Procedures for dealing with the inadvertent discovery of human remains would be consistent with the Native American Graves Protection and Repatriation Act. If human remains were discovered, DOE would notify all Tribes that have expressed an interest in the repatriation of graves as required under Native American Graves Protection and Repatriation Act, including the Shoshone-Bannock, Shoshone, Painte, and the Northwestern band of the Shoshone Nation. These Tribes would then have an opportunity to claim the remains and associated artifacts in accordance with the requirements of Native American Graves Protection and Repatriation Act (DOE, 1995g).

Traffic and Transportation: All onsite shipments of foreign research reactor spent nuclear fuel would be in compliance with ID Directive 5480.3, "Hazardous Materials Packaging and Transportation Safety Requirements." These requirements provide assurance that, under normal conditions, the Idaho National Engineering Laboratory would meet "as low as reasonably achievable" conditions, credible accident situations (those with probability of occurrence greater than 1×10^{-7} per year) would not result in a loss of shielding or containment or a criticality, and an unintentional release of radioactive material would generate a timely response (DOE, 1995g).

Accidents: The DOE would implement the Idaho National Engineering Laboratory emergency response programs, as appropriate, following the occurrence of an accident to prevent or mitigate consequences. These emergency response programs, implemented in accordance with 5500-DOE series orders, typically involve emergency planning, emergency preparedness, and emergency response actions. Participating government agencies with plans that are interrelated with the Idaho National Engineering Laboratory Emergency Plan for Action include: the State of Idaho, Bingham County, Bonneville County, Butte County, Clark County, Jefferson County, the Bureau of Indian Affairs, and Fort Hall Indian Reservation. When an emergency condition exists at a facility, the Emergency Action Director is responsible for recognition, classification, notification, and protective action recommendations. Each emergency response plan utilizes resources specifically dedicated to assist a facility in emergency management. These resources include, but are not limited, to the following (DOE, 1994h):

- Idaho National Engineering Laboratory Warning Communications Center,
- Idaho National Engineering Laboratory Fire Department,
- Facility Emergency Command Centers,

- DOE Emergency Operations Centers,
- County and State Emergency Command Centers,
- medical, health physics, and industrial hygiene specialists,
- protective clothing and equipment (respirators, breathing air supplies, etc.), and
- periodic training exercises and drills within and between the organizations involved in implementing the response plans.

F.4.3 Hanford Site

If the Hanford Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Hanford Site to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2) the Hanford Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory, and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Hanford Site until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Hanford Site under Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly in Phase 2, the Hanford Site could receive TRIGA foreign research reactor spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, Western foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As a Phase 2 site, the Hanford Site would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility constructed on the 200 Area Plateau or the FMEF, which is a partially completed, large, hot cell facility. The new dry storage facility is described in Section 2.6.5.1.1. Description of the FMEF is provided in Appendix F, Section F.3.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Hanford Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

3A. The spent nuclear fuel that was managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Hanford Site where it would be managed at a new dry storage facility constructed either at the 200 Area Plateau or at the FMEF. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel that would be managed in the dry storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (22,700 elements). If the Hanford Site receives TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory or only Western spent nuclear fuel, the dry

storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Hanford Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Hanford Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site and manage it in facilities sized accordingly. The impacts from the management of this lesser amount of spent nuclear fuel would be bounded by analysis option 3A (above).

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Hanford Site would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Hanford Site would be bounded by analysis option 3A (above).

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Hanford Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 3A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years, and therefore the amount of spent nuclear fuel available for acceptance would also be decreased. In this case, the Hanford Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Hanford Site would be bounded by analysis option 3A (above).

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and stored would remain constant. The impacts would be the same as in option 3A (above).

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States as the foreign research reactor operators would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent nuclear fuel in this case cannot be quantified; however, the upper limit, as considered under analysis option 3A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent

nuclear fuel would be taken. The choices do not affect the management impacts at the Hanford Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Hanford Site for Phase 2 until ultimate disposition. For this implementation alternative, an analysis option 3B, which is similar to 3A, is considered as follows:

3B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Hanford Site where it would be managed at a new wet storage facility constructed at either the 200 Area Plateau or the WNP-4 Spray Pond facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements). If the Hanford Site receives only TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory, or only western fuel, the dry storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Hanford Site would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Hanford Site is not considered.

F.4.3.1 Existing Facilities

Existing facilities at the Hanford Site include the FMEF and the WNP-4 Spray Cooling Pond for dry and wet storage, respectively, of foreign research reactor spent nuclear fuel. For this analysis, existing facilities at the Hanford Site were considered essentially as new because of the significant modifications that would be required to use them for foreign research reactor spent nuclear fuel storage. Handling and transfer operations at the FMEF and the WNP-4 Spray Cooling Pond would be used to support new dry and wet storage facilities, respectively. The evaluation of potential environmental impacts is presented in Section F.4.3.2 and reflects the foreign research reactor spent nuclear fuel storage options described in Section F.4.3.

F.4.3.2 New Facilities (Phase 2)

Analysis options 3A and 3B involve the use of new or major additions to existing facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.3.2.1 Dry Storage

Dry storage is associated with analysis option 3A, which would require the construction of a new dry storage facility near the 200 Area Plateau or at the FMEF (FMEF currently has handling and transfer, but

not adequate storage capabilities). The dry storage option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 and Appendix F, Section F.3. There are no environmental impact parameters that would discriminate between the two designs. For the purpose of this analysis the impacts from the larger dry vault design are presented.

F.4.3.2.1.1 Land Use

A new dry storage facility would be located in either the 200 Area Plateau or at the FMEF in the 400 Area. These areas have been generally developed for industrial use. Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land at either area. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at either area would significantly impact land use patterns on the Hanford Site.

F.4.3.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents approximately 7.2 percent of the estimated FY 1995 total expenditures for the Hanford Site (1,288 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be $15.6 million for receipt and handling and $0.6 million for storage. These costs represent approximately 1.2 percent and 0.05 percent of FY 1995 total expenditures for the Hanford Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from direct and secondary construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Hanford Site of approximately 18,500 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to eight persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Hanford Site.

F.4.3.2.1.3 Cultural Resources

No direct impacts on any cultural resources in the 200 Area Plateau would be expected from construction or operation of the new dry storage facility. This site has been surveyed for cultural resources, and no prehistoric or historic archaeological properties were found. No indirect impacts would be anticipated because no known archaeological sites are present within approximately 4 km (2.5 mi) of the 200 Area Plateau. Because the site is in an industrialized area, construction would not alter the historic significance or association with the Manhattan Project and/or Cold War facilities located nearby.

No direct or indirect impacts are expected to any cultural resources of significance to the Yakama Indian Nation, the Confederated Tribes of the Umatilla Indian Reservation, or the Wanapum Band. This is based

on the location of the 200 Area Plateau relative to sacred and culturally important areas which have been identified through ethno-historical research and interviews with elders of bands that formerly used the Hanford Site (DOE, 1995g).

Modification of FMEF for dry storage would be inside the fence of the 400 Area. No cultural resources are known to exist within that area. Because of its location, no cultural resources on the Hanford Site would be disturbed by construction.

F.4.3.2.1.4 Aesthetic and Scenic Resources

Any changes caused by construction and operation of either dry storage facility would be consistent with the existing overall visual environment of the Hanford Site. Topographic features would obstruct both candidate storage sites from the view of populated areas. Although the new dry storage facility could be seen from the farmland bluffs that overlook the Columbia River to the east, these lands are on private property that is not readily accessible to the public. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility (DOE, 1995g).

F.4.3.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Hanford Site that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils and would necessitate the clearing and grading of 3.7 ha (9 acres). Site preparation, land shaping, and grading activities associated with construction would present a slight to moderate erosion hazard, but would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.3.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction include generation of fugitive dust (particulate matter) and smoke from earth moving and clearing operations and emissions from construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Emissions of sulfur dioxide and nitrogen dioxide would result entirely from diesel exhaust. For this analysis, all vehicular emissions were conservatively assumed to occur within 1 year during 200 ten-hour work days. As shown in Table F-56, air quality impacts associated with construction-related activities would be minimal, and compliance with Federal and State ambient air quality standards would not be

adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-56 Estimated Maximum Concentrations of Criteria Pollutants at the Hanford Site Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[a]	*Baseline Concentration*[a]	*Construction Activities*
Hanford Site Boundary ($\mu g/m^3$)				
• Total Suspended Particulate (TSP)	Annual	75	56	0.4
• Particulate Matter (PM_{10})	24-hr	150	81	14
• Particulate Matter (PM_{10})	Annual	50	27	0.4
Workplace (ppmv)				
• Sulfur Dioxide	Annual	52	0.5	0.4
• Nitrogen Dioxide	Annual	100	6,500	200

[a] *Source: DOE, 1995g*

Nonradiological emissions would not be expected during operation of a new dry storage facility for foreign research reactor spent nuclear fuel.

Radiological Emissions: No radiological emissions from construction of a new dry storage facility for foreign research reactor spent nuclear fuel would be expected. Based on fuel drying and storage operations conducted at the Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides, such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides that are released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask; and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility for foreign research reactor spent nuclear fuel were calculated based on the methodology and assumptions described in Appendix F, Section F.6. The radiological consequences of air emissions from the operation of a new dry storage facility at the Hanford Site are discussed in Section F.4.3.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.

F.4.3.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 15,000 million l (3,960 million gal) for the Hanford Site, these amounts represent no more than a 0.04 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Hanford Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Hanford Site. The impact on water quality during operations would also be small. Existing water treatment facilities at the Hanford Site could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Hanford Site would still be well within the design capacities of treatment systems at the Hanford Site. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.3.2.1.8 Ecology

Terrestrial Resources: Vegetation within construction areas would be destroyed during land-clearing activities. Plant species that are dominant on the 200 Area Plateau include: big sagebrush, cheatgrass, and Sanberg's bluegrass. Total area destroyed would amount to about less than one percent of this community on the Hanford Site. Although the plant communities to be disturbed are well-represented on Hanford Site, they are relatively uncommon regionally because of the widespread conversion of shrub-steppe habitats to agriculture. Disturbed areas are generally recolonized by cheatgrass, a nonnative species, at the expense of native plants. Mitigation of these impacts would include minimizing the area of disturbance and revegetating with native species, including shrubs, and establishing a 3:1 acreage replacement habitat in concert with a habitat enhancement plan presently being developed for Hanford Site in general. Adverse impacts to vegetation on Hanford Site would be limited to the project area and vicinity, and would not affect the viability of any plant populations on the Hanford Site (Bergsman et al., 1994).

Construction of the new dry storage facility would have some adverse affect on animal populations. Less mobile animals, such as invertebrates, reptiles, and small animals within the project area would be destroyed during land-clearing activities. Larger mammals and birds in construction and adjacent areas would be disturbed by construction activities and would move to adjacent suitable habitat, and these individual animals might not survive and reproduce. Project facilities would displace about 3.7 ha (9 acres) of animal habitat for the life of the dry storage facility. Revegetated areas (e.g., construction laydown areas and buried pipeline routes) would be reinvaded by animal species from surrounding undisturbed habitats. The adverse impacts of construction are expected to be limited to the project area and vicinity and should not affect the viability of populations on the Hanford Site.

Very small quantities of radionuclides would be released to the atmosphere during dry storage facility operations. No organisms studied to date are reported to be more sensitive than man to radiation. Therefore, the effects of these releases on terrestrial organisms are expected to be minor (Bergsman et al., 1994).

Any impacts to the vegetation and animal communities would be mitigated by minimizing the amount of land disturbed during construction, employing soil erosion control measures during construction activities, and revegetating disturbed areas with native species. These mitigation measures would limit the amount of direct and indirect disturbance to the construction area and surrounding habitats and would speed the recovery process for disturbed lands (Bergsman et al., 1994).

Operational impacts on terrestrial biotic resources would include exposure of plants and animals to small amounts of radionuclides released during operation of the new dry storage facility. The levels of radionuclide exposure would be below those levels that produce adverse effects (Bergsman et al., 1994).

Wetlands: There are no wetlands on or near either candidate storage site (Bergsman et al., 1994).

Threatened and Endangered Species: Construction and operation of the new dry storage facility would remove 3.7 ha (9 acres) of relatively pristine big sagebrush/cheatgrass/Sanberg's bluegrass habitat. This sagebrush habitat is considered priority habitat by the State of Washington because of its relative scarcity in the State and its use as nesting/breeding habitat by loggerhead shrikes, sage sparrows, sage thrashers, burrowing owls, pygmy rabbits, and sagebrush voles (Bergsman et al., 1994).

Loggerhead shrikes, listed as a Federal candidate (Category 2) and State candidate species, forage on the proposed spent nuclear fuel site and are relatively common on the Hanford Site. This species is sagebrush-dependent, as it is known to select primarily tall big sagebrush as nest sites. Construction of the new dry storage facility would remove big sagebrush habitat which would preclude loggerhead shrikes from nesting there. Foreign research reactor spent nuclear fuel site development would also be expected to reduce the value of the site as foraging habitat for shrikes known to nest in adjacent areas (Bergsman et al., 1994).

Sage sparrows and sage thrashers, both State candidate species, occur in mature sagebrush/bunchgrass habitat at the Hanford Site. The sage sparrow was observed on the proposed site in a survey during spring 1994. These species are known to nest primarily in sagebrush. Construction of the new dry storage facility would preclude both of these species nesting there and reduce the site's suitability as foraging habitat for these species (Bergsman et al., 1994).

Dry storage facility construction is not expected to substantially decrease Hanford Site population of loggerhead shrikes, sage sparrows, or sage thrashers because similar sagebrush habitat is still relatively common on the Hanford Site. However, the cumulative effects of constructing the new dry storage facility, in addition to future developments that further reduce sagebrush habitat (causing further fragmentation of nesting habitat), could negatively affect the long-term viability of populations of these species on the Hanford Site (Bergsman et al., 1994).

Burrowing owls, a State candidate species, are relatively common on Hanford Site and nest in abandoned ground squirrel burrows on the 200 Area Plateau. Construction would remove sagebrush and disturb soil, displacing ground squirrels and thus reducing the suitability of the area for nesting by burrowing owls, and would also displace small mammals, which constitute a portion of the prey base for this species. Dry storage facility construction would not be expected to negatively impact the viability of the population of burrowing owls on the Hanford Site, as their use of ground squirrel burrows as nests is not limited to burrows in big sagebrush habitat (Bergsman et al., 1994).

Pygmy rabbits, a Federal candidate (Category 2) and State-listed threatened species, are known to utilize tall clumps of big sagebrush habitat throughout most of their range. However, this species has not recently been observed on the Hanford Site. Construction of the new dry storage facility would therefore reduce the potential for this species' occurrence by removing habitat suitable for its use (Bergsman et al., 1994).

Sagebrush voles, a State minor species, are common on Hanford Site and select burrow sites near sagebrush; however, this species is common only at higher elevations around the Hanford Site. Construction of the new dry storage facility would remove sagebrush habitat, precluding sagebrush voles from utilizing the site. However, construction would not affect the overall viability of sagebrush vole populations on Hanford Site because the majority of the population is found on the Fitzner/Eberhardt Arid Lands Ecology Preserve (Bergsman et al., 1994).

The closest known nests of ferruginous hawks, a Federal candidate (Category 2) and State threatened species, and Swainson's hawk, a State candidate, are 8.5 km (5 mi) and 6.2 km (3.7 mi), respectively, from the 200 Area Plateau. The potential site comprises a portion of the foraging range of these hawks. Construction of the new dry storage facility is not expected to disrupt the nesting activities of these species. However, construction would displace small mammal populations and thus reduce the prey for these birds. The cumulative effects of constructing the new dry storage facility, in addition to future reductions in sagebrush habitat (causing further fragmentation of foraging habitat), could negatively affect the long-term viability of populations of these two species on the Hanford Site (Bergsman et al., 1994).

Piper's daisy, listed as a State sensitive species, is relatively uncommon but widely distributed across the Hanford Site. Piper's daisy occurs in gravelly soils on the 200 Area Plateau. If construction of the new dry storage facility includes disturbing soils in the gravel pit, Piper's daisy would be eliminated in that area. However, because of the species' wide distribution, construction would not be expected to negatively affect the viability of this species on the Hanford Site (Bergsman et al., 1994).

DOE has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the proposed construction sites of foreign research reactor spent nuclear fuel storage facilities at the Hanford Site, as required by the Endangered Species Act.

The modification of FMEF for dry storage would take place within the fenced 400 Area. This area has already been disturbed and no further ecological impacts would be expected.

F.4.3.2.1.9 Noise

Noise generated onsite by construction and operation of a new dry storage facility should not adversely affect the public or the Hanford Site environment. Based on a noise impact analysis for locating a new production reactor at the Hanford Site, ambient noise levels would not exceed the limits set by Washington State or the Environmental Protection Agency. The analysis indicated that any increased traffic along the major roadways from construction and operation of the new production reactor would result in little or no increase in the annoyance level experienced by communities or individuals. As a result, no significant noise impacts from activities associated with the new dry storage facility construction and operation are expected at receptor locations outside the Hanford Site boundary or at residences along the major highways leading to either candidate storage site.

F.4.3.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with the daily operations of the Hanford Site. Again, traffic congestion would not be a significant problem.

Traffic congestion, although moderate at shift changes, would not be noticeably worse due to this level of construction effort.

F.4.3.2.1.11 Occupational and Public Health and Safety

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Hanford Site would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-57 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Hanford Site. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-57 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Hanford Site (Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• FMEF (dry storage)	0.00020	1.0×10^{-10}	0.011	0.0000055
• New Dry Storage Facility	0.00025	1.3×10^{-10}	0.015	0.0000075
Storage at:				
• FMEF (dry storage)	0	0	0	0
• New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 3A involves the receipt and unloading of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-58 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Hanford Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the

Table F-58 Handling-Related Impacts to Workers at the Hanford Site (New Dry Storage)

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	FMEF/New Dry Storage	*FMEF/New Dry Storage*
Phase 2	266/113[a]	0.11/0.05[a]

[a] *The two numbers represent the cask/vault designs respectively.*

administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

F.4.3.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Hanford Site would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-59. These requirements represent a small percent of current requirements for the Hanford Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Hanford Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-59 Annual Utility and Energy Requirements for New Dry Storage at the Hanford Site

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	340,000	800 - 1,000	0.3 percent
Fuel (l/yr)	83,000,000	0	0 percent
Water (l/yr)	15,000,000,000	1,590,000[a] 400,000[b]	0.01 percent[a] 0.003 percent[b]

[a] *During receipt and handling.*

[b] *During storage.*

F.4.3.2.1.13 Waste Management

Construction of a new dry storage facility at the Hanford Site would generate 1,800 m^3 (2,340 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-60. These quantities, represent a very small percent increase above current levels at the Hanford Site. Existing waste management storage and disposal activities at Hanford Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Hanford Site waste management capacities would be minimal.

Table F-60 Annual Waste Generated for New Dry Storage at the Hanford Site

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	240	none	0 percent
Transuranic (m^3/yr)	170	none	0 percent
Solid Low-Level (m^3/yr)	20,000	22[a] 1[b]	0.11 percent[a] 0.005 percent[b]
Wastewater (l/yr)	210,000,000	1,590,000[a] 400,000[b]	0.75 percent[a] 0.2 percent[b]

[a] *During receipt and handling.*

[b] *During storage.*

F.4.3.2.2 Wet Storage

Analysis option 3B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Hanford Site. This storage option would require the construction of a new wet storage facility.

F.4.3.2.2.1 Land Use

A new wet storage facility would be located on the 200 Area Plateau or in conjunction with the WNP-4 Spray Cooling Pond. These areas have already been developed for industrial use. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land at either area. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at either area would significantly impact land use patterns on the Hanford Site.

F.4.3.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 8.7 percent of the estimated FY 1995 total expenditures for the Hanford Site (1,288 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 1.8 percent and 0.3 percent of FY 1995 total expenditures for the Hanford Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at Hanford Site of approximately 18,500 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Hanford Site.

F.4.3.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.3.2.1.3).

The potential for impacting cultural resources would be even less for the WNP-4 Spray Pond because the structures are all essentially in place. Thus, there would be no opportunity for discovery of cultural resources during construction.

F.4.3.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.3.2.1.4).

F.4.3.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.3.2.1.5).

F.4.3.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of 2.8 ha (7 acres) of land. In comparison, 3.7 ha (9 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction (Section F.4.3.2.1.6).

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological-Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operation in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs as an indicator nuclide. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE 5480.1B (DOE, 1989b) for both onsite and offsite concentrations.

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading and storage are provided in Section F.6.6.1.

No radiological emissions would be produced during construction of a new wet storage facility.

F.4.3.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (0.72 million gal), respectively. With an annual average water usage of approximately 15,000 million l (3,960 million gal) for the Hanford Site, these amounts represent an increase of about 0.02 percent and less than 0.005 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Hanford Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Hanford Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Hanford Site could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Hanford Site would still be well within the design capacities of treatment systems at the Hanford Site. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.3.2.2.8 Ecology

Impacts to ecology would be the same as for new dry storage (Section F.4.3.2.1.8).

F.4.3.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.3.2.1.9).

F.4.3.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.3.2.1.10).

F.4.3.2.2.11 Occupational and Public Health and Safety

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Hanford Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and

resulting doses are discussed in Section F.5 of this appendix. Table F-61 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Hanford Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-61 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• WNP-4 Spray Pond	0.00022	1.1×10^{-10}	0.0058	0.0000029
• New Wet Storage Facility	0.00020	1.0×10^{-10}	0.012	0.000006
Storage at:				
• WNP-4 Spray Pond	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
• New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 3B involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from Idaho National Engineering Laboratory and/or Savannah River Site and 193 shipments directly from the ports into a wet storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-62 presents the population dose that would be received by the members of the working crew and the associated risks if that working crew handled the total number of transportation casks at the Hanford Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

F.4.3.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Hanford Site would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-63. These requirements represent a small percent of current requirements for the Hanford Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Hanford Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-62 Handling-Related Impacts to Workers at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

	Worker Population Dose (person-rem)	*Worker Population Risk (LCF)*
	New Wet Storage Facility or WNP-4 Spray Pond	*New Wet Storage Facility or WNP-4 Spray Pond*
Phase 2	109	0.04

Table F-63 Annual Utility and Energy Requirements for New Wet Storage at the Hanford Site

Commodity	*Baseline Site Usage*	*Wet Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	340,000	1,000-1,500	0.44 percent
Fuel (l/yr)	83,000,000	0	0 percent
Water (l/yr)	15,000,000,000	2,700,000[a] 1,500,000[b]	0.02 percent 0.01 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.3.2.2.13 Waste Management

Construction of a new wet storage facility at the Hanford Site would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-64. These quantities, represent a very small percentage increase above current levels at the Hanford Site. Existing waste management storage and disposal activities at Hanford Site could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on existing Hanford Site waste management capacities would be minimal.

Table F-64 Annual Waste Generated for New Wet Storage at the Hanford Site

Waste Form	*Baseline Site Generation*	*Wet Storage Generation*	*Percent Increase*
High-Level Waste (m^3/yr)	240	none	0 percent
Transuranic Waste (m^3/yr)	170	none	0 percent
Solid Low-Level Waste (m^3/yr)	20,000	16[a] 1[b]	0.08 percent 0.005 percent
Wastewater (l/yr)	210,000,000	1,590,000[a] 400,000[b]	0.75 percent 0.2 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.3.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Hanford Site is presented here based on the methodology in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated. Information concerning radiation doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-65 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-65 Frequency and Consequences of Accidents at the Hanford Site

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Dry Storage Accidents[a]					
• Spent Fuel Assembly Breach	0.16	3.0	0.57	42	50
• Dropped Fuel Cask	0.0001	0.26	0.0085	3.0	0.22
• Aircraft Crash w\Fire[b]	NA	NA	NA	NA	NA
Dry Storage Accidents at FMEF					
• Spent Fuel Assembly Breach	0.16	4.7	2.1	46	0.99
• Dropped Fuel Cask	0.0001	0.2	0.032	3.2	0.0049
• Aircraft Crash w\Fire[b]	NA	NA	NA	NA	NA

NA = Not Applicable

[a] *New Dry Storage Facility*

[b] *Aircraft Crash accidents are not applicable to Hanford Site because their frequency of occurrence is less than one every ten million years.*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Hanford Site. These annual risks are multiplied by the maximum duration of the implementation alternative at the Hanford Site to obtain conservative estimates of risks for the Hanford Site. These risk estimates are presented in Table F-66.

Table F-66 Annual Risks of Accidents at the Hanford Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	2.4×10^{-7}	4.6×10^{-8}	0.0034	0.0000032
• Dropped Fuel Cask	1.3×10^{-11}	4.3×10^{-13}	1.5×10^{-7}	8.8×10^{-12}
• Aircraft Crash w\Fire[b]	NA	NA	NA	NA
Dry Storage Accidents at FMEF				
• Spent Nuclear Fuel Assembly Breach	3.7×10^{-7}	1.7×10^{-7}	0.0037	6.4×10^{-8}
• Dropped Fuel Cask	8 x 10-12	1.6×10^{-12}	1.6×10^{-7}	2.5×10^{-13}
• Aircraft Crash with Fire[b]	10^{-12}	NA	NA	NA

NA = Not Applicable

[a] *New Dry Storage Facility*

[b] *Aircraft crash accidents are not applicable to Hanford Site because their frequency of occurrence is less that one every ten million years*

Table F-67 presents the frequency and consequences of the accidents analyzed for the Hanford Site for new wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Hanford Site. These annual risks are multiplied by the

Table F-67 Frequency and Consequences of Accidents at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
New Wet Storage Facility:					
• Spent Nuclear Fuel Assembly Breach	0.16	0.13	0.0033	1.6	0.25
• Accidental Criticality	0.0031	64	14	740	3,600
• Aircraft Crash[a]	NA	NA	NA	NA	NA
WNP-4 Spray Pond:					
• Spent Nuclear Fuel Assembly Breach	0.16	0.15	0.0033	1.3	0.00024
• Accidental Criticality	0.0031	97	76	620	120
• Aircraft Crash[a]	NA	NA	NA	NA	NA

NA = Not Applicable

[a] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

maximum duration of this implementation alternative at the Hanford Site to obtain conservative estimates of risks at the Hanford Site. Table F-68 presents the risk estimates for this implementation alternative.

Table F-68 Annual Risks of Accidents at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
New Wet Storage Facility:				
• Fuel Assembly Breach	1.1×10^{-8}	2.7×10^{-10}	0.00013	1.6×10^{-8}
• Accidental Criticality	1.0×10^{-7}	2.2×10^{-8}	0.0012	0.0000044
• Aircraft Crash[a]	NA	NA	NA	NA
WNP-4 Spray Pond:				
• Fuel Assembly Breach	1.2×10^{-8}	2.7×10^{-10}	0.00011	1.5×10^{-11}
• Accidental Criticality	1.5×10^{-7}	1.2×10^{-7}	0.00096	1.5×10^{-7}
• Aircraft Crash[a]	NA	NA	NA	NA

NA = Not Applicable

[a] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

F.4.3.3.1 Secondary Impact of Radiological Accidents at the Hanford Site

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.3.3. Radiological accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident.

In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Hanford Site, the radiological accident with the highest MEI dose is the fuel assembly breach at a dry storage facility located at the FMEF (Table F-65). For this accident, the MEI dose would be 3.9 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage are expected at the Hanford Site.

F.4.3.4 Cumulative Impacts at the Hanford Site

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions, and current activities at the Hanford Site. A major portion of the presentation is based on information included in the Programmatic SNF&INEL Final EIS (DOE, 1995g), the Management of Spent Nuclear Fuel from the K Basins Draft EIS (DOE, 1995d) and the Safe Interim Storage of Hanford Tank Wastes Final EIS (DOE, 1995c).

Table F-69 summarizes the cumulative impacts for land use, socioeconomics, air quality, occupational and public health and safety, energy and water consumption and waste generation. The table also presents the contributions from the storage of foreign research reactor spent nuclear fuel on the cumulative impacts at the Hanford Site. For the purposes of this analysis, both the contributions from management of foreign research reactor spent nuclear fuel and the cumulative impacts were maximized by selecting the Centralization Alternative of the Programmatic SNF&INEL Final EIS at the Hanford Site.

As shown in Table F-69, the contribution from management of foreign research reactor spent nuclear fuel to the cumulative impacts at the Hanford Site would be minimal. It is concluded, therefore, that the implementation of any of the alternatives (including the Centralization Alternative) for the DOE spent nuclear fuel management program would not be expected to significantly contribute to cnmulative impacts.

F.4.3.5 Unavoidable Adverse Environmental Impacts

Unavoidable impacts associated with foreign research reactor spent nuclear fuel management activities would derive principally from construction activities needed for new storage facilities. There would be displacement of some animals from the construction site and the destruction of plant life within the area scoped for construction [up to 4 ha (10 acres)]. Criteria pollutants and radionuclides, would also be released in up to permitted quantities. Traffic congestion and noise would be expected to increase by a few percent during the construction of major facilities.

F.4.3.6 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of facilities for foreign research reactor spent nuclear fuel facilities at the Hanford Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in construction would probably not be recoverable. These would include

Table F-69 Cumulative Impacts at the Hanford Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	9	84,343[b]	84,352
Socioeconomics (persons)	190[c]/30[d]	3300/1220[e]	3,490[c]/1250[d]
Air Quality (nonradiological)	See Table F-56	NA	(f)
Occupational and Public Health and Safety			
• MEI Dose (rem/yr)	2.5×10^{-7}	0.0000036	0.0000036
LCF (per year)	1.3×10^{-10}	1.5×10^{-9}	1.5×10^{-9}
• Population dose (person-rem/yr)	0.015	0.22	0.235
LCF (per year)	0.0000075	0.00011	0.00011
• Worker Collective dose (person-rem/yr)	8.9[g]	116.5	125.4
LCF (per year)	0.0035	0.0466	0.05
Energy and Water Consumption			
• Electricity (MW-hr/yr)	1,000	495,600	496,600
• Fuel (million l/yr)	0	94.4	94.4
• Water (million l/yr)	2.2	15,004	15,006
Waste Generation			
• High-Level (m^3/yr)	0	354	354
• Low-Level (m^3/yr)	22	33,310	33,332
• Transuranic (m^3/yr)	0	240	240
• Mixed/hazardous (m^3/yr)	0	402	402

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a Laser Interferometer Gravitational-Wave Observatory, decommissioning of unused facilities, site restoration activities interim storage and tank wastes, management of spent nuclear fuel from the K basins, and current activities.*

[b] *Current operational areas constitute 83,767 acres*

[c] *Increase over baseline, during construction activities*

[d] *Increase over baseline, during operation activities*

[e] *Current working force is approximately 18,500 persons*

[f] *Nonradiological ground level cumulative concentrations would be within regulatory standards. 24-hour concentration for fugitive dust may exceed limits during construction of more than one facility simultaneously.*

[g] *The dose is due to the handling of the Foreign Research Reactor Spent Nuclear Fuel during receipt averaged over 30 years*

finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery.

F.4.3.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that Hanford Site could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Hanford Site would be minimal in most environmental media.

Pollution Prevention: DOE is responding to Executive Order 12856 and associated DOE orders and guidelines by reducing the use of toxic chemicals; improving emergency planning, response, and accident notification; and encouraging the development and use of clean technologies and the testing of innovative

pollution prevention technologies. Program components include waste minimization, source reduction and recycling, and procurement practices that preferentially procure products made from recycled materials. The pollution prevention program at the Hanford Site is being formalized in a Hanford Site Waste Minimization and Pollution Prevention Awareness Program Plan (DOE, 1995g).

The foreign research reactor spent nuclear fuel program activities would be conducted in accordance with this plan, and implementation of the pollution prevention and waste minimization plans would minimize impacts of wastes generated during spent nuclear fuel management activities (DOE, 1995g).

Socioeconomics: The level of predicted employment for foreign research reactor spent nuclear fuel activities at the Hanford Site is not large enough in comparison with present Hanford, local, or regional employment to produce a boom-bust impact on the economy (DOE, 1995g).

Cultural Resources: To avoid loss of cultural resources during construction of foreign research reactor spent nuclear fuel facilities on the Hanford Site, a cultural resources survey of the area of interest would be conducted by Pacific Northwest Laboratories Cultural Resources staff. Assuming no such resources were found, construction would proceed. If, however, during construction (earth moving) any cultural resource is discovered, construction activities would be halted and the Pacific Northwest Laboratories Cultural Resources staff called upon to evaluate and determine the appropriate disposition of the find.

To avoid loss of cultural resources during operation, such as unauthorized artifact collection, workers could be educated through programs and briefing sessions to inform them on applicable laws and regulations for site protection. These educational programs would stress the importance of preserving cultural resources and specifics of the laws and regulations for site protection. The exact locations of cultural resources are not identified by the Pacific Northwest Laboratories Cultural Resources group, therefore, any such artifact collection would be in an area discovered by the worker(s) (DOE, 1995g).

Geology: Soil loss would be controlled during construction using standard dust suppression techniques on disturbed soil and by stockpiling with cover where necessary. Following construction, soil loss would be controlled by revegetation and relandscaping of disturbed areas (DOE, 1995g).

Air Resources: To avoid impacts associated with emissions of fugitive dust during construction activities, exposed soils would be treated using standard dust suppression techniques. New facility sources of pollutant emissions to the atmosphere would be designed using best available technology to reduce emissions to "as low as reasonably achievable" levels (DOE, 1995g).

Water Resources: The impacts to surface and groundwater sources could be minimized through recycling of water, where feasible, and with cleanup of excess process water before release to ground or surface water (DOE, 1995g).

Noise: Generation of construction and operations noise would be reduced, as practicable, by using equipment that complies with noise guidelines (40 CFR Parts 201-211). Construction workers and other personnel working in environments exceeding U.S. Environmental Protection Agency-recommended guidelines during spent nuclear fuel storage, construction, or operation would be provided with earmuffs or earplugs approved by the Occupational Safety and Health Administration (29 CFR Part 1910). Because of the remote location of the Hanford Site foreign research reactor spent nuclear fuel activities, there would be no noise impacts with respect to the public for which mitigation would be necessary (DOE, 1995g).

Traffic and Transportation: At sites with increasing traffic concerns, DOE would encourage use of high-occupancy vehicles (such as vans or buses), implementing carpooling and ride-sharing programs, and staggering work hours to reduce peak traffic.

Occupational and Public Health and Safety: Although no radiological impacts on workers or the public were evident from the evaluation of incident-free foreign research reactor spent nuclear fuel activities at Hanford, further improvement in controls to protect both workers and the general public is a continuing activity. The "as low as reasonably achievable" principle would be used for controlling radiation exposure and exposure to hazardous/toxic substances. The Hanford Site would continue to refine its current emergency planning, emergency preparedness, and emergency response programs in place to protect both workers and the public (DOE, 1995g).

Site Utilities and Support Services: No mitigation measures beyond those identified for ground disturbance activities associated with bringing power and water to the foreign research reactor spent nuclear fuel site would appear necessary. In those cases, use of standard dust suppression techniques and revegetation of disturbed areas would mitigate ground disturbance impacts.

Accidents: The Hanford Site maintains an emergency response center and has emergency action plans and equipment to respond to accidents and other emergencies. These plans include training of workers, local emergency response agencies (such as fire departments) and the public communication systems and protocols, readiness drills, and mutual aid agreements. The plans would be updated to include consideration of new foreign research reactor spent nuclear fuel facilities and activities. Design of new facilities to current seismic and other facility protection standards would reduce the potential for accidents, and implementation of emergency response plans would substantially mitigate the potential for impacts in the event of an accident.

F.4.4 Oak Ridge Reservation

If the Oak Ridge Reservation site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Oak Ridge Reservation to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2), the Oak Ridge Reservation would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Oak Ridge Reservation until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Oak Ridge Reservation under Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, in Phase 2, the Oak Ridge Reservation could receive the aluminum-based foreign research reactor spent nuclear fuel managed at the Savannah River site during Phase 1, Eastern foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative.

As a Phase 2 site, the Oak Ridge Reservation would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility to be constructed on the West Bear Creek Valley Site. The location is preferred among the four locations considered in a siting study performed for spent nuclear fuel management (MMES, 1994). Description of the new dry storage facility is provided in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation is based on the above considerations. The analysis options selected do not represent all possible combinations but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

4A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new dry storage facility until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel that would be managed in the new dry storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States discussed in Section 2.2.2 introduce additional analysis options that could be considered for the Oak Ridge Reservation as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Oak Ridge Reservation would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for this amount of spent fuel. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 4A above.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Oak Ridge Reservation would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount of spent nuclear fuel would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Oak Ridge Reservation would be bounded by analysis option 4A above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Oak Ridge Reservation would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 4A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and, therefore, the amount of spent nuclear fuel available for acceptance would also be decreased. In this case, the Oak Ridge Reservation would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Oak Ridge Reservation would be bounded by analysis option 4A above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but

the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in option 4A above.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States as the foreign research reactor operators would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of fuel, in this case, cannot be quantified, however, the upper limit, as considered under analysis option 4A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management impacts at the Oak Ridge Reservation.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Oak Ridge Reservation for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 4B, which is similar to 4A, is considered as follows:

4B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Oak Ridge Reservation would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Oak Ridge Reservation is not considered.

F.4.4.1 Existing Facilities

There are no existing facilities for storing foreign research reactor spent nuclear fuel at Oak Ridge Reservation. Consequently, all potential environmental consequences from foreign research reactor spent nuclear fuel storage are related to new facility construction and operation.

F.4.4.2 New Facilities (Phase 2)

Analysis options 4A and 4B involve the use of new facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.4.2.1.3 Cultural Resources

There are no known historical, archaeological, paleontological, or Native American traditional sites in or around the potential storage site. No impacts to cultural resources are expected from ground disturbance, noise, or air emissions during construction or operation of the facility. Consultation with the Tennessee State Historic Preservation Office prior to project implementation is required by Section 106 of the National Historic Preservation Act of 1966. The State Historic Preservation Office may recommend further studies of the potential storage site to verify that no archaeological areas would be disturbed by construction activities (DOE, 1995g).

F.4.4.2.1.4 Aesthetic and Scenic Resources

Construction and operation of a new dry storage facility for foreign research reactor spent nuclear fuel would have similar impact on aesthetic and scenic resources at the Oak Ridge Reservation as the construction of spent nuclear fuel facilities under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel facilities associated with the Centralization Alternative would consist of a series of industrial buildings set within a 36-ha (90-acre) site. The maximum height of the buildings on the site would not exceed 12.8 m (42 ft) above ground level, or two to three stories. Since the buildings would be set into the south face of Pine Ridge, between Pine Ridge and Chestnut Ridge, the site would not be visible from areas outside the Reservation, with the possible exception of a limited section of Gallaher Road on the west side of the Clinch River, looking east along Bear Creek Valley and the Bear Creek Road which is accessible to the public. The site would be screened by appropriate vegetation so that the public views would not be affected. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility.

F.4.4.2.1.5 Geology

For the most part, geologic impacts from construction activities would be limited to soil disturbance; although in some areas, ripping or blasting of limestone, dolomite, or chert layers might be required. No extensive or unique geologic or mineral resources are found in or around the potential storage site, so no geological impacts would be expected (DOE, 1995g). The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

Because previously undisturbed areas would be used for new construction, some soil impacts from siting a new dry storage facility at the West Bear Creek Valley site would occur as a result of grading. Potential impacts from sediment runoff generated during construction activities would be minimized by implementation of soil erosion and sediment control measures. During operations, impacts to soil resources would be controlled by the planting or landscaping of land surfaces not covered by pavement and buildings (DOE, 1995g).

Major seismic activity and associated mass movement and subsidence are unlikely to occur during the construction or operation because faults in the area have not been active since the late Paleozoic Era (DOE, 1995g).

F.4.4.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction include generation of fugitive dust (particulate matter) and smoke from earth moving and clearing operations and emissions from construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Construction of this facility would require the clearing of approximately 16 ha (40 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years) of the project. Emissions of sulfur dioxide, nitrogen dioxide, and carbon monoxide are assumed to result entirely from diesel exhaust during the construction process. Respirable particulate matter (e.g., PM_{10}) is assumed to be 64 percent of the total suspended particulates estimated for the construction effort. Additionally, wetting controls are assumed to reduce this amount by 50 percent, which is a very conservative estimate.

Table F-70 presents the air quality impacts associated with the construction of a new dry storage facility at the Oak Ridge Reservation. Additionally, this table shows that the ambient impacts would be minimal and compliance with existing Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population. The estimated impacts from construction activities were generated using the Environmental Protection Agency regulatory-approved Industrial Source Complex Short-Term Model, Version 2.0, in conjunction with onsite meteorological data from 1991.

Table F-70 Estimated Maximum Concentrations of Criteria Pollutants at the Oak Ridge Reservation Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[a]	*Baseline Concentration*[b]	*Construction Activities*
Oak Ridge Reservation Boundary ($\mu g/m^3$)[c]				
• Particulate Matter (PM_{10})[c]	24-hr	150	84.9	0.5450
	Annual	50	0.43	0.0144
• Carbon Monoxide	1-hr	40,000	2,748.0	26.756
	8-hr	10,000	2,290.8	3.345
• Sulfur Dioxide	3-hr	1,300	170.3	2.356
	24-hr	365	55.2	0.345
	Annual	80	1.1	0.006
• Nitrogen Oxide	Annual	100	2.1	0.098

[a] *64 percent of total suspended particulates is considered to be respirable particulate matter (e.g., PM_{10}) for the construction activities. The standard refers to the actual PM_{10} standard.*

[b] *Source: DOE, 1995g*

[c] *To convert to $\mu g/ft^3$, multiply by 0.0283*

Nonradiological emissions are not expected during operation of a new dry storage facility.

Radiological Emissions: No radiological emissions from construction of a new dry storage facility for foreign research reactor spent nuclear fuel are expected. Based on fuel drying and storage operations

conducted at Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask, and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility were calculated based on the methodology and assumptions described in Appendix F, Section F.6. The radiological consequences of air emissions from the operation of the dry storage facilities at the Oak Ridge Reservation are discussed in Section F.4.4.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.4.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 3,060 million l (808 million gal) for the Oak Ridge Reservation, these amounts represent no more than a 0.07 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Oak Ridge Reservation.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Oak Ridge Reservation. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Oak Ridge Reservation could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Oak Ridge

Reservation would still be well within the design capacities of treatment systems at the Oak Ridge Reservation. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.4.2.1.8 Ecology

Terrestrial Resources: Radiation doses received by terrestrial biota from foreign research reactor spent nuclear fuel activities would be expected to be similar to those received by man. Although guidelines have not been established for acceptance limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for other species. Evidence indicates that no other living organisms have been identified that are likely to be significantly more radiosensitive than man. Thus, so long as exposure limits protective of man were not exceeded, no significant radiological impact on populations of biota would be expected as a result of foreign research reactor spent nuclear fuel activities at the West Bear Creek Site (DOE, 1995g).

Under the Centralization Alternative, construction of the potential spent nuclear fuel management facility would result in the disturbance of approximately 36 ha (90 acres) [16 ha (40 acres) if foreign research reactor spent nuclear fuel is considered in isolation], or less than 1 percent of the Oak Ridge Reservation. It is assumed that the area to be disturbed includes construction laydown areas, grading, and new buildings, and that the access road or other rights-of-ways have not been included in the total area to be disturbed. Vegetation within the area of the potential site for the spent nuclear fuel management facility would be destroyed during land clearing activities, but may be mitigated by revegetating with native species where possible. Vegetation cover in this area is predominantly oak-hickory forest or pine-hardwood forest. Both forest types are common on the Oak Ridge Reservation and within the region (DOE, 1995g).

Construction of a new dry storage facility would have some adverse effects on animal populations. Less mobile animals, such as amphibians, reptiles, and small mammals, within the project area would be destroyed during land-clearing activities. Larger mammals and birds in construction and adjacent areas would be disturbed by construction activities and would move to nearby suitable habitat. The long-term survival of these animals would depend on whether the area to which they moved was at or below its carrying capacity. Areas that would be revegetated upon completion of construction would be of minimal value to most wildlife, but might be repopulated by more tolerant species (DOE, 1995g).

The Migratory Bird Treaty Act is primarily concerned with the destruction of migratory birds, as well as their eggs and nests. It could be necessary to survey construction sites for the nests of migratory birds prior to construction and/or avoid clearing operations during the breeding season (DOE, 1995g).

Activities associated with operation, such as noise, increased human presence and traffic, and night lighting could affect wildlife living immediately adjacent to the storage site. While these disturbances could cause some sensitive species to move from the area, most animals should be able to adjust (DOE, 1995g).

Wetlands: Construction of a new dry storage facility would likely displace the forested wetlands adjacent to tributaries of Grassy Creek flowing through the potential site. This unavoidable displacement of wetlands would be accomplished in accordance with U.S. Army Corps of Engineers and Tennessee Water Quality Control Administration requirements. The potential also exists to disturb wetlands further downstream through erosion and sedimentation. Such impacts would be controlled through implementation of a soil erosion and sediment control plan. Construction-related discharges to Grassy

Creek would be relatively low and have negligible impacts to wetlands associated with the creek. No impacts to wetlands are anticipated during facility operations (DOE, 1995g).

Construction of a new dry storage facility would require the rechanneling of tributaries to Grassy Creek that cross the potential site, thus causing the loss of this aquatic habitat. In addition, soil erosion due to construction could cause water quality changes (primarily sediment loading) to Grassy Creek and its tributaries. These impacts could be minimized by implementation of soil erosion and sediment control measures. No operational impacts to aquatic resources are anticipated. It is assumed that the potential project would have a water retention pond within the security fence that might provide minimal habitat for amphibians in the area.

Threatened and Endangered Species: No Federally-listed species are expected to be affected. Site surveys would be required to verify the presence of State-listed or other special status species. Land clearing activities could destroy protected plant species, such as purple fringeless orchid and pink lady's-slippers, that may occur within the site. State-listed species including the Cooper's, sharp-shinned, and red-shouldered hawks, the barn owl, and the black vulture, which potentially occur in the area, could be impacted by project activities. Approximately 16 ha (40 acres) of potential nesting and foraging habitat would be lost as a result of construction activities. Because this type of habitat is abundant in the area, the loss would not be expected to affect the viability of populations of these species. However, appropriate steps would be taken to prevent nest disturbance. DOE would consult with the Tennessee Department of Environment and Conservation as appropriate to avoid or mitigate imminent impacts to State-listed species (DOE, 1995g). DOE would also consult with the U.S. Fish & Wildlife Service regarding threatened and endangered species for the proposed construction sites of foreign research reactor spent nuclear fuel storage facilities at the Oak Ridge Reservation. Impacts to threatened and endangered species are not anticipated.

F.4.4.2.1.9 Noise

Noises generated on the Oak Ridge Reservation do not propagate offsite at levels that impact the general population. Thus, the Oak Ridge Reservation noise impacts for both the Centralization and Regionalization by Fuel Type and Geography Alternatives would be those resulting from transportation of personnel and materials to and from the site that affect nearby communities, and those resulting from onsite sources that may affect some wildlife near these sources (DOE, 1995g).

The transportation noises are a function of the size of the work force (e.g., an increased work force would result in increased employee traffic and corresponding increases in deliveries by construction crews). Such noise and activity associated with construction would be expected to have short-term effects on most wildlife. Under the Centralization Alternative, the projected Oak Ridge Reservation work force would increase by about nine percent in the years 2000 to 2002 during peak construction, and decrease thereafter. There would be a corresponding increase in private vehicle and truck trips to the site. The day-night average sound level at 15 m (50 ft) from the roads that provide access to the Oak Ridge Reservation would be expected to increase by less than 1 decibel. No change is expected in the community reaction to noise along these routes. No mitigation of traffic noise impacts is proposed (DOE, 1995g).

F.4.4.2.1.10 Traffic and Transportation

Construction and operation of a new dry storage facility would involve a small increase in the number of employees commuting to the Oak Ridge Reservation and transportation of foreign research reactor spent nuclear fuel and hazardous chemicals within the site.

The maximum reasonably foreseeable scenario for construction and operation traffic occurs under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS. This would occur in 2001, when there would be about 4,200 full time employees and about 409,500 people in the region of influence. Construction and operation employees would contribute little to the future traffic because they represent such a small percentage of the region of influence population growth (DOE, 1995g). This conclusion would also be valid for a new dry storage facility for foreign research reactor spent nuclear fuel.

F.4.4.2.1.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Oak Ridge Reservation would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-71 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Oak Ridge Reservation. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-71 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Oak Ridge Reservation (New Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility	0.089	4.5×10^{-8}	0.085	0.000043
Storage at:				
• New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 4A involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-72 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Oak Ridge Reservation.

The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-72 Handling-Related Impacts to Workers at the Oak Ridge Reservation (New Dry Storage)

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	New Dry Storage	*New Dry Storage*
Phase 2	266/113[a]	0.11/0.05[a]

[a] *The two numbers represent the cask/vault designs respectively*

F.4.4.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Oak Ridge Reservation would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-73. These requirements represent a small percent of current requirements for the Oak Ridge Reservation. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Oak Ridge Reservation is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-73 Annual Utility and Energy Requirements for New Dry Storage at the Oak Ridge Reservation

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	335,800	800 - 1,000	0.3 percent
Fuel (l/yr)	3,600[c]	0	0 percent
Water (l/yr)	3,060,000,000	1,590,000[a] 400,000[b]	0.05 percent[a] 0.01 percent[b]

[a] *During receipt and handling*

[b] *During storage*

[c] *Decatherms/yr of natural gas*

F.4.4.2.1.13 Waste Management

Construction of a new dry storage facility at the Oak Ridge Reservation would generate 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in

Table F-74. These quantities represent a very small percent increase above current levels at the Oak Ridge Reservation. Existing waste management storage and disposal activities at Oak Ridge Reservation could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Oak Ridge Reservation waste management capacities would be minimal.

Table F-74 Annual Waste Generated for New Dry Storage at the Oak Ridge Reservation

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	0	0	0 percent
Transuranic (m^3/yr)	16	0	0 percent
Solid Low-Level (m^3/yr)	6,902	22[a] 1[b]	0.32 percent[a] 0.01 percent[b]
Wastewater (l/yr)	754,000,000	1,590,000[a] 400,000[b]	0.21 percent[a] 0.05 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.4.2.2 Wet Storage

Analysis option 4B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation. This storage option would require the construction of a new wet storage facility.

F.4.4.2.2.1 Land Use

A new wet storage facility would be located in a 36-ha (90-acres) area in the eastern portion of West Bear Creek Valley. The majority of the land in this area can be characterized as vacant, unused, and ready for development. Use of West Bear Creek Valley for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 16 ha (40 acres) of land. This represents about 44 percent of the space designated for foreign research reactor spent nuclear fuel storage; however, this represents only about 0.1 percent of the entire Oak Ridge Reservation. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at any of the areas would significantly impact land use patterns on Oak Ridge Reservation.

F.4.4.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 8.2 percent of the estimated FY 1995 total expenditures for the Oak Ridge Reservation (1,174 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 2 percent and 0.3 percent of FY 1995 total expenditures for the Oak Ridge Reservation. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Oak Ridge Reservation of approximately 17,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Oak Ridge Reservation.

F.4.4.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.4.2.1.3).

F.4.4.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.4.2.1.4).

F.4.4.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.4.2.1.5).

F.4.4.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 3 ha (7 acres) of land. In comparison, approximately 4 ha (10 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction (Section F.4.4.2.1.6).

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double banked high efficiency particulate air filters prior to release to the environment through an exhaust air system. The high efficiency particulate air filters would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilating, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite

concentrations (DOE, 1989b). Air emissions from the wet storage facility are expected to be similar to the air emissions from the CPP-603 at the Idaho National Engineering Laboratory. The annual air emission for the CPP-603 was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas. Radiological emissions from the operation of the wet storage facility were calculated based on the methodology and assumptions used in Section F.6.

F.4.4.2.2.7 Water Resources

The annual water usage during construction and operations of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 3,060 million l (808 million gal) for the Oak Ridge Reservation, these amounts represent an increase of about 0.06 percent and 0.09 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Oak Ridge Reservation.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Oak Ridge Reservation. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Oak Ridge Reservation could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Oak Ridge Reservation would still be well within the design capacities of treatment systems at the Oak Ridge Reservation. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.4.2.2.8 Ecology

Impacts to ecology would be the same as for new dry storage (Section F.4.4.2.1.8).

F.4.4.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.4.2.1.9).

F.4.4.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.4.2.1.10).

F.4.4.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Oak Ridge Reservation would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from routine airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during

storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-75 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Oak Ridge Reservation. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-75 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI DOSE (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Wet Storage Facility	0.060	3.0×10^{-8}	0.061	0.000031
Storage at:				
• New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 4B involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site, and 193 shipments directly from ports into a wet storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-76 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Oak Ridge Reservation. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-76 Handling-Related Impacts to Workers at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	New Wet Storage	*New Wet Storage*
Phase 2	109	0.04

F.4.4.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at Oak Ridge Reservation would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in

Table F-77. These requirements represent a small percent of current requirements for the Oak Ridge Reservation. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Oak Ridge Reservation is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-77 Annual Utility and Energy Requirements for Wet Storage at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Commodity	*Baseline Site Usage*	*Wet Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	335,800	800 - 1,000	0.15 percent
Fuel (l/yr)	3,600[a]	0	0 percent
Water (l/yr)	3,060,000,000	2,700,000[b] 1,500,000[c]	0.09 percent 0.05 percent

[a] *Decatherms/yr of natural gas*

[b] *During receipt and handling*

[c] *During storage*

F.4.4.2.2.13 Waste Management

Construction of a new wet storage facility at the Oak Ridge Reservation would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-78. These quantities represent a very small percentage increase above current levels at the Oak Ridge Reservation. Existing waste management storage and disposal activities at the Oak Ridge Reservation could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on existing the Oak Ridge Reservation waste management capacities would be minimal.

Table F-78 Annual Waste Generated for Wet Storage at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Waste Form	*Baseline Site Generation*	*Wet Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	0	0	0 percent
Transuranic (m^3/yr)	16	0	0 percent
Solid Low-Level (m^3/yr)	6,902	16[a] 1[b]	0.23 percent 0.01 percent
Wastewater (l/yr)	754,000,000	1,590,000[a] 400,000[b]	0.21 percent 0.05 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.4.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Oak Ridge Reservation is presented here based on the methodology in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage

locations and the same accidents at any of the sites evaluated. Information concerning radiation doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-79 presents the frequencies and the consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-79 Frequency and Consequences of Accidents at the Oak Ridge Reservation

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Dry Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	22	42	55	140
• Dropped Spent Nuclear Fuel Cask	0.0001	1.4	0.18	15	0.61
• Aircraft Crash w\Fire	0.000001	2300	180	2900	610

[a] *New Dry Storage Facility*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Oak Ridge Reservation. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks for the Oak Ridge Reservation. These risk estimates are presented in Table F-80.

Table F-80 Annual Risks of Accidents at the Oak Ridge Reservation

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	0.0000018	0.0000034	0.0044	0.0000088
• Dropped Spent Nuclear Fuel Cask	7.0×10^{-11}	9.0×10^{-12}	7.5×10^{-7}	2.4×10^{-11}
• Aircraft Crash w/Fire	1.2×10^{-9}	9.0×10^{-11}	0.0000015	2.4×10^{-10}

[a] *New Dry Storage Facility*

Table F-81 presents the frequency and consequences of the accidents analyzed for each site for wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Oak Ridge Reservation. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks at the Oak Ridge Reservation. Table F-82 presents the risk estimates from this implementation alternative.

F.4.4.3.1 Secondary Impact of Radiological Accidents at the Oak Ridge Reservation

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.4.3. Radiological

Table F-81 Frequency and Consequences of Accidents at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
New Wet Storage Facility					
• Spent Nuclear Fuel Assembly Breach	0.16	0.71	0.20	16	0.68
• Accidental Criticality	0.0031	1,500	3,300	1,400	6,800
• Aircraft Crash	0.000001	380	600	2,900	1,900

Table F-82 Annual Risks of Accidents at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-8}	1.6×10^{-8}	0.0013	4.4×10^{-8}
• Accidental Criticality	0.0000024	0.000005	0.0022	0.0000084
• Aircraft Crash	1.9×10^{-10}	3.0×10^{-10}	0.0000015	7.6×10^{-10}

accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Oak Ridge Reservation, the radiological accident with the highest MEI dose is the aircraft crash into a dry storage facility with fire. For this accident, the MEI dose would be 2,300 mrem. For the air immersion and ground surface pathways only, the dose would be 140 mrem, which is greater than the 100 mrem limit used in this analysis. Local contamination would be likely around the dry storage facility, but is expected to be contained entirely within the boundaries of the Oak Ridge Reservation. Cleanup activities should be small and any impacts to land uses, cultural resources, water quality, and ecology would be reversible. No impacts to national defense or local economies would be expected.

F.4.4.4 Cumulative Impacts at the Oak Ridge Reservation

This section presents the cumulative impacts of the proposed action, potential impacts of other contemplated major DOE actions, and current activities at the site. A major portion of the presentation is based on information included in the Programmatic SNF&INEL Final EIS (DOE, 1995g), the Tritium Supply and Recycling Final EIS (DOE, 1995a), and the Disposition of Surplus Highly Enriched Uranium Draft EIS (DOE, 1995e). Other activities considered for the Oak Ridge Reservation which could affect the site environment have not been determined sufficiently at this time to allow impact evaluation. They

include activities associated with the waste management at the site, storage and disposition of weapons-usable fissile materials, and stockpile stewardship and management program.

Tables F-83 and F-83A summarize the cumulative impacts for land use, socioeconomics, air quality, occupational and public health and safety, energy and water consumption, and waste generation at the site. Table F-83 also presents the contribution from the storage of foreign research reactor spent nuclear fuel on the cumulative impacts at the Oak Ridge Reservation. For the purposes of this analysis, both the contributions from management of foreign research reactor spent nuclear fuel and the cumulative impacts were maximized by selecting the Centralization Alternative of the Programmatic SNF&INEL Final EIS at the Oak Ridge Reservation.

As shown in Table F-83, the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts (under the Centralization Alternative) at the Oak Ridge Reservation would be minimal. The Programmatic SNF&INEL Final EIS concludes that the implementation of any of the alternatives (including the Centralization Alternative) for the DOE spent nuclear fuel management program would not be expected to significantly contribute to cumulative impacts (DOE, 1995g). This conclusion is also valid for the implementation of any of the alternatives considered in this EIS for storage of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation.

F.4.4.5 Unavoidable Adverse Environmental Impacts

Construction of the potential foreign research reactor spent nuclear fuel storage facilities would require the disturbance of approximately 16 ha (40 acres) of mostly forested undeveloped land. Although this represents less than one percent of the undeveloped land on the Oak Ridge Reservation, it would eliminate potential foraging and nesting habitat and would destroy plant species in the area. It would also require the dedication of a reasonably level land parcel that could otherwise accommodate other construction projects.

F.4.4.6 Irreversible and Irretrievable Commitments of Resources

Construction and operation of new foreign research reactor spent nuclear fuel storage facilities would require commitments of electrical energy, fuel, concrete, steel, sand, gravel and miscellaneous chemicals. Most of the water that would be withdrawn from the Clinch River to operate the foreign research reactor spent nuclear fuel facilities would be returned to surface water in the Clinch River watershed, although some evaporative losses would be unavoidable. The land dedicated to the foreign research reactor spent nuclear fuel facilities could become available for other urban uses following closure and decommissioning. However, the soils on the site would have to be amended to support land uses such as agriculture, forestry, or wildlife management.

F.4.4.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Oak Ridge Reservation could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Oak Ridge Reservation would be minimal in most environmental media.

Pollution Prevention: The DOE Oak Ridge Field Office established a Waste Minimization and Pollution Prevention Awareness Plan to reduce the quantity and toxicity of hazardous, mixed, and radioactive wastes generated at the Oak Ridge Reservation. The plan is designed to reduce the possible pollutant releases to the environment and thus increase the protection of employees and the public. All contractors and users that exceed the U.S. Environmental Protection Agency criteria for small-quantity generators are

Table F-83 Cumulative Impacts at the Oak Ridge Reservation

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	40	14,335[b]	14,375
Socioeconomics (persons)	190[b]/30[c]	3,917[b]/930[c]	4,107[b]/960[c]
Air Quality (nonradiological)	See Table F-83A	See Table F-83A	See Table F-83A
Occupational and Public Health and Safety			
• MEI Dose (rem/yr)	0.00009	0.0155	0.0156
LCF (per year)	4.5×10^{-8}	0.0000077	0.0000078
• Population Dose (person-rem/yr)	0.085	94.5	94.6
LCF (per year)	0.000043	0.047	0.047
• Worker Collective Dose (person-rem/yr)	8.9[d]	261.3	270.2
LCF (per year)	0.0036	0.104	0.108
Energy and Water Consumption			
• Electricity (MW-hr/yr)	1,000	4,981,000[e]	4,982,000
• Natural Gas (million m^3/yr)	0	68.64	68.64
• Coal (tons/yr)	0	35,053	35,053
• Diesel Oil (million l/yr)	0	4.83	4.83
• Water (million l/yr)	2.2	68,172	68,174
Waste Generation			
• High-Level (m^3/yr)	0	0	0
• Low-Level (m^3/yr)	22	34,989	35,011
• Transuranic (m^3/yr)	0	16	16
• Mixed/Hazardous (m^3/yr)	0	119,411	119,411

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of the Expended Core Facility, the construction and operation of the Advanced Neutron Source Facility, construction and operation of a Tritium production facility, and surplus highly enriched uranium management activities at the site*

[b] *Increase over baseline (17,000), during construction activities*

[c] *Increase over baseline (17,000), during operation activities*

[d] *The dose is due to the handling of Foreign Research Reactor Spent Nuclear Fuel during receipt averaged over 30 years*

[e] *Major portion of the requirement for electricity by the proposed tritium production facility (3,740,000 MW-hr/yr)*

establishing their own waste minimization and pollution prevention awareness programs. Contractor programs ensure that waste minimization activities are in accordance with Federal, State, and local environmental laws and regulations, and DOE orders (DOE, 1995g).

Additional goals include the promotion and use of nonhazardous materials, establishment of a baseline of waste generation data, calculations of annual reductions of waste generated, and implementation of recycling programs. Goals also include incorporation of waste minimization concepts and technologies in planning and design of new processes and facilities, and in upgrades of existing facilities. A waste minimization task force composed of representatives from each contractor has been established to coordinate waste minimization and pollution awareness activities (DOE, 1995g).

Socioeconomics: To reduce construction- and operation-related impacts, coordination with local communities could address potential impacts from increased labor and capital requirements. The knowledge of the extent and effect of growth due to foreign research reactor spent nuclear fuel

Table F-83A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Oak Ridge Reservation[a]

Pollutant	*Averaging Time*	*Regulatory Standard (μg/m³)*	*Cumulative Concentration[b] (μg/m³)*	
Carbon Monoxide	1-hour	40,000	3,696	(9.2%)
	8-hour	10,000	2,495	(24.9%)
Nitrogen Oxides	Annual	100	13	(13%)
Surfur Dioxide	3-hour	1,300	336.6	(25.9%)
	24-hour	365	5.84	(1.6%)
	Annual	80	3.62	(4.52%)
Particulate Matter (PM_{10})	24-hour	150	88.1	(58.7%)
	Annual	50	0.48	(0.96%)
Total Suspended Particulates	Annual	150	119	(79.3%)

[a] *Concentrations represent activities from: foreign research reactor spent nuclear fuel management, DOE-owned spent fuel management, construction and operation of the Expended Core Facility, construction and operation of the Advanced Neutron Source Facility, construction and operation of a tritium supply and recycling facility, and surplus highly enriched uranium management at the site*

[b] *Numbers in parentheses indicate the percentage of the regulatory standard*

management activities could greatly enhance the ability of affected jurisdictions to plan effectively. Effective planning would address change in levels of service for housing, infrastructure, utilities, transportation, and public services and finances (DOE, 1995g).

To alleviate potential impacts associated with the in-migration of labor, local labor force availability could be increased through various employment training and referral systems currently provided by the Oak Ridge Reservation. The goal of these systems would be to reduce the potential for in-migration of labor to support foreign research reactor spent nuclear fuel management activities (DOE, 1995g).

Water Resources: The potential foreign research reactor spent nuclear fuel storage facilities would have to be located and constructed to minimize floodplain impacts and to avoid floodplains to the maximum extent possible, as required by Executive Order 11988 (Floodplain Management) and DOE rule 10 CFR 1022. Site-specific surveys would be performed to determine locations of flooding elevations more accurately (DOE, 1995g).

Ecology: DOE would consult with the Tennessee Department of Environment and Conservation as appropriate to avoid or mitigate imminent impacts to State-listed species (DOE, 1995g).

Accidents: New foreign research reactor spent nuclear fuel storage facilities would be designed to comply with current Federal, State, and local laws, DOE orders, and industrial codes and standards. This would provide facilities that are highly resistant to the effects of severe natural phenomena, including earthquakes, floods, tornadoes, high winds, as well as credible events as appropriate to the site, such as fires and explosions, and manmade threats to its continuing structural integrity for containing materials (DOE, 1995g).

Emergency preparedness plans have also been prepared for existing facilities and would be revised for new facilities to lower the potential consequences of an accident to workers and the public. All workers receive evacuation training to ensure timely and orderly personnel movement away from high-risk areas. Plans and arrangements with local authorities would also be in place to evacuate the general public that may be at risk of exposure to hazardous materials that are accidentally released (DOE, 1995g).

F.4.5 Nevada Test Site

If the Nevada Test Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Nevada Test Site to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2), the Nevada Test Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory, and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Nevada Test Site until ultimate disposition.

Although the Nevada Test Site has no existing facilities to receive foreign research reactor spent nuclear fuel at the beginning of the policy period, it has facilities that could be modified to receive foreign research reactor spent nuclear fuel within 5 years. These facilities are large hot cells located in the Nevada Research and Development Area on Jackass Flats. Presently these facilities (e.g., E-MAD) have little usage, but some are in acceptable condition. To use the E-MAD facility, a small pool would have to be constructed to be used for transferring the spent nuclear fuel from the transportation casks to containers designed for dry storage. A description of the E-MAD facility is included in Appendix F (Section F.3). The E-MAD facility could be ready within 5 years of the start of the proposed policy period.

The amount of spent nuclear fuel that would be received and managed at the Nevada Test Site under Management Alternative 1, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, during Phase 2, the Nevada Test Site could receive the TRIGA spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, Western foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative.

As a Phase 2 site, the Nevada Test Site would receive and manage foreign research reactor spent nuclear fuel at a newly constructed dry storage facility or a modified E-MAD facility. Description of the new dry storage facility is provided in Section 2.6.5.1.1.

The analysis of potential environmental impacts from management of foreign research reactor spent nuclear fuel at the Nevada Test Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

5A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site, where it would be managed at a new dry storage facility or a modified E-MAD facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new or E-MAD facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel that would be managed would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Nevada Test Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Nevada Test Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for the reduced amount of spent nuclear fuel. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 5A above.

- Under Implementation Subalternative 1b (Section 2.3.1), the Nevada Test Site would receive from the Idaho National Engineering Laboratory and/or the Savannah River Site only HEU. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the storage of this amount of fuel would be bounded by analysis option 5A (above).

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Nevada Test Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 5A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and, therefore, the amount of spent nuclear fuel available for acceptance would also be decreased. In such a case, the Nevada Test Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Nevada Test Site would be bounded by analysis option 5A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2) the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis option 5A.

- Under Implementation Subalternative 3, (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. The various arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States as the foreign research reactor operators would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent fuel, in this case, cannot be quantified; however, the upper limit, as considered under analysis option 5A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management impacts at the Nevada Test Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Nevada Test Site for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 5B, which is similar to 5A, is considered as follows:

5B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements). If the Nevada Test Site receives TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory or only western spent fuel, the wet storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.3.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Nevada Test Site would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Nevada Test Site is not considered.

F.4.5.1 Existing Facilities

Existing facilities considered for foreign research reactor spent nuclear fuel storage at the Nevada Test Site include the E-MAD facility in Area 25. For this analysis, the E-MAD facility was considered essentially as new because of the significant modifications needed to use it for foreign research reactor spent nuclear fuel storage. These modifications could be completed sometime between 1996 and 2006. The potential environmental impacts associated with the modification would be bounded by the impacts associated with the construction of a dry storage facility presented in Section F.4.5.2. Impacts from the operation of the E-MAD facility are presented below.

F.4.5.1.1 Socioeconomics

Potential socioeconomic impacts associated with storage option 5A would be attributable to staffing requirements at the E-MAD facility. The staffing requirements for dry storage would be about 120 full time employees. Considering that the total work force at the Nevada Test Site is approximately 4,000 (DOE, 1995g), the addition of 120 full time employees for foreign research reactor spent nuclear fuel storage is not expected to have any measurable socioeconomic impact in the region of influence.

F.4.5.1.2 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions

assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. For the purpose of these calculations, the refurbished E-MAD facility is treated as a generic dry storage facility. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6. Table F-84 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Nevada Test Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-84 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• E-MAD (dry storage)	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at:				
• E-MAD (dry storage)	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 5A involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-85 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-85 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (person-rem)	*Worker Population Risk (LCF)*
	E-MAD	*E-MAD*
Phase 2	113	0.05

F.4.5.1.3 Material, Utility, and Energy Requirements

The material, utility, and energy requirements for the E-MAD facility are typical of those for dry storage. Table F-86 presents the estimated material, utility and energy consumption for dry storage.

Table F-86 Annual Utility and Energy Requirements for Dry Storage at the Nevada Test Site

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.6 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,138,000,000	1,590,000[b] 400,000[c]	0.14 percent 0.04 percent

[a] *The majority of the energy used at the Nevada Test Site is provided by electricity. Current usage is not available*

[b] *During receipt and handling*

[c] *During storage*

These requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

F.4.5.1.4 Waste Management

The contribution of waste associated with the operation of the E-MAD facility is typical of that for new dry storage (Section F.4.5.2.1.13).

F.4.5.1.5 Air Quality

The contribution of air emissions associated with the operation of the E-MAD facility is typical to that for new dry storage (Section F.4.5.2.1.5).

F.4.5.1.6 Water Resources

The effect of the operation of the E-MAD facility on the water usage is typical to that for new dry storage (Section F.4.5.2.1.7).

F.4.5.2 New Facilities (Phase 2)

Analysis options 5A and 5B involve the use of new facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.5.2.1 Dry Storage

Analysis option 5A involves long-term dry storage of foreign research reactor spent nuclear fuel at the Nevada Test Site. This analysis option would require the construction of a new dry storage facility. The analysis option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 and earlier in this appendix. There are no environmental impact parameters that would discriminate between the two designs.

F.4.5.2.1.1 Land Use

A new dry storage facility would be located in Area 5 in the southeastern portion of the Nevada Test Site. The land in this area can be characterized as sparsely vegetated desert, ready for development. Use of Area 5 for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at any of the areas would significantly impact land use patterns on the Nevada Test Site.

F.4.5.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents approximately 66 percent of the estimated FY 1995 total expenditures for the Nevada Test Site (141 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be $15.6 million for receipt and handling and $0.6 million for storage. These costs represent about 11 percent and 0.5 percent of FY 1995 total expenditures for the Nevada Test Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from direct construction employment on the region of influence would not be significant. In addition, when compared to the projected FY 1995 work force at the Nevada Test Site of approximately 4,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to eight persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Nevada Test Site.

F.4.5.2.1.3 Cultural Resources

There are no known historical, archaeological, paleontological, or Native American traditional sites in or around the potential storage site. No impacts to cultural resources are expected from ground disturbance, noise, or air emissions during facility construction or operation of the new dry storage facility. Consultation with the Nevada State Historic Preservation Office prior to project implementation is required under Section 106 of the National Historic Preservation Act of 1966. The State Historic Preservation Office may recommend further studies of the proposed storage site to verify that no archaeological areas would be disturbed by construction activities (DOE, 1995g).

F.4.5.2.1.4 Aesthetic and Scenic Resources

Construction and operation of a new dry storage facility for foreign research reactor spent nuclear fuel would have less impact on aesthetic and scenic resources at the Nevada Test Site than the construction of

facilities for spent nuclear fuel management under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g).

The proposed spent nuclear fuel facilities under Centralization, when fully constructed and under operation, would consist of a series of industrial buildings set within a 36-ha (90-acre) site. The site would not be visible from areas outside the Nevada Test Site. The new dry storage facility for foreign research reactor spent nuclear fuel would be constructed and operated under similar conditions. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility.

F.4.5.2.1.5 Geology

The new dry storage facility for foreign research reactor spent nuclear fuel would be situated on tertiary volcanic or sedimentary rocks near volcanic or intrusive centers where small to medium-size precious metal deposits could be developed. However, because the Nevada Test Site is closed to mining operations, any precious metal deposits that might exist in or around the potential storage site would not be impacted (DOE, 1995g). Further, no mass movement or subsidence and sediment runoff from land disturbances would be expected (DOE, 1995g). The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.5.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts at the Nevada Test Site associated with the dry storage facility include the generation of fugitive dust from construction activities (e.g., clearing of land, grading, and road preparation) and vehicle emissions from the heavy equipment utilized during the construction phase of the project. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed, barren surfaces.

The construction of this facility would require the clearing of 3.7 ha (7 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years) of the project. Emissions of sulfur dioxide, nitrogen dioxide, and carbon monoxide are assumed to result entirely from diesel exhaust during the construction process. Respirable particulate matter (e.g., PM_{10}) is assumed to be 64 percent of the total suspended particulates estimated for the construction effort. Additionally, wetting controls are assumed to reduce this amount by 50 percent, which is a very conservative estimate.

Table F-87 presents the air quality impacts associated with the construction of the dry storage facility at the Nevada Test Site. Additionally, this table shows that the ambient impacts would be minimal and compliance with existing Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population. The estimated impacts from construction activities were generated using

Table F-87 Estimated Maximum Concentrations of Criteria Pollutants at the Nevada Test Site Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[b]	*Baseline Concentration*[a]	*Construction Activities*
Boundary (μg/m3):[b, c]				
• Particulate Matter (PM_{10})[a]	24-hour	150	84.90	0.0020
	Annual	50	0.43	0.1107
• Carbon Monoxide	1-hour	40,000	2,748.0	26.756
	8-hour	10,000	2,290.8	3.345
• Sulfur Dioxide	3-hour	1,300	170.3	2.356
	24-hour	365	55.2	0.345
	Annual	80	1.1	0.006
• Nitrogen Oxides	Annual	100	[d]	0.098

[a] *Source: (DOE, 1995g)*

[b] *64 percent of total suspended particulates is considered to be respirable particulate matter (e.g., PM_{10}) for the construction activities. The standard refers to the actual PM_{10} standard.*

[c] *To convert to μg/ft^3, multiply by 0.0283*

[d] *No sources indicated*

the U.S. Environmental Protection Agency's regulatory-approved Industrial Source Complex Short-Term Model, Version 2.0 in conjunction with onsite meteorological data from 1991.

Nonradiological emissions are not expected during operation of the new dry storage facility for foreign research reactor spent nuclear fuel. Any emissions associated with dry storage would be directly attributable to front-end wet storage activities only.

Radiological Emissions: No radiological emissions from construction of a new dry storage facility for foreign research reactor spent nuclear fuel are expected. Based on dry fuel drying and storage operations conducted at Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides such as ^{58}Co, ^{60}Co, ^{50}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amount of these radionuclides that are released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask; and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage, small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility for foreign research reactor spent nuclear fuel were calculated based on the methodology and assumptions described in Appendix F, Section F.5. The radiological consequences of air emissions from the operation of the new dry storage facility at the Nevada Test Site are discussed in Section F.4.5.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.5.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 1,138 million l (301 million gal) for the Nevada Test Site, these amounts represent no more than a 0.2 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impacts on water resources at the Nevada Test Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Nevada Test Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Nevada Test Site could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Nevada Test Site would still be well within the design capacities of treatment systems at the Nevada Test Site. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.5.2.1.8 Ecology

Terrestrial Resources: Radiation doses received by terrestrial biota from foreign research reactor spent nuclear fuel activities would be expected to be similar to those received by man. Although guidelines have not been established for acceptance limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for other species. Evidence indicates that no other living organisms have been identified that are likely to be substantially more radiosensitive than man. Thus, so long as exposure limits protective of man were not exceeded, no significant radiological impact on populations of biota would be expected as a result of spent nuclear fuel activities at Area 5 (DOE, 1995g).

Wetlands: Under the Centralization Alternative, construction of a new dry storage facility would result in the disturbance of approximately 36 ha (90 acres) [3.7 ha or (9 acres) for foreign research reactor spent

nuclear fuel], or less than 1 percent of site location. No wetlands are expected to be disturbed because none exist in or around the proposed storage site (DOE, 1995g).

Threatened and Endangered Species: The project area is located within the range of the desert tortoise, a Federally-listed threatened species. Recent pre-activity surveys for other nearby projects have not identified the desert tortoise in the general area of the project site. However, a pre-activity survey for this project would be conducted to determine the presence or absence of the desert tortoise and other species of concern. If present, the desert tortoise could be impacted during construction of the new dry storage facility due to increased vehicular traffic, construction of trenches for utilities, and other temporary construction excavations. Prior to and during construction activities, fencing of the area and removal of tortoises within the fence would decrease the potential to bring harm to the desert tortoise (DOE, 1995g).

DOE has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the proposed construction site of foreign research reactor spent nuclear fuel storage facilities at the Nevada Text Site, as required by the Endangered Species Act.

Construction of a new dry storage facility would have some adverse effects on animal populations. Less mobile animals, such as amphibians, reptiles, and small mammals within the project area would be destroyed during land-clearing activities. Larger mammals and birds in construction and adjacent areas would be disturbed by construction activities and would move to nearby suitable habitat. The long-term survival of these animals would depend on whether the area to which they moved was at or below its carrying capacity. Areas that would be revegetated upon completion of construction would be of minimal value to most wildlife but may be repopulated by more tolerant species (DOE, 1995g).

The Migratory Bird Treaty Act is primarily concerned with the destruction of migratory birds, as well as their eggs and nests. It may be necessary to survey construction sites for the nests of migratory birds prior to construction and/or avoid clearing operations during the breeding season (DOE, 1995g).

Activities associated with operation, such as noise, increased human presence and traffic, and night lighting could affect wildlife living immediately adjacent to the site. While these disturbances may cause some sensitive species to move from the area, most animals should be able to adjust.

F.4.5.2.1.9 Noise

Noises generated on the Nevada Test Site do not propagate offsite at levels that impact the general population. Thus, noise impacts for both the Centralization and Regionalization by Fuel Type and/or Geography Alternatives at the Nevada Test Site would be limited to those resulting from the transportation of personnel and materials to and from the site that affect nearby communities, and those resulting from onsite sources that may affect some wildlife near these sources (DOE, 1995g).

The transportation noises are a function of the size of the work force (e.g., an increased work force would result in increased employee traffic and corresponding increases in deliveries by construction crews). Such noise and activity associated with construction would be expected to have short-term effects on most wildlife (DOE, 1995g).

Under the Centralization Alternative, the projected Nevada Test Site work force would increase by about 48 percent in the years 2000 to 2002, during peak construction, and would decrease thereafter. There would be a corresponding increase in truck, private vehicle, and bus trips to the site. The day-night average sound level at 15 m (50 ft) from U. S. Route 95 would be expected to increase by about 1 decibel.

No change is expected in the community reaction to noise along this route. No mitigation of traffic noise impacts is proposed (DOE, 1995g).

F.4.5.2.1.10 Traffic and Transportation

Construction and operation of a new dry storage facility would involve a small increase in the number of employees commuting to the Nevada Test Site and transportation of foreign research reactor spent nuclear fuel and hazardous chemicals within the site.

The maximum reasonably foreseeable scenario for construction and operations traffic occurs under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS. This would occur in 2001, when there would be about 3,400 full-time employees, and about 1,200,000 people in the region of influence. None of the future baseline levels of service would change due to spent nuclear fuel-related impacts (DOE, 1995g). These conclusions are equally valid for a new dry storage facility for foreign research reactor spent nuclear fuel.

F.4.5.2.1.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-88 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Nevada Test Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-88 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site (New Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Rise (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at:				
• New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 5A involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site, and 193 shipments directly from ports to a new dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with

the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-89 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-89 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (Person-rem)		*Worker Population Risk (LCF)*	
	New Dry Storage Cask	*New Dry Storage Vault*	*New Dry Storage Cask*	*New Dry Storage Vault*
Phase 2	266	113	0.11	0.05

F.4.5.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Nevada Test Site would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water.

The annual utility and energy requirements during operations are shown in Table F-90. These requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-90 Annual Utility and Energy Requirements for New Dry Storage at the Nevada Test Site

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.6 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,138,000,000	1,590,000[b] 400,000[c]	0.14 percent[b] 0.04 percent[b]

[a] *The majority of energy used at the Nevada Test Site is provided by electricity.*

[b] *During receipt and handling*

[c] *During storage*

F.4.5.2.1.13 Waste Management

Construction of a new dry storage facility at the Nevada Test Site would generate 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-91. These quantities represent a very small percent increase above current levels at the Nevada Test Site. Existing waste management storage and disposal activities at the Nevada Test Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Nevada Test Site waste management capacities would be minimal.

Table F-91 Annual Waste Generated for New Dry Storage at the Nevada Test Site

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	0	none	0 percent
Transuranic (m^3/yr)	0	none	0 percent
Solid Low-Level (m^3/yr)	10,845	22[a] 1[b]	0.20 percent[a] 0.01 percent[b]
Wastewater (l/yr)	11,000,000	1,590,000[a] 400,000[b]	14.4 percent[a] 3.6 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.5.2.2 Wet Storage

Analysis option 5B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Nevada Test Site. This storage option would require the construction of a new wet storage facility.

F.4.5.2.2.1 Land Use

A new wet storage facility would be located in Area 5 in the southeastern portion of the Nevada Test Site. The land in this area can be characterized as sparsely vegetated desert, ready for development. Use of Area 5 for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at any of the areas would significantly impact land use patterns on the Nevada Test Site.

F.4.5.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 79.5 percent of the estimated FY 1995 total expenditures for the Nevada Test Site (141 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be substantial. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 16.5 percent and 2.5 percent of FY 1995 total expenditures for the Nevada Test Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be positive.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Nevada Test Site of approximately 4,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Nevada Test Site.

F.4.5.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.5.2.1.3).

F.4.3.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.5.2.1.4).

F.4.5.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.5.2.1.5).

F.4.5.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 3 ha (7 acres) of land. In comparison, approximately 4 ha (10 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction, as presented in Section F.4.5.2.1.6.

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilating, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b).

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the CPP-603 at Idaho National Engineering Laboratory. The annual air emission for the CPP-603 was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas. Radiological emissions from the operation of the wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

F.4.5.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (1.2 million gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 1,138 million l (301 million gal) for the Nevada Test Site, these amounts represent an increase of about 0.17 percent and 0.23 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Nevada Test Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Nevada Test Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Nevada Test Site could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Nevada Test Site would still be well within the design capacities of treatment systems at the Nevada Test Site. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.5.2.2.8 Ecology

Impacts to the ecology would be the same as for new dry storage (Section F.4.5.2.1.8).

F.4.5.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.5.2.1.9).

F.4.5.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.5.2.1.10).

F.4.5.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an

80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-92 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Nevada Test Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-92 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Wet Storage Facility	0.00052	2.6×10^{-10}	0.00052	2.6×10^{-7}
Storage at:				
• New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.4×10^{-12}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 5B involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site and 193 shipments directly from ports into a new wet storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-93 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-93 Handling-Related Impacts to Workers at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	New Wet Storage	*New Wet Storage*
Phase 2	109	0.04

F.4.5.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Nevada Test Site would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-94. These

requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-94 Annual Utility and Energy Requirements for Wet Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Commodity	*Baseline Site Usage*	*Wet Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.84 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,139,000,000	2,700,000[b] 1,500,000[c]	0.23 percent 0.13 percent

[a] *The majority of energy used at the Nevada Test Site is provided by electricity.*

[b] *During receipt and handling*

[c] *During storage*

F.4.5.2.2.13 Waste Management

Construction of a new wet storage facility at the Nevada Test Site would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-95. These quantities represent a very small percentage increase above current levels at the Nevada Test Site. Existing waste management storage and disposal activities at the Nevada Test Site could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on the existing the Nevada Test Site waste management capacities would be minimal.

Table F-95 Annual Waste Generated for Wet Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Waste Form	*Baseline Site Generation*	*Wet Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	0	none	0 percent
Transuranic (m^3/yr)	0	none	0 percent
Solid Low-Level (m^3/yr)	10,845	16[a] 1[b]	0.15 percent 0.01 percent
Wastewater (l/yr)	11,000,000	1,590,000[a] 400,000[b]	14.5 percent 3.6 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.5.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Nevada Test Site is presented here based on the methodology in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated. Information concerning radiological doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-96 presents the frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-96 Frequency and Consequences of Accidents at the Nevada Test Site

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Dry Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	1.7	0.31	1.5	20
• Dropped Fuel Cask	0.0001	0.11	0.0014	0.40	0.089
• Aircraft Crash w/Fire	1×10^{-6}	180	1.2	250	87

[a] *E-MAD and New Dry Storage Facility*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Nevada Test Site. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks for the Nevada Test Site. These risk estimates are presented in Table F-97.

Table F-97 Annual Risks of Accidents at the Nevada Test Site

	Consequences			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	1.4×10^{-7}	2.5×10^{-8}	0.00012	0.0000013
• Dropped Fuel Cask	5.5×10^{-12}	7.0×10^{-14}	2.0×10^{-8}	3.6×10^{-12}
• Aircraft Crash w/Fire	9.0×10^{-11}	6.0×10^{-13}	1.3×10^{-7}	3.5×10^{-11}

[a] *E-MAD and New Dry Storage Facility*

Table F-98 presents the frequency and consequences of the accidents analyzed for each site for wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Nevada Test Site. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks at the Nevada Test Site. Table F-99 presents the risk estimates from this implementation.

F.4.5.3.1 Secondary Impact of Radiological Accidents at the Nevada Test Site

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.5.3. Radiological

Table F-98 Frequency and Consequences of Accidents at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
• Spent Nuclear Fuel Assembly Breach	0.16	0.054	0.0016	0.33	0.10
• Accidental Criticality	0.0031	88	15	54	1,300
• Aircraft Crash	1×10^{-6}	29	4.2	61	290

Table F-99 Annual Risks of Accidents at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

	Consequences			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
• Spent Nuclear Fuel Assembly Breach	4.2×10^{-9}	1.3×10^{-10}	0.000026	6.4×10^{-9}
• Accidental Criticality	1.4×10^{-7}	2.3×10^{-8}	0.000084	0.000016
• Aircraft Crash	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	1.2×10^{-10}

accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Nevada Test Site, the radiological accident with the highest MEI dose is the aircraft crash into a dry storage facility with fire (Table F-96). For this accident, the MEI dose would be 180 mrem. For the air immersion and ground surface pathways only, the dose would be 1.0 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage would be expected at the Nevada Test Site.

F.4.5.4 Cumulative Impacts at the Nevada Test Site

The section presents the cumulative impacts of the proposed action, potential impacts of other contemplated DOE actions, and current activities at the site. A major portion of the presentation is based on information included in the Programmatic SNF&INEL Final EIS (DOE, 1995g) and the Tritium Supply and Recycling Final EIS (DOE, 1995a). The Programmatic SNF&INEL Final EIS includes the quantitative impacts from a proposed Expended Core Facility at the site. The Nevada Test Site is also considered in the storage and disposition of weapons-usable fissile materials program which could affect the site environment. The impacts from this program have not been determined sufficiently at this time to allow impact evaluation.

Table F-100 Cumulative Impacts at the Nevada Test Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	9	314,393[b]	314,402
Socioeconomics (persons)	190[c]/30[d]	2,662[c]/1,000[d]	2,852[c]/1,030[d]
Air Quality (nonradiological)	See Table F-100A	See Table F-100A	See Table F-100A
Occupational and Public Health and Safety			
• MEI Dose (rem/yr)	7.6×10^{-7}	0.00031	0.00031
LCF (per year)	3.8×10^{-10}	1.55×10^{-7}	1.55×10^{-7}
• Population Dose (person-rem/yr)	0.00093	0.095	0.095
LCF (per year)	4.7×10^{-7}	.00047	0.000047
• Worker Collective Dose (person-rem/yr)	8.9[e]	81	89.9
LCF (per year)	0.0036	0.032	0.035
Energy and Water Consumption			
• Electricity (MW-hr/yr)	1,000	4,019,000[f]	4,020,000
• Fuel (million l/yr)	0	6,129	6,129
• Water (million l/yr)	2.2	2,563	2,565
Waste Generation			
• High-Level (m^3/yr)	0	0	0
• Low-Level (m^3/yr)	22	44,578	44,600
• Mixed/Hazardous (m^3/yr)	0	252	252
• Transuranic (m^3/yr)	0	16	16

[a] *Other activities include existing activities, DOE-owned spent fuel management activities, construction and operation of an Expended Core Facility, and construction and operation of a tritium production facility.*

[b] *Existing developed land area is 314,000 acres*

[c] *Increase over baseline (3,300 persons) during construction activities*

[d] *Increase over baseline (3,300 persons) during operation activities*

[e] *The dose is due to the handling of foreign research reactor spent nuclear fuel during receipt, averaged over 30 years*

[f] *Major portion is the requirement for electricity by the tritium production (accelerator) facility (3,740,000 MW-hr/yr)*

Tables F-100 and F-100A summarize the cumulative impacts for land use, socioeconomics, air quality, occupational and public health and safety, energy and water consumption and waste generation at the site. Table F-100 also presents the contribution from the storage of foreign research reactor spent nuclear fuel to the cumulative impacts at the Nevada Test Site. For the purposes of this analysis, both the contributions from management of foreign research reactor spent nuclear fuel and the cumulative impacts were maximized by selecting the Centralization Alternative of the Programmatic SNF&INEL Final EIS at the Nevada Test Site.

As shown in Table F-100, the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts (under the Centralization Alternative) at the Nevada Test Site would be minimal. The Programmatic SNF&INEL Final EIS concludes that the implementation of any of the alternatives (including the Centralization Alternative) for the DOE spent nuclear fuel management program would not be expected to significantly contribute to cumulative impacts (DOE, 1995g). This conclusion is also valid for the implementation of any of the alternatives considered in this EIS for storage of foreign research reactor spent nuclear fuel at the Nevada Test Site.

Table F-100A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Nevada Test Site[a]

Pollutant	*Averaging Time*	*Regulatory Standard (μg/m³)*	*Cumulative Concentration[b] (μg/m³)*
Carbon Monoxide	1-hour	40,000	2,815 (7%)
	8-hour	10,000	2,306 (23%)
Nitrogen Oxides	Annual	100	4.2 (4.2%)
Sulfur Dioxide	3-hour	1300	173.6 (13.3%)
	24-hour	365	55.5 (15.2%)
	Annual	80	1.1 (1.3%)
Particulate Matter (PM_{10})	24-hr	150	85 (56.6%)
	Annual	50	0.54 (1.1%)

[a] *Concentrations represent activities from: foreign research reactor spent nuclear fuel management, DOE-owned spent nuclear fuel management, construction and operation of an Expended Core Facility, and construction and operation of a tritium production and recycling facilities*

[b] *Number in parentheses indicate the percentage of the Regulatory Standard*

F.4.5.5 Unavoidable Adverse Environmental Impacts

Construction of the potential new foreign research reactor spent nuclear fuel storage facilities would require the disturbance of approximately 4 ha (10 acres) of undeveloped land. Although this represents less than one percent of the undeveloped land on the Nevada Test Site, it would eliminate potential terrestrial wildlife habitat, including habitat potentially suitable for the Federally-listed desert tortoise. It would also require the dedication of a small land parcel potentially suitable for other construction projects, but similar land parcels are abundant on the Nevada Test Site.

F.4.5.6 Irreversible and Irretrievable Commitments of Resources

Construction and operation of new foreign research reactor spent nuclear fuel facilities would require commitments of electrical energy, fuel, concrete, steel, sand, gravel and miscellaneous chemicals. Groundwater to operate the foreign research reactor spent nuclear fuel facilities would be withdrawn from an aquifer that is presently experiencing localized overdraught. Further studies would be necessary to quantify any irreversible effects on future groundwater availability attributable to spent nuclear fuel withdrawals from that aquifer. The land dedicated to the foreign research reactor spent nuclear fuel facilities would become available for other rural uses following closure and decommissioning.

F.4.5.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Nevada Test Site could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the site would be minimal in most environmental media.

Pollution Prevention: The DOE Nevada Field Office published a Waste Minimization and Pollution Prevention Awareness Plan in June 1991 to reduce the quantity and toxicity of hazardous, mixed, and radioactive wastes generated at DOE Nevada Field Office facilities. The plan is designed to reduce the possible pollutant releases to the environment and thus increase the protection of employees and the

public. All DOE Nevada Field Office contractors and the Nevada Test Site users that exceed the Environmental Protection Agency criteria for small-quantity generators are establishing their own waste minimization and pollution prevention awareness programs that are implemented by the DOE Nevada Field Office plan. Contractor programs ensure that waste minimization activities are in accordance with Federal, State, and local environmental laws and regulations, and DOE orders (DOE, 1995g).

Additional goals include the promotion and use of nonhazardous materials, establishment of a baseline of waste generation data, calculations of annual reductions of waste generated, and implementation of regulatory programs. Goals also include incorporation of waste minimization concepts and technologies in planning and design of new processes and facilities and in upgrades of existing facilities. A waste minimization task force composed of representatives from each contractor and the Nevada Test Site user has been established to coordinate DOE/Nevada Test Site waste minimization and pollution awareness activities (DOE, 1995g).

Socioeconomics: To reduce construction- and operation-related impacts, possible coordination with local communities could address potential impacts from increased labor and capital requirements. The knowledge of the extent and effect of growth due to foreign research reactor spent nuclear fuel management activities could greatly enhance the ability of affected jurisdictions to plan effectively. Effective planning would address changes in levels of service for housing, infrastructure, utilities, transportation, and public services and finances (DOE, 1995g).

To alleviate potential impacts associated with the in-migration of labor, local labor force availability could be increased through various employment training and referral systems currently provided by the Nevada Test Site. The goal of these systems would be to reduce the potential for in-migration of labor to support foreign research reactor spent nuclear fuel management activities (DOE, 1995g).

Cultural Resources: Consultation with the Nevada State Historic Preservation Office prior to project implementation is required under Section 106 of the National Historic Preservation Act of 1966. The State Historic Preservation Office may recommend that further archaeological studies be conducted throughout the construction area to verify that there are no archaeological sites subject to disturbance (DOE, 1995g).

Water Resources: The foreign research reactor spent nuclear fuel facilities would have to be located and constructed to minimize floodplain impacts and to avoid floodplains to the maximum extent possible, as required by Executive Order 11988 (Floodplain Management) and DOE orders. Site-specific surveys would be performed to determine locations of flooding elevations more accurately (DOE, 1995g).

Accidents: The foreign research reactor spent nuclear fuel storage facilities would be designed to comply with current Federal, State, and local laws, DOE orders, and industrial codes and standards. This would provide facilities that are highly resistant to the effects of severe natural phenomena, including earthquakes, floods, tornadoes, and high winds, as well as credible events appropriate to the site, such as fires and explosions and manmade threats to its continuing structural integrity for containing materials (DOE, 1995g).

An emergency preparedness plan would also be developed to lower the potential consequences of an accident to workers and the public. All workers receive evacuation training to ensure timely and orderly personnel movement away from high-risk areas. Plans and arrangements with local authorities would also be in place to evacuate the general public that may be at risk of exposure to hazardous materials accidentally released (DOE, 1995g).

F.5 Occupational Radiation Impacts from Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel

Occupational exposure to gamma radiation would depend largely on the operational history of the spent nuclear fuel elements to be stored in the facility and the length of time that these elements have been allowed to decay from the time that they were taken out of the reactor until they were placed in the cask for shipment to the storage facility. Normally, the decay time for fuel elements is established so that the gamma heating in the transportation cask is within specification and the radiation field on the outside cask surface is 200 mrem per hour or less. Special shipments can be made, however, with higher cask surface radiation fields, provided other requirements are met. Radiation exposures to personnel during receiving operations and surveys would depend on the level of radiation that is measurable on the exterior (surface) of the transportation cask. These initial operations are anticipated to provide the majority of personnel exposure since the remaining operations would be remote and could take advantage of the shielding built into the facility.

Realistic annual occupational radiation exposure estimates for facility operation can be performed once the following have been established:

- determination of accurate decay-time averaged values for the spent nuclear fuel,
- development of shielding characteristics for transportation casks for the spent nuclear fuel to be shipped to the facility,
- definition of personnel requirements for each of the individual operations to be accomplished within the facility, and
- completion of a time-motion study for the spent nuclear fuel element movement through the preliminary design of the facility.

The analyses in this appendix are based on a best estimate of the above conditions. The potential impacts are given in doses per cask shipment, so that the results can be simply multiplied by the total number of shipments for each potential storage arrangement.

Wet Storage: Occupational radiation exposure from the receipt, handling and storage of foreign research reactor spent nuclear fuel at a wet storage facility is treated in a generic way for all potential management sites, since the activities are essentially identical regardless of where the facility is located. It is based on actual handling experience of spent nuclear fuel at the Idaho National Engineering Laboratory and the Savannah River Site.

The workers involved with each cask were assumed to include the shipping agent, shift foremen, health physics technicians, and equipment operators. The equipment operators include onsite workers who remove each cask from its shipping container and transport it to the receiving bay, and those who perform most of the actual labor involved thereafter, such as transferring the spent nuclear fuel to storage and decontaminating the empty cask prior to returning it to the owners. Thus, while the assessment does not distinguish between them, the operators are a diverse group of workers whose distinct duties make it unlikely that the same operators could receive all of the calculated individual doses discussed below. As a result, it was assumed that the two foremen and two operators involved in handling the spent nuclear fuel casks outside the receiving facility would be different than the two foremen and two operators working inside the facility (the health physics technicians were assumed to be the same). This provides a conservative estimate of 12 workers.

In order to estimate the occupational radiation doses from the handling of foreign research reactor spent nuclear fuel transportation casks at the spent nuclear fuel management sites, it was necessary to develop a curve of dose rate versus distance for these casks. Historical data based on 44 research reactor spent nuclear fuel transportation cask receipts at either the Savannah River Site or the Idaho National Engineering Laboratory were obtained and evaluated. This historical data showed an average measured dose rate of approximately 2.3 mrem per hour at 1 m (3.3 ft) from the surface of the transportation cask. One cask, however, was measured to be 20 mrem per hour at 1 m (3.3 ft). To encompass this historical data, including the highest measured dose rate cask, an analysis was performed that assumed a dose rate of 23 mrem per hour at 1 m (3.3 ft) from the cask surface. It should be noted that, in the unlikely event that a higher dose rate transportation cask was received at the management site, radiological control procedures for as low as reasonably achievable limits would be utilized to ensure that the worker doses would be minimized. Dose rate reduction is usually accomplished by a combination of restrictions on time, distance from the source, and the provision of additional radiation shielding.

The plot of bounding transportation cask dose rate versus distance in Figure F-50 was developed using the ZYLIND computer code and appropriate conservative methodology. ZYLIND (RSIC, 1990) is a shielding computer code that uses the point kernel method to calculate photon dose rates from a cylindrical source and shield geometry. ZYLIND was developed in Germany in 1989 and then released to the Oak Ridge National Laboratory Radiation Shielding Information Center. ZYLIND has been extensively validated by comparison to measured dose rates from several hundred cylindrical containers with radioactive materials. ZYLIND calculated dose rates that were conservative and within 10 to 20 percent of the measured dose rates. ZYLIND allows the photon energy source to be divided into up to 20 energy groups from 0 to 10 million electron volts (Mev), allows up to eight materials regions, and includes mass attenuation and dose buildup information for a wide range of shielding materials.

The methodology used in calculating bounding transportation cask dose rates had four underlying assumptions. First, it was assumed that the dose rate at 1 m (3.3 ft) from the cask surface is 23 mrem per hour. Second, neutron dose rates from foreign research reactor spent nuclear fuel were assumed to be negligible and the only dose was assumed to be due to gamma (photon) radiation. A third assumption was that the foreign research reactor spent nuclear fuel source term inside a cask could be conservatively simulated by a single 1.0 Mev gamma energy group. Traditional NRC source terms (DiNunno et al, 1962) for spent nuclear fuel fission products assume an average gamma energy of 0.7 Mev. By using 1.0 Mev, the average gamma energy is expected to be conservatively bounded. Finally, it was assumed that the use of a point kernel cylindrical source-shield computer code (i.e., ZYLIND) would conservatively calculate the dose rates from a transportation cask.

The principal inputs for the calculation were the ZYLIND computer code manual (Radiation Shielding Information Center ZYLIND) and the U.S. Department of Transportation Certificates of Competent Authority for seven transportation cask designs that are likely to be used for the shipment of foreign research reactor spent nuclear fuel. These seven designs are: TN7, GNS-11, LHRL-120, NAC/LWT, PEGASE (IU-04), BMI-1, and GE-2000. These transportation casks are described in Appendix B, Section B.2. The U.S. Department of Transportation Certificates of Competent Authority provided geometry data on the cask inside cavity dimensions and the thickness and material composition of shielding adjacent to the cavity for each design.

With the cask geometry information, a set of ZYLIND calculations was performed for each design. An initial 1.0 Mev gamma source was estimated and ZYLIND was executed to calculate the dose rate at 1 m (3.3 ft) from the cask surface. This source was iterated upon until the 1 m (3.3 ft) dose rate equaled 23 mrem per hour. After this source was determined, the same source and cask geometry were rerun to calculate the dose rate at distances of from 0-50 m (0-164 ft) from the cask surface. This process was

repeated for each of the transportation cask designs. The resulting dose rates at distance for each cask design were compared and the highest dose rate response at all distances was synthesized from this data to produce Figure F-50.

Table F-101 shows the actual dose rates encountered during receipt and handling for essentially all of the foreign research reactor spent nuclear fuel casks, which are expected to be one to two orders of magnitude lower than the limit, based on actual experience with foreign research reactor spent nuclear fuel in the past.

Table F-102 presents the wet storage collective dose for unloading one transportation cask using time, distance, and personnel data from the Idaho National Engineering Laboratory and the dose rate curve in Figure F-50. The total worker dose per transportation cask was calculated to be 0.31 person-rem. The actual distances for each worker are based on conservative estimates of actual work experience that would reflect an as low as reasonably achievable Radiation Protection Program as required by DOE regulation (10 CFR 835).

Generic Dry Cask Storage: The receipt, handling, and storage occupational radiation doses (deep dose equivalents) for dry storage are also treated in a generic way, since the operation of the general facility designed for dry storage would be the same at any management site. The assessment is based on Pressurized Water Reactor spent nuclear fuel from the reactor's spent nuclear fuel storage pool to an NRC-licensed dry storage facility at the Calvert Cliffs nuclear power plant in Maryland. The system employed is the horizontal module system (NUHOMS), which was selected for this assessment for two reasons: (1) it is a current, regulatory-approved design that is readily available for foreign research reactor spent nuclear fuel dry storage, and (2) the worker dose rates calculated for the system are among the highest of the current systems now in use for storage of commercial spent nuclear fuel. As a result, the system analysis provides a reasonably conservative estimate for storage of foreign research reactor spent nuclear fuel in NUHOMS and a reasonable upperbound assessment for all other foreign research reactor spent nuclear fuel generic dry storage. The Calvert Cliffs Safety Analysis Report does not identify each category of worker associated with receipt, handling, transfer, and storage of spent nuclear fuel. As a result, for this assessment, the doses (deep dose equivalent) were assumed to be the same for all workers (titled "operators" in the assessment). However, it would appear from the work activities that it cannot be the same operators who support each of the activities. As a result, job titles comparable to those considered for wet storage have been defined in order to determine the average worker dose per cask. Therefore, it is assumed that the following categories of workers are involved: foremen (2), health physics technicians (2), equipment operators for the storage pool activities (2), and different equipment operators for the one-site transport and transfer of the spent nuclear fuel from the transfer cask to the dry storage cask (3), welders (2), helium leak test technician (1), and dye penetrant test technician (1). Each of the distances listed is the average distance for all of the workers involved in each one of the 25 specific activities associated with receipt, handling, transfer, and dry storage. Thus, for example, the first activity [loading fuel into the container (dry shielded canister)], would involve four workers (one foreman, one health physics technician, and two operators) in the Spent Fuel Pool area. The results indicate that the collective dose to the working crew of 13 would be 1.5 person-rem per NUHOMS cask transfer. A transfer cask load is approximately equal to the foreign research reactor spent nuclear fuel inventory of eight transportation casks.

IFSF (Dry Vault) Specific Dry Storage: Based on data provided by the Idaho National Engineering Laboratory, Table F-103 was generated to present the occupational dose for unloading one transportation cask into the IFSF. The collective dose to unload one transportation cask into the IFSF was calculated to be 0.32 person-rem. This dose is considered representative of a generic dry vault storage facility.

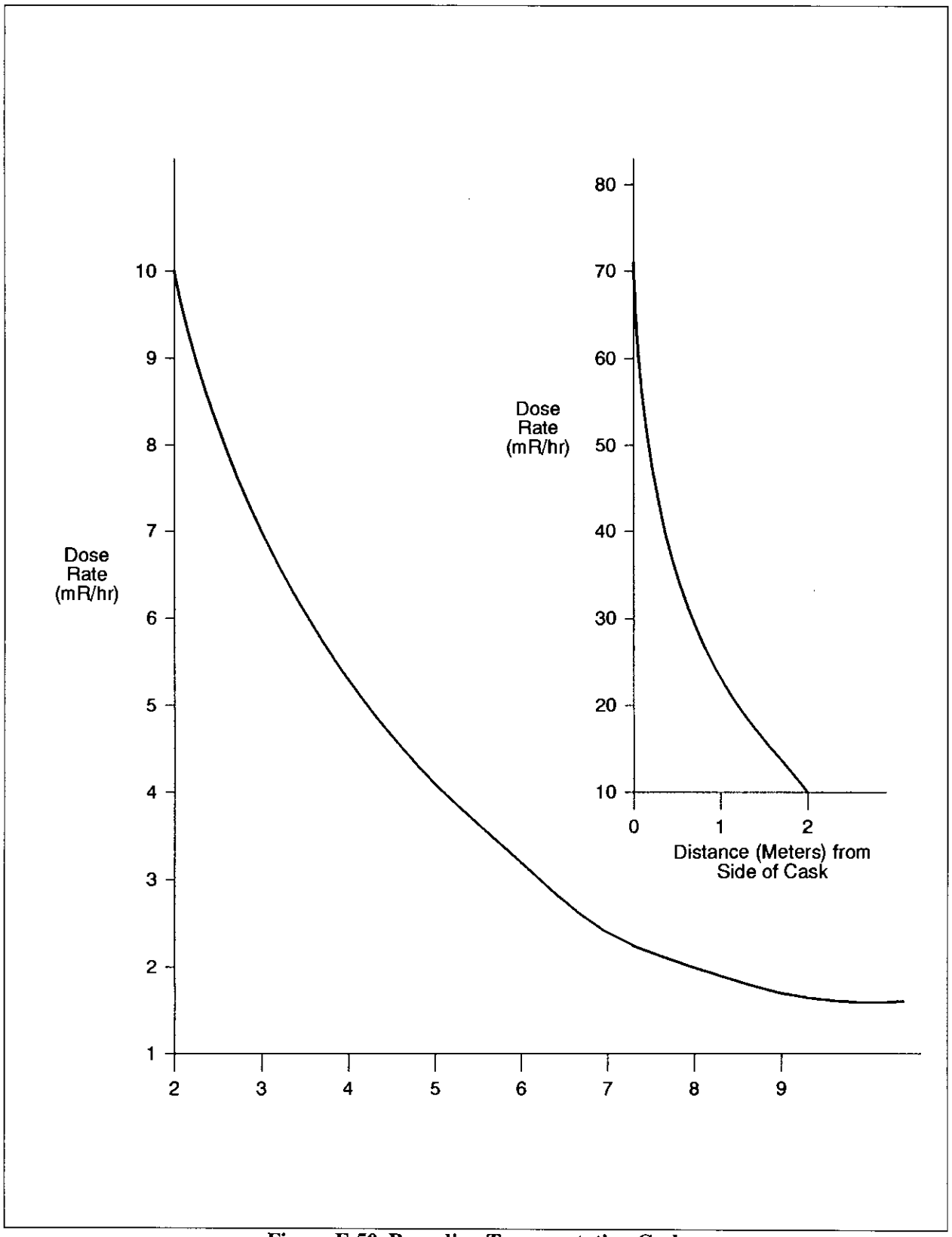


Figure F-50 Bounding Transportation Cask
(Dose Rate at a Distance Normalized to 10 millirem/hour at 2 meters [6.6 feet])

Table F-101 Actual Foreign Research Reactor Spent Nuclear Fuel Transportation Cask Dose Rate Measurements

Date	*Cask Model*	*Fuel*	*Side Cask Measured Dose Rate (mrem/hr) at 1 m (3.3 ft)*
Savannah River Site-Provided Data			
10/2/94	IU04-PEGASE		0.4
10/2/94	PEGASE		2.08
10/2/94	TN-7		8.4
10/2/94	GNS-11		1.2
12/18/87	PEGASE	DR3	1.5
1/26/89	PEGASE	DR3	0.6
12/31/87	PEGASE	ORPHEE	1.4
10/30/86	PEGASE	ORPHEE	0.5
9/1/87	PEGASE	SILOE	1.0
11/5/87	PEGASE	SILOE	0.3
12/30/87	PEGASE	SILOE	0.3
2/7/89	TN-7/2	RHF	15.0
8/16/88	TN-7/2	RHF	20.0
9/25/81	SWED.R2-B/23	AAR	0.9
1/20/89	TN-1	HFR	6.0
7/20/88	TN-1	HFR	0.5
6/8/88	GNS-11	FRJ-2	8.0
8/30/88	GNS-11	FRJ-2	0.8
8/30/88	GNS-11	FRJ-2	10.0
2/27/86	GOSLAR NO.1	FRG	2.0
11/12/86	GOSLAR NO.1	FRG	1.0
2/26/86	GOSLAR NO.2	FRG	0.2
3/10/80	GOSLAR NO.1	ASTRA	0.2
3/14/80	GOSLAR NO.2	ASTRA	0.1
1/29/86	BMI-1	RINC	0.5
2/6/86	BMI-1	RINC	0.4
6/5/84	BMI-1	U.VA.	0.5
6/11/84	BMI-1	U.VA.	0.5
9/11/84	BMI-1	U.MICH.	0.1
10/13/87	BMI-1	U.MICH.	0.1
7/14/81	BMI-1	U.MICH.	1.0
10/6/87	BMI-1	U.MICH.	0.1
Idaho National Engineering Laboratory-Provided Data			
	BMI-1	CORNELL-TRIGA	3.5
	BMI-1	BERKLEY-TRIGA	0.5
	BMI-1	MICHIGAN-TRIGA	1.0
	BMI-1	BERKLEY-TRIGA	0.5
	GE-700	BNL-HFBR	1.0
	GE-700	BNL-HFBR	1.0
	GE-700	BNL-HFBR	1.0
	GE-700	U. OF MISSOURI	0.1
	BMI-1	CORNELL-TRIGA	3.5
	BMI-1	MICHIGAN-TRIGA	0.1
	BMI-1	HANFORD-TRIGA	1.0
	BMI-1	HANFORD-TRIGA	1.0

Table F-102 Worker Dose Assessment for Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel in Wet Storage

Exposed Workers	*A. Exposure Distance (m)*	*B. Dose Rate (mrem/hr)*	*C. Exposure Time (minutes/cask)*	*D. Dose/Cask-Person (mrem)*	*E. Number of Exposed Workers*	*F. Collective Dose (Person-rem)*
Transport Receipt						
Shipping Agent	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Health Physics Tech	1.0	23.0	5	1.9E+00	1	1.9E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			15	3.6E+00		3.6E-03
Guards	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Remove Container Cover						
Foreman	10.0	1.5	20	5.0E+01	1	5.0E-04
	5.0	4.2	10	7.0E+01	1	7.0E-04
Subtotal			30	1.2E+00		1.2E-03
Operators	0.3	50.0	15	1.3E+01	3	3.8E-02
	0.3	50.0	10	8.3E+00	1	8.3E-03
	1.0	23.0	1	3.8E-01	1	3.8E-04
	2.0	10.0	10	1.7E+00	2	3.3E-03
Subtotal			36	2.3E+01		5.0E-02
Survey Cask						
Health Physics Tech	0.3	50.0	45	3.8E+01	1	3.8E-02
Subtotal			45	3.8E+01		3.8E-02
Removal of Impact Limiters						
Foreman	5.0	4.2	50	3.5E+00	1	3.5E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	5.0	4.2	45	3.2E+00	1	3.2E-03
	0.3	50.0	15	1.3E+01	1	1.3E-02
Subtotal			60	1.6E+01		1.6E-02
Operators	0.3	50.0	10	8.3E+00	3	2.5E-02
	0.5	36.0	60	3.6E+01	2	7.2E-02
Subtotal			70	4.4E+01		9.7E-02
Move Cask						
Equipment Operators	0.3	50.0	5	4.2E+00	1	4.2E-03
Subtotal			5	4.2E+00		4.2E-03
Removal of Cask from Transport						
Foreman	8.00	2.1	45	1.6E+00	1	1.6E-03
	2.00	10.0	10	1.7E+00	1	1.7E-03
	1.00	23.0	5	1.9E+00	1	1.9E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	8.00	2.1	30	1.1E+00	1	1.1E-03
	2.00	10.0	20	3.3E+00	1	3.3E-03
	1.00	23.0	10	3.8E+00	1	3.8E-03
Subtotal			60	8.2E+00		8.2E-03
Equipment Operators	4.00	5.3	59	5.2E+00	2	1.0E-02
	1.00	23.0	1	3.8E-01	1	3.8E-04
Subtotal			60	5.6E+00		1.1E-02

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Testing & Verification of Integrity						
Health Physics Tech	5.00	4.2	30	2.1E+00	1	2.1E-03
	2.00	10.0	28	4.7E+00	1	4.7E-03
	0.3	50.0	2	1.7E+00	1	1.7E-03
Subtotal			60	8.4E+00		8.4E-03
Operators	0.30	50.0	30	2.5E+01	2	5.0E-02
	4.00	5.3	30	2.7E+00	2	5.3E-03
Subtotal			60	2.8E+01		5.5E-02
Movement of Cask to Unloading Pool and Immersion						
Operators	0.30	50.0	2	1.7E+00	3	5.0E-03
	NA	0.1	58	9.7E-02	2	1.9E-04
Subtotal			60	1.8E+00		5.2E-03
Cask Unloading/Inspection/Storage						
Foreman	NA	0.1	240	4.0E-01	1	4.0E-04
Subtotal			240	4.0E-01		4.0E-04
Safeguards	NA	0.1	240	4.0E-01	1	4.0E-04
Subtotal			240	4.0E-01		4.0E-04
Operators	NA	0.1	240	4.0E-01	5	2.0E-03
Subtotal			240	4.0E-01		2.0E-03
Removal of Cask from Unloading Pool						
Operators	NA	0.1	60	1.0E-01	2	2.0E-04
Subtotal			60	1.0E-01		2.0E-04
Replacement of Cask of Transport and all subsequent operators assumed to be no exposure greater than background						
	NA	0.0	90	0	5	0
Subtotal				0		0
Total				**4.4E+01 (Max.)**		**3.1E-01**

Table F-103 Worker Dose Assessment for Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel in a Dry Storage Facility (Irradiated Fuel Storage Facility) or Generic Vault

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Transport Receipt						
Shipping Agent	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Health Physics Tech	1.0	23.0	5	1.9E+00	1	1.9E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			15	3.6E+00		3.6E-03
Guards	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00	1	1.1E-03
Remove Container Cover						
Foreman	10.0	1.5	20	5.0E-01	1	5.0E-04
	6.0	1.2	10	7.0E-01	1	7.0E-04
Subtotal			30	1.2E+00		1.2E-03
Operators	0.3	50.0	15	1.3E+01	3	3.8E-02
	0.3	50.0	10	8.3E+00	1	8.3E-03

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
	1.0	23.0	1	3.8E-01	1	3.8E-04
	2.0	10.0	10	1.7E+00	2	3.3E-03
Subtotal			36	2.3E+01		5.0E-02
Survey Cask						
Health Physics Tech	0.3	50.0	45	3.8E+01	1	3.8E-02
Subtotal			45	3.8E+01		3.8E-02
Removal of Impact Limiters						
Foreman	5.0	4.2	50	3.5E+00	1	3.5E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	5.0	4.2	45	3.2E+00	1	3.2E-03
	0.3	50.0	15	1.3E+01	1	1.3E-02
Subtotal			60	1.6E+01		1.6E-02
Operators	0.3	50.0	10	8.3E+00	3	2.5E-02
	0.5	36.0	60	3.6E+01	2	7.2E-02
Subtotal			70	4.4E+01		9.7E-02
Removal of Cask from Transport to Transfer Cart						
Foreman	8.0	2.1	45	1.6E+00	1	1.6E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
	1.0	23.0	5	1.9E+00	1	1.9E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	8.00	2.1	30	1.1E+00	1	1.1E-03
	2.00	10.0	20	3.3E+00	1	3.3E-03
	1.00	23.0	10	3.8E+00	1	3.8E-03
Subtotal			80	8.2E+00		8.2E-03
Equipment Operators	4.00	5.3	59	5.2E+00	2	1.0E02
	1.00	23.0	1	3.8E-01	1	3.8E-04
Subtotal			60	5.6E+00		1.1E-02
Testing & Verification of Integrity, Lid Bolt Removal						
Foreman	4.00	5.3	60	5.3E+00	1	5.3E-03
Subtotal			60	5.3E+00		5.3E-03
Health Physics Tech	5.00	4.2	30	2.1E+00	1	2.1E-03
	2.00	10.0	28	4.7E+00	1	4.7E-03
	0.30	50.0	2	1.7E+00	1	1.7E-03
Subtotal			60	8.4E+00		8.4E-03
Operators	0.30	50.0	30	2.5E+01	2	5.0E-02
	4.00	5.3	30	2.7E+00	2	5.3E-03
Subtotal			60	2.8E+01		5.5E-02
Movement of Cask into Handling Cove						
Foreman	4.00	5.3	60	5.3E+00	1	5.3E-03
Subtotal			60	5.3E+00		5.3E-03
Operators	4.0	5.3	10	8.8E-01	2	1.8E-03
	NA	0.1	50	8.3E-02	2	1.7E-04
Subtotal			60	9.7E-01		1.9E-03
Cask Unloading/Inspection/Storage						
Foreman	NA	0.1	480	8.0E-01	1	8.0E-04
Subtotal			480	8.0E-01		8.0E-04
QA Inspector	NA	0.1	480	8.0E-01	1	8.0E-04
Subtotal			480	8.0E-01		8.0E-04

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Operators	NA	0.1	480	8.0E-01	2	1.6E-03
Subtotal			480	8.0E-01		1.6E-03
Removal of Cask from Handling Cove						
Foreman	NA	0.1	60	1.0E-01	1	1.0E-04
Subtotal			60	1.0E-01		1.0E-04
Operators	NA	0.1	60	1.0E-01	2	2.0E-04
Subtotal			60	1.0E-01		2.0E-04
Replacement of Cask of Transport and All Subsequent Operators Assumed to be No Exposure Greater than Background	NA	0.0	90	0	5	0
Subtotal			90	0		0
Total				**4.4E+01 (Max.)**		**3.2E-01**

DSC = Dry Shielded Canister

Transfer Between Storage Facilities: The collective doses were calculated for loading fuel into a pod, a dry vault (i.e., the IFSF), and dry cask (i.e., Calvert Cliffs NUHOMS) or during transfer between these facilities. It was assumed that larger commercial spent nuclear fuel transportation casks are used for intersite and intrasite movement of foreign research reactor spent nuclear fuel within the United States. Their capacity is approximately four times that of the foreign research reactor spent nuclear fuel transportation casks from overseas. It was also assumed that the transfer cask for the dry cask design has a capacity which is approximately eight times that of the overseas foreign research reactor spent nuclear fuel transportation casks.

F.6 Evaluation Methodologies and Assumptions for Incident-Free Operations and Hypothetical Accidents at Management Sites

Appendix F.6 describes only the methodologies and assumptions used for estimating radiation exposure (doses) to individuals and the general public from releases of radioactivity during incident-free operations and hypothetical accidents at potential management sites. The descriptions of similar evaluations for ground and marine transportation and port accidents are documented in Appendix E and Appendix D.

F.6.1 Analysis Methods for Evaluation of Radiation Exposure

F.6.1.1 General

An evaluation of incident-free operations and hypothetical accidental radioactive material releases at the proposed storage sites was performed to assess the impact of possible radiation exposure to individuals and the general population. The analysis assumes that the same operations are being carried out at different potential storage locations. The impact of the same radioactive material releases was evaluated at all potential sites. This approach provides a consistent method for comparing the effects of the proposed alternative actions.

F.6.1.2 Exposure Impacts to Be Estimated

The impact of radiation exposure (dose) to the following individuals and the general population is calculated for incident-free operation of the spent nuclear fuel storage facility and for accident conditions:

- ***Worker:*** An individual located 100 m (330 ft) from the radioactive material release point.[2] The dose to the worker is calculated for the 50th-percentile meteorology only (DOE, 1992a).
- ***MEI:*** A theoretical individual living at the storage site boundary and receiving the maximum exposure.
- ***NPAI:*** At some storage sites, highways used by the public may cross the Federal reservation where foreign research reactor spent nuclear fuel operations could be conducted. Consequently, these analyses included evaluation of the exposure to a theoretical motorist who might be stranded on such a highway at the time of an accident. Based on experience from emergency exercises, emergency response teams would be able to evacuate such an individual within 2 hours, so this was the exposure time used in the calculations.
- ***General population*** within an 80 km (50 mi) radius of the facility.

The doses to the NPAI, MEI and general population are calculated for the 50th- and 95th-percentile meteorological conditions. The details of exposure times for MEI, NPAI, worker, and general public are given in Section F.6.4.1.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using the following potential pathways:

- external direct exposure from immersion in the airborne radioactive material (air immersion),
- external direct exposure from radioactive material deposited on the ground (ground surface),
- internal exposure from inhalation of radioactive aerosols and suspended particles (inhalation), and
- internal exposure from ingestion of contaminated terrestrial food and animal products (ingestion).

The radiation dose is estimated by the GENII (Version 1.485) computer program (Napier et al., 1988) in a manner recommended by the International Commission on Radiological Protection in Publications 26 and 30 (ICRP, 1977; ICRP, 1979-1982). Committed dose equivalents [3] are calculated for organs such as the gonads, breasts, red bone marrow, lungs, thyroid, bone surface, liver, lower large intestine, upper large

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2 *For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (300 ft) for elevated release.*

3 *The definitions of committed dose equivalents, committed effective dose equivalents, and total effective dose equivalents are consistent with those given in 10 CFR Part 835, "Occupational Radiation Protection; Final Rule," (DOE, 1993a).*

intestine, small intestine, and stomach. Weighting factors are used for various body organs to calculate weighted or committed effective dose equivalent (EDE) from radiation inside the body due to inhalation or ingestion. The committed EDE value is the summation of the committed dose equivalent to the specific organ weighted by the relative risk to that organ compared to an equivalent whole-body exposure.

The program also estimates deep-dose equivalent for the external exposure pathways (immersion in the radioactive material and exposure to ground contamination) and a 50-year committed EDE for the internal exposure pathways. The sum of the deep-dose equivalent for external pathways and the committed EDE for internal pathways is called the cumulative dose or "total EDE" in this EIS and is also estimated by the GENII program.

The exposure from ingestion of contaminated terrestrial food and animal products is calculated on a yearly basis. However, it is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose exceeds the protective action guidelines for use in the event of radiological accidents (EPA, 1991). No reduction of exposure due to protective actions was accounted for in this analysis, however. This results in a conservative approach that may overestimate health effects within an exposed population, but allows for consistent comparisons between alternatives.

F.6.1.3 Evaluation of Health Effects

Health effects are calculated from the exposure results. The risk factors used for calculations of health effects are taken from International Commission on Radiological Protection Publication 60 (ICRP, 1991) (see Table F-104). From this list only the factors associated with the fatal cancers were used in the analysis. Other factors are given as additional information for completeness.

Table F-104 Risk Estimators for Health Effects from Ionizing Radiation

Effect	*Risk Factor (probability/rem)*[a]	
	Worker	*General Population*
Fatal cancer (all organs)	4.0×10^{-4}	5.0×10^{-4}
Weighted nonfatal cancer[b]	8.0×10^{-5}	1.0×10^{-4}
Weighted genetic effects[b]	8.0×10^{-5}	1.3×10^{-4}

[a] *For high individual exposures (20 rem), the risk factors are multiplied by a factor of two.*

[b] *In determining a means of assessing health effects from radiation exposure, the International Commission on Radiological Protection has developed a weighting method for nonfatal cancers and genetic effects. These factors are provided here for information only and were not used in this analysis. Genetic effects can only be applied to population, not to individuals.*

F.6.1.4 Population

Population distributions specific to each site were used for the evaluations. The population distributions were obtained from each site. The population information was obtained in 16 compass directions and 10 radial distances from the likely location of a foreign research reactor spent nuclear fuel storage facility to an 80 km (50 mi) total distance.

F.6.1.5 Meteorology

Meteorology specific to each site was used in the evaluation. The site-specific meteorological data was prepared, or acquired from each candidate storage site, in the form of joint frequency distribution in terms

of percentage of time that the wind blows in specific directions (i.e., south, south-southwest, southwest, etc.) for the given midpoint (or average) wind speed class and atmospheric stability. Accident consequence calculations were performed using 50th- and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95th-percentile condition represents relatively low probability meteorological conditions which produce higher calculated exposures, and is defined as that condition which is not exceeded more than 5 percent of the time. GENII determines 50th- and 95th-percentile meteorological conditions using site-specific joint frequency distribution weather data.

F.6.1.6 Computer Programs

The following computer programs were used to evaluate the radiation exposure to the specified individuals and the general population.

GENII: The GENII code (Napier et al., 1988) was used to model both acute and chronic releases to the atmosphere. This code was developed by the Pacific Northwest Laboratory to incorporate the internal dosimetry models recommended in International Commission on Radiological Protection Publication 26 (ICRP, 1977) and Publication 30 (ICRP, 1979-1982) into environmental pathway analysis models in use at the Pacific Northwest Laboratory. This code has been used by the Pacific Northwest Laboratory and other laboratories in site-wide dosimetry calculations. It has been extensively validated and quality assured.

ORIGEN2: ORIGEN2 (Croff, 1980) is a computer code system for calculating the buildup and decay of radioactive materials (fission products, actinides, and activation products). The code input was modeled to describe the HEU and low enriched uranium (LEU) research reactor nuclear fuel system and used neutronic cross-section data that are distinct to these fuels. ORIGEN2 has been used extensively by the Argonne National Laboratory in the RERTR program in estimating nuclide inventories of irradiated fuels. The code and the specific neutronic cross-section parameters for HEU and LEU fuels were acquired from the Argonne National Laboratory. ORIGEN2 is widely used and accepted throughout the nuclear industry.

F.6.2 Screening/Selection of Accidents for Detailed Examination

Accidents considered for inclusion in the detailed analyses are similar to those analyzed in the Programmatic SNF&INEL Final EIS for the spent nuclear fuel storage facility operations (DOE, 1995g). The analyzed accident scenarios in the Programmatic SNF&INEL Final EIS for each potential storage site were reviewed to identify the bounding accidents to be considered in this EIS. The review included accidents initiated by natural phenomena (earthquakes, tornadoes, hurricanes, etc.) and accidents initiated from human or equipment failure (fires, explosions, aircraft crashes, transportation accidents, and terrorism).

A review of accidents indicates that only severe accident conditions could result in a release of radioactive material to the environment or an increase in radiation levels. Some types of accidents, such as procedure violations, spills of small volumes of water containing radioactive particles, and most other types of common human error may occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material or radiation to result in a significant release to the environment or a meaningful increase in radiation levels. Stated another way, the very low consequences associated with these events produce smaller risks than those for the accidents analyzed, even when combined with a higher probability of occurrence. Consequently, they have not been included in the results presented in this EIS.

Accidents initiated at nearby facilities, either by other activities unrelated to spent nuclear fuel handling or storage or during construction of a wet or dry storage facility, would not produce effects more severe than the sequence of events being analyzed. This is because foreign research reactor spent nuclear fuel undergoing examination or in the process of being stored would not need special conditions or uninterrupted operator attention to prevent overheating or to maintain containment or shielding. Therefore, evacuation in response to an accident at some other facility would not compromise integrity of the spent nuclear fuel.

The potential for common-cause accidents at a storage facility has been considered. It is possible for natural phenomena, like an earthquake, to produce more than one accident at a site causing a situation that results in a release of radioactive material into the atmosphere or an increase in radiation levels due to loss of shielding. However, the probability of two or more accidents having maximum consequences occurring concurrently is less than the probability of the individual events. For example, if an earthquake affected the wet storage facility, a crane might fail causing damage to stored spent nuclear fuel, and the water pool might drain. The impacts for this could be conservatively estimated by summing the consequences. Similarly, consequences from spent nuclear fuel facilities within a DOE site could be combined to conservatively estimate site-wide impacts. But again, the probability of a common-cause event resulting in this number of consequences is lower than the probability of individual accidents because, due to separation distances, the severity of impact will vary between facilities. The existing security measures in effect at the management sites would essentially preclude any sabotage or terrorist activity. Further, any acts of terrorism are expected to result in consequences which are bounded by the results of accidents analyzed. Thus, no specific analyses of the results of terrorist acts were conducted.

Based on the above, the review identified the following bounding accident scenarios:

- criticality caused by human error during operation, equipment failure, or earthquake;
- mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a fuel); and
- accident involving an impact by either an internal or external initiator with and without an ensuing fire.

F.6.3 Accident Scenarios Considered

A total of six bounding accident scenarios for the handling and storage of foreign research reactor spent nuclear fuel were identified for detailed analysis. Each of these accident scenarios was evaluated at each storage location using identical source terms. As described below, three of the bounding accident scenarios apply to wet storage and three apply to dry storage.

F.6.3.1 Wet Storage Bounding Accident Scenarios

Three hypothetical accident scenarios were evaluated for foreign research reactor spent nuclear fuel stored in water pools: (a) fuel element breach (i.e., cutting into the fuel region) or mechanical damage due to operator error, (b) an accidental criticality, and (c) an aircraft crash into the water pool facility. In addition to these three scenarios, a dropped fuel cask was also considered to be a foreseeable accident. However, as will be seen in Section F.6.4.4.4, the consequences of this accident are bounded by the cutting into a fuel region scenario. Therefore, a dropped fuel cask was not evaluated in detail.

F.6.3.2 Dry Storage Bounding Accident Scenarios

Three hypothetical accidents were evaluated for foreign research reactor spent nuclear fuel handled in dry storage: (a) fuel element breach (i.e., cutting into the fuel region) or mechanical damage during examination work and handling, (b) dropping of a fuel cask, and (c) an aircraft crash with ensuing fire in the dry storage facility. No credible mechanism was identified for an accident criticality in dry storage.

F.6.4 Bounding Accident Evaluation

F.6.4.1 Basic Assumptions

The analysis of airborne releases from hypothetical accidents is performed using the GENII Version 1.485 computer program. Unless otherwise stated, the following conditions were used when performing calculations. In most cases, these are the default conditions in the GENII program.

Meteorological Data:

- Fiftieth- and 95th-percentile meteorological conditions for each storage site were defined using site-specific joint frequency distribution weather data.
- The release is assumed to occur at ground level (0 m).
- Mixing layer height is 1,000 m (3,280 ft). Airborne materials freely diffuse in the atmosphere near ground level in what is known as the mixing depth. A stable layer exists above the mixing depth which restricts vertical diffusion above 1,000 m.
- Wet deposition is zero (it is assumed that no rain occurs to accelerate deposition and reduce the size of area affected by the release).
- Dry deposition of the cloud is modeled. During movement of the radioactive plume, a fraction of the radioactive material in the plume is deposited on the ground due to gravitational forces. The deposited material no longer contributes to the air immersion dose from the plume, but now contributes as exposure from ground surface radiation and ingestion.
- The quantity of deposited radioactive material is proportional to the material particle size and deposition velocities (in m/sec) used in the GENII code as follows:

solids = 0.001	halogens = 0.01	noble gases = 0.0
cesium = 0.001	ruthenium = 0.001	

- If radioactive releases occur through a stack, then additional plume dispersion can be accounted for by considering the beneficial effects of jet plume rise. In this analysis, jet plume rise is ignored.
- When released gases have a heat content, the plume can disperse more quickly. In this calculation, buoyant plume effects are ignored.

Inhalation Data:

- Breathing rate is 330 cm^3/sec (20.1 in^3/sec) for the worker and the NPAI; 270 cm^3/sec (16.5 in^3/sec) for people at the site boundary and beyond (the MEI and the general population).
- Particle size is 1.0 micro-meter (micron).
- The internal exposure period is 50 years for the individual organs and tissues evaluated.
- Exposure during passage of the entire plume is assessed for the MEI and the general public. Exposures to the worker and NPAI are discussed below.
- Inhalation exposure factors are based on International Commission on Radiological Protection Publication 30 (ICRP, 1979-1982).

Mitigating Factors:

For the MEI and members of the general public residing at the site boundary and beyond, no allowances are made for any preventive or mitigative actions that would limit their exposure. These individuals are assumed to be exposed to the contaminated plume during the entire period of its passage, as it travels downwind from the accident site. Similarly, no action is taken to prevent these people from continuing their normal daily routine, including ingestion of the potentially contaminated terrestrial food and animal products. It is assumed, however, that the public would spend approximately 30 percent (about 8 hours) of the day within their homes or other buildings. Therefore, the exposure of the general public to radiation from contaminated ground surface is reduced appropriately. Calculations were done on a yearly basis to determine the effective annual dosage from inhalation, external exposure, and ingestion, and an associated dose commitment extending over a 50-year period from initiation of intake (NRC, 1977a).

Onsite workers would be trained to take quick, decisive action during an accident. These individuals would be trained to quickly evacuate the affected area and move to well-defined "relocation" areas on the facility. Therefore, it is assumed that workers would be exposed to only 5 minutes of the radioactive plume as they move to relocation centers. Once the plume has moved offsite and downwind, the workers would be instructed to walk to vehicles waiting to evacuate them from the site. It is assumed that an additional 15 minutes would be required to evacuate the workers from the contaminated area and, therefore, the workers would receive a total of 20 minutes of exposure to radioactive material deposited on the ground. No ingestion of contaminated foods is assumed for these individuals.

Individuals that may be traversing the site in a vehicle (i.e., NPAI) would be evacuated from the affected area within 2 hours. This is based on the availability of security personnel at all locations to oversee the removal of collocated workers and travelers in a safe and efficient manner. Therefore collocated workers and travelers would be exposed to the entire contaminated plume as it travels downwind for a period not to exceed 2 hours. Similarly, the radiation from the deposited radioactive materials would be limited to a 2-hour period. No ingestion of contaminated foods is assumed for these individuals.

Table F-105 provides the individual exposure times used in the accident analyses presented later in this appendix.

Table F-105 Estimated Individual Exposure Times

Exposure Type	*Worker (100 m)*	*NPAI*	*MEI/General Public*
To Plume	5 min	100% of release time up to 120 min	100% of release time
To Fallout on Ground Surface	20 min	120 min	0.70 yr
To Food	NA	NA	1 yr

F.6.4.2 Source Term

The source term is the amount of respirable radioactive material, in terms of Ci (curies), that are released to the air. The airborne source term is typically estimated by the following five-component linear equation:

Source Term = MAR x DR x ARF x RF x LPF

where:

MAR = Material-at-Risk (g or Ci),

DR = Damage Ratio,

ARF = Airborne Release Fraction (or Airborne Release Rate for continuous release),

RF = Respirable Fraction, and

LPF = Leak Path Factor.

***MAR*:** The MAR is the amount of radionuclides (in g or Ci of activity for each radionuclide) available to be acted upon by a given physical stress (i.e., an accident). The MAR is specific to a given process in the facility of interest. It is not necessarily the total quantity of material present, but is that amount of material in the scenario of interest postulated to be available for release.

***DR*:** This is the fraction of material exposed to the effects of the energy/force/stress generated by the postulated event. For the bounding accident scenarios discussed in this document, the value of DR is assumed to be one (i.e., all exposed material is released), unless otherwise specified.

***ARF*:** This is the fraction of the material that becomes airborne due to the accident. Generic ARF values from DOE sources (Elder et al., 1986; DOE, 1994d) are used in this document unless other values more appropriate to a particular accident scenario are used for ARF. The values for ARF are summarized in Table F-106.

***RF*:** This is the fraction of the material, with particle sizes of 10 micro-meters (microns) or less (DOE, 1994d) that could be retained in the respiratory system following inhalation. The term RF is applied only for the inhalation pathway.

***LPF*:** The LPF accounts for the action of removal mechanisms, such as containment systems, filtration, deposition, etc., to reduce the amount of airborne radioactivity that is ultimately released to occupied spaces of the facility or to the environment. An LPF of 1.0 (i.e., no reduction) is assigned in accident scenarios involving a major failure of confinement barriers.

Table F-106 Release Fractions[a] for Various Release Mechanisms

Material	*Release Mechanisms*		
	Fuel breach[a]	*Fire*	*Criticality Accident*[b]
Gas			1.0
Noble Gas	1.0	1.0	
Krypton	0.3	1.0	
Other Noble Gas	0.1	1.0	
Halogens	0.1	1.0	0.25[d]
Iodine-129	0.25		
Solids			
Volatile	0.01[c]	2.5×10^{-4} [d,e]	
Nonvolatile	0.01	2.5×10^{-6} [d,e]	

Source: DOE, 1995g

[a] *As recommended in Elder et al., 1986.*

[b] *Regulatory Guide values (NRC, 1977b, 1979b, and 1988b).*

[c] *Actually semi-volatile (cesium, rhodium, antimony, selenium, technetium, and tellurium); review on a case-by-case basis.*

[d] *Includes release fraction, respirable fraction and plate-out.*

[e] *Data from DOE, 1995g.*

F.6.4.3 Description of Radiological Accident Scenarios and Generic Parameters

As discussed previously, the accident screening and selection process led to selection of six bounding accident scenarios involving radioactive materials. Appropriate assumptions also have been discussed regarding meteorological parameters, dispersion parameters, dose estimates, and emergency response and protective actions. Each of the accident scenarios is described in the following text according to the major headings listed below:

- Description of Accident,
- Development of Radioactive Source Term, and
- Dose Calculations and Results.

The contents of these sections and a summary of the generic parameters used follow.

Description of Accident provides a basis for accident selection and discusses possible initiating events. A qualitative assessment of scenario likelihood is provided.

Development of Radioactive Source Term describes the assumptions that apply to the development of the resulting source term. Specifically, it discusses the various multipliers (defined earlier in this section) that convert the MAR to the source term.

These multipliers have the following values:

- DR is 1.0, unless otherwise specified.
- ARF is taken from Table F-106, or clearly stated if different.
- LPF is 1.0 for a major failure of confinement barriers.

Dose Calculations and Results relates the computer modeling to the specific accident scenario, and documents the results. Specifically, these subsections accomplish the following:

- describe assumptions and unique input parameters (other than the source term) used in the computer model,
- document the computer model output in terms of exposure to radionuclides for individuals and for the general population within a 80 km (50 mi) radius, and
- assess the potential for health effects.

Unless otherwise specified, the meteorological/dispersion parameters and estimated exposure times summarized are used in the dosimetry calculations for specific accident scenarios. Under some circumstances, facility worker exposures could be either greater or less than these nominal values.

F.6.4.4 Accident Scenario Descriptions and Source Terms

F.6.4.4.1 Fuel Element Breach

***Description of Conditions*:** Fuel element mechanical damage due to handling during examination, such as accidentally cutting into the fuel region, was assessed. This hypothetical accident results from inadvertent cutting across the fuel region when cropping off the aluminum and nonfuel ends of a fuel unit. All noble gas isotopes are postulated to be released to the facility building and escape to the environment. The majority of the volatile and solid nuclides are likely to be retained in the fuel or the facility exhaust filters. The resulting airborne release to the environment was evaluated.

***Likelihood*:** The frequency of this scenario is estimated to be 0.16/yr (DOE, 1995g). This frequency estimate is based on historical operation data (one event in 6 years) for a spent nuclear fuel storage facility. This estimate is conservative for the case of foreign research reactor spent nuclear fuel storage because the majority of the spent nuclear fuel elements are expected to be cropped prior to their emplacement in a transportation cask at a foreign research reactor. Nevertheless, this estimate is retained for the evaluation of the potential risk associated with the handling and preparation of foreign research reactor spent nuclear fuel for storage in both a dry and a wet storage facility.

***Source Term*:** Conditions used in developing the source term are as follows:

- Only one spent nuclear fuel element is damaged. This is because only one spent nuclear fuel element is being handled at a time.

If the spent nuclear fuel cutting accident occurs in a dry cell (dry storage), the following assumptions apply:

- All (100 percent) of the noble gases available for release are released to the atmosphere. Here, it was assumed that all noble gases in an irradiated fuel element would be released. This is conservative, since foreign research reactor fuels are dispersion fuels in which the gaseous fission products are essentially trapped within the fuel matrix. This is different than for commercial reactor fuel, where gaseous fission products collect in the gap between the fuel and its sealed metal fuel rod and are readily released if the rod is damaged.

- Twenty-five percent of the halogens in the spent nuclear fuel are released to the environment. This is also conservative for the reason stated above.
- One percent of the particulate fission products is released to the dry cell from the spent nuclear fuel element, and 99.9 percent removed prior to release to the environment by the normally installed high-efficiency particulate air filters. The use of 99.9 percent efficiency is conservative, since normal efficiency of installed high-efficiency particulate air filters is greater than 99.99 percent.
- Cesium (Cs) and Ruthenium (Ru) behave like particulate fission products.
- The release to the environment occurs at a constant rate over a 15-minute period.

If the spent nuclear fuel cutting accident occurs under water (wet storage) the following assumptions apply:

- All (100 percent) of the noble gases available for release are released to the environment.
- Twenty-five percent of the halogens available for release will be released to the pool, and only 10 percent of this amount will be released to the air. This additional reduction is due to the fact that halogen gases dissolve in the water as they escape (leak out) from the failed fuel. Based on solubility alone, it is expected that all iodines are dissolved in the water pool before they get to the pool surface. In spite of this fact, for the purposes of the analyses, it was assumed that 2.5 percent of halogens available for release will be released to the atmosphere.
- There is no particulate fission product release to the environment. All particulates are retained in the pool water.
- Since only gaseous fission products are released to the air inside the facility, installed high-efficiency particulate air filters would not provide additional reduction in the amount of material released to the environment.
- The release to the environment occurs at a constant rate over a 15-minute period.

F.6.4.4.2 Accidental Criticality

Description of Conditions: In this hypothetical accident scenario, an accidental uncontrolled chain reaction producing 1 x 10^{19} fissions is postulated. The 10^{19} fission criticality is a very conservative assumption for the spent nuclear fuel pool. This assumption is only applicable to liquid processes (such as uranium reprocessing) as stated in Regulatory Guides 3.33 and 3.34 (NRC, 1979a and 1979b). This criticality is assumed to consist of an initial burst of 10^{18} fissions in 0.5 seconds, followed at 10 minute intervals for the next 8 hours by a burst of 2 x 10^{17} fissions, for a total of 10^{19} fissions. The total yield for a moderated solid system, as applicable to the spent nuclear fuel in a wet pool, is estimated to be on the order of 10^{18} fissions. This is because the initial criticality will disrupt the critical geometry and no further criticality burst will occur.

The criticality occurs in the water pool and the spent nuclear fuel remains covered in the water. The fission products released include those specified in Regulatory Guide 3.34 (NRC, 1979b) from the criticality over an 8-hour period, plus fission products existing in the fuel as a result of its original use in the foreign research reactor. Removal of fission products by the pool water is considered in the analysis.

Criticality is not considered in the dry storage because the licensing design basis for spent nuclear fuel dry storage design facilities precludes the consideration of any criticality accident by design. The design must demonstrate, through rigorous structural and criticality analyses, that the likelihood of a criticality is incredible or unforeseeable. No effective moderator, such as water, exists in a dry storage design; and, even if flooded, it remains subcritical.

***Likelihood*:** The frequency of this scenario is estimated at 3.1 x 10^{-3} per year (DOE, 1995g). The estimation of this frequency was conservatively based on a statistical evaluation considering that no accidental criticality event with spent nuclear fuel storage has occurred (DuPont, 1983b). This frequency is estimated by considering both the various process-related upset conditions and the natural phenomena hazard (i.e., earthquake and tornadoes) initiated criticality events. The magnitude of fission yield for such a criticality accident was estimated to range from about 5 x 1017 to 1 x 10^{19} fissions. The historical criticality accidents at different DOE facilities dealing with spent nuclear fuels indicate a much smaller fission yield than that evaluated here. The frequency of an accidental criticality of the magnitude evaluated here is estimated to be between one and two orders of magnitude less than the estimated frequency.

***Source Term*:** Conditions used in developing the source term are as follows:

- The fractions of the fission products from damaged spent nuclear fuel elements released to the building are 100 percent of the noble gases, 25 percent of the halogens, 0.1 percent of the Ru, and 0.05 percent of the Cs and remaining solids (NRC, 1977b, 1979b, and 1988b).
- Fission products from 10 spent nuclear fuel elements damaged in the criticality accident are also released in addition to the gaseous fission products created by the criticality event.
- A high-efficiency particulate air filter removes 99.9 percent of the solid fission products that were released to the air inside the facility before they enter the environment.
- The release to the environment occurs at a constant rate over a 15-minute period. This is conservative as compared to the 8-hour release allowed in Regulatory Guide 3.34 (NRC, 1979b).

F.6.4.4.3 Aircraft Crash

Dry Storage:

***Description of Conditions*:** A hypothetical aircraft accident scenario was developed for the dry storage option. This accident is analyzed only at storage sites that have a likelihood of accident occurrence greater than 10^{-7} per year. The consequences of this accident are expected to bound all other dry storage accident scenarios involving an impact that results in fire. The aircraft crash accident is postulated to cause damage to a single transfer container in the dry unloading cell in a modular vault storage facility. Engineering experience indicates that most of the aircraft structure is stopped by the dry storage building structure. Only a heavy dense jet engine rotor shaft is expected to be capable of penetrating the building and damaging the container. Due to the severity of the impact, it was assumed that the cask is breached and the fuel elements in the cask are damaged. The release of fission products occurs due to the impact and resultant fire (i.e., from aviation fuel).

The accident scenario for a dry cask storage facility is similar to that of a modular vault facility. The aircraft crash analysis is the only accident scenario applicable to a dry cask storage. In this scenario, it is

expected that the concrete structure which houses the storage canisters is sufficiently rugged that it can survive an aircraft accident with no significant damage to the spent nuclear fuel.

***Likelihood*:** The frequency of this scenario is site dependent. DOE, as part of the Programmatic SNF&INEL Final EIS, has performed calculations of aircraft crash hit frequencies at potential storage sites (i.e., Savannah River Site, Idaho National Engineering Laboratory, Oak Ridge Reservation, Hanford Site, and Nevada Test Site) for naval fuel (DOE, 1995g). The reported crash frequencies are: 2×10^{-6} per year for the Savannah River Site, 1×10^{-6} per year for the Oak Ridge Reservation, 4×10^{-7} per year for Nevada Test Site, 7×10^{-8} per year for the Idaho National Engineering Laboratory, and 4×10^{-8} per year for the Hanford Site. These frequency estimates were based on the number of commercial air carriers and military aircraft passing within a 10-mile radius of the proposed storage location at these sites. The calculations for the Idaho National Engineering Laboratory also included potential hazards from a nearby airport. These calculations were performed very conservatively, by considering that all the overflights within the 10-mile radius will pass directly over the storage location at each site.

A new assessment of aircraft impact probabilities for the Idaho National Engineering Laboratory chemical processing plant indicates a frequency of aircraft crash into a dry storage facility the size of the IFSF of about 2.6×10^{-10} per year from overflights and 3.5×10^{-7} per year from airport-related flights near the plant (WINCO, 1994). (The IFSF effective area is five times that considered in the evaluation for the naval fuel storage area, which represents the critical areas containing spent nuclear fuel. Therefore both results are consistent, from the overall crash frequency point of view at the Idaho National Engineering Laboratory.)

In order to provide an understanding of the rationale used in this EIS for this scenario, an overview of the aircraft crash analysis approach is presented. In general, the aircraft crash hit frequency is calculated based on four factors: number of flight operations (takeoff, landing, overflight), aircraft crash rate, facility effective area, and an assumption of crash area distribution. Several models are currently used to estimate the hit frequency. The results of these models are driven by the assumptions regarding the target area and crash area distribution. For example, assuming that overflights (high or low altitude) pass over the facility inherently assumes that the crash area distribution is a straight line. This overestimates the frequency by at least a factor of 10 (approximate width of an airway). In calculating effective area, the analysis considers that an aircraft can hit a facility either directly (falling on the building, footprint area), by skidding into the building (skid area), or in an angular impact (shadow area). Depending on the assumptions of skid length and the angular approach of a crash terminating aircraft, the sum of the latter two areas may contribute between 80 to 95 percent of the total effective area. It is important to note that aircraft that fall vertically with the greatest impact contribute between 1 and 10 percent to the overall crash rate. Therefore, for the majority of cases, the aircraft will hit the ground before it hits the facility.

Based on the above summary, it is considered that frequencies reported in the Programmatic SNF&INEL Final EIS are conservative by at least a factor of 10 for all sites except the Idaho National Engineering Laboratory. Nonetheless, for the purposes of analyses and consistency, this EIS will consider frequencies similar to those used in the Programmatic SNF&INEL Final EIS. The potential aircraft crash frequency at the Oak Ridge Reservation, the Nevada Test Site, the Idaho National Engineering Laboratory, and the Savannah River Site is conservatively set at 10^{-6} per year. This scenario will not be applicable to the Hanford Site, where the estimated frequency is less than 10^{-7} per year.

***Source Term*:** Conditions used in developing the source term are as follows:

- Only one transfer cask containing 20 spent nuclear fuel elements would be damaged by the impact and the resultant fire. This is based on the fact that, if an aircraft hits the building,

only the transfer cask is susceptible to damage by the crash. The stored casks are protected by a three-foot concrete shield, and therefore would not be affected by the crash. Based on a conservative estimate of the duration of the transfer operation, the transfer cask could be damaged by the accident only one percent of the time.

- Of the available fission products, 100 percent of the noble gases, 100 percent of the halogens, 2.5 percent of the cesium, and 0.025 percent of the remaining solids are released to the environment. The overall, respirable fractions of fission products released to the environment are consistent with that given in Table F-106 for a fire scenario.

- The release to the environment occurs at a constant rate over a 15-minute period.

- No filtration by high-efficiency particulate air filters is assumed.

For dry cask storage, it was assumed that the ruggedness of the overall dry cask structure is similar to that of a transportation cask. Based on this assumption, the accident source terms were assumed to be similar to that of aluminum-based spent nuclear fuel source terms for the highest severity accident (cask damage and fire) utilized in the RADTRAN accident analysis (DOE, 1995g). The overall source terms for this scenario include: 63 percent of noble gases, 6×10^{-3} percent of halogens, 1×10^{-3} percent of cesium, 2.4×10^{-4} percent of ruthenium, and 1×10^{-5} percent of other solid fission products available in a dry cask.

Wet Storage:

***Description of Conditions*:** Impact into water pools by aircraft with resulting damage to the spent nuclear fuel elements stored inside the pool was evaluated. The hypothetical accident might damage the fuel either by the aircraft directly striking it or by the aircraft causing sufficient damage to the building to cause part of the building to collapse and strike the fuel. Fission products are released from the spent nuclear fuel units into the water pool, however, the pool water is not released to the environment. An aircraft crash into a water pool would not produce enough force to cause the pool to leak because the walls of the water pool are constructed of thick reinforced concrete with earth surrounding them, making them very strong. In addition, based on the discussion provided above, it was judged unlikely that an aircraft would impact the water pool at an angle steep enough to expose the floor of the pool or the walls of the pool below the water level to direct impact.

***Likelihood*:** The same frequency as discussed above will be used for an aircraft crash into a wet storage facility.

***Source Term*:** Conditions used in developing the source term are as follows:

- It was estimated that about 140 spent nuclear fuel elements would damaged. This estimate was based on the consideration of the size of spent nuclear fuel allowing fuel stacking and an assumption that only one percent of the upper stacked fuel will be damaged.

- Of the available fission products, 100 percent of the noble gases and 25 percent of the halogens are released to the pool water. Due to the presence of pool water, a reduction of the halogen release by a factor of 10 occurs prior to release to the environment.

- The pool water is not expected to be lost and the solid fission products from ruptured and damaged fuel elements remain in the water. However, for the purposes of this analysis, it was conservatively assumed that 0.01 percent of the solid fission products (including Cs and Ru) released from the damaged fuel elements to the pool would be displaced upon

impact. Only one percent of released solid fission products would become airborne and released to the environment. This assumption considers that, upon impact, a percentage of the spent nuclear fuel fails, the solid fission products enter the pool, and only finely crushed particulates are splashed out of the pool in the same timeframe that the aircraft hits the water.

- The release to the environment occurs at a constant rate over a 15-minute period.
- Spent nuclear fuel elements remained covered in the water pool.
- The building confinement is assumed to have failed; no filtration by high-efficiency particulate air filters is assumed.

F.6.4.4.4 Fuel Cask Drop

Dry Storage:

Description of Conditions: Mechanical damage due to handling during examination, such as dropping of the spent nuclear fuel cask during transfer, was assessed. The fuel casks are certified to result in no failure for a specific drop height, (free drop from 9 m [30 ft] height onto an unyielding surface), and under no circumstances will the cask be moved above such height during operations within a storage facility. Nevertheless, it was assumed that, upon cask drop, the seals of the cask would fail, releasing the gaseous fission products from the damaged fuel inside the cask to the facility building and the environment. All of the nonvolatile and solid nuclides are assumed to be retained in the fuel or the facility high-efficiency particulate air filters. The resulting airborne release to the environment was evaluated.

Likelihood: The frequency of this scenario is estimated at 10^{-4} per year (DOE, 1995g). This estimate is considered to be an upper bound for this scenario.

Source Term: Conditions used in developing the source term are as follows:

- Only one fuel cask is involved. This is because only one fuel cask is being handled at a time. For the purposes of this analysis, it was assumed that an equivalent of one spent nuclear fuel element inside the cask is damaged, and its gaseous fission products are released inside the cask. This assumption is conservative, since the fuel is secured inside the cask and the cask is not expected to be damaged.
- All (100 percent) of the gaseous fission products and 25 percent of halogens from the damaged fuel element are released to the atmosphere.
- None of the particulate fission products are released to the environment.
- Cs and Ru behave like particulate fission products.
- The release to the environment occurs at a constant rate over a 15-minute period.

Wet Storage:

The source term for a fuel cask drop is similar to that for the fuel element breach scenario in a wet storage facility. The gaseous fission products released inside the cask are vented under

water (or in the pool). Since the estimated frequency of this scenario is less than that of the fuel element breach, no specific analysis for this scenario was performed.

F.6.5 Incident-Free Operation Source Terms

This section details the assumptions and the evaluation process used to determine the risk of radiological emissions generated during different activities in incident-free operation of a storage facility. The incident-free operation emissions consist of two parts: transient (i.e., emissions from gaseous release during receipt and unloading of the transportation casks), and steady state (i.e., emissions from spent nuclear fuel in storage). Since only mechanically sound spent nuclear fuel elements are shipped, no radioactive releases are expected during transit. To ensure this, the spent nuclear fuel elements are checked prior to shipment to identify and separate any damaged fuel elements. The damaged fuel elements are then encapsulated and prepared for shipment. In spite of the fact that no spent nuclear fuel elements have ever failed during transit, it was assumed that one percent of the spent nuclear fuel elements will arrive failed and release gaseous fission products (noble gases and halogens) into the cask. Depending on the type of storage facility, the receipt and unloading of the transportation casks could occur in a dry cell or a wet pool. Unloading operations in a dry cell causes all gaseous fission products to be released to the building and eventually to the environment. If the unloading process occurs in a wet pool, a majority of the halogen gases will be absorbed in the water; only 10 percent of halogens will be released to the environment. The building high-efficiency particulate air filters will not be effective for halogens and noble gases. During the unloading process, all spent nuclear fuel elements are checked to ensure that they are mechanically sound. If a damaged fuel element is found, it is encapsulated in a can before it is placed in wet or dry storage. The potential annual radiological releases from failed fuel elements during the unloading process were estimated based on the gaseous inventories of bounding fuels (see Appendix B, Section B.1.4) ad the associated number of fuels expected over the acceptance period. The receipt and unloading process of foreign research reactor spent nuclear fuel from abroad is expected to last 13 years (see Section 2.2.1). It was assumed that failed fuel would release 100 percent of its noble gases and 25 percent of its halogens. This assumption is consistent with that used in the accident analysis.

The steady state emissions from a new wet storage facility are assumed to be similar to those released from the RBOF facility at the Savannah River Site. Although the emissions at the RBOF facility may not be a good representation of the foreign research reactor spent nuclear fuel, RBOF has the most foreign research reactor spent nuclear fuel elements stored in its pool; and as such, was considered to provide the best approximation of the expected release. Based on the emission data from RBOF, the steady-state emissions from a wet storage facility are assumed to be about 2×10^{-7} curies of Cesium-137 per year (DOE, 1995g). This is a conservative assumption. For existing wet storage facilities, the radiation exposure to the MEI and the general public were estimated based on the combined radionuclide atmospheric emissions originating from current conditions of the facilities and that expected from foreign research reactor spent nuclear fuel. At Savannah River Site, the average annual atmospheric emissions from the existing fuels at L-reactor disassembly basin are estimated to be 254 curies of tritium and 6.49×10^{-5} curies of Cesium-137 (Shedrow, 1994b) over Phase 1 of the policy period. The assumption is that the foreign research reactor spent nuclear fuel would be stored temporarily (about 10 years) in the wet pool until a more permanent dry storage facility is built. The annual atmospheric radiological emissions from RBOF and BNFP wet pools are similar to those that are currently released from RBOF and which were used for a new facility. The annual atmospheric radiological emissions from the Idaho National Laboratory's FAST wet storage facility were assumed to be similar to that of a new wet storage facility. This facility has been designed and built according to current codes and regulations.

The steady-state emissions from a dry storage facility are considered to be zero. This is because the fuel will be checked to ensure that it is mechanically sound (i.e., no damage) before it is placed into dry storage, and the dry storage canisters that house the fuel are sealed.

F.6.6 Dose Calculations and Results

F.6.6.1 Source Terms

Tables F-107 and F-108 provide the incident-free operation and accident source terms. The source terms for the annual emissions from the unloading process were calculated based on the assumption that a constant annual rate of fuel mix with bounding radionuclide inventories (as defined in Appendix B, Section B.1) is received over the acceptance period. The fission products in a BR-2 type spent nuclear fuel element were used as the MAR in the accident analysis source term calculations (see Appendix B for more details). Four fuel categories were defined in Appendix B: BR-2, NRU, RHF, and TRIGA. BR-2 fuel type constitutes the majority of the foreign research reactor spent nuclear fuels. In addition, since spent nuclear fuels come in different sizes and lengths, use of the BR-2 spent nuclear fuel in the accident analysis means involvement of a larger number of spent nuclear fuels in each accident. For example, in the source term calculations for an aircraft crash accident involving a transfer cask, it was assumed that the cask would contain 20 BR-2 spent nuclear fuel elements. If the cask contained NRU elements, there would be five elements in the cask; and if it contained RHF elements, there would be only four elements per cask. For the generic wet storage case, the bounding spent nuclear fuel is considered to have been cooled at least 300 days prior to shipment. In the case of dry storage, the fuel has been cooled for at least 3 years.

Table F-107 Annual Emission Releases From Storage Facilities

	Releases During Unloading in		*Steady State*	
Isotope	*Dry Cell (Ci)*	*Wet Pool (Ci)*	*Dry Storage (Ci)*	*Wet Storage (Ci)*
Tritium	39.6	39.6	---	---
Krypton-85	1.14×10^{3}	1.14×10^{3}	---	---
Iodine-129	4.87×10^{-4}	4.87×10^{-5}	---	---
Iodine-131	9.12×10^{-4}	9.12×10^{-5}	---	---
Xenon-131	2.01×10^{-5}	2.01×10^{-5}	---	---
Cesium-137	---	---	0.0	2.20×10^{-7}

Table F-108 Accident Source Terms (Curies)

	Dry Storage[a]			*Wet Storage*		
Isotope	*Fuel Element Breach*	*Dropped Fuel Cask*	*Aircraft Crash with Fire*	*Accident Criticality*[b]	*Fuel Element Breach*	*Aircraft Crash*
Tritium	2.40	2.12	42.4	24.0	2.40	336
Krypton 85	68.6	59.7	1,190	686	68.6	9,610
Iodine 129	3.04×10^{-5}	2.93×10^{-5}	2.34×10^{-3}	3.04×10^{-4}	3.04×10^{-6}	4.26×10^{-4}
Iodine 131	8.88×10^{-8}		0.0	2.20	8.88×10^{-9}	1.24×10^{-6}
Xenon 131m	3.71×10^{-5}		0.0	8.24×10^{-2}	3.71×10^{-5}	5.19×10^{-3}
Strontium 89	1.13×10^{-2}		1.42×10^{-6}	5.67×10^{-3}		0.159
Strontium 90	5.78×10^{-3}		2.73×10^{-2}	2.89×10^{-3}		8.09×10^{-2}
Yttrium 90	5.78×10^{-3}		2.73×10^{-2}	2.89×10^{-3}		8.09×10^{-2}
Yttrium 91	2.03×10^{-2}		8.35×10^{-6}	1.01×10^{-2}		0.284
Zirconium 95	2.97×10^{-2}		3.31×10^{-5}	1.49×10^{-2}		0.416
Niobium 95	6.11×10^{-2}		7.15×10^{-5}	3.06×10^{-2}		0.856
Ruthenium 103	2.47×10^{-3}		1.12×10^{-8}	2.47×10^{-3}		3.46×10^{-2}
Rhodium 103m	2.47×10^{-3}		1.12×10^{-8}	2.47×10^{-3}		3.46×10^{-2}

	Dry Storage[a]			Wet Storage		
Isotope	Fuel Element Breach	Dropped Fuel Cask	Aircraft Crash with Fire	Accident Criticality[b]	Fuel Element Breach	Aircraft Crash
Ruthenium 106	5.97×10^{-3}		6.70×10^{-3}	5.97×10^{-3}		8.36×10^{-2}
Rhodium 106m	5.97×10^{-3}		6.70×10^{-3}	5.97×10^{-3}		8.36×10^{-2}
Tin 123	1.19×10^{-4}		8.30×10^{-6}	5.93×10^{-5}		1.66×10^{-3}
Antimony 125	2.47×10^{-4}		7.10×10^{-4}	1.24×10^{-4}		3.46×10^{-3}
Tellurium 125m	5.89×10^{-5}		1.74×10^{-4}	2.94×10^{-5}		8.24×10^{-4}
Tellurium 127m	2.46×10^{-4}		1.55×10^{-5}	1.23×10^{-4}		3.45×10^{-3}
Tellurium 129m	5.25×10^{-5}		2.95×10^{-11}	2.63×10^{-5}		7.35×10^{-4}
Cesium 134	4.56×10^{-3}		1.10	2.28×10^{-3}		6.38×10^{-2}
Cesium 137	5.72×10^{-3}		2.73	2.86×10^{-3}		8.01×10^{-2}
Cerium 141	1.59×10^{-3}		3.53×10^{-10}	7.97×10^{-4}		2.23×10^{-2}
Cerium 144	8.67×10^{-2}		6.25×10^{-2}	4.33×10^{-2}		1.21
Praseodymium 144	8.67×10^{-2}		6.25×10^{-2}	4.33×10^{-2}		1.21
Promethium 147	1.34×10^{-2}		3.78×10^{-2}	6.71×10^{-3}		0.188
Promethium 148m	2.10×10^{-5}		1.70×10^{-10}	1.05×10^{-5}		2.94×10^{-4}
Europium 154	1.72×10^{-4}		7.30×10^{-4}	8.61×10^{-5}		2.41×10^{-3}
Europium 155	3.61×10^{-5}		1.32×10^{-3}	1.81×10^{-5}		5.06×10^{-4}
Uranium 234	2.50×10^{-10}		1.80×10^{-9}	1.27×10^{-10}		3.55×10^{-9}
Uranium 235	3.80×10^{-9}		1.90×10^{-8}	1.90×10^{-9}		5.37×10^{-8}
Uranium 238	9.50×10^{-11}		4.70×10^{-10}	4.70×10^{-11}		1.33×10^{-9}
Plutonium 238	1.78×10^{-5}		8.75×10^{-5}	8.92×10^{-6}		2.50×10^{-4}
Plutonium 239	5.11×10^{-7}		2.57×10^{-6}	2.56×10^{-7}		7.16×10^{-6}
Plutonium 240	3.33×10^{-7}		1.68×10^{-6}	1.67×10^{-7}		4.67×10^{-6}
Plutonium 241	7.89×10^{-5}		3.56×10^{-4}	3.94×10^{-5}		1.10×10^{-3}
Americium 241	1.10×10^{-7}		1.85×10^{-6}	5.50×10^{-8}		1.54×10^{-6}
Americium 242m	2.92×10^{-10}		1.44×10^{-9}	1.46×10^{-10}		4.08×10^{-9}
Americium 243	1.20×10^{-9}		6.00×10^{-9}	6.01×10^{-10}		1.68×10^{-8}
Curium 244	3.69×10^{-7}		2.25×10^{-7}	1.85×10^{-7}		5.17×10^{-6}
Curium 242	4.86×10^{-7}		6.40×10^{-8}	2.43×10^{-7}		6.81×10^{-6}
Krypton 83m				160		
Krypton 85m				150		
Krypton 87				990		
Krypton 88				650		
Krypton 89				42,000		
Xenon 133m				1.80		
Xenon 133				27.0		
Xenon 135m				2,200		
Xenon 135				360		
Xenon 137				49,000		
Xenon 138				13,000		
Iodine 132				275		
Iodine 133				40.0		
Iodine 134				1,100		
Iodine 135				120		

[a] *Source terms are those of modular dry vault storage. The dry cask source terms for accident scenarios are the same or smaller than those of modular dry vault storage, therefore, the modular dry vault storage source term values are considered to be bounding values for the impact evaluations.*

[b] *Particulate source terms (from Strontium 89 to Curium 242) are 1000 times higher, if a facility does not have or has an ineffective, high efficiency particulate air filters. This condition is applicable to the Savannah River Site Wet Storage at RBOF and L-Reactor Disassembly Basin.*

The incident-free operation source terms for chemical separation at the Savannah River Site and the Idaho National Engineering Laboratory were taken from the Interim Management of Nuclear Materials Final EIS (DOE, 1995b) and the Programmatic SNF&INEL Final EIS (DOE, 1995g), respectively. Accident source terms for the chemical separation process were not developed for foreign research reactor spent nuclear fuel. It was considered that the consequences of chemical separation operations-related accidental scenarios are similar to those identified and analyzed in the above documents.

F.6.6.2 Site-Specific Parameters

Several site-specific parameters were required as input to the computer models. The site-specific parameters deal with meteorology, individual and general population food consumption rates, food production locations, and distances and directions of individuals and populations with respect to release locations. The food consumption rates apply only to the MEI and the population dose calculations as indicated in Table F-105. Site-specific food consumption rates consistent with those used in the Programmatic SNF&INEL Final EIS (DOE, 1995g) were utilized. Different contaminated food consumption rates were used at each site because the rate at each site is calculated based on the food production rate within an 80 km (50 mi) radius and the amount of supplemental food (uncontaminated food) that is imported from outside of the 80 km (50 mi) radius. If food production around the site is not sufficient for the population consumption rate, then uncontaminated food is imported. Otherwise, the consumed food is assumed to be contaminated.

F.6.6.3 Results

Tables F-109 through F-116 provide summaries of the consequences, in terms of mrem and/or person-rem, of postulated accident doses to the MEI, NPAI, worker and the public. Except for the worker, where the dose is calculated using the 50th-percentile meteorology, dose calculations were performed for both the 50th- and the 95th-percentile meteorologies using the assumptions and input values discussed above. The accident scenarios and source terms, as described earlier in this appendix, were generically applied to new dry and wet storage facilities. For the existing facilities at each management site, the assumptions and the related source terms were adjusted to conform to the conditions of each facility. Two types of results were provided for the offsite residents (MEI and population). Because protective action guidelines (EPA, 1991) specify mitigative actions to prevent consumption of contaminated food, the dose to offsite residents is reported for all pathways (i.e., external, inhalation, and ingestion) and without the ingestion pathway (i.e., external and inhalation). It should be noted that, as stated earlier, no reduction of exposure to the plume or to contaminated ground surface as a result of early evacuation of offsite populations due to protective action guidelines was accounted for in this analysis.

The analyses were performed for a generic wet and a generic dry storage facility at the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site, as well as for site-specific locations (BNFP, L-Reactor Basin area, and RBOF at the Savannah River Site, and FMEF and WNP-4 Spray Pond at the Hanford Site). The consequences of accident scenarios for the IFSF (dry), CPP-749 (dry) and FAST (wet) storage areas at the Idaho National Engineering Laboratory are considered to be equal to those of a generic dry and a generic wet storage facility, respectively. The consequences of accident scenarios for E-MAD at Nevada Test Site are considered to be similar to that of a generic dry storage facility at Nevada Test Site. For the RBOF and the L-Reactor disassembly basin, the criticality accident source terms were adjusted to conform with the conditions assumed in the Basis for Interim Operation reports for these facilities (WSRC, 1995b and 1995c), where no credit was taken for high efficiency particulate air filters after a criticality accident.

Table F-109 Summary of the Accident Analysis Dose Assessments at the Savannah River Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents - H-Area									
Fuel Assembly Breach	0.16	Dose/event	0.24	0.068	9.2	0.055	0.0043	28	0.62
		Dose/yr	0.038	0.011	1.5	0.0088	0.00069	4.5	0.099
		LCF[b]	1.9×10^{-8}	5.5×10^{-9}	0.00075	4.4×10^{-9}	3.5×10^{-10}	1.8×10^{-6}	0.000050
Dropped Fuel Cask	0.0001	Dose/event	0.018	0.00034	0.55	0.0039	0.000024	0.28	0.011
		Dose/yr	1.8×10^{-6}	3.4×10^{-8}	0.000055	3.9×10^{-7}	2.4×10^{-9}	0.000028	1.1×10^{-6}
		LCF[b]	9.0×10^{-13}	1.7×10^{-14}	2.8×10^{-8}	2.0×10^{-13}	1.2×10^{-15}	1.1×10^{-11}	5.5×10^{-10}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	40	0.29	1300	8.9	0.019	120	87
		Dose/yr	0.000040	2.9×10^{-7}	0.0013	8.9×10^{-6}	1.9×10^{-8}	0.00012	0.000087
		LCF[b]	2.0×10^{-11}	1.5×10^{-13}	6.5×10^{-7}	4.5×10^{-12}	9.5×10^{-15}	4.8×10^{-11}	4.4×10^{-8}
New Wet Storage Accidents - H-Area									
Fuel Assembly Breach	0.16	Dose/event	0.0070	0.00039	0.23	0.0016	0.000027	0.14	0.016
		Dose/yr	0.0011	0.000062	0.037	0.00026	4.3×10^{-6}	0.0022	0.00026
		LCF[b]	5.5×10^{-10}	3.1×10^{-11}	0.000019	1.3×10^{-10}	2.2×10^{-12}	8.8×10^{-10}	1.3×10^{-7}
Accidental Criticality	0.0031	Dose/event	17	9.5	370	4.0	0.69	1600	15
		Dose/yr	0.053	0.030	1.2	0.012	0.0021	5.0	0.047
		LCF[b]	2.7×10^{-8}	1.5×10^{-8}	0.00060	6.0×10^{-9}	1.1×10^{-9}	2.0×10^{-6}	0.000024
Aircraft Crash	1×10^{-6}	Dose/event	4.1	0.98	150	0.92	0.061	400	10
		Dose/yr	4.1×10^{-6}	9.8×10^{-7}	0.00015	9.2 x 10-7	6.1×10^{-8}	0.00040	0.000010
		LCF[b]	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	4.6×10^{-13}	3.1×10^{-14}	1.6 x 10-10	5.0×10^{-9}

Table F-109A Summary of the Accident Analysis Dose Assessments at the Savannah River Site Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Dry Storage Accidents - H- Area						
Fuel Assembly Breach	0.16	Dose/event	0.053	3.1	0.012	0.0011
		Dose/yr	0.0085	0.050	0.0019	0.00018
		LCF[b]	4.3×10^{-9}	0.000025	9.5×10^{-10}	9.0×10^{-8}
Dropped Fuel Cask	0.0001	Dose/event	0.00024	0.015	0.000055	0.00090
		Dose/yr	2.4×10^{-8}	1.5×10^{-6}	5.5×10^{-9}	9.0×10^{-8}
		LCF	1.2×10^{-14}	7.5×10^{-10}	2.8×10^{-15}	4.5×10^{-11}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	0.91	55	0.20	0.037
		Dose/yr	9.1×10^{-7}	0.000055	2.0×10^{-7}	3.7×10^{-8}
		LCF	4.6×10^{-13}	2.8×10^{-8}	1.0×10^{-13}	1.9×10^{-11}
New Wet Storage Accidents - H-Area						
Fuel Assembly Breach	0.16	Dose/event	0.00027	0.017	0.000062	0.0010
		Dose/yr	0.000043	0.0027	9.9×10^{-6}	0.00016
		LCF	2.2×10^{-11}	1.4×10^{-6}	5.0×10^{-12}	8.0×10^{-8}
Accidental Criticality	0.0031	Dose/event	9.9	240	2.5	5.8
		Dose/yr	0.031	0.74	0.0078	0.018
		LCF	1.6×10^{-8}	0.00037	3.9×10^{-9}	9.0×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	0.76	46	0.18	3.1
		Dose/yr	7.6×10^{-7}	0.000046	1.8×10^{-7}	3.1×10^{-6}
		LCF	3.8×10^{-13}	2.3×10^{-8}	9.0×10^{-14}	1.6×10^{-9}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-110 Summary of the Accident Analysis Dose Assessments at the Idaho National Engineering Laboratory Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	1.3	0.67	15	0.093	0.062	28	0.83
		Dose/yr	0.21	0.11	2.4	0.015	0.0099	4.5	0.13
		LCF[b]	1.1×10^{-7}	5.5×10^{-8}	0.0012	7.5×10^{-9}	5.0×10^{-9}	1.8×10^{-6}	0.000065
Dropped Fuel Cask	0.0001	Dose/event	0.074	0.0033	0.83	0.0052	0.00032	0.12	0.047
		Dose/yr	7.4×10^{-6}	3.3×10^{-7}	0.000083	5.2×10^{-7}	3.2×10^{-8}	0.000012	4.7×10^{-6}
		LCF	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	2.6×10^{-13}	1.6×10^{-14}	4.8×10^{-12}	2.4×10^{-9}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	180	2.9	2000	13	0.27	120	110
		Dose/yr	0.00018	2.9×10^{-6}	0.0020	0.000013	2.7×10^{-7}	0.00012	0.00011
		LCF	9.0×10^{-11}	1.5×10^{-12}	1.0×10^{-6}	6.5×10^{-12}	1.4×10^{-13}	4.8×10^{-11}	5.5×10^{-8}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	0.0016	0.0036	0.43	0.0028	0.00036	0.14	0.025
		Dose/yr	0.00026	0.00058	0.069	0.00045	0.000058	0.022	0.0040
		LCF	1.3×10^{-10}	2.9×10^{-10}	0.000035	2.3×10^{-10}	2.9×10^{-11}	8.8×10^{-9}	2.0×10^{-6}
Accidental Criticality	0.0031	Dose/event	28	30	140	3.4	12	1800	12
		Dose/yr	0.087	0.093	0.43	0.011	0.037	5.6	0.037
		LCF	4.4×10^{-8}	4.7×10^{-8}	0.00022	5.5×10^{-9}	1.9×10^{-8}	2.2×10^{-6}	0.000019
Aircraft Crash	1×10^{-6}	Dose/event	22	9.8	250	1.6	0.88	400	14
		Dose/yr	0.000022	9.8×10^{-6}	0.00025	1.6×10^{-6}	8.8×10^{-7}	0.00040	0.00014
		LCF	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	8.0×10^{-13}	4.4×10^{-13}	1.6×10^{-10}	7.0×10^{-8}

Table F-110A Summary of the Accident Analysis Dose Assessments at the Idaho National Engineering Laboratory Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.23	2.7	0.017	0.15
		Dose/yr	0.037	0.43	0.0027	0.024
		LCF[b]	1.9×10^{-8}	0.00022	1.4×10^{-9}	0.000012
Dropped Fuel Cask	0.0001	Dose/event	0.0010	0.013	0.000079	0.00076
		Dose/yr	1.0×10^{-7}	1.3×10^{-6}	7.9×10^{-9}	7.6×10^{-8}
		LCF	5.0×10^{-14}	6.5×10^{-10}	4.0×10^{-15}	3.8×10^{-11}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	4.0	0.45	0.29	2.6
		Dose/yr	4.0×10^{-6}	0.000045	2.9×10^{-7}	2.6×10^{-6}
		LCF	2.0×10^{-12}	2.3×10^{-8}	1.5×10^{-13}	1.3×10^{-9}
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.0012	0.014	0.000090	0.00085
		Dose/yr	0.00019	0.0022	0.000014	0.00014
		LCF	9.5×10^{-11}	1.1×10^{-6}	7.0×10^{-12}	7.0×10^{-8}
Accidental Criticality	0.0031	Dose/event	17	26	2.6	5.4
		Dose/yr	0.053	0.081	0.0081	0.017
		LCF	2.7×10^{-8}	0.000041	4.1×10^{-9}	8.5×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	3.4	39	0.25	2.1
		Dose/yr	3.4×10^{-6}	0.000039	2.5×10^{-7}	2.1×10^{-6}
		LCF	1.7×10^{-12}	2.0×10^{-8}	1.3×10^{-13}	1.1×10^{-9}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-111 Summary of the Accident Analysis Dose Assessments at the Hanford Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	3.0 0.48 2.4×10^{-7}	0.57 0.091 4.6×10^{-8}	42 6.7 0.0034	0.15 0.024 1.2×10^{-8}	0.061 0.0098 4.9×10^{-9}	50 8.0 3.2×10^{-6}	2.0 0.32 0.00016
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.26 0.000026 1.3×10^{-11}	0.0085 8.5×10^{-7} 4.3×10^{-13}	3.0 0.00030 1.5×10^{-7}	0.011 1.1×10^{-6} 5.5×10^{-13}	0.00031 3.1×10^{-8} 1.6×10^{-14}	0.22 0.000022 8.8×10^{-12}	0.15 0.000015 7.5×10^{-9}
Aircraft Crash w/Fire[c]	NA	---	NA	NA	NA	NA	NA	NA	NA
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.13 0.021 1.1×10^{-8}	0.0033 0.00053 2.7×10^{-10}	1.6 0.26 0.00013	0.0064 0.0010 5.0×10^{-10}	0.00035 0.000056 2.8×10^{-11}	0.25 0.040 1.6×10^{-8}	0.078 0.013 6.5×10^{-6}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	64 0.20 1.0×10^{-7}	14 0.044 2.2×10^{-8}	740 2.3 0.0012	4.8 0.015 7.5×10^{-9}	12 0.037 1.9×10^{-8}	3600 11 4.4×10^{-6}	55 0.17 0.000085
Aircraft Crash[c]	NA	---	NA	NA	NA	NA	NA	NA	NA

Table F-111A Summary of the Accident Analysis Dose Assessments at the Hanford Site Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	0.30 0.048 2.4×10^{-8}	6.5 1.0 0.00050	0.015 0.0024 1.2×10^{-9}	0.31 0.050 0.000025
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.0039 3.9×10^{-7} 2.0×10^{-13}	0.029 2.9×10^{-6} 1.5×10^{-9}	0.000071 7.1×10^{-9} 3.6×10^{-15}	0.0015 1.5×10^{-7} 7.5×10^{-11}
Aircraft Crash w/Fire[c]	NA	---	NA	NA	NA	NA
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.0016 0.00026 1.3×10^{-10}	0.032 0.0051 2.6×10^{-6}	0.000079 0.000013 6.5×10^{-12}	0.0018 0.00029 1.5×10^{-7}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	7.9 0.025 1.3×10^{-8}	180 0.56 0.00028	2.0 0.0062 3.1×10^{-9}	27 0.084 0.000042
Aircraft Crash[c]	NA	---	NA	NA	NA	NA

NA = Not Applicable

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

[c] *Aircraft crash accidents are not applicable to the Hanford Site since their frequency of occurrence is less than 10^{-7}/yr.*

Table F-112 Summary of the Accident Analysis Dose Assessments at the Oak Ridge Reservation Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	***Risk***	***MEI (mrem)[a]***	***NPAI (mrem)***	***Population (person-rem)***	***MEI (mrem)***	***NPAI (mrem)***	***Worker (mrem)***	***Population (person-rem)***
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	22 3.5 1.8×10^{-6}	42 6.7 3.4×10^{-6}	55 8.8 0.0044	2.1 0.34 1.7×10^{-7}	9.4 1.5 7.5×10^{-7}	140 22 8.8×10^{-6}	8.4 1.3 0.00065
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	1.4 0.00014 7.0×10^{-11}	0.18 0.000018 9.0×10^{-12}	15 0.0015 7.5×10^{-7}	0.14 0.000014 7.0×10^{-12}	0.042 4.2×10^{-6} 2.1×10^{-12}	0.61 0.000061 2.4×10^{-11}	2.3 0.00023 1.2×10^{-7}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event Dose/yr LCF	2300 0.0023 1.2×10^{-9}	180 0.00018 9.0×10^{-11}	2900 0.0029 1.5×10^{-6}	220 0.00022 1.1×10^{-10}	41 0.000041 2.1×10^{-11}	610 0.00061 2.4×10^{-10}	440 0.00044 2.2×10^{-7}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.71 0.11 5.5×10^{-8}	0.20 0.0032 1.6×10^{-8}	16 2.6 0.0013	0.068 0.011 5.5×10^{-9}	0.046 0.0074 3.7×10^{-9}	0.68 0.11 4.4×10^{-8}	2.5 0.40 0.00020
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	1500 4.7 2.4×10^{-6}	3300 10 5.0×10^{-6}	1400 4.3 0.0022	230 0.71 3.6×10^{-7}	910 2.8 1.4×10^{-6}	6800 21 8.4×10^{-6}	210 0.65 0.00033
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF	380 0.00038 1.9×10^{-10}	600 0.00060 3.0×10^{-10}	2900 0.0029 1.5×10^{-6}	29 0.000029 1.5×10^{-10}	130 0.00013 6.5×10^{-11}	1900 0.0019 7.6×10^{-10}	120 0.00012 6.0×10^{-8}

Table F-112A Summary of the Accident Analysis Dose Assessments at the Oak Ridge Reservation Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	***Risk***	***MEI (mrem)[a]***	***Population (person-rem)***	***MEI (mrem)***	***Population (person-rem)***
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	9.8 1.6 8.0×10^{-7}	29 4.6 0.0023	0.96 0.15 7.5×10^{-8}	4.4 0.70 0.00035
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.038 3.8×10^{-6} 1.9×10^{-12}	0.13 0.000013 6.5×10^{-9}	0.0039 3.9×10^{-7} 2.0×10^{-13}	0.021 2.1×10^{-6} 1.1×10^{-9}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event Dose/yr LCF	180 0.00018 9.0×10^{-11}	500 0.00050 2.5×10^{-7}	17 0.000017 8.5×10^{-12}	76 0.000076 3.8×10^{-8}
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.042 0.0067 3.4×10^{-9}	0.14 0.022 0.000011	0.0043 0.00069 3.5×10^{-10}	0.023 0.0037 1.9×10^{-6}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	1100 3.4 1.7×10^{-6}	1100 3.4 0.0017	180 0.56 2.8×10^{-7}	150 0.47 0.00024
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF	140 0.00014 7.0×10^{-11}	420 0.00042 2.1×10^{-7}	13 0.000013 6.5×10^{-12}	61 0.000061 3.1×10^{-8}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-113 Summary of the Accident Analysis Dose Assessments at the Nevada Test Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	1.7	0.31	1.5	0.052	0.0046	20	0.038
		Dose/yr	0.27	0.050	0.24	0.0083	0.00074	3.2	0.0060
		LCF[b]	1.4×10^{-7}	2.5×10^{-8}	0.00012	4.2×10^{-9}	3.7×10^{-10}	1.3×10^{-6}	3.0×10^{-9}
Dropped Fuel Cask	0.0001	Dose/event	0.11	0.0014	0.40	0.0033	0.000026	0.089	0.010
		Dose/yr	0.000011	1.4×10^{-7}	0.000040	3.3×10^{-7}	2.6×10^{-9}	8.9×10^{-6}	1.0×10^{-6}
		LCF	5.5×10^{-12}	7.0×10^{-14}	2.0×10^{-8}	1.7×10^{-13}	1.3×10^{-15}	3.6×10^{-12}	5.0×10^{-10}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	180	1.2	250	5.6	0.020	87	6.2
		Dose/yr	0.00018	1.2×10^{-6}	0.00025	5.6×10^{-6}	2.0×10^{-8}	0.000087	6.2×10^{-6}
		LCF	9.0×10^{-11}	6.0×10^{-13}	1.3×10^{-7}	2.8×10^{-12}	1.0×10^{-14}	3.5×10^{-11}	3.1×10^{-9}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	0.054	0.0016	0.33	0.0017	0.000029	0.10	0.0084
		Dose/yr	0.0086	0.00026	0.053	0.00027	4.6×10^{-6}	0.016	0.0013
		LCF	4.2×10^{-9}	1.3×10^{-10}	0.000026	1.4×10^{-10}	2.3×10^{-12}	6.4×10^{-9}	6.5×10^{-7}
Accidental Criticality	0.0031	Dose/event	88	15	54	6.9	1.1	1300	1.9
		Dose/yr	0.27	0.047	0.17	0.021	0.0034	4.0	0.0059
		LCF	1.4×10^{-7}	2.3×10^{-8}	0.000084	1.1×10^{-8}	1.7×10^{-9}	0.000016	3.0×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	29	4.2	61	0.92	0.067	290	1.6
		Dose/yr	0.000029	4.2×10^{-6}	0.000061	9.2×10^{-7}	6.7×10^{-8}	0.00029	1.6×10^{-6}
		LCF	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	4.6×10^{-13}	3.4×10^{-14}	1.2×10^{-10}	8.0×10^{-10}

Table F-113A Summary of the Accident Analysis Dose Assessments at the Nevada Test Site Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.78	0.26	0.024	0.0066
		Dose/yr	0.13	0.042	0.0038	0.0011
		LCF[b]	6.2×10^{-8}	0.000021	1.9×10^{-9}	5.3×10^{-7}
Dropped Fuel Cask	0.0001	Dose/event	0.0031	0.0011	0.00011	0.000033
		Dose/yr	3.1×10^{-7}	1.1×10^{-7}	1.1×10^{-8}	3.3×10^{-9}
		LCF	1.6×10^{-13}	5.5×10^{-11}	5.5×10^{-15}	1.7×10^{-12}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	13	4.5	0.41	0.12
		Dose/yr	0.000013	4.5×10^{-6}	4.1×10^{-7}	1.2×10^{-7}
		LCF	6.5×10^{-12}	2.3×10^{-9}	2.1×10^{-13}	6.0×10^{-11}
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.0036	0.0013	0.00012	0.000037
		Dose/yr	0.00058	0.00021	0.000019	5.9×10^{-6}
		LCF	2.9×10^{-10}	1.1×10^{-7}	9.5×10^{-12}	3.0×10^{-9}
Accidental Criticality	0.0031	Dose/event	55	5.4	5.8	0.70
		Dose/yr	0.17	0.017	0.018	0.0022
		LCF	8.5×10^{-8}	8.5×10^{-6}	9.0×10^{-9}	1.1×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	11	3.7	0.35	0.096
		Dose/yr	0.000011	3.7×10^{-6}	3.5×10^{-7}	9.6×10^{-8}
		LCF	5.5×10^{-12}	1.9×10^{-9}	1.8×10^{-13}	4.8×10^{-11}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-114 Summary of the Accident Analysis Dose Assessments at the Barnwell Nuclear Fuels Plant Wet Storage Facility[a] at the Savannah River Site - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)[b]*	*NPAI (mrem)*	*Population (Person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (Person-rem)*
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	0.018	0.00099	0.028	0.0055	0.00027	0.00080	0.0033
		Dose/yr	0.0056	0.00016	0.0045	0.00088	0.000043	0.00013	0.00053
		LCF[c]	2.8×10^{-9}	8.0×10^{-11}	2.3×10^{-6}	4.4×10^{-10}	2.2×10^{-10}	5.2×10^{-11}	2.7×10^{-7}
Accidental Criticality	0.0031	Dose/event	80	75	44	42	45	75	5.6
		Dose/yr	0.25	0.23	0.14	0.13	0.14	0.23	0.017
		LCF	1.3×10^{-7}	1.2×10^{-7}	0.000070	6.5×10^{-8}	7.0×10^{-8}	9.2×10^{-8}	8.5×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	92	31	23	11	3.9	70	2.3
		Dose/yr	0.00092	0.000031	0.000023	0.00011	3.9×10^{-6}	0.000070	2.3×10^{-6}
		LCF	4.6×10^{-10}	1.6×10^{-11}	1.2×10^{-8}	5.5×10^{-11}	2.0×10^{-12}	2.8×10^{-10}	1.2×10^{-9}

Table F-114A Summary of the Accident Analysis Dose Assessments at the Barnwell Nuclear Fuels Plant Wet Storage Facility[a] at the Savannah River Site - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)[b]*	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.00072	0.0021	0.00024	0.00021
		Dose/yr	0.00012	0.00034	0.000038	0.000034
		LCF[c]	6.0×10^{-11}	1.7×10^{-7}	1.9×10^{-12}	1.7×10^{-8}
Accidental Criticality	0.0031	Dose/event	64	27	37	3.7
		Dose/yr	0.20	0.084	0.12	0.012
		LCF	1.0×10^{-7}	0.000042	6.0×10^{-8}	6.0×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	17	6.9	2.0	0.68
		Dose/yr	0.000017	6.9×10^{-6}	2.0×10^{-6}	6.8×10^{-7}
		LCF	8.5×10^{-12}	3.5×10^{-9}	1.0×10^{-12}	3.4×10^{-10}

[a] *Emissions will be released through an elevated stack for Accidental Criticality and Fuel Assembly Breach accidents.*

[b] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[c] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-115 Summary of the Accident Analysis Dose Assessments at the Receiving Basin for Offsite Fuels and L-Reactor Basin Wet Storage Facilities at the Savannah River Site-All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Wet Storage Accidents - RBOF									
Fuel Assembly Breach	0.16	Dose/event	0.0070	0.00039	0.23	0.0016	0.000027	0.14	0.016
		Dose/yr	0.0011	0.000062	0.037	0.00026	4.3×10^{-6}	0.0022	0.00026
		LCF[b]	5.5×10^{-10}	3.1×10^{-11}	0.000019	1.3×10^{-10}	2.2×10^{-12}	8.8×10^{-10}	1.3×10^{-7}
Accidental Criticality	0.0031	Dose/event	130	44	4800	30	2.9	16000	310
		Dose/yr	0.40	0.14	14.9	0.093	0.0090	50	0.96
		LCF	2.0×10^{-7}	7.0×10^{-8}	0.0074	4.7×10^{-8}	4.5×10^{-9}	0.000020	0.00048
Aircraft Crash	1×10^{-6}	Dose/event	4.1	0.98	150	0.92	0.061	400	10
		Dose/yr	4.1×10^{-6}	9.8×10^{-7}	0.00015	9.2×10^{-7}	6.1×10^{-8}	0.00040	0.000010
		LCF	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	4.6×10^{-13}	3.1×10^{-14}	1.6×10^{-10}	5.0×10^{-9}
Wet Storage Accidents- L-Reactor Basin[c]									
Fuel Assembly Breach/	0.16	Dose/event	0.0093	0.00097	0.14	0.0011	0.00015	0.11	0.022
		Dose/yr	0.0015	0.00016	0.022	0.00018	0.00024	0.018	0.0035
		LCF	7.4×10^{-10}	8.0×10^{-11}	0.000011	8.8×10^{-11}	1.2×10^{-11}	7.1×10^{-9}	1.8×10^{-6}
Accidental Criticality	0.0031	Dose/event	170	120	3000	21	21	14000	440
		Dose/yr	0.527	0.37	9.3	0.065	0.065	43	1.4
		LCF	2.6×10^{-7}	1.9×10^{-7}	0.0047	3.3×10^{-8}	3.3×10^{-8}	0.000017	0.00070
Aircraft Crash	1×10^{-6}	Dose/event	4.2	2.6	93	0.60	0.39	70	14
		Dose/yr	4.2×10^{-6}	2.6×10^{-6}	0.000093	6.0×10^{-7}	3.9×10^{-7}	0.000070	0.000014
		LCF	2.1×10^{-12}	1.3×10^{-12}	4.7×10^{-8}	3.0×10^{-13}	2.0×10^{-13}	2.8×10^{-11}	7.0×10^{-9}

Table F-115A Summary of the Accident Analysis Dose Assessments at the Receiving Basin for Offsite Fuels and L-Reactor Basin Wet Storage Facilities at the Savannah River Site-External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Wet Storage Accidents - RBOF						
Fuel Assembly Breach	0.16	Dose/event	0.00027	0.017	0.000062	0.0010
		Dose/yr	0.000043	0.0027	9.9×10^{-6}	0.00016
		LCF[b]	2.2×10^{-11}	1.4×10^{-6}	5.0×10^{-12}	8.0×10^{-8}
Accidental Criticality	0.0031	Dose/event	38	1900	8.8	120
		Dose/yr	0.12	5.9	0.027	0.37
		LCF	5.9×10^{-8}	0.0029	1.4×10^{-8}	0.00019
Aircraft Crash	1×10^{-6}	Dose/event	0.76	46	0.18	3.1
		Dose/yr	7.6×10^{-7}	0.000046	1.8×10^{-7}	3.1×10^{-6}
		LCF	3.8×10^{-13}	2.3×10^{-8}	9.0×10^{-14}	1.6×10^{-9}
Wet Storage Accidents - L-Reactor Basin[c]						
Fuel Assembly Breach	0.16	Dose/event	0.00034	0.010	0.000041	0.0016
		Dose/yr	0.000054	0.0016	6.6×10^{-6}	0.00026
		LCF	2.7×10^{-11}	8.0×10^{-7}	3.3×10^{-12}	1.3×10^{-7}
Accidental Criticality	0.0031	Dose/event	50	1200	6.5	170
		Dose/yr	0.16	3.72	0.020	0.53
		LCF	7.8×10^{-8}	0.0019	1.1×10^{-8}	0.00026
Aircraft Crash	1×10^{-6}	Dose/event	0.77	28	0.11	4.2
		Dose/yr	7.7×10^{-7}	0.000028	1.1×10^{-7}	4.2×10^{-6}
		LCF	3.9×10^{-13}	1.4×10^{-8}	5.5×10^{-14}	2.1×10^{-9}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event per year.*

Table F-116 Summary of the Accident Analysis Dose Assessments for the Fuel Material Examination Facility Dry Storage and WNP-4 Wet Storage Facilities at the Hanford Site - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents at FMEF[b]									
Fuel Assembly Breach	0.16	Dose/event	4.7	2.1	46	0.42	0.25	0.99	5.7
		Dose/yr	0.75	0.34	7.4	0.067	0.040	0.16	0.91
		LCF[c]	3.7×10^{-7}	1.7×10^{-7}	0.0037	3.4×10^{-8}	2.0×10^{-8}	6.4×10^{-8}	0.00046
Dropped Fuel Cask	0.0001	Dose/event	0.2	0.032	3.2	0.017	0.0017	0.0049	0.41
		Dose/yr	0.00002	3.2×10^{-6}	0.00032	1.7×10^{-6}	1.7×10^{-7}	4.9×10^{-7}	0.000041
		LCF	8×10^{-12}	1.6×10^{-12}	3.2×10^{-7}	8.5×10^{-13}	8.5×10^{-14}	2.5×10^{-13}	2.1×10^{-8}
Aircraft Crash w/Fire[d]	NA	---	NA	NA	NA	NA	NA	NA	NA
Wet Storage Accidents at WNP-4[b]									
Fuel Assembly Breach	0.16	Dose/event	0.15	0.0033	1.3	0.018	0.00060	0.00024	0.13
		Dose/yr	0.024	0.00053	0.21	0.0029	0.000096	0.000038	0.021
		LCF	1.2×10^{-8}	2.7 x 10-10	0.00011	1.5×10^{-9}	4.8×10^{-11}	1.5×10^{-11}	0.000011
Accidental Criticality	0.0031	Dose/event	97	76	620	20	45	120	160
		Dose/yr	0.3	0.24	1.9	0.062	0.14	0.37	0.50
		LCF	1.5×10^{-7}	1.2×10^{-7}	0.00096	3.1×10^{-8}	7.0×10^{-8}	1.5×10^{-7}	0.00025
Aircraft Crash[d]	NA	---	NA	NA	NA	NA	NA	NA	NA

Table F-116A Summary of the Accident Analysis Dose Assessments for the Fuel Material Examination Facility Dry Storage and WNP-4 Wet Storage Facilities at the Hanford Site - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Dry Storage Accidents at FMEF[b]						
Fuel Assembly Breach	0.016	Dose/event	0.46	6.6	0.041	0.79
		Dose/yr	0.074	1.1	0.0066	0.12
		LCFc	3.7×10^{-8}	0.00055	3.3×10^{-9}	0.000060
Dropped Fuel Cask	0.0001	Dose/event	0.0028	0.04	0.00025	0.0057
		Dose/yr	2.8×10^{-7}	4.0×10^{-6}	2.5×10^{-8}	5.7×10^{-7}
		LCF	1.4×10^{-13}	2.0×10^{-9}	1.2×10^{-14}	2.9×10^{-10}
Aircraft Crash w/Fire[d]	NA	---	NA	NA	NA	NA
Wet Storage Accidents at WNP-4[b]						
Fuel Assembly Breach	0.16	Dose/event	0.0023	0.032	0.00028	0.0034
		Dose/yr	0.00037	0.0051	0.000045	0.00054
		LCF	1.8×10^{-9}	2.6×10^{-6}	2.2×10^{-11}	2.7×10^{-7}
Accidental Criticality	0.0031	Dose/event	32	180	12	120
		Dose/yr	0.099	0.56	0.037	0.37
		LCF	5.0×10^{-9}	0.00028	1.9×10^{-8}	0.00019
Aircraft Crash[d]	NA	---	NA	NA	NA	NA

NA = Not Applicable

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Emissions will be released through an elevated stack for Fuel Assembly Breach, Dropped Fuel Cask, and Accidental Criticality Accidents.*

[c] *Point Estimate of Latent Cancer Fatalities event/yr.*

[d] *Aircraft Crash accidents are not applicable to the Hanford Site since their frequency of occurrence is less than 10^{-7} event/yr.*

Table F-117 provides a summary of the consequences of radiation exposure to the public and to the MEI from emissions in wet storage (generic and existing), and dry storage (generic and existing).

Table F-117 Normal Release Dose Assessments and Latent Cancer Fatalities at Storage Sites

	MEI Dose (mrem/yr)	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Savannah River Site				
Receipt/Unloading at:				
RBOF	1.1×10^{-4}	5.5×10^{-11}	5.7×10^{-3}	2.8×10^{-6}
L-Reactor Basin	7.3×10^{-5}	3.7×10^{-11}	4.6×10^{-3}	2.3×10^{-6}
BNFP	6.5×10^{-4}	3.3×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Dry Storage Facility	1.8×10^{-4}	9.0×10^{-11}	8.6×10^{-3}	4.3×10^{-6}
New Wet Storage Facility	1.1×10^{-4}	5.5×10^{-11}	5.7×10^{-3}	2.8×10^{-6}
Storage at:				
RBOF	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
L-Reactor Basin[a]	3.6×10^{-4}	1.8×10^{-10}	2.2×10^{-2}	1.1×10^{-5}
BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
Idaho National Engineering Laboratory				
Receipt/Unloading at:				
IFSF (dry storage)	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
FAST (wet storage)	3.8×10^{-4}	1.9×10^{-10}	3.1×10^{-3}	1.6×10^{-6}
CPP-749 (dry storage)	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Dry Storage Facility	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Wet Storage Facility	3.8×10^{-4}	1.9×10^{-10}	3.1×10^{-3}	1.6×10^{-6}
Storage at:				
IFSF (dry storage)	0	0	0	0
FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
CPP-749 (dry storage)	0	0	0	0
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
Hanford Site				
Receipt/Unloading at:				
FMEF (dry storage)	2.0×10^{-4}	1.0×10^{-10}	1.1×10^{-2}	5.5×10^{-6}
WNP-4 Spray Pond (wet storage)	2.2×10^{-4}	1.1×10^{-10}	5.8×10^{-3}	2.9×10^{-6}
New Dry Storage Facility	2.5×10^{-4}	1.3×10^{-10}	1.5×10^{-2}	7.5×10^{-6}
New Wet Storage Facility	2.0×10^{-4}	1.0×10^{-10}	1.2×10^{-2}	6.0×10^{-6}
Storage at:				
FMEF (dry storage)	0	0	0	0
WNP-4 Spray Pond (wet storage)	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}
Oak Ridge Reservation				
Receipt/Unloading at:				
New Dry Storage Facility	8.9×10^{-2}	4.5×10^{-8}	8.5×10^{-2}	4.3×10^{-5}
New Wet Storage Facility	6.0×10^{-2}	3.0×10^{-8}	6.1×10^{-2}	3.1×10^{-5}
Storage at:				
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}

	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Nevada Test Site				
Receipt/Unloading at:				
E-MAD (dry storage)	7.6×10^{-4}	3.8×10^{-10}	9.3×10^{-4}	4.7×10^{-7}
New Dry Storage Facility	7.6×10^{-4}	3.8×10^{-10}	9.3×10^{-4}	4.7×10^{-7}
New Wet Storage Facility	5.2×10^{-4}	2.6×10^{-10}	5.2×10^{-4}	2.6×10^{-7}
Storage at:				
E-MAD (dry storage)	0	0	0	0
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.0×10^{-12}

[a] *L-Reactor basin doses are due to existing conditions; the foreign research reactor spent nuclear fuel contribution would be six orders of magnitude smaller*

F.6.7 Accident Scenarios Involving Target Materials

A review of the hypothetical accident scenarios analyzed for spent nuclear fuel indicates that only the aircraft crash with fire accident is applicable to the target materials. The frequency of occurrence of an accident involving target materials is estimated to be 3 percent of the 1×10^{-6} per year frequency figure used in the spent nuclear fuel accident analysis. This is because the number of transfer casks that would involve target material is less than 3 percent of that used for 22,700 spent nuclear fuel elements. Therefore, the frequency of this scenario is less than 10^{-7} per year, and is considered to be unforeseeable. Nonetheless, this accident was analyzed and its consequences at potential storage locations were summarized in Table F-118. The frequency of this accident is set conservatively at 10^{-7} per year.

The process by which target materials are prepared for shipment [i.e., drying and canning of the target material solutions, (see Appendix B, Section B.1.5)] releases all gaseous fission products (noble gases and halogens). In addition, the cans in which target materials would be packed do not require any further cutting when they are received in a storage facility. A review of the hypothetical accident scenarios analyzed for spent nuclear fuel indicates that only the aircraft crash with fire accident would be applicable to the target materials. The cans are never cut, and there are no gaseous fission products; therefore, fuel element breach and fuel cask drop scenarios would not be applicable. In addition, should there be an aircraft crash into the wet storage pool where the target material is stored; or, if an accidental criticality in the pool were to occur, the radioactivity releases would be bound by that of the spent nuclear fuel analyzed for these accidents. This is because the amount of radioactive inventory per target material can is very small compared to that in the bounding spent nuclear fuel. In addition, any releases from the target cans would be absorbed in the pool.

Therefore, a scenario involving an aircraft crash into a dry storage facility with ensuing fire was analyzed for the target materials. The scenario assumptions are similar to those described in Section F.6.4.4.3. Because of the size of each can, it was assumed that the transfer cask involved in the accident would contain 40 cans of target materials containing maximum radionuclide inventories, (i.e., 40 cans of 200 grams of ^{235}U per can cooled for at least 3 years). The overall respirable release fraction is assumed to be 5×10^{-3} (Neuhauser and Kanipe, 1993). Table F-119 shows the radioactivity release source terms for this accident.

Table F-118 Summary of the Accident Analysis Dose Assessments for the Aircraft Crash Accident with Fire Involving Target Material - All Pathways

			95 Percent Meteorology			*50 Percent Meteorology*			
Site[c]	*Frequency (event/yr)*	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
NTS		Dose/event	180	28	120	5.6	0.45	2000	3.0
		Dose/yr	0.000018	2.8×10^{-6}	0.000012	5.6×10^{-7}	4.5×10^{-8}	0.00020	3.0×10^{-7}
	1×10^{-7}	LCF[b]	9.0×10^{-12}	1.4×10^{-12}	6.0×10^{-9}	2.8×10^{-13}	2.3×10^{-14}	8.0×10^{-11}	1.5×10^{-10}
ORR		Dose/event	2400	4000	3700	230	910	14000	560
		Dose/yr	0.00024	0.00040	0.00037	0.000023	0.000091	0.0014	0.000056
	1×10^{-7}	LCF	1.2×10^{-10}	2.0×10^{-10}	1.9×10^{-7}	1.2×10^{-11}	4.6×10^{-11}	5.6×10^{-10}	2.8×10^{-8}
INEL		Dose/event	130	63	1500	9.3	5.7	2700	84
		Dose/yr	0.000013	6.3×10^{-6}	0.00015	9.3×10^{-7}	5.7×10^{-7}	0.00027	8.4×10^{-6}
	1×10^{-7}	LCF	6.5×10^{-12}	6.3×10^{-12}	7.5×10^{-8}	4.7×10^{-13}	2.9×10^{-13}	1.1×10^{-10}	4.2×10^{-9}
SRS		Dose/event	26	6.3	970	5.8	0.41	2700	66
		Dose/yr	2.6×10^{-6}	6.3×10^{-7}	0.000097	5.8×10^{-7}	4.1×10^{-8}	0.00027	6.6×10^{-6}
	1×10^{-7}	LCF	1.3×10^{-12}	3.2×10^{-13}	4.9×10^{-8}	2.9×10^{-13}	2.1×10^{-14}	1.1×10^{-10}	3.3×10^{-9}

Table F-118A Summary of the Accident Analysis Dose Assessments for the Aircraft Crash Accident with Fire Involving Target Material - External and Inhalation Pathways

			95 Percent Meteorology		*50 Percent Meteorology*	
Site[c]	*Frequency (event/yr)*	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
NTS		Dose/event	64	22	2.1	0.57
		Dose/yr	6.4×10^{-6}	2.2×10^{-6}	2.1×10^{-7}	5.7×10^{-8}
	1×10^{-7}	LCF[b]	3.2×10^{-12}	1.1×10^{-9}	1.1×10^{-13}	2.9×10^{-11}
ORR		Dose/event	870	2500	83	560
		Dose/yr	0.000087	0.00025	8.3×10^{-6}	0.000056
	1×10^{-7}	LCF	4.4×10^{-11}	1.3×10^{-7}	4.2×10^{-12}	2.8×10^{-8}
INEL		Dose/event	20	230	1.4	12
		Dose/yr	2.0×10^{-6}	0.000023	1.4×10^{-7}	1.2×10^{-6}
	1×10^{-7}	LCF	1.0×10^{-12}	1.2×10^{-8}	7.0×10^{-14}	6.0×10^{-10}
SRS		Dose/event	4.4	270	1.0	19
		Dose/yr	4.4×10^{-7}	0.000027	1.0×10^{-7}	1.9×10^{-6}
	1×10^{-7}	LCF	2.2×10^{-13}	1.4×10^{-8}	5.0×10^{-12}	9.5×10^{-10}

NTS = Nevada Test Site; ORR = Oak Ridge Reservation; INEL = Idaho National Engineering Laboratory; SRS = Savannah River Site

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/ yr.*

[c] *Aircraft crash accidents are not applicable to the Hanford Site since their frequency of occurrence is much less than 10^{-7} event/yr.*

Table F-119 Target Materials Aircraft Crash with Fire Accident Source Terms

Isotope	*Curies*
Strontium-89	2.4×10^{-4}
Strontium-90	3.1×10^{0}
Yttrium-90	3.1×10^{0}
Yttrium-91	1.4×10^{-3}
Zirconium-95	5.2×10^{-3}
Niobium-95	1.1×10^{-2}
Rubidium-103	2.1×10^{-6}
Rubidium-106	7.9×10^{-1}
Ruthenium-103m	2.1×10^{-6}
Tin-123	1.1×10^{-3}
Antimony-125	8.2×10^{-2}
Tellurium-125m	2.0×10^{-2}
Tellurium-127m	2.2×10^{-3}
Tellurium-129m	6.0×10^{-9}
Cesium-134	6.5×10^{-3}
Cesium-137	3.0×10^{-1}
Cerium-141	7.4×10^{-8}
Cerium-144	7.7×10^{0}
Presidium-144	7.7×10^{0}
Promethium-147	6.3×10^{0}
Promethium-148m	2.5×10^{-9}
Europium-154	1.4×10^{-3}
Europium-155	5.2×10^{-2}
Uranium-234	1.4×10^{-7}
Uranium-235	8.3×10^{-5}
Uranium-238	1.5×10^{-6}
Plutonium-238	3.3×10^{-6}
Plutonium-239	6.2×10^{-4}
Plutonium-240	1.4×10^{-5}
Plutonium-241	1.3×10^{-4}
Americium-241	6.9×10^{-7}
Americium-242m	4.4×10^{-12}
Americium-243	3.1×10^{-12}
Curium-242	6.8×10^{-12}
Curium-244	3.2×10^{-12}

F.7 Costs

The cost of implementing the proposed action is analyzed in this section. For the purpose of the cost analysis, the alternatives described in Section 2.1 of the EIS were adjusted to reflect the Record of Decision on the Programmatic SNF&INEL Final EIS (DOE, 1995g) issued in May 1995. According to this Record of Decision, if foreign research reactor spent nuclear fuel is managed in the United States, the aluminum-based portion would be managed at the Savannah River Site and the TRIGA portion would be managed at the Idaho National Engineering Laboratory. The cost analysis also considers the financing arrangements discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS that would affect the cost to the United States. The cost information is presented as follows:

F.7.1 Summary of Cost Information

F.7.2 Costs of Individual Program Components

F.7.3 Interpreting the Minimum Program Costs

F.7.4 Interpreting the Other Cost Factors

F.7.1 Summary of Cost Information

This section presents total costs for the proposed policy and implementation alternatives that would impact the costs. The costs are presented in two parts: 1) minimum discounted costs (base case) for the well-defined program components and integration approaches, and 2) "other cost factors" that are likely but sufficiently uncertain that they cannot be directly included in the minimum discounted costs. The costs are shown as net present values in a consistent accounting framework.

Several important factors are used when estimating costs. These factors are as follows:

- *Site- and Implementation-Specific Facilities* - All costs for management in the United States are for facilities that exist or are planned at either the Savannah River Site or the Idaho National Engineering Laboratory. Costs are allocated to the program in proportion to the share of foreign research reactor spent nuclear fuel managed or transferred at each facility. This allocation of capital and operating costs within larger programs results in lower costs to the program than would be the case for the use of facilities dedicated to foreign research reactor spent nuclear fuel.

- *Schedule of Activities* - For all management alternatives (except total management overseas), all spent nuclear fuel is shipped, managed for 40 years, and disposed (either as spent nuclear fuel or as reprocessing waste) on schedules that are appropriate for the selected facilities.

- *Discount Rate* - The base case costs are discounted to 1996 at the rate specified by the Office of Management and Budget for the year ending February 1996. This rate is 4.9 percent real. The base case costs for management outside the United States are discounted at a 3 percent real rate of interest. This rate is estimated to be the long-term real rate of interest that can be expected on a trust fund outside the United States. If the net present value of the costs of the program are received in 1996, a hypothetical trust fund invests the money at the real discount rate so that future expenditures are made out of principal and accrued interest.

- *Net Present Value* - Net present value is a figure-of-merit for decision-making on the basis of life-cycle cost, not a value used for establishing budgets or cash flows. All costs are shown in constant 1996 dollars discounted to 1996. This means that the costs for the duration of the program, expressed as a net present value, are due and payable on January 1, 1996, not in the year the costs are incurred.

- *Timing of Expenses* - All costs are assumed to be incurred on the last day of each year of the 40-year management period. The principal and accrued interest in the trust funds (at the net present value of the program costs) are exactly sufficient to meet the costs as they are incurred.

- *Timing of Payments* - Deferring payments beyond January 1, 1996 increases the payments required (either from reactor operators or the United States Congress) by a factor based on the discount rate and the deferral. Pro-forma full-cost recovery fees are shown for payments made on December 31 of each of the 13 receipt years (1996 through 2008).

- *Inflation and Escalation* - Costs are expressed in constant 1996 dollars in this analysis, so the effects of inflation are eliminated. No costs are escalated in real terms.

- *Ultimate Disposition* - Estimated costs for geologic disposal of intact spent nuclear fuel or waste from chemical separation are included to provide a complete life-cycle cost analysis.

F.7.1.1 Scenarios Analyzed

For the purpose of the cost analysis, six scenarios were analyzed. The scenarios reflect the alternatives that affect cost directly, are consistent with the Record of Decision of the Programmatic SNF&INEL Final EIS (DOE, 1995g) and include the costs for ultimate disposal. The six cost scenarios are:

1. *Management Alternative 1 (Storage)* - Storage of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site with new dry or wet storage facilities; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory at existing wet or dry storage facilities.

2. *Management Alternative 1 (revised to incorporate chemical separation)* - Chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

3. *Management Alternative 1 (revised to incorporate a new technology)* — Implementation of a new treatment and/or packaging technology for aluminum-based foreign research reactor spent nuclear fuel and target material at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

4. *Target Material* - Storage of target material at the Savannah River Site. This scenario provides the cost differential that can be used to assess the cost of managing target material in addition to the foreign research reactor spent nuclear fuel in Management Alternative 1 storage and chemical separation scenarios.

5. *Management Alternative 2* - Management of all foreign research reactor spent nuclear fuel overseas. This scenario reflects a combination of reprocessing and dry storage overseas. Countries with the capability to accept the waste from reprocessing are assumed to have their spent nuclear fuel reprocessed. The rest use dry storage.

6. *Management Alternative 3* - Chemical separation of a portion of the aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; reprocessing of the remainder of aluminum-based foreign research reactor spent nuclear fuel overseas; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

By varying the quantities of material managed in different ways in the United States and overseas, different cost scenarios can be generated. The costs of these variations are bounded by the costs of the scenarios described above. For instance, a management alternative that includes acceptance of target material into the United States would be represented by a combination of Scenarios 1 and 4 or 2 and 4.

The implementation alternatives under Management Alternative 1 related to alternative amounts of foreign research reactor spent nuclear fuel eligible under the policy (Section 2.2.2.1), and alternative policy durations (Section 2.2.2.2), were not considered separately in the cost analysis because they are bounded

by the cost scenarios analyzed. These implementation alternatives reduce the amount of foreign research reactor spent nuclear fuel eligible under the policy.

The implementation alternative under Management Alternative 1 related to alternative locations for taking title to the foreign research reactor spent nuclear fuel (Section 2.2.2.4) was not considered because it does not affect the cost analysis.

F.7.1.2 Minimum Program Costs

Table F-120 shows the minimum discounted program costs (base case) for the six scenarios defined above. These costs cover all foreign research reactor spent nuclear fuel shipments, management over 40 years, and geologic disposal. Uncertainties (risks) and escalation are zero. The schedule for activities in Europe under Management Alternative 3 is similar to that in the United States but not exactly the same. Reprocessing takes place over 13 years at Dounreay (the same timespan used for chemical separation at the Savannah River Site) although it could be completed at Dounreay in 9 or 10 years. Dounreay's charges for reprocessing are based on 1996 costs, not costs for 1996 through 2008 averaged over the 13-year period (as was done for the Savannah River Site). Geologic disposal takes place in 2025 through 2030 in Europe and 2030 through 2035 in the United States. Costs are discounted at 3 percent for the portion to be managed overseas and at 4.9 percent for the portion to be managed in the United States.

Table F-120 Minimum Program Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Net Present Value*
1. Management Alternative 1 (Storage)	725/775[a]
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625
3. Management Alternative 1 (revised to incorporate a New Technology)[b]	625-950
4. Target Material	35
5. Management Alternative 2	1,250
6. Management Alternative 3	675

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

Because of the uncertainties involved with the implementation of the new technology, the cost for Scenario 3 is presented as a range as discussed in Appendix F, Section F.7.2.9. Also, shipping costs in Scenario 3 include the assumption that of the total number of cask shipments, only 38 cask shipments would be accepted at the West Coast.

F.7.1.3 Other Cost Factors

There are four important sources of cost risk (excluding escalation) that are not part of the minimum costs in Table F-120. Table F-121 shows the likely values (risks) for these factors, taking into account the absolute values of the uncertainties and their probability of occurrence. A brief summary of these cost factors follows the table.

The other cost factors summarized in Table F-121 are as follows:

1. *Systems Integration and Logistics Risks* - Significant risks exist in the details of the policy implementation. The implementation of the policy would involve up to 41 foreign countries, up to 13 years of receipts, dozens of foreign ports, up to ten domestic ports, two U.S.

Table F-121 Other Cost Factors
(Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	Cost Factors				
	Systems Integration & Logistics	*Component Risks*	*Non-Program Risks*	*3% Discount Rate*	*Range*
1. Management Alternative 1 (Storage)[a]	100	75	35	175	385
2. Management Alternative (revised to incorporate Chemical Separation)	100	±15	10	125	200-250
3. Management Alternative 1 (revised to incorporate a New Technology)[b,c]	100	75	35	225	435
4. Target Material	5	5	0	25	35
5. Management Alternative 2	100+	±500	1000	250	350-1850
6. Management Alternative 3	100	±10	150	75	315-335

[a] *It is assumed that risks are the same for dry or wet storage options.*

[b] *It is assumed that risk factors are the same as Management Alternative 1 (Storage).*

[c] *Includes target material.*

management sites, and possibly several new facilities. Technical and procedural bottlenecks could arise in many areas.

2. *Component Risks* - Significant risks exist for specific components of the foreign research reactor spent nuclear fuel program, e.g., the adequacy of the characterization of spent nuclear fuel for interim storage, the methods of spent nuclear fuel disposal, the cost allocation at existing and new facilities, and development of new technology.

3. *Non-Program Risks* - Significant risks exist for components of other programs that affect the implementation of the foreign research reactor spent nuclear fuel EIS, e.g., escalating repository costs, adoption of monitored retrievable storage, and differences in facility utilization plans between this EIS and those of other EISs affecting the Savannah River Site and the Idaho National Engineering Laboratory. For Scenario 5, the risks are that no spent nuclear fuel infrastructure exists in more than half of the eligible countries and that no geologic disposal program exists in most of the eligible countries.

4. *Discount Rate Risks* - Significant risks exist that the discount rate required by the Office of Management and Budget for the year ending February, 1996 (4.9 percent real) will be reduced to a more historically representative level (e.g., 3 percent) in some future annual update. The base case costs for management outside the United States are discounted at a 3 percent rate. The use of a high discount rate is particularly risky because 1) revenues are likely to be fixed (in $/kgTM) early in the program while expenses are variable and uncertain, and 2) revenues received from the reactor operators during the 1996 through 2008 shipping period will almost certainly exceed the costs of management activities during that period. Mathematically, the excess revenues are placed in a trust fund that compounds interest at the discount rate. If the discount rate exceeds the rate at which funds actually compound, then outyear program costs (e.g., disposal) could not be met from the principal and accrued interest in the trust fund. A reduction in the discount rate from 4.9 percent to 3.0 percent has a larger impact on the program than any of the technical or systems integration risks.

F.7.1.4 Potential Total Costs

Table F-122 combines the base case costs with the "other cost factors" to provide a realistic expectation of the potential total costs of the program, excluding escalation. The "other cost factors" are divided into technical factors and discount rate-related factors. This table also shows the cumulative percentage effect on the minimum discounted program costs of real escalation at a rate of 1 percent per year over 40 years.

Table F-122 Potential Total Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Minimum Program Cost*	*Other Cost Factors (Technical)*	*Other Cost Factors (Discount Rate)*	*Potential Total Cost, No Escalation*	*1% Real Escalation, Cumulative*
1. Management Alternative 1 (Storage)	725/775[a]	210	175	≈1,100	+11%
2. Management Alternative 1 (revised to incorporate chemical separation)	625	85-145	125	≈900	+9%
3. Management Alternative 1 (revised to incorporate a new technology)[c]	625-950	210	225	≈1,050-1,400	10%-11%
5. Management Alternative 2	1250	600-1600	250	2,100-3,100	+13%
6. Management Alternative 3[b]	675	225-275	75	≈1,000	+9%

[a] *Dry/Wet new storage facilities.*

[b] *The total cost risk to the United States is less than 1/2 the total cost risk since a large portion of the activities under this alternative would occur overseas.*

[c] *Includes target material.*

Table F-122 shows that the net present value of the potential total costs of implementing the program in the United States, including an estimate of program risks but excluding escalation, range from about $900 for Scenario 2 to $1.4 billion for Scenario 3.

Costs for storing foreign research reactor spent nuclear fuel overseas are highly speculative. In addition, the overseas storage costs are always higher than the more centralized management alternatives because of the extremely high cost of safely and securely managing and disposing of small quantities of spent nuclear fuel in dozens of countries.

The program costs presented in Tables F-120, F-121, and F-122 are in constant 1996 dollars, discounted to 1996. This implies that funds required to cover these costs are received in 1996 and explicitly or implicitly placed in a trust fund. If payments into the trust fund are deferred, then they must be larger than if they had been received on January 1, 1996. For example, if payments are made in 13 equal annual installments every December 31 over the 1996 through 2008 shipping and receiving period, then the constant-dollar payments must increase by 37 percent. A composite of payment schedules, e.g., 13 years for developed country reactor operators and pay-as-you-go (for the United States) for all other costs, including developing country costs, has the effect of increasing the required constant-dollar payments by as much as 25 to 50 percent.

F.7.1.5 Cost to the United States

The cost of the proposed policy to the United States would depend on the type of financing arrangement that DOE adopts in implementing the policy and the discount rate at which revenues from reactor operators accrue interest. Alternative financing arrangements are discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS. Briefly, the financing arrangements considered are:

1. United States bears the full cost of the program for developing countries and charges a competitive fee to developed countries.

2. United States bears the full cost for all countries (*no* fee).

3. United States charges a *full-cost-recovery* fee to all countries.

4. United States bears the full cost of the program for developing countries and charges a *full-cost-recovery* fee to developed countries.

From a practical standpoint, the U.S. cost under financing arrangement 3 above would be zero. The issue would be whether any foreign countries would participate in the program if full-cost recovery exceeded a competitive fee. The first and fourth arrangements are functionally similar, the U.S. cost resulting from the difference in the *competitive versus the full-cost-recovery fee.* The U.S. cost under the second arrangement (no fee) would be the total program cost as discussed earlier. Any fees established by the United States will take place pursuant to a Federal Register notice after the Record of Decision for this EIS.

Table F-123 shows costs to the United States for the minimum program in each of the cost scenarios analyzed (except target material) under a variety of fee schedules. Adding target material to Scenarios 1, 2, 5, or 6 would increase the cost by 3 to 4 percent. Fees of $2,000/kgTM, $5,000/kgTM, $7,500/kgTM, and $10,000/kgTM, including a pass-through of shipping charges (all expressed in constant 1996 dollars and levelized over 13 years), are used to provide a range of estimates for the cost to the United States. These fees do not imply that reactor operators would pay them for management in Europe or the United States, or that the fee established by the United States will be one of these values. They are used for illustration only and suggest a bounding range, exclusive of technical risk factors, discount rate adjustments, and escalation. The cost to the United States, presented in Table F-123, is the sum of: 1) the cost of managing the foreign research reactor spent nuclear fuel from the developing countries, including shipping, and 2) the difference between the revenues received for management of developed country foreign research reactor spent nuclear fuel and the total program cost of managing developed country foreign research reactor spent nuclear fuel, excluding shipping. Including shipping in the U.S. management costs allows management costs for the United States and the United Kingdom to be presented on a comparable basis.

Table F-123 shows that for minimum discounted program costs and fees charged to developed country reactor operators levelized over 13 years, costs to the United States for management of foreign research reactor spent nuclear fuel (and target material in Scenario 3) could range from several hundred million dollars at a fee of $2,000/kgTM to a profit for fees of $7,500/kgTM to $10,000/kgTM. The cost of managing the spent nuclear fuel from the developing countries (including shipping) adds roughly $100 million more to the cost borne by the United States. Excluding Scenario 5, for which all costs and fees are speculative, the table shows that costs to the United States in Management Alternative 3 are significantly lower than for Management Alternative 1. The savings to the United States exist because the United States bears none of the cost of Spent Nuclear Fuel Management in Europe except the cost of blending down the HEU at Dounreay.

If fees in the $2,000 to $10,000 per kgTM range (levelized $1996 dollars) are established and charged over 13 years, the costs to the United States would be as estimated in Table F-123 (excluding target materials) plus any additional cost factors not incorporated in the minimum program costs. These additional cost factors are: 1) technical risks, 2) discount rate-related risks, and 3) escalation. Table F-122 shows that

Table F-123 Costs to the United States for the Minimum Program Under Various Scenarios and Fee Structures (Millions of 1996 Dollars, Net Present Value of Costs in 1996, Fees Levelized Over 1996-2008 Period)

Scenario[a]	Full-Cost Recovery[b]	Levelized Shipping Fee $/kgTM	Levelized Management Fee (excluding shipping) $/kgTM	Net Present Value For Levelized Fee[c] (Developed Countries Only)				No Fee[d]	Total (excluding shipping)
				$2,000/ kgTM	$5,000/ kgTM	$7,500/ kgTM	$10,000/ kgTM	Developed Countries	
1. Management Alternative 1 (Storage)	100	1,500	6,500	325	100	(75)	(250)	475	575
2. Management Alternative 1 (revised to incorporate Chemical Separation)	90	1,500	5,800	275	50	(125)	(300)	425	525
3. Management Alternative 1 (revised to incorporate a New Technology)	90-110	1,700	5,600-9,200	275-550	50-325	(150)-125	(325)-(50)	425-700	500-800
5. Management Alternative 2[e]	500+							1,250 +	1,750+
6. Management Alternative 3[f]	85	1,500	6,000	225	75	(50)	(175)	300	375

[a] *The total mass (kgTM) of foreign research reactor spent nuclear fuel in the various scenarios is approximately as follows: Aluminum-based plus TRIGA: 115,000 kgTM; from developing countries: 15,000 kgTM; from developed countries: 100,000 kgTM; to Dounreay in Management Alternative 3: 37,000 kgTM. The total mass of target material is approximately 3,400 kgTM aluminum-based equivalent and essentially all from developed countries.*

[b] *Full-cost recovery from developed countries only. The United States bears the costs of the developing countries in these cases.*

[c] *Net present value of costs to the United States for management fees paid in 13 equal annual installments on December 31 of the years 1996 through 2008. Add costs in column labeled "Full-Cost Recovery" to generate total cost to the United States (developed and developing countries).*

[d] *As above, implicitly paid by the taxpayers in 13 equal annual installments (to maintain consistency with the payment period of the reactor operators), excluding shipping. The net present value of shipping in Scenarios 1 [Management Alternative 1 (Storage)] and 2 [Management Alternative 1 (revised to incorporate chemical separation)] is $140 Million. The net present value of shipping to the United States only in Scenario 6 is $90 Million. The net present value of shipping in Scenario 3 [Management Alternative 1 (revised to incorporate a new technology)] is $160 Million.*

[e] *There is no defined basis for the charges to the United States for non-U.S. management. Costs to the United States under Management Alternative 2 assume that the United States absorbs the cost to construct and operate independent foreign research reactor spent nuclear fuel storage installations (including all supporting safety, security, transport, health physics, etc. infrastructure) for the 22 countries with no commercial nuclear power programs and that the United States partially subsidizes the other countries, depending on developmental status, commercial nuclear power infrastructure, and other factors.*

[f] *U.S. component of Management Alternative 3 only. Revenues paid to the United States exclude shipping charges. Costs to the United States for management in Europe consist only of the charge to blend down the HEU to LEU ($20 million). European reactor operators using Dounreay are assumed to bear all other costs.*

technical risks could add roughly $100 to $200 million to the costs borne by the United States. Discount rate-related risks are of a similar size. Escalation risks are more uncertain but could be in the same range.

F.7.2 Costs of Individual Program Components

This section provides details on program costs for each of the scenarios outlined in section F.7.1.

F.7.2.1 Programmatic Cost Assumptions

Table F-124 shows programmatic assumptions about costs and the basis for the cost calculations.

Table F-124 Programmatic Assumptions and Bases

Variable	*Assumption*	*Basis*
Year Dollars	1996	Standardized to first year of program.
Discount Rate for Management in the United States	4.9 percent real	Required by Office of Management and Budget for programs beginning between February 1995 and February 1996.
Discount Rate for Management in Europe	3.0 percent real	Representative of long-run average in larger Western European economies.
Rounding of Totals	$25 million	Highlights differences between programs that typically differ by $100 million. No implication of precision.
Component Contingencies	Included in base costs	Standard costing assumption
Program Risks	Not included in base costs	Logistical complexity of program could add 10-15 percent to total costs.
Uncertainties	Not included in base costs	
Risk-adjustment	Not included in base costs	
Escalation	Not included in base costs	
Costs incurred over what period	40 years (1996 to 2035) in United States 35 years (1996 to 2020) in United Kingdom	Maximum length of interim storage
Repository Shipping	2030 to 2035 in United States 2025 to 2030 in United Kingdom	Storage maximum in United States and United Kingdom
Qualification of fuel types for disposal	$10M per type allocated to the program, 5 types in program	Idaho National Engineering Laboratory estimate. The foreign research reactor spent nuclear fuel program is estimated to be responsible for three types of aluminum-based spent nuclear fuel and two types of TRIGA spent nuclear fuel. The types are related to the repository program characteristics.

F.7.2.2 Individual Program Components

The proposed foreign research reactor spent nuclear fuel program consists of many components. Table F-125 outlines the components of the five cost analysis scenarios described in Section F.7.1. Detailed discussions of the individual program components follow the table.

F.7.2.3 Logistics and Program Management

Under Management Alternative 1, the United States would undertake a program where the maximum requirements begin with the shipping of an estimated 837 casks of foreign research reactor spent nuclear

Table F-125 Applicability of Specific Cost Components to the Cost Analysis Scenarios

Component	*Appendix Section*	*Management Alternative 1 (Storage)*	*Management Alternative 1 (revised to incorporate Chemical Separation)*	*Management Alternative 1 (revised to incorporate a New Technology)*	*Management Alternative 2*	*Management Alternative 3*	*Target Material*
Programmatic Assumptions	F.7.2.1	x	x		x	x	x
Logistics and Program Management	F.7.2.3	x	x		x	x	x
Shipping Spent Nuclear Fuel to the United States	F.7.2.4	x	x			x	x
Shipping Spent Nuclear Fuel to the United Kingdom	F.7.2.5				x	x	
Interim Storage at the Savannah River Site	F.7.2.6	x					x
Interim Storage at the Idaho National Engineering Laboratory	F.7.2.7	x	x			x	
Chemical Separation at the Savannah River Site	F.7.2.8		x			x	
New Technology	F.7.2.9			x			x
Reprocessing in the United Kingdom	F.7.2.10				x	x	
High-Level Waste Vitrification and Separation Waste Storage	F.7.2.11		x		x	x	
Disposal of Spent Nuclear Fuel	F.7.2.12	x			x		x
Disposal of Vitrified High-Level Waste	F.7.2.13		x		x	x	
Storage or Reprocessing Overseas	F.7.2.14				x	x	

fuel and 140 casks of target material (977 casks in total) from dozens of ports in 41 countries to as many as 10 ports in the United States and one or more border crossings from Canada. Consistent with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995g), aluminum-based foreign research reactor spent nuclear fuel would be delivered to the Savannah River Site and TRIGA foreign research reactor spent nuclear fuel would be delivered to the Idaho National Engineering Laboratory. Approximately 815 casks (including target material) would be shipped to the Savannah River Site and 162 to the Idaho National Engineering Laboratory.

Once the foreign research reactor spent nuclear fuel was in transit to the United States and especially once title had been transferred to the United States, numerous regulations covering safety, health, and environmental compliance would take effect. It is estimated that the direct cost of coordinating shipping, ensuring regulatory compliance, providing program documentation, conducting inspections in the United States and overseas, and providing overall program logistical support is about $5 million per year during the active shipping period (exclusive of shipping costs). This cost would be lower if material is shipped to the United Kingdom (Management Alternative 3). For Management Alternative 1, the discounted cost over the 13-year receipt period in the United States would be approximately $50 million. Costs for logistics and program management during the non-receiving period (years 14 through 40) are assumed to be modest and are accounted for as part of the management costs at the U.S. site.

F.7.2.4 Shipping to the United States

Shipping the foreign research reactor spent nuclear fuel and target material to the United States requires an estimated 977 cask shipments. Of this, 837 cask shipments would contain spent nuclear fuel and 140 cask shipments would contain target material.[4] The shipping period would be thirteen years, beginning in 1996. Discounted total shipping costs to and from the United States are estimated at about $140 million for the spent nuclear fuel and about 10 percent more if target material is included.[5] Under Management Alternative 3, where approximately one-third of the spent nuclear fuel casks are shipped to the United Kingdom, costs for shipping the remaining casks to the United States are about $90 million. Costs include cask rental, inland freight by truck in the United States and overseas, ocean transport to and from the United States (except for shipments from Canada, which would go by truck), port handling, security, insurance, administration, and contingencies. Logistics and program management is described in Section F.7.2.3.

The technical requirements and costs associated with shipping differ depending on the point of origin of the spent nuclear fuel. Costs are estimated separately for seven countries and/or regions of the world: Europe, Australia, Japan, Asia (excluding Australia and Japan), Canada, Other Atlantic, and Other Pacific. This section discusses technical issues associated with shipping the foreign research reactor spent nuclear fuel and target material to the United States from each region.

- **Europe** — European regulations for inland freight and ocean freight shipments of spent nuclear fuel have become very strict in recent years and are virtually certain to become more stringent. Requirements for permits, cross-border shipping, consolidation for ocean shipping, and other factors have driven the cost per cask in an unconsolidated movement to far more than that for inland freight in the United States. Considering the European Community's vessel requirements for the spent nuclear fuel shipped in 1995 under the Urgent Relief Environmental Assessment (DOE, 1994i), it is prudent for costing purposes to assume that shipment by chartered vessel rather than regularly scheduled commercial vessel would be required. (Shipment by purpose-built vessel is not likely to be required.) The cost of chartering a ship capable of carrying spent nuclear fuel casks from Europe to the United States' East Coast and handling the casks at the port and on-board is approximately $400,000. This cost can be spread over a maximum of 6 to 8 casks per vessel. For costing purposes, the EIS assumes 6 casks per vessel and two European ports-of-call. European nations account for an estimated 505 casks, 393 containing aluminum-based spent nuclear fuel, 14 containing target material, and 98 containing TRIGA spent nuclear fuel.

- **Australia** — Australia owns a single, large spent nuclear fuel transportation cask and, thus, does not generate a cask rental charge as part of the spent nuclear fuel program. (A charge for a typical transportation cask is assessed, however, to show true costs to undertake the program.) Australia is unlikely to require chartered shipping. Inland freight charges are moderate. Australia would account for 9 casks, all of which would contain aluminum-based foreign research reactor spent nuclear fuel. For cost analysis, the

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4 *The number of casks required for target material could be reduced by more than half if the material was converted to an oxide form prior to shipping. The estimate of 140 casks is based on a conservative estimate of shipping the material as a calcine.*

5 *Shipping target material increases costs much less than proportionately because most of the target material is in Canada.*

Australian cask is assumed to be shipped as part of larger shipments from Asia. These shipments would carry 6 casks per vessel and call on three ports per transit to the United States.

- **Japan** — The Japan Atomic Energy Research Institute owns two casks that would be used for spent nuclear fuel accepted by the United States. Because the Japan Atomic Energy Research Institute is near the port of export for Japan, inland freight charges would be negligible. Japan would likely require chartered vessels (at least as far as Europe for shipments of spent nuclear fuel to the United States via Europe). Shipment by chartered vessel would be approximately $450,000, or $225,000 per cask. It is estimated that Japan would ship approximately 110 casks to the United States (99 aluminum-based and 11 TRIGA). Japan could choose to acquire more casks to reduce its cost per ocean transit. As with Australia, a cask charge is assigned to show true program costs.

- **Asia** (excluding Australia and Japan) — Asian nations (excluding Japan) would be expected to have relatively low inland freight costs. It is unclear if Asian nations would require chartered vessels. Asian nations (excluding Australia and Japan) account for an estimated 62 casks (23 containing aluminum-based spent nuclear fuel, 1 containing target material, and 38 containing TRIGA spent nuclear fuel).

- **Canada** — For cost analysis, all Canadian shipments (approximately 116 casks of aluminum-based spent nuclear fuel and 125 casks of target material) are assumed to come by truck to the Savannah River Site. Cask rental and inland freight charges reflect the shipping times and distances for long overland routes. Shipping by rail is also feasible.

- **Other Atlantic** — All other nations nearer the Atlantic Ocean than the Pacific Ocean are assumed to have characteristics similar to those of Asia (excluding Australia and Japan) but lower ocean shipping costs because of greater proximity to the United States. Shipments from Mexico would come by sea, since the Mexican spent nuclear fuel is located in the southern part of the country. The Other Atlantic nations are not likely to require chartered vessels. Other Atlantic nations account for 38 casks, 23 of which would contain aluminum-based spent nuclear fuel and 15 of which would contain TRIGA spent nuclear fuel.

- **Other Pacific** — All other nations nearer the Pacific Ocean than the Atlantic Ocean are assumed to have characteristics similar to those of Asia (excluding Australia and Japan) but lower ocean shipping costs because of greater proximity to the United States. Because the Other Pacific countries are on the western coast of South America (which is significantly closer to the southeastern United States than the northwestern United States) and because all the spent nuclear fuel from these countries is aluminum-based, the EIS assumes that all shipments from Other Pacific countries will go by sea to an East Coast port via the Panama Canal. The Other Pacific nations would not be likely to require chartered vessels. Other Pacific nations account for 12 casks, all of which would contain aluminum-based spent nuclear fuel.

Table F-126 summarizes the cost of shipping a single spent nuclear fuel cask from various parts of the world to the United States in the configuration considered most likely by this EIS. The base case assumes the use of charter ships. The discounted cost of overseas shipping to the United States (including overland shipping from Canada and including target material) is shown in the table as $158 million (summing the bottom row). Of the 977 shipments, 827 originate either in Canada or in ports nearer the U.S. East Coast

Table F-126 Representative Shipping Costs to/from the United States for a Spent Nuclear Fuel Cask (Thousands of 1996 Dollars per Cask and Millions of 1996 Dollars for the Program, including Target Material)

Activity/Cost	*Europe*	*Australia*	*Japan*	*Other Asia*	*Canada*	*Other Atlantic*	*Other Pacific*
Charter	Y	Y	Y	Y	N/A	Y	Y
U.S. Coast	East	West	West	West	N/A	East	East
Charter Cost $k	400	550	550	500	N/A	300	300
Casks/Charter	6	See Other Asia	6	6	N/A	6	6
Ports-of-Call	2	See Other Asia	1	3	N/A	3	3
Total Rental Charges, $k/Cask	51	48	42	66	21	60	66
Inland Freight, Country, Site, and Overland Route Weighted, $k/Cask	37	38	41	30	25	26	38
Insurance, Security, Administration, Cask Return, $k/Cask	51	49	58	70	36	49	49
$k/Cask, Excluding Contingency	224	253	239	246	86	232	246
Number of Casks (Aluminum)	393	9	99	23	116	23	12
Number of Casks (TRIGA)	98	0	11	38	0	15	0
Number of Casks (Target Material)	14	0	0	1	125	0	0
Number of Casks (Total)	505	9	110	62	241	38	12
of which, from Developing Countries	72	0	0	53	0	38	12
Total Cost, including 15% Contingency, $M	130	2	30	18	24	10	3
Discounted Cost ($M)	95	2	22	13	17	7	2

(including 12 cask shipments from the West Coast of South America). The remaining 150 cask shipments originate in ports nearer the U.S. West Coast. Assuming shipments to the nearest U.S. coast, regardless of the type of spent nuclear fuel, an estimated 113 shipments of TRIGA spent nuclear fuel received at East Coast ports and an estimated 132 shipments of aluminum-based spent nuclear fuel received at West Coast ports would be shipped overland to the appropriate management site. Key issues in analyzing shipping costs follow the table.

- **Use of Chartered Ships** - Chartered vessels are used for conservatism in costing. The cost increase from using charters rather than regularly scheduled commercial vessels (for those countries likely to permit shipping by regularly scheduled commercial liners) is approximately $10 million. Europe and Japan would require charters in any case. Canada would transport by land. These three regions/countries account for more than 80 percent of the proposed shipments. No additional port-related costs are assigned if military ports are used.

- **Casks Per Ocean Shipment** - The costs in Table F-126 assume that European and Japanese shipments are made by a chartered vessel carrying six casks consolidated at two ports in Europe and one in Japan. For Japan, this charter loading implies the acquisition of more casks or the use of commercial casks. The nine casks (total) from Australia are assumed to be part of larger shipments from Asia. Even though Japan and Australia would most likely use their own casks, a cask rental charge is shown to reflect true program costs regardless of where the costs are borne and to reflect the likely requirement for new casks by Japan. Reducing the number of casks per shipment from Japan increases shipping costs by $20 million.

Shipments from the rest of the world (excluding Europe and Canada) are assumed by charter at the rate of 6 casks per vessel and 3 ports-of-call (i.e., two casks per country). Adding ports-of-call increases costs in transit (by about $20,000 per port-of-call and $20,000 per day in transit between ports) but saves money on balance by increasing the number of casks on the ship. Reducing the shipments from Asia (excluding Australia, Japan) and the Other Atlantic and Other Pacific countries to 2 casks and 1 port-of-call would increase program costs by $12 million.

- **Shipping to Distant Coasts and Sites** -- The cost of shipping the foreign research reactor spent nuclear fuel depends on which ports were selected and from where they would be accepting the shipments. The dynamics of the program are that roughly 75 percent of the foreign research reactor spent nuclear fuel is aluminum-based (and therefore would be destined for the Savannah River Site, on the United States' East Coast) and roughly 75 percent of the foreign research reactor spent nuclear fuel (excluding Canadian spent nuclear fuel) is in countries on the Atlantic side of the United States. While the 75 percent aluminum-based spent nuclear fuel and the 75 percent Atlantic spent nuclear fuel are not identical, there is sufficient overlap to create a situation where shipping all the spent nuclear fuel directly to a United States East Coast port and then distributing the TRIGA spent nuclear fuel to the Idaho National Engineering Laboratory by land would be only about 5 percent ($8 million) more expensive than shipping the spent nuclear fuel to the nearest port and then overland to the appropriate site. The cost of overland shipping by truck from an eastern port to the Idaho National Engineering Laboratory for a shipment that would logically arrive at an eastern port is less than the cost of ocean shipping to a western port to minimize the overland transit by truck.

- **Receipt Rates at the Savannah River Site and the Idaho National Engineering Laboratory** -- To accept all the foreign research reactor spent nuclear fuel within the proposed 13-year period requires, on average, cask receipts of almost six casks per month (seven per month if target material is included). Splitting the spent nuclear fuel by fuel type, consistent with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995g), implies receipt of 4 to 5 casks per month of aluminum-based spent nuclear fuel at the Savannah River Site and about one cask per month of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory.[6] About 1 cask per month of target material would also be received at the Savannah River Site.

- **Cask Rental Charges** -- Truck casks rent for approximately $1,500 per day on long-term lease. Shorter-term rentals are appreciably more expensive (EG&G, 1994b). Table F-126 incorporates the $1,500 per day rate for a long-term lease. The use of the smaller truck casks (compared to rail casks) permits savings in ocean shipping, short overland transport (although this could change in response to high charter costs), and security. The cost to acquire a new truck cask has been increasing steadily and is now approaching $2 million. The time from ordering to delivery exceeds 1 year. Because of the limited market for casks and the risk of constructing a cask for which there is no long-term demand, potential cask owners and lessors would place a high fixed charge rate on an investment in new casks for the foreign research reactor program. For a 20-year operating life, the fixed

6 *The weighted-average number of spent nuclear fuel elements per cask is estimated to be slightly more than 27. The sites are limited by cask receipt rates, not elements per cask. Some casks would have as many as 120 elements. Others would have one element. Most would have about 27 to 30 elements.*

charge rate would be at least 30 percent. For a fixed charge rate of 30 percent, a $2 million cask must rent for $600,000 per year, or approximately $1,650 per day on a yearly lease.

- **Cask Shipment and Rental Periods** -- The average time required to complete a round-trip shipment depends on the area of cask origin, the number of casks shipped at one time, the number of ports-of-call made enroute to the United States, inland shipping in the United States, and turnaround time at the sites. Excluding Canada, round-trip cask shipment periods range from an average of less than 40 days for a cask from the Atlantic coast of South America to the southeast coast of the United States (with an ultimate destination of the Savannah River Site) to more than 60 days for a cask from Australia to the same ultimate destination (either via a Pacific port and an overland transit to the Savannah River Site or via a passage through the Panama Canal to an Atlantic Port).

The base costs cover two ocean transits, port handling in two countries, shipment to and from the cask lessor, and overland transport from the ports to and from the sites and reactor facilities. Cask handling at the sites is estimated separately.

- **Contingencies** -- Over the past few years, the cost of almost all phases of international spent nuclear fuel shipping has risen sharply. Also, European regulations regarding ocean shipping of nuclear cargoes have tightened dramatically. While these costs are built into the values in Table F-126, potentially large additional contingencies are not. These contingencies include escalating cask lease rates; partially filled casks; higher inland freight charges in the United States; dedicated rail shipping in the United States; consolidation limitations in Asia, South America, or Africa; and additional security. On the other hand, the single largest contingency -- the use of charter ships -- has been added to the base case. Consideration of the magnitude of the contingencies suggests a contingency factor of about 15 percent. This factor applies to the shipping component of the program only, not the impacts on the program logistics or integration from delays in shipping, barriers erected by the States, etc. These program-level impacts are discussed separately in Section F.7.4.

F.7.2.5 Shipping to the United Kingdom

Shipping to the United Kingdom is less expensive than shipping to the United States. Cost estimates provided by the United Kingdom Atomic Energy Authority for this EIS are about $30,000 per cask from Europe (Scullion, 1995). This compares to more than $200,000 per cask estimated for shipments from Europe to the United States. The estimates for shipping to the United Kingdom reflect the large savings from the very short ocean transit from continental Europe (and thus the vessel charter cost), the ocean transit and site turn-around periods (and thus the cask rental time), the inland freight charges for shipping a short distance in the United Kingdom, and the reduced administrative, insurance, and security costs for the shorter activity.

It is possible that the estimated cost for shipments to the United Kingdom is understated in comparison to the U.S. costs for at least two reasons. First, no detailed analysis of the cost components similar to that in Table F-126 was conducted and thus some costs, especially indirect costs, such as administration, may have been omitted. Second, costs for shipments to the United States have increased sharply in recent years. Costs for recent shipments to the United States were higher than anticipated and may not be reflected in the estimated costs to ship from Europe to the United Kingdom.

F.7.2.6 Storage at the Savannah River Site

Consistent with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995g), approximately 17,800 aluminum-based spent nuclear fuel elements could be received and managed at the Savannah River Site. These elements would be stored or chemically separated. Under Implementation Alternative 1c to Management Alternative 1, target material equal to about 600 aluminum-based elements could also be received and stored at the Savannah River Site. The cost to receive and store the target material is proportional to the ratio of target material (expressed in element-equivalents, e.g., cans) to spent nuclear fuel elements (i.e., about 3.4 percent). Costs in this section refer to the basic implementation of Management Alternative 1 (17,800 spent nuclear fuel elements and no targets).

Storage at the Savannah River Site would consist of two phases: Phase-1 storage in existing facilities and Phase-2 storage in new facilities. Logistically, the base case for Management Alternative 1 (storage) is as follows:

- At the start of the implementation period, aluminum-based spent nuclear fuel would be shipped to the Savannah River Site and wet-stored in RBOF and the L-Reactor disassembly basin.

- At about the same time, construction would begin on a staging and characterization facility and an interim dry or wet storage facility at the Savannah River Site. The staging facility would be designed to receive and transfer all the foreign research reactor spent nuclear fuel (and other nuclear materials, including domestic research reactor spent nuclear fuels). The dry or wet storage facility would be designed to store the spent nuclear fuel (and possibly target material) until the spent nuclear fuel and target material were prepared for shipment to the repository. The new facilities would be commissioned in 2003, accept off-site receipts of foreign research reactor spent nuclear fuel through 2008, and on-site transfers (of all aluminum-based materials, not just foreign research reactor spent nuclear fuel) from the RBOF and the L-Reactor disassembly basin through about 2008 or 2009. If commissioning of the new storage facility is delayed to 2005, transfers from existing basins would continue through about 2010.

- At some point in the 2015 to 2035 time period, the stored spent nuclear fuel would be prepared for repository disposal in as-yet unspecified repository-qualified canisters. Cost estimates are based on a repository packaging and shipping period of 2030 to 2035.

Table F-127 shows the annual costs for storage of 17,800 foreign research reactor spent nuclear fuel elements at the Savannah River Site during Phase 1 and Phase 2, where Phase 2 storage is dry (WSRC, 1995c). Receiving and storing target material would add $20 million (discounted) to expenditures at the Savannah River Site and $35 million (discounted) to the total costs. The key assumptions used to generate the costs in Table F-127 are discussed below.

- Annual operating costs for round-the-clock operations at RBOF and L-Reactor disassembly basin are allocated to the foreign research reactor spent nuclear fuel program in proportion to the share of foreign research reactor spent nuclear fuel mass transferred to or from the basins relative to total cask transfers at RBOF, and L-Reactor disassembly basin in each year until all of the foreign research reactor spent nuclear fuel has been transferred to dry storage (about 2008 or 2009). Unit costs are assumed fixed in each year. Thus, allocable costs scale in proportion to the amount of foreign research reactor material received at the basins.

Table F-127 Storage of Aluminum-Based Spent Nuclear Fuel at the Savannah River Site, Including Phase 2 Dry Storage (Millions of 1996 Dollars)

Year	*Basin Costs*	*Capital Costs-Staging*	*Operating Costs-Staging*	*Capital Costs-Storage Facility*	*Capital Costs-Storage Canisters*	*Operating Costs-Storage Site*	*Decontamination & Decommissioning*
1996	16	1					
1997	20	5					
1998	23	5					
1999	24	6		1			
2000	21	18		1			
2001	15	22		2			
2002	15	24		2			
2003	15	5	3	1	9	3	
2004	15		3	1	9	3	
2005	19		3	1	9	3	
2006	16		3	1	9	3	
2007	16		3	1	9	3	
2008	16		3	1	9	3	
2009	6	1	3	1		3	
2010		1	1			3	
2011		4	1			2	
2012		4	1			2	
2013		3	1			2	
2014		2	1			2	
2015-2029			1/yr.			2/yr.	
2030-2034			3/yr.			3/yr.	
2035			3			3	9
Total Costs (Undiscounted)	238	103	65	14	55	77	9
NPV	174	74	23	9	34	28	1

- A staging facility would be constructed for operation in 2003 (although it could be deferred until 2005). The primary functions of the staging facility would be to: 1) accept on-site transfers and off-site receipts, 2) characterize the spent nuclear fuel, 3) transfer the spent nuclear fuel from the received casks to interim dry or wet storage, and 4) transfer the spent nuclear fuel from interim dry or wet storage to repository-qualified canisters. If processing of the spent nuclear fuel prior to disposal were required, e.g., melting and diluting the material, additional facilities would be added to the staging facility.

  Excluding a melt-and-dilute facility, the staging facility is estimated to have a discounted cost of $150 million. Because the foreign research reactor spent nuclear fuel program would share the facility with other programs, costs are allocated to the foreign research reactor spent nuclear fuel program in proportion to the average of: 1) foreign research reactor spent nuclear fuel to other spent nuclear fuel received and staged to dry or wet storage over the period of facility operations (2003 through 2035), and 2) foreign research reactor spent nuclear fuel to other spent nuclear fuel staged from storage to repository casks over the period of facility operations. Using this approach, about 43 percent of the capital and operating costs of the facility would be allocable to the foreign research reactor spent nuclear fuel program. Foreign research reactor spent nuclear fuel represents about 57 percent of the total aluminum-based spent nuclear fuel to be managed at the Savannah

River Site over the 1996 through 2035 period (by MTR-equivalents) but only 28 percent of the total aluminum-based spent nuclear fuel received initially at the new staging and characterization facility. The unweighted average of these two percentages is 43 percent. Because most of the foreign research reactor spent nuclear fuel arrives prior to the operation of the new staging and characterization facility, the foreign research reactor spent nuclear fuel bears a disproportionately high share of the operating costs of RBOF and L-Reactor disassembly basin and a disproportionately low share of the capital and operating costs of the new staging and characterization facility.

- A new dry or wet storage facility would be constructed for operation in 2003. A dry storage facility would consist of a pad, fence, canisters, and storage overpacks. It would operate through 2035. The canisters used at the dry facility would not necessarily be qualified for repository disposal. During the storage phase, the canisters would be loaded with approximately 228 elements apiece.[7]

For cost analysis, the spent nuclear fuel is assumed to be taken out of storage over the period 2030 through 2035 for transfer to repository-qualified canisters. The actual timing of the transfers could be earlier but cannot be specified at present. The undiscounted cost of the facility per canister is $530,000 for the canister itself, $110,000 for the storage overpack, and $10,000 for a share of the pad (Stroupe, 1995). Undiscounted fixed costs for the facility, about 57 percent of which would be allocable to the foreign research reactor spent nuclear fuel program, include $24 million for security fencing, other fixed facility costs, licensing, etc.

It is estimated that about 57 percent of the $100 million discounted cost of a wet storage pool constructed from 1996 through 2002 would be allocated to the foreign research reactor spent nuclear fuel program.

Operations and maintenance and safeguards and security costs of about $3.2 million per year are allocated about 57 percent to the foreign research reactor spent nuclear fuel program (WSRC, 1995c). For the wet storage pool, operations and maintenance costs are about $5 million per year higher than at the dry facility. These costs are also allocated about 57 percent to the foreign research reactor spent nuclear fuel program.

The net present value of the allocated expenditures to receive and dry-store the aluminum-based spent nuclear fuel at the Savannah River Site is the sum of the net present values on the bottom row of Table F-127 or about $350 million for dry storage (and about $400 million for wet storage, not shown). Additional expenditures at the Savannah River Site would be incurred to receive and store target material ($20 million) and qualify the aluminum-based spent nuclear fuel for the repository ($25 million). The site-specific components of the other cost factors described in Section F.7.4 (e.g., additional characterization requirements, materials processing prior to repository packaging) would also be extra.

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7 *Aluminum-based elements can be related according to equivalent MTR elements. The precise loading level is 255 MTR-equivalent elements. Excluding the RHF elements from France, the average aluminum-based foreign research reactor spent nuclear fuel element is equal to 1.12 MTR-equivalents. RHF elements are rated at 20 MTR-equivalents each. The weighted-average MTR-equivalent from the MTR type elements and RHF elements is 21,400. The mass of aluminum-based elements is 101,300 kgTM. This value is based on 85 RHF elements at 110 kg (243 lb) each, 2,650 NRU elements (from Canada) at 5.7 kg (12.6 lb) each, and 15,064 MTR-type elements at 5.1 kg (11.2 lb) each. Target material is excluded from these calculations. Including target material would increase the total to 104,700 kgTM excluding the can.*

F.7.2.7 Storage at the Idaho National Engineering Laboratory

In the base scenarios involving United States acceptance of foreign research reactor spent nuclear fuel, approximately 4,900 TRIGA elements would be shipped to the Idaho National Engineering Laboratory for storage in existing facilities.

The Idaho National Engineering Laboratory would store the TRIGA spent nuclear fuel in the IFSF until the spent nuclear fuel was transferred to canisters for shipping to the repository. Table F-128 shows annual operating costs for the Idaho National Engineering Laboratory to dry-store approximately 4,900 TRIGA elements at the IFSF. (The Idaho National Engineering Laboratory could also wet-store the TRIGA elements at the FAST facility for about twice the cost as at IFSF.) The discounted total cost using the IFSF facility for storage is approximately $30 million. Qualification of TRIGA spent nuclear fuel for repository disposal would add another $15 million.

To complete the transfers from existing storage facilities to repository-qualified dry storage canisters, the Idaho National Engineering Laboratory might eventually require a new staging facility similar to that at the Savannah River Site. The Idaho National Engineering Laboratory is deferring construction of this facility until the repository waste acceptance criteria are available some time after 2000. Based on the share of TRIGA spent nuclear fuel relative to all material to be dry-stored at the Idaho National Engineering Laboratory and geologically disposed, the foreign research reactor spent nuclear fuel program would be allocated no more than $10 million of the capital cost of a staging facility whose discounted total cost would be less than $150 million. Allocable operating costs would be the same as shown in the column for repository canister loading. For cost analysis, repository loading and shipping is assumed to take place in 2030. Actual loading could take place earlier but cannot be specified at present.

F.7.2.8 Chemical Separation at the Savannah River Site

Implementation of Management Alternative 1 (revised to incorporate chemical separation) at the Savannah River Site could take place in different ways. One bounding case is to assume that existing and new facilities are used in essentially the same way as in Management Alternative 1 (storage). RBOF and L-Reactor disassembly basin are used for receiving and lag-storage, a new staging and characterization facility is required for repository loading of aluminum-based material received after the completion of chemical separation operations, one of the Canyons (F- or H-Canyon) is used at a moderate rate, new dry or wet storage facilities are required, etc. This option can be viewed as the separation of foreign research reactor spent nuclear fuel within a larger program to store and directly dispose non-foreign aluminum-based spent nuclear fuel.

The other bounding case can be viewed as the separation of foreign research reactor spent nuclear fuel within an accelerated program to chemically separate all of the accumulated aluminum-based materials and medium-term receipts. In this case, RBOF continues to be used for 40 years for receipts, characterization, storage and repository loading; no new staging and characterization facility is constructed; receipts of aluminum-based spent nuclear fuel from domestic sources are accelerated; one of the Canyons (F- or H-Canyon) is used at an accelerated pace; and no new dry or wet storage facilities are required.

In either case, chemical separation continues to around 2008 to 2010, at which point the canyons are shut down. In the first case, however, enough material remains on site and due to be received that a large-scale storage program (including a new staging and characterization facility) is required. In the second case, very little separable material remains on site and only about 5 casks per year are due to be received at the

Table F-128 Storage of TRIGA Spent Nuclear Fuel at the Idaho National Engineering Laboratory (Millions of 1996 Dollars)

Year	*Capital Costs -Staging*	*Transfers*	*IFSF Operations*	*Repository Canister Loading*	*Repository Canisters*	*Operations & Maintenance*	*Decontamination &Decommissioning*
1996		.4	1				
1997	1	.4	1				
1998		.4	1				
1999		.4	1				
2000		.4	1				
2001		.4	1				
2002		.4	1				
2003		.4	1				
2004		.4	1				
2005		.4	1				
2006		.4	1				
2007		.4	1				
2008		.4	1				
2009			1				
2010			1				
2011			1				
2012			1				
2013	3		1				
2014	4		1				
2015	3		1				
2016-2029			1/yr.				
2030			1	5	10	.1/yr	
2031-2035							2
Total Costs (Undiscounted)	11	5	35	5	10	4	2
NPV	5	4	17	1	2	2	0

time the Canyons are shut down. In this latter case, existing facilities can handle all program functions, including repository canister loading.

The first case is used for cost analysis purposes in this Appendix. This case is more probable, since it is more conservative with respect to selection of separation as an alternative and more conservative with respect to costs.

In either case, uranium (but not plutonium) is chemically separated from fission products at one of the canyons at the Savannah River Site.[8] In this EIS, costs for operations at F-Canyon are used since they are slightly higher than costs at H-Canyon (about $25 million). The credit for recovered uranium is the same in either case.

For either Management Alternative 1 (revised to incorporate chemical separation) (17,800 elements) or Management Alternative 3 (12,200 elements), the following assumptions apply:

8 *HEU cannot be chemically separated from LEU. Plutonium can be chemically separated from uranium and fission products but it is not the intention of the Savannah River Site or this EIS to do so. The amount of plutonium in the foreign research reactor spent nuclear fuel is negligible.*

- Basin operations continue until all material can be transferred out of the basins to the Canyons. Foreign research reactor spent nuclear fuel is out of the basins in 2006 under Management Alternative 1 (revised to incorporate chemical separation) and 2005 under Management Alternative 3. From 2003 forward, all receipts take place at the new staging and characterization facility.

- Canyon operations would take place over a maximum of 13 years (1998 through 2010). Actual operations could be completed by about 2009 under Management Alternative 1 (revised to incorporate chemical separation) and as soon as the final shipments were received under Management Alternative 3. The selection of a canyon or canyons will be specified in the Interim Management of Nuclear Materials EIS (DOE, 1995b) and a facilities utilization study currently in process at the Savannah River Site. If no processing is selected in either of those studies, none will be selected for foreign research reactor spent nuclear fuels.

- All Canyon operations that apply to foreign research reactor spent nuclear fuel apply to other, similar materials at the Savannah River Site, i.e., domestic research reactor spent nuclear fuel and DOE and government aluminum-based spent nuclear fuel. Canyon capacity is shared according to the MTR-equivalents of material in each category (foreign and non-foreign) on-site in each year.

- All spent nuclear fuel would be separated incrementally to at-risk materials under the Interim Management of Nuclear Materials EIS (DOE, 1995b).[9] Processing of at-risk materials would be completed in 2002 or early 2003. Spare dissolver capacity would be available for aluminum-based spent nuclear fuels at a rate of 720 MTR-equivalents in 1998 and 1999 and 2880 MTR-equivalents thereafter. Aluminum-based spent nuclear fuel could be separated incrementally over the period 1998 through 2002 or early 2003. Costs (at F- and H-Canyon) for all aluminum-based spent nuclear fuels would be $10.4 million in 1996, $13 million in 1997, $18 million in 1998 and 1999, $5 million in 2000, $3 million in 2001, $13 million in 2002, and $32 million per year thereafter.

- Recovered HEU is blended down to LEU for sale to a commercial power reactor operator. The value of the LEU is assumed to be 85 percent of the value of fresh LEU for power reactors. A value of $5,000 per MTR-equivalent is used for the sales revenue.

- A penalty is assessed on any programs that defer phasedown of Canyon operations beyond the point that the Canyon would be "deinventoried." At some point after 2002, separation of research reactor aluminum-based spent nuclear fuel may become the final base mission at a Canyon. If so, continued operations would incur a deferral penalty estimated at $11 million in the first year of deferral, $22 million in the second year, and $33 million per year thereafter.[10]

Table F-129 shows the costs allocated to the foreign research reactor spent nuclear fuel program for activities at the Savannah River Site under Management Alternative 1 (revised to incorporate chemical

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9 *At-risk materials are materials that require stabilization through processing to ensure long-term safety and security.*

10 *A further penalty for deferring "deactivation" is not assessed. The Savannah River Site does not believe that continued operations through about 2008 to 2010 will defer the transition from deinventoried status to deactivated status.*

Table F-129 Chemical Separation Costs at the Savannah River Site Under Management Alternative 1 (Revised to Incorporate Chemical Separation)

Year	*Allocated Receiving, Lag Storage, Facilities Support*	*Spare Dissolver Capacity (MTR-equivalents)*	*Approximate Foreign Research Reactor Share of Dissolver (percent)*	*Allocated Canyon Operations Cost*	*Allocated Canyon Deferral Penalty*	*Allocated HEU Credit*
1996	16	0				
1997	22	0				
1998	25	720	72	13		3
1999	26	720	72	13		3
2000	27	2880	72	4		10
2001	22	2880	72	2		10
2002	23	2880	72	10		10
2003	17	2880	72	23	8	10
2004	16	2880	72	23	16	10
2005	19	2880	72	23	24	10
2006	8	2880	72	23	24	10
2007	0	2880	72	23	24	10
2008	1	2880	72	23	24	10
2009	1	2880	60	19	20	9
2010	1	2880	0	21	20	3
2011-2035	0-2/yr.					
Total Costs (Undiscounted)	242			199	139	107
NPV	178			127	81	70

separation). The foreign research reactor spent nuclear fuel program incurs costs at the percentage shown in the fourth column. This percentage is approximately the spare dissolver capacity allocated to the foreign research reactor program in each year. The change in the percentage at the end occurs because less than proportional dissolver capacity is required to complete the foreign research reactor spent nuclear fuel processing.

Table F-130 shows the same information as Table F-129, adjusted for the shipment of 5,600 aluminum-based elements to Dounreay, Scotland (Management Alternative 3). Table F-130 shows significant cost reductions at the Savannah River Site for basin operations (and related non-processing activities). Separation operations are shown ending in 2007 even though receipts continue through 2008. This is a function of the dissolver capacity available through 2010 and the reduced quantity of foreign research reactor spent nuclear fuel at the Savannah River Site under Management Alternative 3. As a practical matter, the cost impact of stretching out the processing through 2008 is insignificant.

Expenditures at the Savannah River Site for receiving and storing the target material and at the Idaho National Engineering Laboratory for receiving and storing the TRIGA spent nuclear fuel are the same as in the Management Alternative 1 (storage). These values are about $20 million (out of a total of about $35 million) and $50 million, respectively.

F.7.2.9 New Technology

Under Implementation Alternative 7 of Management Alternative 1 (discussed in Section 2.2.2.7 of the EIS), DOE would initiate a development program to select a new treatment and/or packaging technology

Table F-130 Chemical Separation Costs at the Savannah River Site Under Management Alternative 3

Year	Allocated Receiving, Lag Storage, Facilities Support	Spare Dissolver Capacity (MTR-equivalents)	Approximate Foreign Research Reactor Share of Dissolver Capacity (Percent)	Allocated Canyon Operations Cost	Allocated Canyon Deferral Penalty	Allocated HEU Credit
1996	12	0				
1997	17	0				
1998	20	720	62	11		
1999	22	720	62	11		2
2000	22	2880	62	3		2
2001	16	2880	62	2		9
2002	16	2880	62	8		9
2003	13	2880	62	20	7	9
2004	11	2880	62	20	14	9
2005	7	2880	62	20	20	9
2006	0	2880	62	20	20	9
2007	0	2880	6	2	2	1
2008	0	2880	0	0	0	0
2009	0	2880	0	0	0	0
2010	0	2880	0	0	0	0
2011-2035	.1-.5/yr.					
Total Costs (Undiscounted)	173			117	63	68
NPV	129			80	39	47

which would then be constructed and operated to manage the foreign research reactor fuel. A number of different technologies will be considered before one or more are selected for further development.

In addition to the uncertainty as to which technology(ies) will be chosen, there are other cost uncertainties including: the repository disposal fee, the need for new facilities and the requirements needed for managing domestic fuel. To account for these uncertainties, a range of costs have been developed. The costs range from about $950 million (undiscounted) or $625 million (discounted) to about $1.75 billion (undiscounted) or $950 million (discounted).

F.7.2.10 Reprocessing in the United Kingdom

Under Management Alternative 3, approximately 5,600 aluminum-based spent nuclear fuel elements would be shipped to the United Kingdom Atomic Energy Authority's facility at Dounreay, Scotland for reprocessing.[11] The remaining 12,200 aluminum-based elements would be chemically separated at the Savannah River Site (Section F.7.2.8).[12] The TRIGA spent nuclear fuel would be stored at the Idaho National Engineering Laboratory (Section F.7.2.7).

11 Equal to about 7,900 MTR-equivalents, including 85 RHF elements at 20 MTR-equivalents apiece.

12 Equal to about 13,600 MTR-equivalents, including the 2,650 Canadian NRU elements and all other elements (excluding the French RHF elements) at 1.12 MTR- equivalents apiece.

The number of elements to be reprocessed at Dounreay is based on the number of spent nuclear fuel elements in countries with commercial nuclear power programs and the clear capability to manage the reprocessing wastes.[13] The reprocessing waste from Dounreay is returned to the countries of origin. More generally, Management Alternative 3 can be viewed as chemical separation of approximately 2/3 of the aluminum-based foreign research reactor spent nuclear fuel elements in the United States and 1/3 in the United Kingdom.

Table F-131 shows the United Kingdom Atomic Energy Authority's currently estimated costs to reprocess aluminum-based spent nuclear fuel elements. The cost for conversion assumes a downblending ratio of 2:1 (i.e., one unit of depleted uranium at 0 percent enrichment is added to each unit of separated uranium at 40 percent enrichment to produce two units of uranium at 20 percent enrichment). The costs in Table F-131 are converted from British Pounds to United States Dollars at a rate of 1.55 dollars per pound. Using these costs, the discounted cost to ship, receive, reprocess, and dispose of the wastes from 5,600 aluminum-based spent nuclear fuel elements on a schedule similar to that at the Savannah River Site and to obtain LEU metal fuel is approximately $265 million. At a discount rate of 3 percent, this is equivalent to about $7,000/kgTM, including a charge of about $700/kgTM for blending down the separated HEU to LEU.

Table F-131 Costs at Dounreay (1996 Dollars)

Activity	*Cost*
Transport Casks to/from Dounreay	$31,000/cask @ 2 casks per shipment
Receive & Unload Casks	$7,700/cask
Reprocess and produce cementous intermediate-level waste	$5,750/kgTM (HEU only)
Convert Uranyl Nitrate to Metal	$4,500/kg uranium metal (or $2,800/kg UO_2 for oxide)
Value of Metallic Uranium	$15,000/kg uranium
Store U-235	$1,550/kg U-235
Store intermediate-level waste	$1,550 per 500 l (132 gal) drum per year (containing 10 kg (22 lb) of spent nuclear fuel wastes)
Transport intermediate-level waste to originating country	$2,600/drum
Geologic disposal of intermediate-level waste	$31,000/drum

Source: All Costs (except value of metallic uranium) (Scullion, 1995).

Foreign research reactor operators may prefer to view their costs as the sum of the undiscounted current costs for shipping, reprocessing, and uranyl nitrate conversion to metal (without downblending to LEU) plus the discounted costs for interim storage of uranium, interim storage of reprocessing waste, and geologic disposal of reprocessing waste. Assuming a 3 percent discount rate for the outyear costs, and excluding the value of recovered metal uranium, the reactor operator would estimate a current cost of about $9,500/kgTM, excluding the value of the recovered uranium and $7,200/kgTM including the value of the recovered uranium. At a zero percent discount rate, which is reasonable if the reactor operator wants to incorporate a risk-adjustment for long-term unknowns like geologic disposal, the current costs are about $12,700/kgTM. The value of recovered metal HEU is credited to Dounreay to make it consistent with the value of the recovered LEU at the Savannah River Site. Blend-down at Dounreay would cost about $700/kgTM on a current cost basis. Since these cost estimates are based on current costs (i.e., 1996 dollars in 1996) rather than the current fraction of a series of costs (i.e., 1/13 of 13 years' worth of constant costs over the 1996 through 2008 period at the Savannah River Site), they are exposed to escalation.

13 Belgium, France, Germany, Italy, Spain, Switzerland, United Kingdom.

In comparing these costs to costs for managing the spent nuclear fuel in the United States, the key technical differences the reactor operator sees are: 1) receipt of converted fresh metal (with or without downblending), and 2) receipt of cementous waste. How the reactor operator prices receipt of the waste product compared to the charge estimated by the United Kingdom Atomic Energy Authority would have a major impact on the attractiveness of doing business with the United Kingdom versus the United States.

Table F-132 shows the annual cash flows from the European component of Management Alternative 3, including downblending to LEU. Escalation is not included in Table F-132. Net present value is calculated at a 3 percent real discount rate.

Table F-132 Annual Cash Flows from Europe Under Management Alternative 3 (Dollars in Millions)

Year	*Shipping & Receipt*	*Reprocessing*	*Conversion*	*Product Value* [a]	*Store U-235*	*Store Intermediate - Level Waste*	*Transport Intermediate- Level Waste*	*Dispose of Intemediate - Level Waste*
1996	.7	16.6	3.8	6.5	.1	0	0	0
1997	.7	16.6	3.8	6.5	.1	0	0	0
1998	.7	16.6	3.8	6.5	.1	0	0	0
1999	.7	16.6	3.8	6.5	.1	.8	0	0
2000	.7	16.6	3.8	6.5	.1	1.2	0	0
2001	.7	16.6	3.8	6.5	.1	1.5	0	0
2002	.7	16.6	3.8	6.5	.1	1.9	0	0
2003	.7	16.6	3.8	6.5	.1	2.3	0	0
2004	.7	16.6	3.8	6.5	.1	2.7	0	0
2005	.7	16.6	3.8	6.5	.1	3.1	0	0
2006	.7	16.6	3.8	6.5	.1	3.5	0	0
2007	.7	16.6	3.8	6.5	.1	3.8	0	0
2008	.7	16.6	3.8	6.5	.1	4.2	0	0
2009-2025	0	0	0	0	0	4.2/yr	0	0
2026-2030	0	0	0	0	0	0	1.4/yr	16.6/yr
Total Cash Flows (Undiscounted)	10	215	49	84	2	118	8	100
NPV	8	176	40	69	1	65	3	38

[a] *Cost Savings*

F.7.2.11 High-level Waste Vitrification and Separation Waste Storage

The chemical separation operations at the Savannah River Site generate low-level waste and high-level waste that must be managed. Low-level waste is converted to a saltstone material and stored on-site. High-level waste is converted into a borosilicate glass log (vitrification) at the Defense Waste Processing Facility. Costs for the Defense Waste Processing Facility function, including low-level waste handling, are $1.77 million per Defense Waste Processing Facility log. Each log is equal to 300 MTR-equivalents or about 268 typical aluminum-based foreign research reactor spent nuclear fuel elements. The 17,800 aluminum-based elements in the proposed foreign research reactor spent nuclear fuel program equate to about 21,400 MTR-equivalents, including the 85 French RHF elements at 20 MTR-equivalents per element, but excluding target material. This generates 72 Defense Waste Processing Facility logs. Four Defense Waste Processing Facility logs are inserted into one waste package canister (i.e., 18 canisters of glass logs for the alternative). Each canister has an estimated cost of $480,000. Total discounted costs to prepare the high-level waste for geologic disposal are about $65 million. Disposal costs are extra.

F.7.2.12 Transportation to the Repository

The canisters of foreign research reactor spent nuclear fuel or vitrified high-level waste must be shipped to a geologic repository for ultimate disposal. For cost estimation only, the approximate distances from the Idaho National Engineering Laboratory and the Savannah River Site to the candidate repository at Yucca Mountain, Nevada, were used. (No claim regarding the suitability of Yucca Mountain is implied by this assumption.) Using these distances, and assuming shipments by truck, the undiscounted and discounted total costs of repository shipping are estimated at $19 million and $3 million, respectively for all spent nuclear fuel and about $1 million (undiscounted) if the aluminum-based spent nuclear fuel is converted to vitrified high-level waste (EG&G, 1994b).

Shipping costs are sensitive to aluminum-based spent nuclear fuel canister loadings. At approximately 14.4 kg (31.7 lb) U-235 per canister on average, the foreign research reactor spent nuclear fuel program (excluding target material) requires about 602 canisters for aluminum-based spent nuclear fuel and 16 canisters for TRIGA spent nuclear fuel. Reductions in the canister loading translate directly into more shipments. Shipping costs are not highly sensitive to high-level waste disposal packaging. In the base case, converting all the aluminum-based foreign research reactor spent nuclear fuel to vitrified high-level waste generates about 18 canisters.

F.7.2.13 Disposal of Spent Nuclear Fuel

The base case method of disposing of foreign research reactor spent nuclear fuel is intact in poisoned canisters. Preliminary costs to dispose of intact foreign research reactor spent nuclear fuel were developed for DOE's Office of Civilian Radioactive Waste Management in November, 1995 (TRW, 1995). Table F-133 shows the canister loadings, number of canisters, and canister costs (not repository costs) for the internal criticality packaging strategy of 14.4 kg (31.7 lb) U-235 per disposal canister.

Table F-133 Internal Criticality Packaging Strategy: Number Required and Cost Detail (in Thousands of 1996 Dollars)

Fuel Type	*U-235 Max kg/Element*	*Max (kg/package) U-235*	*Max Element/ Package*	*Package Cost ($1000)*	*Actual Element/ Package*	*Number Elements*	*Number Packages*	*Package Cost $1000*
LE MTR	0.42	43	102	114	48	4,838	101	11,490
HE MTR	0.42	14.4	34	92	24	5,692	237	21,819
Dense MTR	0.84	14.4	17	92	24	296	12	1,134
LE Tube	0.40	43	108	114	48	1,149	24	2,729
HE Tube	0.40	14.4	36	114	48	2,928	61	6,954
LE Cluster	0.49	43	88	215[a]	48	1,695	35	7,557
HE Cluster	0.49	14.4	29	114[a]	24	1,097	46	5,211
RHF	9.20	14.4	1	57.1	1	86	86	4,911
LE TRIGA	0.038	43	1132	315[b]	1008	3,834	4	1,198
HE TRIGA	0.133	14.4	108	92[b]	96	1,106	12	1,060
Total						22,721	618	$64,0964

Source: TRW, 1995

[a] *Cost category shifted up 1 to (approximately) account for the fact that the longer length of the Cluster elements precludes their being stacked in three layers (so that the ratio is only 4 to 1).*

[b] *Cost category shifted down by 2 to account for the fact that the smaller width (diameter) of the TRIGA elements permits 16 to fit into the same cross-sectional area as PWR assembly in the commercial waste package.*

The discounted canister-related cost of the packaging strategy displayed in Table F-133 is $11 million. The cost to dispose of the canisters depends on the size of the canisters and the loading levels. An estimate for disposal of full-size (i.e., commercial-type) spent nuclear fuel canisters prepared by the Idaho National Engineering Laboratory equated to $1.8 million per canister in 1994$ in 1994 (Stroupe, 1995), including transportation to the repository. This translated into $2.07 million per canister in 1996$ in 1996, the baseline cost for this EIS. These canisters contained 120 MTR-equivalents of aluminum-based spent nuclear fuel and 500 TRIGA spent nuclear fuel elements.

For the much smaller canisters and lower loading levels shown in Table F-133, a total undiscounted disposal cost (excluding transportation) of $373 million is estimated (TRW, 1995). This translates into an implied charge per canister of approximately $100 thousand for canisters containing aluminum-based spent nuclear fuel and $150 thousand for canisters containing TRIGA spent nuclear fuel. Assuming that repository development costs (1/3 of total repository charges) are incurred from 1996 through 2029 and repository emplacement costs (2/3 of total repository charges) are incurred in 2030 through 2035, the discounted cost of the disposal program (excluding the canisters) is approximately $110 million. About 95 percent of this charge is for disposal of aluminum-based spent nuclear fuel. Discounted total costs for intact disposal of aluminum-based spent nuclear fuel and TRIGA spent nuclear fuel including canister are approximately $125 million.

Total program costs are highly sensitive to the timing of disposal. Accelerating disposal to the 2015 to 2020 time period (rather than 2030 to 2035) reduces undiscounted costs by $50 million but increases discounted costs by $50 million. The savings arise from fewer years of storage prior to repository loading. The discounted cost penalty arises because the large outyear costs for repository development and emplacement lose 15 years of discounting.

F.7.2.14 Disposal of Vitrified High-level Waste

High-level waste is vitrified in the Savannah River Site Defense Waste Processing Facility. The borosilicate glass logs are inserted into waste packages (i.e., metal canisters similar to that used to dispose of commercial spent nuclear fuel) and disposed geologically. The cost to dispose of each waste package is estimated at $1.61 million, including transportation to the repository. At four Defense Waste Processing Facility logs per waste package, the aluminum-based foreign research reactor spent nuclear fuel would generate about 18 waste packages. Discounted disposal costs would be about $10 million. The discounted cost to dispose of the 35 TRIGA spent nuclear fuel canisters at the same loading as described in Section F.7.2.13 is about $10 million. Discounted costs increase by $15 million and undiscounted costs decrease by $25 million due to accelerating repository disposal by 15 years.

F.7.2.15 Storage or Reprocessing Overseas (Management Alternative 2)

For the purpose of the cost analysis, the primary steps in Management Alternative 2 are as follows:

- Each country retains its spent nuclear fuel.
- The countries with commercial nuclear power reprocessing programs reprocess their spent nuclear fuel. The other countries dry-store their spent nuclear fuel.
- Spent nuclear fuel is geologically disposed in an unspecified manner at multiple sites.

Discounted costs for Management Alternative 2 are estimated (very roughly) at $1.25 billion. Costs for this alternative are highly speculative since there is no basis for estimating how most countries would manage their spent nuclear fuel individually or collectively or what types of facilities or approaches they would (or could) select. Of the 41 countries in the proposed foreign research reactor spent nuclear fuel program, 22 have no commercial nuclear power infrastructure to support either a storage or a reprocessing program. These 22, and most of the remaining 19, have no clear program for geologic disposal. Since no country inside or outside the proposed foreign research reactor spent nuclear fuel program has offered to store or dispose of the spent nuclear fuel from other countries, there is no obvious method by which most of the countries in the program could manage their spent nuclear fuel. The costs shown here assume substantial cost penalties from the establishment of up to 22 new spent nuclear fuel storage installations, including all supporting infrastructure.

Reprocessing at the United Kingdom Atomic Energy Authority's facility at Dounreay, Scotland is already an option for Euratom countries that can accept the return of the reprocessing waste. If the United Kingdom Atomic Energy Authority were to reprocess all the material in the foreign research reactor spent nuclear fuel program, including fuels for which it has no current commercial capability, direct costs would exceed $1 billion. Logistics would be highly problematic, however, since the United Kingdom Atomic Energy Authority would require at least 35 years to complete the task at its currently offered capacity. The limited number of other facilities that could reprocess commercial spent nuclear fuel, e.g., the French facility at Marcoule, have not made any commitments to do so. The technical and cost uncertainties associated with disposal of either spent nuclear fuel or high-level waste are entirely speculative but must be considered extremely high.

Overall, there is no basis for assuming that distributed management of the spent nuclear fuel and, in particular, distributed geologic disposal of the spent nuclear fuel or high-level waste, could be accomplished at a cost remotely resembling that of the United States or any other country with a large-scale commercial nuclear power infrastructure.

F.7.3 Interpreting the Minimum Program Costs

Table F-120 (Section F.7.1.2) showed the minimum discounted program costs for the five bounding scenarios. The table showed that for the discount rates appropriate for the U.K. and U.S. portions of the program, hybrid chemical separation/reprocessing of aluminum-based spent nuclear fuel in the United States and the United Kingdom (Management Alternative 3) was about as costly as chemical separation of aluminum-based spent nuclear fuel in the United States alone. Either of the chemical separation/reprocessing approaches was substantially less costly than storing and directly disposing of all the spent nuclear fuel in the United States.

In interpreting the minimum discounted program costs, note that important components of the costs of multiple alternatives are fixed or nearly fixed. Table F-134 shows this relationship. For example, shipping to the United States is the same whether all the spent nuclear fuel is stored or separated. This means that the differences between the costs for the key management function (i.e., storage and disposal or chemical separation and disposal) are substantially larger (in percentage terms) than the differences between the total costs of an implementation alternative. It also means that risks in the unique components of the various implementation alternatives will have an outsized impact on the relative costs of the alternatives.

Table F-134 shows that the undiscounted costs for Management Alternative 1 (storage) exceed $1.4 billion, excluding target material and all other cost and risk factors. The undiscounted costs for Management Alternative 1 (revised to incorporate chemical separation) are approximately $1 billion. Undiscounted costs for Management Alternative 3 are about $1.1 billion. A substantial portion of the cost

premium for Management Alternative 3 is due to diseconomies of scale in using the Savannah River Site for two-thirds of the aluminum-based foreign research reactor spent nuclear fuel rather than all of the foreign research reactor spent nuclear fuel. Although not shown in Table F-134, use of a 4.9 percent discount rate for the European component of Management Alternative 3 would generate total costs that are indistinguishable from those under Management Alternative 1 (revised to incorporate chemical separation).

Table F-134 Composition of Minimum Program Costs for Spent Nuclear Fuel Management, 1996 Dollars

	Logistics & Program Management	*Shipping to United States*	*P-1 Operations at SRS*	*P-2 Operation at SRS, including Repository Fuels Qualification*	*P-1/P-2 Operations at INEL, Including Repository Fuels Qualification*	*Ship & Dispose Spent Nuclear Fuel*	*Reprocess Aluminum-Based Spent Nuclear Fuel (Net)*	*Stabilize, Ship & Dispose High-Level Waste*	*Dounreay (Including Blend Down, Net)*	*Total*
Management Alternative 1 (Storage, Dry)[a] Undiscounted	65	194	238	354	92	486	0	0	0	1428
Discounted	47	141	174	195	47	123	0	0	0	727
Management Alternative 1 revised to incorporate Chemical Separation[a] Undiscounted	65	194	190	55	92	13	231	150	0	989
Discounted	47	141	148	31	47	3	138	68	0	623
Management Alternative 1 revised to incorporate a New Technology[b] Undiscounted	65-70	+218	263	222-376	92	123-715	0	0	0	983-1,734
Discounted	47-52	158	191	129-241	47	54-244	0	0	0	626-933
Management Alternative 3[a] Undiscounted	44	124	136	37	92	13	113	100	417	1076
Discounted	32	90	108	21	47	3	73	45	263	682

[a] *No target material*

[b] *Includes target material*

Variations in the share of aluminum-based spent nuclear fuel that is stored/disposed in the United States versus separated/disposed in the United States shift the program costs within the boundary points established in Table F-134 for implementation alternatives to Management Alternative 1. This shift is non-linear, as summarized below:

- Spent Nuclear Fuel Repository Qualification - The proposed foreign research reactor spent nuclear fuel program would be responsible for the costs to repository-qualify an estimated three types of aluminum-based spent nuclear fuel. The discounted cost to characterize these three fuel types is approximately $25 million. (See Table F-123). If all the aluminum-based foreign research reactor spent nuclear fuel were separated, there would be no repository qualification and no charge to the program. Whether separation of less than

all the aluminum-based spent nuclear fuel eliminates one or more fuel types from qualification requirements would depend on when the fuel was received, where each fuel type appeared on the prioritization for separation, and how long separation continues.

- Canyon Operating Costs - Canyon operating costs allocated to the foreign research reactor spent nuclear fuel program are at a minimum during the years when processing is incremental to processing under the Interim Management of Nuclear Materials EIS (1998 to 2002) and higher afterwards. Switching from incremental costing to average variable costing increases annual costs from as little as $1 million for 2,880 MTR-equivalents to about $32 million. Including the phase-down penalty (Section F.7.2.8) increases the cost by approximately $33 million per year. The timing of the switch from incremental costing to average variable costing (and thus the impact on the foreign research reactor spent nuclear fuel program) depends on decisions made under the Interim Management of Nuclear Materials EIS (DOE, 1995b) and a facilities utilization study underway at the Savannah River Site. The timing of any deferral penalty is subjective. It depends on whether other missions for the Canyons have been identified and whether plans to deinventory the Canyon used by the foreign research reactor spent nuclear fuel program have been developed. It is clear that Canyon operations costs allocable to the foreign research reactor spent nuclear fuel program per year or per MTR-equivalent would be much higher after 2002 than before 2002 but it is not certain how much higher or when they would become higher. This uncertainty prevents a linear estimation of separation costs according to the quantity of material processed. Section F.7.4 discusses this issue in more detail.

- Staging and Characterization Facility Capital Costs -- The Savannah River Site plans to construct a staging facility to transfer spent nuclear fuel from the existing wet basins to interim dry storage and ultimately to repository canisters. The unallocated discounted capital cost of this facility exceeds $150 million. There is no necessarily correct way to allocate the capital costs of this facility since it supports multiple components of multiple programs and is sized according to joint requirements of multiple programs. Section F.7.2.6 described the cost allocation approach used in this EIS. Approaches that could increase the costs allocated to the foreign research reactor spent nuclear fuel program are also plausible.

- Basin Operating Costs -- The Savannah River Site has estimated the costs to operate RBOF and L-Reactor disassembly basin over a roughly 10-year period at a round-the-clock operations level but has no generalized relationship that permits continuous variation in basin costs according to the number of elements received or stored. Costs depend on the timing of the receipts, the amount of characterization and canning, intra-site and inter-site transfer requirements, the variability in year-to-year staffing, and other factors.

Section F.7.4 outlines four additional groups of factors of significance in using the minimum program costs in Table F-120 as a decision basis for the program.

F.7.4 Interpreting the Other Cost Factors

Table F-120 showed the minimum discounted cost for the five bounding scenarios. The costs in Table F-120 include component contingencies but they do not include system risks, component and non-component risks, or the effects of discount rate changes.[14] Table F-121 showed these latter factors for the five scenarios. Detailed discussions are presented below. Real escalation is excluded from all costs in both tables.

F.7.4.1 Systems Integration and Logistics

The minimum program costs include the contingencies related to individual components of the program, e.g., shipping, basin operations, storage, transfers, and disposal. The minimum program costs do not include systems integration or logistics risks. The proposed foreign research reactor spent nuclear fuel program involves 41 foreign countries (a majority of which have no commercial nuclear power program), dozens of foreign ports, 13 years of receipts, up to 10 domestic ports, as many as 250 cross-country spent nuclear fuel shipments, at least two management sites, and developmental technologies (especially repository disposal technologies). Substantial systems integration bottlenecks could arise in many technical areas, e.g., insufficient casks to ship at the required rate or at the estimated loadings; vulnerability-related shutdowns at existing facilities; requirements for on-site canning prior to cask loading; unplanned requirements for dry storage characterization or conditioning; unexpected facilities requirements for meeting the repository waste acceptance criteria; delays in repository acceptance; and so forth, including normal project (not component) contingencies.

Substantial bottlenecks could also arise in many procedural areas, e.g., incompatibilities with Naval programs at the Idaho National Engineering Laboratory; requirements for on-site inspections by the International Atomic Energy Agency; constraints on shipments, duration of shipments, shipment routes, or quantities of materials shipped pursuant to agreements with the states, and so forth. Because the list of technical and procedural issues that could delay and complicate the program is both long and highly plausible, it is realistic to expect costs to increase above the component-level minimums that make up Table F-120. This risk is estimated at 10 to 15 percent of minimum discounted program costs.

F.7.4.2 Program Component Risks

Several key components of the foreign research reactor spent nuclear fuel program are uncertain. This section discusses the most important probability-adjusted uncertainties (risks).

- ***The method of disposal of spent nuclear fuel-*** The base case assumption is that aluminum-based HEU spent nuclear fuel and HEU TRIGA spent nuclear fuel can be loaded into poisoned canisters and disposed at the equivalent of 14.4 kg (31.7 lb) U-235 per canister. This packing density could be unacceptable to the repository program. Processing the uranium into an isotopically neutral mass (1 percent U-235) would require construction of a new melt-and-dilute facility. Construction and operation of this facility

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14 Contingencies refer to costs that are certain to occur based on historical experience with programs of similar maturity. These costs are grouped under the term "contingency" because they cannot be line-itemized. Uncertainties refer to changes in the costs of individual components or the overall program that might occur due to unknown changes in regulations, technical conditions, operational status, etc. They are assigned a probability based on their likelihood. Thus, contingencies will occur--they just cannot be line-itemized; uncertainties may occur-- they are adjusted for their probability of occurrence and expressed as risks.

could add $100 million or more to the cost of spent nuclear fuel disposal. Processing the spent nuclear fuel to avoid severe mass limitations on disposal is considered a high probability event.

- ***The adequacy of limited characterization of the spent nuclear fuel*** - There is technical uncertainty about the requirements for characterizing and conditioning the spent nuclear fuel before storing it. At the Savannah River Site, the characterization stage consists of checking the history of the spent nuclear fuel and its paperwork (documentation), visual inspection, gamma scanning (to verify the presence and amount of fissile material), and a leak detection test ("sipping") to determine if any fission products are escaping from the spent nuclear fuel elements. Canning would be limited to degraded elements only. If more extensive characterization and canning is required, new hot cells may be required. Allocable discounted costs to add and operate a hot cell at the staging facility are on the order of $100 million. The requirement for additional characterization and conditioning is a moderately probable event.

- ***Bottlenecks at the Defense Waste Processing Facility*** - Complete separation of aluminum-based spent nuclear fuel at the Savannah River Site generates about 72 Defense Waste Processing Facility logs at a cost of $1.77 million per log. The Savannah River Site estimates that for capital costs of about $100 million and operating costs of about $40 million per year, it could remove bottlenecks at the Defense Waste Processing Facility such that the cost would decline to $1.0 million per year. For the foreign research reactor spent nuclear fuel program, the allocated cost of the capital and operating requirements to relieve the bottleneck is a few million dollars. The discounted savings would be in the range of $50 million. The likelihood that the foreign research reactor spent nuclear fuel program would realize these savings is low to moderate.

- ***Failure to commercially sell the recovered uranium*** - The Savannah River Site might not be allowed to blend-down the recovered HEU for sale as power reactor fuel. DOE, for example, could choose to safeguard the HEU and isolate its chemical separation operations from the commercial power market. This would cost the foreign research reactor spent nuclear fuel program an additional $70 million. The likelihood that the foreign research reactor spent nuclear fuel program would fail to recover this value is low.

F.7.4.3 Non-Program Risks

The key non-program risk is that the cost of repository disposal increases across the board due to a change in scope (not due to escalation within the existing scope). The repository cost allocation used in this EIS assumes no monitored retrievable storage and one geologic repository. If either of these assumptions is incorrect, the cost of the repository component of the program would increase by about 20 percent. If both are incorrect, the cost of the repository component of the program would increase by about 40 percent. These increases translate into cost increases for geologic disposal of intact foreign research reactor spent nuclear fuel of about 5 to 10 percent. The cost of the chemical separation alternative (including disposal of TRIGA spent nuclear fuel) would increase by about 1 to 2 percent.

Cost escalation in the base repository program would also increase the allocated costs for the foreign research reactor spent nuclear fuel program. This type of cost escalation is highly speculative but could be in the tens of millions of dollars for the storage alternatives. Escalation is treated separately from other cost risks.

A second non-program risk is that one or more of the EISs that relate to materials management and facilities use at the Savannah River Site or the Idaho National Engineering Laboratory (besides the foreign research reactor spent nuclear fuel EIS) leads to legal or regulatory action that delays all site activities and throws the foreign research reactor spent nuclear fuel program off-schedule or out of the planned facilities.

F.7.4.4 Discount Rates

This EIS uses the real discount rate specified by the Office of Management and Budget for long-term government projects evaluated in the year ending February 1996 (OMB, 1995). The specified rate, 4.9 percent, is historically high. It compares to Office of Management and Budget rates of 3.8 percent, 4.5 percent, and 2.9 percent for the years ending in February of 1993, 1994, and 1995, respectively (OMB 1992; OMB, 1993; OMB, 1994). It also compares to measured real, long-term government interest rates of 3.2 percent, 2.9 percent, 4.1 percent, and 3.4 percent (through 1995 quarter 2), respectively for the years 1992, 1993, 1994, and 1995 (FRB Cleveland, 1995). Finally, it compares to a Congressional Budget Office estimate of 2 percent for government projects independent of the period and duration (Hartman, 1990).

Unlike the United States, the United Kingdom issues some debt instruments that are the equivalent of inflation-adjusted treasury securities. In recent years, these have yielded between 2 and 5 percent. The rate as of mid January, 1996 was approximately 3.6 percent. The United Kingdom is also currently considering the required discount rate (i.e., real rate of return) on trust funds to provide for decommissioning commercial nuclear power plants. Although no decision has been reached, the government supports a 6 percent rate while the United Kingdom nuclear utilities support a 2 percent rate.

The appropriate discount rate for the analysis is the risk-free rate at which funds received today can be invested to cover future expenses. Since receipt of the revenues precede expenditures, a conservative rate is low. This is the reverse of the more common situation where a high rate is used to discount future receipts compared to current expenses. Moreover, where fixed and certain revenues precede variable and uncertain expenses, the need for a conservative (i.e., low) discount rate is even greater. At a 3 percent discount rate, the discounted cost of Management Alternative 1 (storage) increases by $175 million (to $900 million). The cost of Management Alternative 1 (revised to incorporate chemical separation) increases by $120 million. The effect on the storage alternative is much greater because the high out-year costs for repository canisters and repository emplacement are much more prominent at the lower discount rate.

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DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix G
Background Documents

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

THE WHITE HOUSE

Office of the Press Secretary

For Immediate Release September 27, 1993

FACT SHEET
NONPROLIFERATION AND EXPORT CONTROL POLICY

The President today established a framework for U.S. efforts to prevent the proliferation of weapons of mass destruction and the missiles that deliver them. He outlined three major principles to guide our nonproliferation and export control policy:

- Our national security requires us to accord higher priority to nonproliferation, and to make it an integral element of our relations with other countries.

- To strengthen U.S. economic growth, democratization abroad and international stability, we actively seek expanded trade and technology exchange with nations, including former adversaries, that abide by global nonproliferation norms.

- We need to build a new consensus — embracing the Executive and Legislative branches, industry and public, and friends abroad — to promote effective nonproliferation efforts and integrate our nonproliferation and economic goals.

The President reaffirmed U.S. support for a strong, effective nonproliferation regime that enjoys broad multilateral support and employs all of the means at our disposal to advance our objectives.

Key elements of the policy follow.

Fissile Material

The U.S. will undertake a comprehensive approach to the growing accumulation of fissile material from dismantled nuclear weapons and within civil nuclear programs. Under this approach, the U.S. will:

- Seek to eliminate where possible the accumulation of stockpiles of highly-enriched uranium or plutonium, and to ensure that where these materials already exist they are subject to the highest standards of safety, security, and international accountability.

- Propose a multilateral convention prohibiting the production of highly-enriched uranium or plutonium for nuclear explosives purposes or outside of international safeguards.

- Encourage more restrictive regional arrangements to constrain fissile material production in regions of instability and high proliferation risk.

- Submit U.S. fissile material no longer needed for our deterrent to inspection by the International Atomic Energy Agency.

- Pursue the purchase of highly-enriched uranium from the former Soviet Union and other countries and its conversion to peaceful use as reactor fuel.

- Explore means to limit the stockpiling of plutonium from civil nuclear programs, and seek to minimize the civil use of highly-enriched uranium.

- Initiate a comprehensive review of long-term options for plutonium disposition, taking into account technical, nonproliferation, environmental, budgetary and economic considerations. Russia and other nations with relevant interests and experience will be invited to participate in this study.

The United States does not encourage the civil use of plutonium and, accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. The United States, however, will maintain its existing commitments regarding the use of plutonium in civil nuclear programs in Western Europe and Japan.

Export Controls

To be truly effective, export controls should be applied uniformly by all suppliers. The United States will harmonize domestic and multilateral controls to the greatest extent possible. At the same time, the need to lead the international community or overriding national security or foreign policy interests may justify unilateral export controls in specific cases. We will review our unilateral dual-use export controls and policies, and eliminate them unless such controls are essential to national security and foreign policy interests.

We will streamline the implementation of U.S. nonproliferation export controls. Our system must be more responsive and efficient, and not inhibit legitimate exports that play a key role in American economic strength while preventing exports that would make a material contribution to the proliferation of weapons of mass destruction and the missiles that deliver them.

Nuclear Proliferation

The U.S. will make every effort to secure the indefinite extension of the Non-Proliferation Treaty in 1995. We will seek to ensure that the International Atomic Energy Agency has the resources needed to implement its vital safeguards responsibilities, and will work to strengthen the IAEA's ability to detect clandestine nuclear activities.

Missile Proliferation

We will maintain our strong support for the Missile Technology Control Regime. We will promote the principles of the MTCR Guidelines as a global missile nonproliferation norm and seek to use the MTCR as a mechanism for taking joint action to combat missile proliferation. We will support prudent expansion of the MTCR's membership to include additional countries that subscribe to international nonproliferation standards, enforce effective export controls and abandon offensive ballistic missile programs. The United States will also promote regional efforts to reduce the demand for missile capabilities.

The United States will continue to oppose missile programs of proliferation concern, and will exercise particular restraint in missile-related cooperation. We will continue to retain a strong presumption of denial against exports to any country of complete space-launch vehicles or major components.

The United States will maintain its general policy of not supporting the development or acquisition of space-launch vehicles in countries outside the MTCR.

For MTCR member countries, we will not encourage new space-launch vehicle programs, which raise questions on both nonproliferation and economic viability grounds. The United States will, however, consider exports of MTCR-controlled items to MTCR member countries for peaceful space launch programs on a case-by-case basis. We will review whether additional constraints or safeguards could reduce the risk of misuse of space launch technology. We will seek adoption by all MTCR partners of policies as vigilant as our own.

Chemical and Biological Weapons

To help deter violations of the Biological Weapons Convention, we will promote new measures to provide increased transparency of activities and facilities that could have biological weapons applications. We call on all nations — including our own — to ratify the Chemical Weapons Convention quickly so that it may enter into force by January 13, 1995. We will work with others to support the international Organization for the Prohibition of Chemical Weapons created by the Convention.

Regional Nonproliferation Initiatives

Nonproliferation will receive greater priority in our diplomacy, and will be taken into account in our relations with countries around the world. We will make special efforts to address the proliferation threat in regions of tension such as the Korean peninsula, the Middle East and South Asia, including efforts to address the underlying motivations for weapons acquisition and to promote regional confidence-building steps.

In Korea, our goal remains a non-nuclear peninsula. We will make every effort to secure North Korea's full compliance with its nonproliferation commitments and effective implementation of the North-South denuclearization agreement.

In parallel with our efforts to obtain a secure, just, and lasting peace in the Middle East, we will promote dialogue and confidence-building steps to create the basis for a Middle East free of weapons of mass destruction. In the Persian Gulf, we will work with other suppliers to contain Iran's nuclear, missile, and CBW ambitions, while preventing reconstruction of Iraq's activities in these areas. In South Asia, we will encourage India and Pakistan to proceed with multilateral discussions of nonproliferation and security issues, with the goal of capping and eventually rolling back their nuclear and missile capabilities.

In developing our overall approach to Latin America and South Africa, we will take account of the significant nonproliferation progress made in these regions in recent years. We will intensify efforts to ensure that the former Soviet Union, Eastern Europe, and China do not contribute to the spread of weapons of mass destruction and missiles.

Military Planning and Doctrine

We will give proliferation a higher profile in our intelligence collection and analysis and defense planning, and ensure that our own force structure and military planning address the potential threat from weapons of mass destruction and missiles around the world.

Conventional Arms Transfers

We will actively seek greater transparency in the area of conventional arms transfers and promote regional confidence-building measures to encourage restraint on such transfers to regions of instability. The U.S. will undertake a comprehensive review of conventional arms transfer policy, taking into account national security, arms control, trade budgetary and economic competitiveness considerations.

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United States Government | Department of Energy

memorandum

DATE: DEC 28 1994

REPLY TO
ATTN OF: EM-37

SUBJECT: Analysis of a Potential New Processing Facility in the Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement

TO: Jill E. Lytle
Deputy Assistant Secretary for Waste Management, EM-30

Based on a series of meetings held between staff from EM-4 and EM-30 during early December, I request that you take immediate action to include in the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Environmental Impact Statement (EIS) an alternative to initiate development work leading to a decision on whether to construct and operate a new SNF processing facility. The following parameters apply to this additional alternative:

- Any new facility would be capable of changing the FRR SNF into a form suitable for geologic disposal, without necessarily separating the fissile materials. A number of alternative processes would ultimately be considered for use in such a facility. Examples of these potential processes should be briefly discussed in the EIS.
- Due to the need for further research and development before the design of such a facility could be selected, the discussion of a new facility will be highly conceptual and programmatic in nature. Further NEPA analysis would be required prior to any decision to construct such a facility.
- Any new facility would be designed to operate safely and to minimize waste volumes, toxicity, and mobility.
- Any new facility would meet or exceed current environmental requirements.
- The alternative should consider construction of a potential new facility at all five of the sites considered for other FRR SNF management activities.
- The discussion should describe the range of quantities of spent fuel that such a facility might be designed to handle (hypothetically, from as little as just the foreign research reactor spent fuel that might be accepted under the FRR SNF EIS to a maximum of all of DOE's spent fuel).
- The design and operation of a new facility would be consistent with U.S. nuclear weapons nonproliferation policies, including the requirements of Presidential Decision Directive 13 regarding reprocessing.

- This alternative considering development potentially leading to a new SNF processing facility is to be in addition to the analysis of chemical separation of the FRR SNF that is already included in the FRR SNF EIS.
- In addition, consideration should be given to utilizing the National Academy of Science to assess the feasibility of using a new facility to produce a waste form that will meet the waste acceptance criteria for a geologic repository.

I recognize that incorporation of this alternative into the FRR SNF EIS at this stage in the development of the EIS will result in approximately a two week delay in the completion of the draft of the EIS. The draft was originally scheduled to be issued for public review and comment by the end of December 1994 and has recently been delayed about two weeks to resolve internal DOE comments. This change will result in a further delay and release of the draft FRR SNF EIS for public review and comment by no earlier than February 1995. This will probably result in a delay in the completion of the final FRR SNF EIS from June 1995 until July 1995. I understand that any delay in the completion of the FRR SNF EIS is likely to raise some objections among the FRR operators. Nevertheless, I consider that it is essential to evaluate this proposed new processing facility and that the small additional delay is acceptable.

Thomas P. Grumbly
Assistant Secretary for
Environmental Management

DEPARTMENT OF STATE
WASHINGTON

October 26, 1992

Dear Mr. Secretary:

During the 1992 International Meeting on Reduced Enrichment for Research and Test Reactors (RERTR) in Denmark, participants voiced very strong concern regarding the apparent reluctance of the Department of Energy to renew the Off Site Fuels Policy, to take back spent research reactor fuel from abroad.

Since 1978, the United States has encouraged countries to convert from the use of high enriched fuel (HEU) to low enriched fuel (LEU). This effort constitutes a key element of U.S. nuclear non-proliferation policy, which has been accepted with some reluctance by other countries, since it entails additional effort and expense on their part. Historically, the Off Site Fuels Policy has been an integral part of the conversion effort, which is perceived by countries as essential to meet reactor operating licensing requirements for disposition of spent fuel and to assure that their research reactor spent fuel is disposed of in a safe and reliable manner.

I fully recognize that renewal of this program will require DOE to resolve difficult and complex budgetary, environmental and technical issues. However, for a variety of reasons, I believe it is essential for DOE to move promptly to renew its policy of taking back foreign research reactor fuel.

We have worked hard for many years to reestablish the position of the United States as a reliable partner in nuclear commerce. We should not forfeit this effort by appearing uncertain about a policy which we have long supported and which is so critical to our non-proliferation objective of eliminating HEU from commercial use.

Clearly, we also do not want to forfeit the significant nuclear non-proliferation gains which have resulted from the RERTR program and our agreement to take back foreign research reactor spent fuel. Limiting the use and location of HEU abroad serves the security interests of both the United States

The Honorable
James D. Watkins,
Secretary of Energy.

and the international community as a whole. Hence, it is particularly disturbing to hear that some countries are considering halting their conversion programs, and even reverting to the use of HEU fuels in the event the United States does not agree to take back U.S.-supplied LEU spent fuel.

Over the past four years, we have maintained a dialogue with DOE concerning the importance of the spent fuel policy. Given the urgent need to resolve this matter, I strongly urge that DOE move quickly to reassure other governments that their spent fuel needs will be fully addressed and that we will continue to honor our commitments to them.

Sincerely,

Larry

Lawrence S. Eagleburger
Acting Secretary

UNITED STATES ARMS CONTROL AND DISARMAMENT AGENCY
Washington, D.C. 20451

THE DIRECTOR

07 DEC 1992

MEMORANDUM FOR THE SECRETARY OF ENERGY

SUBJECT: Reducing Foreign Inventories of U.S.-Supplied Highly Enriched Uranium

For many years the United States has encouraged reduced use of highly enriched uranium (HEU) for civil purposes as a key component of U.S. nuclear nonproliferation policy. This effort has met with some success, and the civil use of HEU has diminished, bringing reduced stockpiles and reduced transportation and diversion risks. An important incentive for foreign users of U.S.-supplied HEU to convert their reactors to low enriched uranium (LEU) fuel was the United States' program to take back the spent fuel.

Recent historic political developments have also presented opportunities for further reducing stockpiles of HEU abroad, thereby further promoting our nuclear nonproliferation objectives. We are arranging to purchase 500 metric tons of HEU from Russia for peaceful uses. South Africa has ended its HEU production and has offered to sell its stockpile to the United States.

I believe we should consolidate these gains and encourage further reduction of civil HEU use. It is essential to act soon to avoid damaging the longstanding and successful U.S. program that encouraged foreign operators to convert HEU research reactors fueled by the United States to the use of LEU fuel. Without appropriate action, some foreign operators might decide against conversion and others may switch back to HEU fuel. Moreover, new foreign suppliers of HEU may emerge.

In this regard, I have three recommendations:

1. Conclude contractual arrangements with appropriate foreign organizations to take back U.S.-supplied research reactor fuel following any necessary environmental determination.

2. Examine the feasibility of additional incentives that would be helpful or necessary toward ensuring the conversion

of those reactors for which alternative LEU fuels have been identified. A general review of the conversion program may be appropriate in any event in view of the recent amendment to the Atomic Energy Act which severely restricts future HEU licensing. In regard to that legislation, we would also support efforts to reestablish the LEU target development program for production of medical isotopes.

3. Ensure that the United States will make South Africa an attractive offer for its HEU.

I do not underestimate the difficulties posed by these recommendations. However, actions such as these would maintain and strengthen a longstanding and successful U.S. policy of reducing HEU stockpiles abroad -- a policy which will continue to promote global nuclear nonproliferation objectives.

Ronald F. Lehman II

Ronald F. Lehman II

THE DIRECTOR GENERAL

1993-07-01

Dear Madam Secretary,

Since 1978, the United States has encouraged countries to convert the cores of their research and test reactors from the use of highly enriched uranium (HEU) to nuclear fuels of low enriched uranium (LEU). This effort, initiated by President Carter, was an important element of the U. S. non-proliferation policy throughout most of the 1980s and was fully supported through the Reduced Enrichment for Research and Test Reactors (RERTR) programme by the International Atomic Energy Agency. The expiration of the U. S. Department of Energy's Off-Site Fuels Policy (the Policy) in 1988 has led to a crisis for the operators of research reactors in many countries where the laws are such that continuation of licensing and/or purchase of new nuclear fuels is contingent upon a resolution of spent fuel management problems. This situation is exacerbated for many reactor operators who complied with the wishes of the U. S. and converted their cores to LEU. They now have interim storage pools filled with irradiated HEU fuels and are trying to cope with a greater throughput of LEU fuels. The anticipated announcement that the U.S. DOE will renew the Policy and in due course begin the take back of research reactor fuels of U.S. origin from around the world will be very much welcomed by the Agency and many of its Member States.

However, because of the problems of spent fuel management facing the operators of many research reactors the Agency urges the earliest implementation of the Policy renewal. Some of these research facilities are the only sources of radioisotope production for medical uses in the countries in question, but face imminent closure unless they can resolve their problems of spent fuel management quickly. The Agency has initiated programmes to advise them, but the real solution for most of them is to return their irradiated research reactor fuels of U. S. origin. It is understood that the renewal of the Policy will require the solution of difficult and complex budgetary, environmental, transportation, legal and technical issues. Nevertheless, the Agency is confident that when the resources of the U. S. DOE are brought to bear on these problems that they will be resolved as soon as possible.

The Honourable Hazel O'Leary
Secretary of Energy
Washington, DC 20585
United States of America

Limiting the use and location of HEU fuels throughout the world remains a valuable objective and will serve the security interests of all nations. The Agency stands ready to help in any way it can consistent with its mandate and budgetary constraints.

Yours sincerely,

Hans Blix

THE SECRETARY OF STATE

WASHINGTON

July 2, 1993

Dear Madam Secretary:

I am writing to urge your personal support for renewal by the Department of Energy of the Off Site Fuels Policy for the acceptance of spent research reactor fuel from abroad.

The Department of State has strongly supported this policy because of its importance in gaining foreign cooperation in converting reactors from highly enriched uranium (HEU) to low enriched (LEU) fuel under the aegis of the Reduced Enrichment in Research and Test Reactors (RERTR) Program.

We recall Secretary Watkins confirmed in 1992 that the Department of Energy proposed to renew the Off Site Fuels Policy, but with the caveat that meeting the requirements of the National Environmental Policy Act (NEPA) could take as long as 2 to 3 years. We are concerned, however, about reports of substantial delays in the amendment of the existing Environmental Assessment, an essential early step in the NEPA process.

Foreign research reactor operators are reportedly highly concerned about a perceived change in DOE policy and have threatened to withdraw from further RERTR cooperation and to seek resumption of HEU supply from sources such as Russia.

A breakdown of the international consensus on conversion of research and test reactors to LEU and a return to an HEU fuel economy would undermine 15 years of intensive U.S. non-proliferation efforts on this matter and substantially reduce the ability of the U.S. to influence nuclear policy in bilateral and international fora.

In light of current developments, I urge your support for early reaffirmation by DOE to other governments of our continued commitment as a reliable supplier to fully address their spent fuel needs.

Sincerely,

Warren Christopher

Warren Christopher

The Honorable
Hazel R. O'Leary,
Secretary of Energy.

The Secretary of Energy
Washington, DC 20585

July 13, 1993

The Honorable Warren Christopher
Secretary of State
Washington, D.C. 20520

Dear Mr. Secretary:

This is in response to your letter dated July 2, 1993, urging my support for renewal of the Department of Energy's policy for the acceptance of spent research reactor fuel from abroad.

The Department of Energy remains committed to the Reduced Enrichment for Research and Test Reactors (RERTR) program, and to the proposal to establish a policy for the return of U.S. origin spent fuel from foreign research reactors. In response to your letter, and other inquiries we have received on this subject, we have taken a hard look at how we can expedite actions in these areas. We have decided on a three-tiered approach, as follows:

1. For any foreign research reactor spent fuel returns for which we can mutually agree that a bona fide emergency exists, the Department of Energy will join with you in consulting with the Council on Environmental Quality on the implementation of alternative arrangements for compliance with environmental review requirements pursuant to the emergency provisions of the Council on Environmental Quality's regulations implementing the National Environmental Policy Act (40 CFR 1506.11).

2. In order to be able to respond to any near-term situation in which the expiration of the Department's acceptance of foreign research reactor spent fuel may threaten the Reduced Enrichment for Research and Test Reactors Program, the Department has begun an expeditious Environmental Assessment of the proposed return of sufficient spent fuel to eliminate that threat. It is proposed that any near-term spent fuel returns would be conducted under the terms and conditions of the enclosed proposed policy and be limited to approximately 550 spent fuel elements which can be stored in existing DOE capacity. This Environmental Assessment is scheduled to be completed by September 1993, and, if appropriate, a proposed Finding of No Significant Impact will be issued for public review by no later than September 30, 1993. Our goal is to complete the National Environmental Policy Act review process of this proposed limited foreign research reactor spent fuel acceptance by the end of this calendar year.

3. For the longer term, the Department will undertake preparation of an Environmental Impact Statement that addresses the proposed return of all U.S. origin foreign research reactor spent fuel, as specified in the enclosed proposed policy. A notice of intent for preparation of this Environmental Impact Statement is in preparation and should be issued in August 1993. The Department intends to issue the draft of the Environmental Impact Statement for public review by no later than the

end of December 1994, and the final Environmental Impact Statement by the end of June 1995.

We cannot continue to address this issue in a business as usual manner. The actions outlined above reflect our determination to move forward promptly and our acknowledgement of the need for a new definition of national security - one that includes both nonproliferation and environmental concerns. To provide added emphasis to the urgency of this effort, the Department requests that the Department of State participate as a cooperating agency in preparation of this environmental documentation.

In conclusion, the Department is committed to work with you and representatives of the Council on Environmental Quality at any time that you consider an emergency situation may be developing. In the meantime, we are proceeding as expeditiously as possible on the actions outlined above.

Sincerely,

Hazel R. O'Leary

Hazel R. O'Leary

Enclosure

DRAFT

Proposed Foreign Research Reactor Spent Nuclear Fuels Acceptance Policy

13 July 1993

PURPOSE - This proposed Department of Energy policy would support United States nonproliferation policy, including one of its key elements, the Reduced Enrichment Research and Test Reactors Program. It would provide opportunities and incentives for research reactor operators in foreign countries holding United States origin spent nuclear fuel containing highly enriched uranium to return that spent nuclear fuel to the United States for storage and eventual geologic disposal. This proposed policy is intended to support the United States nonproliferation objective of eliminating United States origin highly enriched uranium from research reactor use. It is also consistent with Section 903(a) of the Energy Policy Act of 1992, which places further restrictions on the export of highly enriched uranium from the United States. This proposed policy would provide incentives to encourage and assist developing countries (defined below) in returning their United States origin highly enriched uranium research reactor spent nuclear fuel to the United States for storage and disposal. For developed countries, the policy would allow return of United States origin research reactor spent nuclear fuel to the United States for storage and disposal on a full-cost-recovery basis.

PROPOSED POLICY - The United States proposes to adopt a policy under which:

1. For developing countries (i.e., those eligible for assistance under the United Nations Assistance Program), the United States would offer to accept United States origin research reactor spent nuclear fuel containing highly enriched uranium for storage and disposal in the United States. The United States would reimburse the developing country for costs incurred in transportation of the spent nuclear fuel from the developing country to a receipt facility in the United States. Upon acceptance of the spent nuclear fuel in the United States, the United States would assume all responsibility for the spent nuclear fuel, including storage of the spent nuclear fuel in the United States, any preparation of the spent nuclear fuel for disposal, all transportation in the United States subsequent to spent nuclear fuel acceptance, and ultimate geologic disposal of the spent nuclear fuel in the United States.

2. For developed countries, the United States would offer to accept all United States origin research reactor spent nuclear fuel containing highly enriched uranium for storage, preparation for disposal, and eventual geologic disposal in the United States. Such acceptance would be conducted on a full-cost-recovery basis, with the developed country responsible for transportation of the spent nuclear fuel to a designated receipt facility in the United States and paying the United States the full cost of all storage, all transportation within the United States subsequent to spent nuclear fuel acceptance, disposal preparation, and ultimate geologic disposal.

3. To encourage the conversion of foreign research reactors currently using United States origin highly enriched uranium fuels to low enriched uranium fuels, the United States would offer to accept for storage and ultimate disposal certain United States origin low enriched uranium research reactor spent nuclear fuel. Specifically, low enriched uranium research reactor spent nuclear fuel of United States origin would be accepted for a ten year period following implementation of this policy from reactors that have already converted, or that were constructed to use and operate with low enriched uranium fuels. United States origin low enriched uranium research reactor spent nuclear fuel exported to research reactors that convert within five years of the effective date of this policy would also be accepted for a ten-year period following their initial order for low enriched uranium fuel.

   The acceptance of low enriched uranium research reactor spent nuclear fuel from developed and developing countries would be conducted on the same terms as stated in 1 and 2 above for highly enriched uranium research reactor spent nuclear fuel.

CONDITIONS

1. This proposed policy would apply only to receipt of spent research reactor nuclear fuel of United States origin.

2. Ownership of the spent nuclear fuel would be transferred to the United States upon acceptance of the spent nuclear fuel by the United States at a designated receipt and inspection facility in the United States.

3. All transportation within a developing country and to the United States receipt facility would be the responsibility of the developing country, but would be paid for by the United States (subject to United States approval of the transportation arrangements and costs).

   All transportation within a developed country and to the United States receipt facility would be the responsibility of, and would be paid for by, the developed country.

4. Criteria concerning the required condition of the spent nuclear fuel would be published by the United States as part of the announcement of this policy, to clarify conditions for acceptance of the spent nuclear fuel. In general terms, all spent nuclear fuel to be accepted by the United States would be required to be either intact and free of defects or canned to ensure the ability to safely contain and manage the spent nuclear fuel.

5. For developed countries, the fee to be paid to achieve full cost recovery would be established by the Department prior to entering into the agreements to accept the spent nuclear fuel. This fee would be based on estimates of the cost of the storage and disposal activities that would be required. The fee

schedule would be updated annually to account for items such as inflation, and experience with the program.

TERMINATION - This policy of accepting low enriched uranium research reactor spent nuclear fuel would expire ten years after the effective date of this policy (or ten years following placement of an order for low enriched uranium research reactor fuel to replace highly enriched uranium research reactor fuel, if such an order is placed within five years of the effective date of this policy). Therefore, countries and research reactor operators that plan to take advantage of this policy for spent nuclear fuel containing low enriched uranium should begin planning for their own national or regional means of storage and disposal of low enriched uranium research reactor spent nuclear fuel for use following termination of this policy.

The proposed policy for accepting research reactor spent nuclear fuel containing highly enriched uranium of United States origin would encourage all countries to return this United States origin research reactor spent nuclear fuel as soon as possible.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 03 1991

Mr. John J. Easton, Jr.
Assistant Secretary of Energy
International Affairs and Energy Emergencies
U.S. Department of Energy
Washington, DC 20585

Dear Mr. Easton:

I am responding to your letter of May 31, 1991, requesting the comments of the Nuclear Regulatory Commission on issues related to the Department of Energy's consideration of renewing the Off-Site Fuels Policy.

The NRC staff believes that it is in the best interest of the United States to allow spent U.S.-origin high enriched uranium (HEU) fuel from domestic and foreign research reactors to be returned to DOE for processing and storage. Such a take-back policy reduces certain safeguards, physical security and safety concerns associated with the indefinite, long-term storage of irradiated HEU fuel in diverse locations. It would, of course, also alleviate the serious lack of spent fuel storage capacity being experienced by several research facilities, including ones in Japan and several European countries. In this regard, however, we assume that in implementing a resumption of the DOE policy to accept spent HEU fuel, the U.S. would not diminish its pressure on foreign countries to continue their best efforts to convert remaining HEU-fueled research reactors to low enriched uranium (LEU) fuel.

In the same vein, it would appear useful for DOE also to extend, beyond the expiration date of December 31, 1992, its offer to take back spent U.S.-origin LEU research reactor fuel from domestic and foreign users. DOE's current examination of the Off-Site Fuels Policy will no doubt address the question of whether or not this commitment is essential to U.S. efforts to minimize the use of HEU fuel in research reactors abroad. Your analysis of this and other aspects of the policy will be of great interest to NRC and can be expected to influence our future export licensing activities.

I trust that these general comments are useful.

Sincerely,

James M. Taylor
Executive Director for Operations

THE SECRETARY OF STATE

WASHINGTON

December 12, 1995

Dear Madam Secretary:

As we move to the final stages of preparing the Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (the EIS), I want to reaffirm the critical need for implementing this policy. The spent fuel acceptance policy which the EIS supports is central to our goal of preventing the spread of nuclear weapons -- and therefore to a major national security objective of this Administration.

One of the key elements of the President's nonproliferation policy has been to minimize and eventually to eliminate the use of high enriched uranium (HEU) in civil world commerce. I greatly appreciate the efforts that you personally have made to reinvigorate the Reduced Enrichment for Research and Test Reactors (RERTR) program which is designed to convert research reactors around the world from using HEU to LEU fuel.

As you know, the willingness of research reactor operators to support this vital program depends on our willingness to assist them with disposition of the spent fuel produced from nuclear materials which the United States originally supplied. Failure to implement this policy successfully would deal a crippling blow to our efforts to minimize the commercial use of HEU.

I hope that you will proceed soon to publish the final EIS and to begin implementing the policy using available practical and appropriate means to ensure the expeditious implementation of the program.

You and your staff are to be commended for the outstanding effort made in preparing the EIS. The Department of State stands ready to cooperate with you in whatever way we can.

Sincerely,

Warren Christopher

Warren Christopher

The Honorable
Hazel R. O'Leary,
Secretary of Energy

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix H
General Provisions of Transportation Planning for the Shipments of Foreign Research Reactor Spent Nuclear Fuel

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

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Appendix H
General Provisions of Transportation Planning for the Shipments of Foreign Research Reactor Spent Nuclear Fuel

H.1 Overview

This appendix is prepared to provide a description of the transportation, emergency response, security, and communications planning that would typically occur prior to any acceptance of foreign research reactor spent nuclear fuel under Management Alternative 1 or Management Alternative 3. Appendix H expands upon Sections 2.0 (Proposed Action and Alternatives) and 2.7 (Characteristics of Emergency Management and Response) in this Environmental Impact Statement (EIS). The information in this appendix is based on U.S. Department of Energy (DOE), State, Tribal, and local authorities' composite experience in planning for several successful radioactive material shipping campaigns, such as Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel and Cesium 137 Capsule Return Program shipments. This appendix is not meant to be all-inclusive as each shipping campaign differs slightly because of the material being shipped, the transportation mode, and the level of involvement of State, Tribal, and local governments.

The implementation of Management Alternative 1 or Management Alternative 3 would involve an ongoing and interactive planning process between DOE, States, Tribes, local authorities, other Federal agencies, the shipper of record/shipper's agent, and the carrier. In past shipping campaigns, these participants have developed an overall "Transportation Plan." This Plan is a blueprint for transportation, emergency response, security, and communications operations that would take place prior to, during, and after the completion of a shipment. Agreements between the various parties are detailed in the Plan. What follows is a description of the main elements of such a Plan. Additional information is available in the *DOE Program Manager's Guide to Transportation Planning* (DOE, 1995).

H.2 Transportation Considerations

Transportation planning integrates a wide range of expertise and requirements, including program management, material handling and packaging, transportation operations (traffic management), key governmental involvement, public information, environmental safety and health, and emergency preparedness. Planning would be clarified in a Transportation Plan that would document the planned logistics for foreign research reactor spent nuclear fuel shipments. The focus of this Plan would be operational; e.g., the handling, packaging, and transport of the foreign research reactor spent nuclear fuel shipment through sequential steps resulting in safe transport of this material to a management site. The plan would include organizational responsibilities of DOE, foreign research reactor operator/shipper of record, corridor jurisdictions and other Federal agencies. It would contain information on shipment schedules, the port(s) of entry, the mode of transport from port(s) of entry to a selected management site, an illustration of the shipment route, emergency plans and contacts, and communications strategies. The plan would include graphic representation of schedules of requirements for pre-, during, and post-shipment activities showing number of days to prepare, load, ship, unload and return the empty cask.

H.2.1 Organizational Roles and Responsibilities

DOE Headquarters: DOE Headquarters sets overall policy for the spent nuclear fuel program and for transportation, resolves policy questions, issues guidance, and provides information for use in transportation activities. DOE Headquarters provides a management team that offers general guidance and technical assistance to the field office implementing the program activities. As the responsible government agency for this program, DOE would ensure overall program coordination with involved organizations and agencies as outlined in this appendix. DOE is committed to providing corridor jurisdictions with technical assistance to help prepare for any shipments, and to supporting the management sites in implementing this program if foreign research reactor spent nuclear fuel is accepted into the United States. Prior to the implementation of a foreign research reactor spent nuclear fuel acceptance policy, DOE would coordinate with corridor jurisdiction authorities to resolve issues related to transport, emergency response, security, and communications management. DOE Headquarters would ensure, through contractual agreement, that the foreign research reactor operators comply with DOE's transportation, emergency response, security, and communications provisions, which exceed U.S. Nuclear Regulatory Commission (NRC) and/or U.S. Department of Transportation regulatory requirements.

Management Sites: The management site would be responsible for overall program management for shipments to the storage facility and transfer of spent nuclear fuel elements from the foreign research reactor to the management site. The management site would require shipping activities to be in compliance with applicable regulations and considerations. When necessary based on shipping experience and reactor operator capabilities, DOE would have a team of specialists from the management site travel to foreign research reactor facilities to verify that spent nuclear fuel and shipping arrangements meet DOE's transportation acceptance requirements. The management site would coordinate all communications activities with DOE Headquarters, other Federal agencies, and corridor jurisdiction authorities. Unloading and cross-site movement to a temporary storage facility would be conducted in accordance with management site operating procedures for acceptance and unloading of spent nuclear fuel.

Reactor Operators: The reactor operator would either perform the duties of the "Shipper of Record", as specified below, or contract with a shipping firm to act as the "Shipper of Record."

Shipper of Record: The Shipper of Record would submit a composite transportation physical security plan to the NRC, for domestic transport, which would contain a U.S. Department of Transportation-approved or State-designated route for highway or rail shipments. At DOE's request, the Shipper of Record would also develop its transportation planning document, which would address considerations such as vessel selection, pre-notifications, import authorizations, port arrangements, U.S. customs clearance, carrier arrangements, schedule, and emergency response. The reactor operator and/or the Shipper of Record are often referred to as the licensee. The licensee refers to the license that has been granted by the NRC to handle spent nuclear fuel.

Other Federal Agencies: Other Federal agencies would cooperate with DOE to ensure safe transportation of the foreign research reactor spent nuclear fuel. These agencies include:

> ***Department of Transportation:*** The U.S. Department of Transportation would perform rail track inspections; ensure that each shipment was in compliance with the regulations for the transport of spent nuclear fuel; and serve as the U.S. Competent Authority in the review of foreign transportation packaging certificates. The U.S. Department of Transportation also maintains a list of State-designated alternative highway routes.

NRC: The NRC would be responsible for review and approval of the composite transportation physical security plan, which would include the routes submitted by the foreign research reactor operators or the shipper of record. The NRC could perform radiological monitoring of the transportation packages at the port.

Department of Defense: If a military port-of-entry were utilized, organizational responsibilities would be defined and agreed upon in advance between DOE and the U.S. Department of Defense. Interagency Agreements would be established between DOE and the appropriate U.S. Department of Defense element to define the provisions under which DOE would be allowed to ship through the U.S. Department of Defense port. Security and safeguard measures would comply with military requirements in addition to NRC requirements.

Corridor Jurisdictions: State, Tribal, and local authorities have primary responsibility for the health and welfare of their citizens. Most State, Tribal, and local emergency preparedness organizations have, as a minimum, the basic capabilities to respond to a transportation emergency. In addition, the following authorities could be exercised:

- Corridor jurisdictions could exercise responsibility for vehicle and equipment inspections.
- Corridor jurisdictions would be responsible for notifying DOE of any road, rail or weather conditions that could affect a shipment crossing their jurisdiction.
- Corridor jurisdictions would designate a central point of contact.
- Corridor jurisdictions would interact with local officials on information and emergency planning activities.
- If highway transport were utilized, DOE, corridor jurisdictions, and the carrier would jointly establish policies regarding bad weather/road conditions and safe parking procedures.

Carrier: The carrier would be responsible for safely transporting the spent nuclear fuel packages and transport containers from the port of entry to the management site. If a shipment were to be completed by rail, this responsibility would include ensuring that rail lines and equipment were properly inspected and in good operating order, following NRC approval of the composite transportation physical security plan, and coordinating with the corridor jurisdictions to arrange inspections as required. If shipment were by highway, this responsibility could include obtaining required State permits, using the designated highway route-controlled quantity route, and arranging for vehicle inspections, as required. The carrier would be required to have a transport plan addressing considerations such as emergency recovery, transportation regulation compliance, two-way communication with rail/truck operators, and subcontractor emergency response plans.

H.2.2 Advance Information

Prior to the transport of spent nuclear fuel within or through a State, the Shipper of Record would notify the Governor or the Governor's designee in writing.

H.2.3 Tracking of Shipment

For shipments of foreign research reactor spent nuclear fuel, DOE policy would require that a satellite tracking/communication system be used.

H.2.4 Implementation of United States/International Atomic Energy Agency Agreement

The International Atomic Energy Agency is an agency of the United Nations headquartered in Vienna, Austria. The International Atomic Energy Agency establishes standards for radioactive materials transport. These model regulations may be adopted by individual nations. The emphasis of the International Atomic Energy Agency model regulations is on package integrity. The NRC and the U.S. Department of Transportation both periodically review and revise their regulations to bring them into general accord with the International Atomic Energy Agency regulations.

H.2.5 Packaging Description

The packaging for transporting spent nuclear fuel is designed to provide containment of its contents as required by international and Federal regulation. Type B packages, used to transport spent nuclear fuel, are designed to protect and retain their contents in both normal and severe accident conditions. Foreign-licensed Type B package designs are reviewed and approved for acceptance by the U.S. Department of Transportation. Figure H-1 displays a transportation cask typically used to transport spent nuclear fuel.

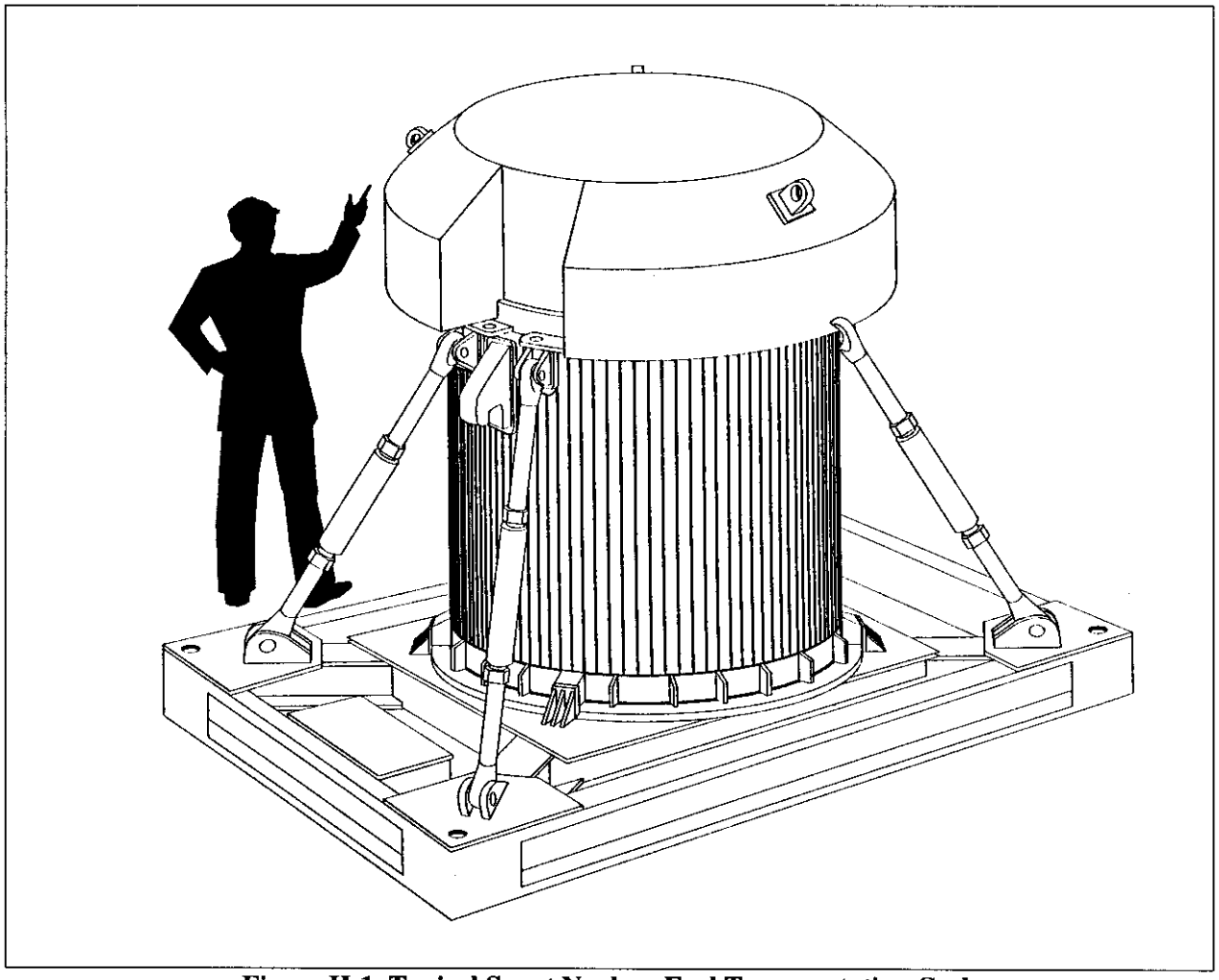

Figure H-1 Typical Spent Nuclear Fuel Transportation Cask

H.3 Emergency Preparedness Considerations

Emergency preparedness for transport of spent nuclear fuel is a vital part of the transportation planning process. Corridor jurisdictions having authority over areas through which these shipments would pass have primary responsibility for protecting the public and the environment and for establishing incident command in the unlikely event that an emergency should occur involving the shipments. DOE would work with State, Tribal, and local authorities to complement existing emergency preparedness capabilities. The carrier for these shipments would be responsible for providing emergency response assistance and recovery/restoration actions, if required. DOE would provide technical operations advice and radiological monitoring assistance to civil authorities and carriers of these shipments, when requested.

H.3.1 Emergency Preparedness

Corridor jurisdiction authorities have primary responsibility for the health and welfare of their citizens. Most States maintain specialized teams capable of responding to hazardous materials incidents. Through the capabilities these teams currently possess for dealing with potential accidents involving other hazardous materials (i.e., hazardous chemicals), they should already have the capability to deal with most plausible accidents involving spent nuclear fuel. Nevertheless, to assist in planning and preparedness for an unlikely but theoretically possible transportation emergency involving any foreign research reactor spent nuclear fuel shipments, DOE would offer a variety of emergency response resources and information to complement existing emergency preparedness programs, and also maintains a comprehensive emergency management system, particularly for radiological emergencies. The emergency management system includes training courses, Regional Coordinating Offices, and DOE Radiological Assistance Program teams.

Corridor Jurisdictions Hazardous Materials or Radiological Response Teams: Most corridor jurisdictions maintain specialized hazardous materials response teams that could be activated to provide technical assistance and mitigation during emergencies. State teams are activated at the request of an Incident Commander or other appropriate State or local authority.

Carrier Emergency Response Assistance: The carrier would provide technical response assistance to corridor jurisdiction responders as required by event scene conditions.

DOE Radiological Assistance Program: DOE's Radiological Assistance Program teams are administered by eight Regional Coordinating Offices. Each Regional Coordinating Office has access to radiological monitoring and assessment capabilities to provide assistance in radiological emergencies. Additional DOE technical experts are available to provide advice on material characteristics and mitigation, packaging and its tie-downs, and radiological monitoring and assessment requirements.

H.3.2 Notifications and Communications

In the unlikely event of a transportation incident involving foreign research reactor spent nuclear fuel, the carrier operator would notify the appropriate State and local authorities, the carrier dispatch center, and the management site communication center. The management site would also inform the appropriate corridor jurisdiction authorities. In any case, State, Tribal, and local authorities would be tied into the transportation plan's communication network arranged for the shipment.

The management site would serve as the designated communication center. DOE policy for transporting foreign research reactor spent nuclear fuel requires that a satellite communication system be used. The carrier may also provide its own transportation tracking system. Further details for the notification, communications, and other responsibilities of the communication center would be outlined in the Shipper of Record's transportation plan.

H.3.3 Emergency Response

If an accident requiring emergency response were to occur, the following emergency-related roles and responsibilities would be provided:

***Carrier Response*:** If an accident were to occur, the carrier operator and/or escort would notify local emergency response personnel as predetermined in both the DOE and Shipper of Record's Transportation Plans. They would also undertake first aid actions and initial incident scene control, provide assistance to first responders, and other emergency actions as defined in the carrier's emergency plan. The carrier operator would provide technical response assistance to State, Tribal, and local responders as required by event scene conditions.

***First Responders*:** Local emergency response personnel would respond to the incident scene when notified by a predetermined notification network initiated by the "Initial Responder." Their first action would be the evaluation of the accident scene (with assistance from the "Initial Responder") for the presence of radiological or other hazards. The response personnel would then act to reduce the hazard and control the event scene. The coordination of the accident scene would typically be under the Incident Command System utilized by local fire departments.

Responders would have information sources available to them in the form of the Department of Transportation *Emergency Response Guidebook* and by available emergency response information accompanying the shipping papers and normally available to responders in the vehicle, or accessible via the satellite communications and tracking system. Response to other hazards identified at the accident scene would be guided by information also contained in the Emergency Response Guidebook or other appropriate protective measures and response guidelines. In all cases, the Incident Commander would be a corridor jurisdiction authority. If States and local responders have additional procedures that provide more specific guidance, responders would follow those procedures.

Local organizations typically involved in first on-scene response include:

- Law enforcement,
- Emergency medical services,
- Cognizant transportation department,
- Hazardous materials team, and
- Fire services.

***Responder Support*:** Response organizations would arrive at the incident scene to support and assist the initial and first on-scene responders as requested by the "First Responders." The Incident Commander would coordinate the actions of these trained personnel coming from agencies within or outside the initial response jurisdiction. Requests for resource augmentation could be performed through local mutual aid associations as part of a response support network. Utilization of response resources would be based upon

communication and coordination with State, Tribal, and local agencies, and shipper/carrier response personnel. For incidents that exceed the capabilities of local, State, or Tribal government, Federal assistance could be requested from DOE. DOE could provide Radiological Assistance Program teams that include personnel and instruments for radiological monitoring, provide medical advice, and request assistance from other Federal agencies.

"Responder Support" assistance could involve traffic and/or access control, support of incident mitigation activities, appropriate notification (hospitals, mutual-aid emergency management system), technical assistance for assessment of health risks, and coordination of emergency health services or technical assistance for assessment of environmental risks and coordination of emergency planning for cleanup and recovery as defined in applicable emergency response plans. Organizations typically involved in responder support include:

- Fire services,
- Law enforcement,
- Cognizant transportation department,
- Hazardous materials team,
- Health protection oversight,
- Environmental oversight, and
- DOE Radiological Assistance Program team.

***Package Recovery Actions*:** If package recovery, repositioning, or placement on another vehicle were required, provisions for necessary service would be prearranged. The carrier would have primary responsibility for package and transporter recovery operations. These activities would not begin until the emergency phase of any accident was terminated, following a decision that no radiological or other hazard was present. Recovery planning is initiated prior to termination of an emergency. DOE would assist the carrier in recovery operations where appropriate. Specific procedures for shipping cask recovery would be included in the shipper of record's agent/carrier's transportation plan. Corridor jurisdictions could exercise highway vehicle inspection authority before permitting the recovery vehicle to continue to a management site. Organizations typically involved in recovery actions include:

- Shipper's representative
- Carrier representatives,
- Management sites representative,
- DOE contractor representatives, and
- State, Tribal, and local authorities.

Cleanup and Incident Scene Restoration: In over forty years of experience with spent nuclear fuel shipments, there has never been an incident in which a spent nuclear fuel transportation cask has released any of its contents, even as a result of an accident. In the unlikely event that there was a release of radioactive material, DOE would be ready to provide whatever assistance was needed to respond to the situation. On the other hand, cleanup of nonradioactive aspects of an accident and scene restoration are a

part of any accident response. The carrier would have primary responsibility for cleanup and site restoration following an emergency and would provide the necessary resources. Cleanup planning could be initiated prior to termination of an emergency. Standards for such actions would be established by regulation and by authorities in the affected jurisdiction(s). Organizations typically involved in cleanup/restoration oversight actions include:

- Carrier representatives,
- Carrier cleanup/restoration representatives,
- Management site representative,
- DOE contractor representatives,
- Federal environmental oversight, and
- State, Tribal, and local authorities.

Carriers are financially responsible for accident response. The carrier is responsible for maintaining $5 million of insurance to cover costs incurred from an accident. Cost incurred by local first responders (firefighters, police, etc.,) to an incident scene are part of the carrier's financial responsibility. Further, the Price-Anderson Amendment Act ensures coverage of cost incurred beyond the $5 million carrier limit for spent nuclear fuel accidents. The Price-Anderson Act was partly established to ensure that funds are available to compensate the public for personal injury and property damage caused by the release of radioactivity (NRC, 1988). Such coverage would only take effect if the Price-Anderson Act conditions are met such as the resultant damage from the release of radioactivity during the accident exceeded the liability protection of the carrier.

H.4 Security Considerations

Domestic transportation of the foreign research reactor spent nuclear fuel would be regulated by the U.S. Department of Transportation and NRC. The objectives of the security measures employed during foreign research reactor spent nuclear fuel shipments would be to minimize the possibilities for radiological sabotage of spent nuclear fuel shipments and facilitate the location and recovery of spent nuclear fuel shipments that could have come under the control of unauthorized persons. The elements of the security systems to be considered when developing a transportation plan and are briefly summarized below.

***Security Packaging*:** Type B packages are used to transport spent nuclear fuel and would have a seal, which is not readily breakable, and which, while intact, would be evidence that the package has not been opened by unauthorized persons.

***General Security Concept*:** The physical security system would include procedures for coping with circumstances that threaten deliberate damage to a spent nuclear fuel shipment and with other safeguards emergencies; instructions for each escort would be developed. These procedures would include detection of the abnormal presence of unauthorized persons, vehicles, or vessels in the vicinity of a spent nuclear fuel shipment or upon detection of a deliberately induced situation that would have the potential for damaging a spent nuclear fuel shipment; monitoring the progress of the spent nuclear fuel shipment to notify the appropriate agencies in the event a safeguards emergency should arise; and maintaining a written log by the escorts and communications center personnel for each spent nuclear fuel shipment. Arrangements would be made with local law enforcement agencies along the routes of road and rail shipments, and at U.S. ports where vessels carrying spent nuclear fuel shipments were docked, for their

response to a security event or a call for assistance. Advance NRC approval of the U.S. Department of Transportation or State-designated alternative routes used for road and rail shipments of spent nuclear fuel, and of any U.S. ports where vessels carrying spent nuclear fuel shipments were scheduled to stop would be required. Shipments would be planned so that scheduled intermediate stops would be avoided to the extent practicable; at least one escort would maintain visual surveillance of the shipment during periods when the shipment vehicle was stopped, or the shipment vessel was docked. Shipment escorts would make calls to the communications center at least every two hours to advise of the status of the shipment for road and rail shipments, and for sea shipments while shipment vessels were docked at U.S. ports. These escorts (other than members of local law enforcement agencies, or ship's officers serving as unarmed escorts) would have successfully completed required training. In addition to NRC licensee requirements, DOE and the licensee could develop extra-regulatory guidelines, as necessary. These guidelines would be established in the overall transportation plan.

***Shipment by Sea*:** The management site would provide a representative at the shipment point of origin to observe the preparation and loading of the material. NRC could also send a representative to the point-of-origin to inspect transport packages and conduct radiological surveys prior to departure.

Advance NRC approval of the routes used for road and rail shipments of spent nuclear fuel, and of any U.S. ports where vessels carrying spent nuclear fuel shipments were scheduled to stop is required. The local law enforcement agencies, at U.S. ports where the ship was docked, would be contacted and arrangements would be made for their response in the event of an emergency situation concerning the spent nuclear fuel.

A shipment vessel, while docked at a U.S. port within a heavily populated area, would be protected by two armed escorts stationed on board the shipment vessel, or stationed on the dock at a location that would permit observation of the shipment vessel; or a member of a local law enforcement agency equipped with normal local law enforcement agency radio communications, who would be stationed on board the shipment vessel, or on the dock at a location that would permit observation of the shipment vessel.

A shipment vessel, while within U.S. territorial waters or while docked at a U.S. port not within a heavily populated area, would be accompanied by an escort (e.g., an officer of the shipment vessel's crew), who would assure that the shipment were unloaded only as authorized by the Shipper of Record.

The escorts would have the capability of communicating with the communications center and local law enforcement agencies through the use of a radiotelephone, or other NRC-approved equivalent means of two-way voice communications.

If a military port were used, applicable requirements would be established in advance among the U.S. Department of Defense, Shipper of Record, DOE, and NRC to provide the appropriate level of security, which would meet or exceed the security provided at commercial ports by the provisions discussed above.

***Shipments by Highway*:** For any shipment of spent nuclear fuel, the Shipper of Record would provide physical protection in compliance with a plan established under requirements prescribed by the NRC or equivalent requirements approved by the Association Administrator for Hazardous Materials Safety. An NRC regulated shipment requires advance approval of the Department of Transportation or State-designated agencies for alternative routes that would be used for highway shipments of spent nuclear fuel. Arrangements would be made with local law enforcement agencies along the transportation route for response to security events or calls for assistance while the shipment was within their jurisdiction. Highway shipments would be accompanied by escorts. Escorts are required by regulations to ride in the shipment vehicle (truck cab) or in separate vehicles maintaining visual sight at all times. Additionally,

escorts would have the capability to communicate with their communications center, each other, and the local law enforcement agency. The local law enforcement agency would be able to direct a prompt response to counter a threat to the spent nuclear fuel. The transport vehicle would be equipped with a feature approved by the NRC, which would allow the driver or on-board escort to immobilize the cab or cargo-carrying portion of the vehicle should an attempt be made to seize control of the vehicle. Both the driver and the on-board escort would have appropriate training.

***Shipment by Rail*:** The NRC requires advance approval of the routes used for rail shipments of spent nuclear fuel. A rail shipment car within a heavily populated area would be accompanied by two armed escorts (who are often members of the local law enforcement agency). At least one escort would be stationed at a location on the train that would permit observation of the shipment car while in motion.

A rail shipment car outside heavily populated areas would be accompanied by at least one escort stationed at a location on the train that would permit observation of the shipment car while in motion. Escorts would have the capability to communicate with the communications center and the cognizant local law enforcement agency.

***Reporting of Safeguards Events*:** The Shipper of Record would notify the NRC Operations Center within one hour of discovery of credible threat (e.g., theft or significant physical damage), entry of unauthorized person, or failure/degradation in a safeguard system. Other notification time criteria would be put in place for less significant safeguards events.

H.5 Communication Considerations

In the case of extended or high visibility shipping campaigns, such as in the implementation of Management Alternative 1 or Management Alternative 3, a written public and media communications plan would be included in the overall Transportation Plan. The communication plan would generally be completed by public affairs personnel at the management sites (in cooperation with State, Tribal, and local authorities). The purpose of this plan is to ensure the exchange of accurate and timely information among the foreign research reactor operators (and their agents, if applicable), the States and Tribes, other Federal agencies, the public, and the media.

H.5.1 Public and Media Communication

The public and media communication plan for a foreign research reactor spent nuclear fuel shipment would establish public information points of contact for each agency and jurisdiction through which a shipment would pass, coordinate public education and information activities in those jurisdictions, promote communication among shipment participants by keeping State, Tribal, and local points of contact informed, and detail the procedures and sources of information in the event of an accident or other incident. To accomplish these goals, the communication plan could include the following:

***Roles and Responsibilities*:** The roles and responsibilities of the management site, the States and Tribes, other Federal agencies, the Shipper of Record or the shipper's agent, the land carrier, the port authorities, and regional State association (if any) would be detailed.

***Congressional and State Notification*:** DOE Headquarters Congressional and Intergovernmental Liaison Office would make appropriate Congressional notifications and respond to requests from Congressional staff and elected officials for additional information. Prior to the transport of spent nuclear fuel within or through a State, the Governor of that State would be notified in writing.

***Stakeholder, State and Tribal Notification*:** The management site, States, and Tribes generally provide for appropriate stakeholder and special interest group notification, as needed, and would respond to requests from stakeholders for additional information consistent with NRC regulations concerning safeguards.

***Media Interactions*:** Media interactions would be coordinated by the management site and would take place prior to the shipment, during the shipment, at the port of entry, after arrival at the management site, and at the completion of the shipment. These activities could range from the press briefings prior to the first shipment to a complete listing of State, Tribal, and local media print and broadcast media, to broad distribution of press releases, radio and TV news spots with accompanying factsheets, videos, and other information materials.

Public Interactions: The management site would interact with the public by answering questions, conducting briefings and meetings, and disseminating material. The States and Tribes often choose to distribute additional information packages through their emergency response organizations, civic and service organizations, local government agencies, or other special interest groups.

***Emergency Procedures*:** In the unlikely event of an accident or an incident, the management site, the Shipper of Record or its agent, and the State, Tribal, and local authorities have a coordinated approach to dealing with information about the emergency, providing timely information to the press and to the pre-established State and local contacts.

References

DOE (U.S. Department of Energy), 1995, *Program Manager's Guide to Transportation Planning*, Office of Transportation, Emergency Management, and Analytical Services of the Office of Environmental Management, Washington, DC, June.

NRC (U.S. Nuclear Regulatory Commission), 1988, *The Price-Anderson System*, NUREG BR-0079, Office of Nuclear Reactor Regulation, Washington, DC.