NUREG – 1935
SAND2008P - XXXX

State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

Main Report *(DRAFT)*

Manuscript Completed: XXXX
Date Published: XXXX

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

E/21

Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

ABSTRACT

The evaluation of accident phenomena and the offsite consequences of severe reactor accidents has been the subject of considerable research by the U.S. Nuclear Regulatory Commission (NRC) over the last several decades. As a consequence of this research focus, analyses of severe accidents at nuclear power reactors are more detailed, integrated, and realistic than at any time in the past. A desire to leverage this capability to address excessively conservative aspects of previous reactor accident analysis efforts was a major motivating factor in the genesis of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project. By applying modern analysis tools and techniques, the SOARCA project seeks to provide a body of knowledge that will support an informed public understanding of the likely outcomes of severe nuclear reactor accidents. The primary objective of the SOARCA project is to provide a best-estimate evaluation of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet. To accomplish this objective, the SOARCA project utilized integrated modeling of accident progression and offsite consequences using both state-of-the-art computational analysis tools as well as best modeling practices drawn from the collective wisdom of the severe accident analysis community.

This Page Intentionally Left Blank

ACKNOWLEDGEMENTS

The contributions of the following individuals in preparing this document are gratefully acknowledged.

Jon Ake	U.S. Nuclear Regulatory Commission
Jonathan Barr	U.S. Nuclear Regulatory Commission
Nathan E. Bixler	Sandia National Laboratories
Jeffrey D. Brewer	Sandia National Laboratories
Terry Brock	U.S. Nuclear Regulatory Commission
Shawn P. Burns	Sandia National Laboratories
Scott Elkins	U.S. Nuclear Regulatory Commission
Randall O. Gauntt	Sandia National Laboratories
Ata Istar	U.S. Nuclear Regulatory Commission
Joseph A. Jones	Sandia National Laboratories
Mark T. Leonard	dycoda, LLC
Jocelyn Mitchell	U.S. Nuclear Regulatory Commission
Andrew J. Nosek	U.S. Nuclear Regulatory Commission
Mark Orr	U.S. Nuclear Regulatory Commission
Robert Prato	U.S. Nuclear Regulatory Commission
Jason Schaperow	U.S. Nuclear Regulatory Commission
F. Joseph Schelling	Sandia National Laboratories
Abdul Sheikh	U.S. Nuclear Regulatory Commission
Richard Sherry	U.S. Nuclear Regulatory Commission
Martin Stutzke	U.S. Nuclear Regulatory Commission
Randolph Sullivan	U.S. Nuclear Regulatory Commission
Charles G. Tinkler	U.S. Nuclear Regulatory Commission
Kenneth C. Wagner	Sandia National Laboratories

This Page Intentionally Left Blank

TABLE OF CONTENTS

FIGURES

TABLES

ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feedwater
ATWS	Anticipated Transient Without Scram
BWR	Boiling-Water Reactor
CDF	Core Damage Frequency
CFR	*Code of Federal Regulations*
CST	Condensate Storage Tank
DC	Direct Current
DHS	Department of Homeland Security
ECCS	Emergency Core Cooling System
ECST	Emergency Condensate Storage Tank
EPZ	Emergency Planning Zone
ETE	Evacuation Time Estimate
FR	*Federal Register*
HPCI	High Pressure Coolant Injection
HPI	High Pressure Injection
ICRP	International Commission on Radiological Protection
IPEEE	Individual Plant Examination – External Events
ISLOCA	Interfacing Systems Loss-of-Coolant Accident
LCF	Latent Cancer Fatality
LOCA	Loss Of Cooling Accident
LOOP	Loss Of Offsite Power
LPI	Low Pressure Injection
LTSBO	Long-Term Station Blackout
LWR	Light-Water Reactor
MSIV	Main Steam Isolation Valve
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
NRF	National Response Framework
OREMS	Oak Ridge Evacuation Modeling System
ORO	Offsite Response Organizations
PGA	Peak Ground Acceleration
PRA	Probabilistic Risk Assessment
PWR	Pressurized-Water Reactor
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RWST	Refueling Water Storage Tank
SAE	Site Area Emergency
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SG	Steam Generator

SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratories
SOARCA	State-of-the-Art Reactor Consequence Analysis Project
SPAR	Simplified Plant Analysis Risk
SRV	Safety Relief Valve
STCP	Source Term Code Package
STSBO	Short-Term Station Blackout
TD-AFW	Turbine Driven Auxiliary Feedwater
TISGTR	Thermally Induced Steam Generator Tube Rupture
TSC	Technical Support Center
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) and its contractor Sandia National Laboratories (SNL) performed the State-of-the-Art Reactor Consequence Analysis (SOARCA) project to develop the best estimates of the offsite radiological health consequences for potential severe reactor accidents at the U.S. operating nuclear power plants.

Background

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by NRC, the nuclear power industry, and the national and international nuclear energy research communities. As part of an NRC effort to assess plant response to security-related events, updated analyses of severe accident progression and offsite consequences were completed in the mid 2000s utilizing the wealth of accumulated research and incorporating detailed, integrated, and more realistic modeling than past analyses. The results of the security-related studies confirmed what has been suspected—that some past studies of plant response and offsite consequences for nonsecurity events were conservative and outdated. It was evident that updated and more realistic estimates were needed.

Objective

NRC initiated the SOARCA project to develop a body of knowledge regarding the realistic outcomes of potential severe reactor accidents to enable NRC to better communicate severe-accident-related aspects of nuclear safety to stakeholders including Federal, State, and local authorities; the nuclear power industry; and the general public. The corresponding and supporting objectives for the SOARCA project include:

- Incorporation of state-of-the-art integrated modeling of severe accident behavior and offsite consequence modeling that includes the insights of some 25 years of research into severe accident phenomenology and health effects of radiation.

- Incorporation of plant changes, such as power uprates and higher core burnup, that are not reflected in earlier assessments; training and emergency procedures enhancements; security-related enhancements described in Title 10, Section 50.54(hh) of the *Code of Federal Regulations*; and offsite emergency response improvements.

- Evaluation of the potential benefits of the security-related mitigation enhancements intended to prevent core damage or to reduce offsite release should one occur.

- Update of the quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," and NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants."

Approach

NRC used current probabilistic risk assessment (PRA) information to select accident scenarios for the project and employed state-of-the-art computer modeling tools to model the accident progression and, for scenarios in which accidents proceed to core damage and containment failure, the radioactive material that could potentially be released into the environment. The staff then assessed those releases to develop the best estimates of the potential health consequences to the public. The SOARCA project incorporated the following:

- Plant-specific severe reactor accident progression taking into account the plant's current design configuration.
- Plant-specific containment failure including the timing, location, and size of the failure.
- Operator actions based on emergency operating procedures, severe accident management guidelines, and any available additional mitigation measures.
- Site-specific meteorological conditions and updated population data.
- Site-specific emergency planning assumptions including evacuation and sheltering.

The central focus of the project was to introduce the use of a detailed, best-estimate, quantification of offsite consequences based on current scientific knowledge and plant capabilities. The essence of the analysis methodology was the application of the integrated severe accident progression modeling tool, the Method for Estimation of Leakage and Consequence of Release (MELCOR) code, and the offsite consequence modeling tool, the MELCOR Accident Consequence Code System, version 2 (MACCS2) code.

Scenario Selection

NRC determined that a rigorous and realistic evaluation of the more important severe accident scenarios would provide better and more detailed accident consequence information than a less intense assessment of a larger number of scenarios. With this in mind, the project set technical criteria to determine which scenarios were important and focused its resources accordingly. Thus, for SOARCA, we elected to analyze the accident scenarios with a core damage frequency (CDF) equal to or greater than ($\geq$) 10^{-6} per reactor-year (1 in a million). In addition, we included scenarios associated with events involving containment bypass or leading to an early failure of the containment with CDF $\geq 10^{-7}$ per reactor-year (1 in 10 million) because those scenarios have an inherent potential for higher consequences (and risk). The scenarios identified in SOARCA included both internally and externally initiated events. The externally initiated events included events for which seismic, fire, and flooding initiators were grouped together. The timeline of operator actions, in response to the externally initiated events, was developed assuming the initiator was a seismic event because the seismic initiator was judged to be the most severe in terms of timing and with respect to how much equipment would be available to mitigate the accident.

Accident Progression

SOARCA includes detailed modeling of behavior associated with long-term containment pressurization, Mark I liner failure, induced steam generator tube rupture (SGTR), hydrogen combustion, and core concrete interactions. Other than the magnitude of the radiological release, a major impact on offsite consequences is derived from the timing of the offsite release. In this respect, we examined SOARCA scenarios with both the timing of core damage and the timing of containment failure in mind. In preparation for the modeling of accident progression and offsite consequences, the staff had extensive cooperation from the licensees of the pilot plants to develop high-fidelity plant system and containment models, define operator actions, and develop models for simulation of scenario-specific and site-specific emergency planning.

Mitigation

The modeling included the mitigation strategies treated in current PRA models, Emergency Operating Procedures (EOPs), and Severe Accident Management Guidelines (SAMGs) as well as the additional equipment and mitigation strategies required by NRC following the terrorist attacks of September 11, 2001, to further improve severe accident mitigation capability (10 CFR 50.54 [hh]). To assess whether it was reasonable to believe that the additional mitigation measures can be taken and to give credit for their use, NRC conducted tabletop exercises and plant walkdowns of the scenarios with the licensees' plant operators, PRA analysts, and other staffs. The set of analyses that credited the use of the mitigation measures was considered the "Mitigated" analyses and the SOARCA baseline analyses. The SOARCA project also analyzed the same groups of scenarios assuming the event proceeded without the use of the additional mitigation measures to assess the benefits of the various mitigation measures and to provide some basis for comparison to past analyses of severe accidents. This set of analyses was considered the "Unmitigated" analyses.

Offsite Consequences

The SOARCA project developed latent cancer fatality estimates representing the range of health effects corresponding to the models proposed by national and international radiation and health effects organizations. The modeling of latent cancer fatality has been an issue of considerable controversy because evidence regarding risk in the low-dose region is inconclusive. Thus, in SOARCA, we assumed the linear no-threshold (LNT) model (a basic assumption in many regulatory applications) and a range of truncation doses in estimating the cancer fatalities. Dose truncation values used for SOARCA included 10 mrem/year representing a small dose and the trivial dose below which the International Commission on Radiological Protection suggests avoiding summing doses; 620 mrem/year, representing background radiation levels in the environment; and 5 rem/year with a 10 rem lifetime cap, representing the Health Physics Society Position Statement in "Radiation Risk in Perspective," August 2004. NRC determined that to facilitate communication and understanding, an appropriate measure of offsite consequences was one in which consequences are framed within a risk context (individual early fatality or latent cancer fatality risk) to relate to the NRC's Safety Goal quantitative health objectives. In addition, because the accident scenarios were of very low frequency, it was essential that the consequence estimates be coupled with frequency to provide a proper perspective. Thus, the

SOARCA offsite radiological consequence estimates for each scenario are expressed as the average individual likelihood of an early fatality or latent cancer fatality conditional to the occurrence of the severe reactor accident (i.e., expressed as a risk metric factoring in the frequency of the scenario). The results showed that individual latent cancer fatality conditional risk is very low and continues to further decrease beyond 50 miles. For this reason, the SOARCA analysis included predictions of individual latent cancer fatality risk for two distance intervals from the site: 0 to 10 miles and 0 to 50 miles. The distance intervals of 10 and 50 miles are historically used for cost-benefit analysis, severe accident mitigation alternatives evaluations, and the emergency planning zones (EPZs).

Pilot Plants

The SOARCA project completed the analysis for two plants that are typical examples of the two types of commercial nuclear power plants used in the United States: the Peach Bottom Atomic Power Station (Peach Bottom), a boiling-water reactor (BWR) near Lancaster, Pennsylvania, and the Surry Power Station (Surry), a pressurized-water reactor (PWR) near Newport News, Virginia. The application of the SOARCA screening criteria to the available level 1 PRA information for the two plants identified two basic types of scenarios: station blackouts (SBO) and containment bypass scenarios. This result supported the inherent adequacy of the selection criteria and the adequacy of the scope of scenarios analyzed. SBO scenarios are representative of a broad class of events in PRA—loss of heat removal events. Selection of SBO events in SOARCA meant that we have covered that broader class of transients involving a loss of heat removal. In considering reactor coolant pump seal leakage, the SBO for the PWR also includes, in part, the effect of a small loss-of-coolant accident (LOCA). In addition, by the selection of SBO scenarios, we included the effects of loss-of-containment heat removal (fan coolers) and containment spray systems, which are all electrically powered to remove airborne radionuclides. For Surry, we included an interfacing systems LOCA (ISLOCA) scenario to reasonably bound events involving a LOCA inside containment for that plant. Medium and large loss-of-inventory accidents were well below the selection criteria of CDF $\geq 10^{-6}$ per reactor-year and thus not included. For Peach Bottom, the medium and large LOCAs had frequencies of 1×10^{-9} and 2×10^{-9} per reactor-year, respectively. For Surry, they had frequencies of 7×10^{-10} and 6×10^{-8} per reactor-year, respectively.

Results

The results of the SOARCA project showed that all the scenarios analyzed could reasonably be mitigated. The mitigation measures, once taken, were adequate to prevent core damage or reduce radiological releases. The security-related measures to provide alternative ac power and portable diesel-driven pumps were especially helpful in counteracting SBO scenarios. For the ISLOCA scenario, installed equipment was adequate to prevent core damage owing to the time available for operators to take actions. Mitigation measures resulted in no core damage for all scenarios analyzed except for the Surry short-term SBO (STSBO) as well as for the case where the STSBO resulted in a thermally induced steam generator tube rupture (TI-SGTR). In the case of the STSBO, the mitigation measure was sufficient to enable flooding of the containment through the containment spray system to cover core debris and, thus, no containment failure occurred. For the TI-SGTR, the predicted individual latent cancer conditional risk was small,

1×10^{-10} per reactor-yr, assuming LNT. The "unmitigated" analyses showed that core damage was delayed for several hours, and several more hours of delay—about 20 hours for the BWR and 45 hours for the PWR—occurred before the onset of offsite radiological release due to containment failure. The predicted individual latent cancer conditional risks for an individual located within 10 miles of the site were in the range of 7×10^{-11} to 6×10^{-10} per reactor-year for the Peach Bottom scenarios and in the range of 2×10^{-11} to 7×10^{-10} per reactor-year for the Surry scenarios.

Comparison to Previous Studies

We compared the SOARCA results to those of previous studies. In the most widely referenced NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," 1982 [1] scenario, it was assumed that a major release (identified as the SST1 release) occurs in 1½ hours. The SOARCA results showed that ample time is available for operators to take mitigation actions if initial efforts are assumed unsuccessful. Even in the case of the most rapid events (i.e., the unmitigated STSBO where core damage began in 1 to 3 hours), reactor vessel failure was delayed for roughly 8 hours allowing time for restoration of cooling and prevention of vessel failure. In SOARCA, containment failure and radiological release are delayed for 8 hours (BWR) or 24 hours (PWR). For the bypass events, substantial delays occurred or, in the case of the thermally induced steam generator tube rupture, the radiological release was substantially reduced. The SOARCA results predicted no early fatalities except in only one case when the scenario was assumed to be unmitigated, and even then the early fatality was essentially zero. In contrast, NUREG/CR-2239 predicted 92 early fatalities for Peach Bottom and 45 early fatalities for Surry for the SST1 source term. For latent cancer fatality results, the exact basis for the NUREG/CR-2239 estimates could not be recovered, but literature searches and sensitivity analyses with MACCS2 suggested that these estimates are for the population within 500 miles of the site. Given this uncertainty, SOARCA does not make a direct comparison to the NUREG/CR-2239 latent cancer fatality estimate.

We also compared the SOARCA sequences to those identified as important to risk in NUREG-1150, V., "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," 1990 [2] for the Surry and Peach Bottom plants. We found that, with one exception, in SOARCA we addressed the more likely and important sequences identified in NUREG-1150. The one exception, a sequence identified in NUREG-1150 that was not included in the SOARCA project, involved an extreme earthquake that directly results in a large breach of the reactor coolant system (large LOCA), a large breach of the containment, and an immediate loss of safety systems. We concluded that this sequence was not appropriate for consideration as part of SOARCA for a number of reasons. Foremost, the state of quantification of such extreme and low-frequency seismic events is poor, considerable uncertainty exists in the quantification of the seismic loading condition itself, and a detailed soil-structure interactions analysis was not performed for the plant (and its equipment) response to the seismic loads. The analysis of the plant's components to the seismic acceleration—commonly referred to as fragility analysis—is a key component, and the lack of detailed analysis in this area made current consideration of this event incompatible with the thrust of SOARCA, which is the performance of detailed, realistic analyses. Moreover, recent experience at nuclear plants in Japan strongly suggests that nuclear

↳ (Kashiwazaki-Kariwa, 2007)

plant designs possess inherently greater capability to withstand the effects of extremely large earthquakes.

Conclusion

The results of the SOARCA project provide a more realistic picture and a better understanding of potential offsite health consequences in the unlikely event of a severe reactor accident. The results represent a major change from the way people perceive severe reactor accidents and their likelihood and consequences. The specific conclusions derived from the SOARCA pilot project are as follows:

- The use of mitigation measures that are not credited in the current PRA was sufficient to delay or prevent core damage and, when core damage occurred, to delay, reduce, or prevent offsite radiological release. This was primarily because of the time available for operator actions and the redundancy and diversity of equipment.

- When the scenarios were analyzed without credit for the use of the mitigation measures that are not credited in current PRA, the accidents progressed more slowly and resulted in smaller releases than past treatments generally indicated. The individual early fatality risk was essentially zero, and the average individual latent cancer risks were very low.

- Latent cancer fatality predictions, using the LNT assumption, were generally dominated by long-term exposure to small annual doses (~500 mrem) in conjunction with the emergency planning return criteria.

- Bypass events did not pose higher cancer fatality risk; higher conditional risk was offset by the lower scenario frequency.

- Explicit consideration of seismic impacts on emergency response (e.g., loss of bridges, traffic signals, and delayed notification) did not significantly impact risk predictions.

Insights

An important insight derived from the SOARCA analysis is that severe accidents progress much more slowly than earlier treatments indicated. The reasons for this are principally twofold: (1) research and development of better phenomenological modeling has produced a much more protracted and delayed core degradation transient with substantial delays of reactor vessel failure and (2) all aspects of accident scenarios receive more realistic treatment, which includes more complete modeling of plant systems and often yields delays in core damage and radiological release. In general, bounding approaches have been used in past simplified treatments using qualitative logical models. In SOARCA, where specific self-consistent scenarios are analyzed in an integral fashion using MELCOR, the result is that accident conditions or attributes that contribute to a more severe response in one area may produce an ameliorating effect in another area. The analysis also indicated that individual latent cancer risk estimates generally decrease with increasing distance in large part due to plume dispersion and fission product deposition closer to the site.

1.0 INTRODUCTION

This document describes the U.S. Nuclear Regulatory Commission's (NRC's) state-of-the-art, realistic assessment of the accident progression, radiological releases, and offsite consequences for important severe accident sequences.

1.1 Background

The evaluation of accident phenomena and offsite consequences of severe reactor accidents has been the subject of considerable research by NRC, the nuclear power industry, and the international nuclear energy research community. Most recently, with Commission guidance and as part of plant security assessments, updated analyses of severe accident progression and offsite consequences were completed using the wealth of accumulated research. These analyses are more detailed (in terms of the fidelity of the representation and resolution of facilities and emergency response), realistic (in terms of the use of currently accepted phenomenological models and procedures), and integrated (in terms of the intimate coupling between accident progression and offsite consequence models).

The results of these recent studies have confirmed what was suspected but not well quantified—namely, that some past studies of plant response and offsite consequences were conservative to the point that predictions were not useful for characterizing results, communicating to the public, or guiding public policy. The subsequent misuse and misinterpretation of these estimates further suggests that communication of risk attributable to severe reactor accidents should be based on realistic estimates of the more likely outcomes. Moreover, as a result of past risk assessments and in response to the terrorist attacks of September 11, 2001, nuclear plants have made additional safety enhancements that reduce the risk of severe accidents as portrayed in earlier NRC assessments.

In addition to the improvements in understanding and calculational capabilities that have resulted from these studies, numerous influential changes have occurred in the training of operating personnel and the increased use of plant-specific capabilities. These changes include:

- The transition from event-based to symptom-based Emergency Operating Procedures for the boiling-water and pressurized-water reactor designs.
- The performance and maintenance of plant-specific probabilistic risk assessments (PRAs) that cover the spectrum of accident scenarios.
- The implementation of plant-specific, full-scope control room simulators to train operators.
- An industrywide technical basis, owners-group-specific guidance and plant-specific implementation of the Severe Accident Management Guidelines.
- Proceduralized use of plant-specific systems required under Title 10, Section 50.54(hh) of the *Code of Federal Regulations*.

- Improved phenomenological understanding of influential processes such as
  - in-vessel steam explosions
  - Mark I containment drywell shell attack
  - dominant chemical forms for fission products
  - direct containment heating
  - hot leg creep rupture
  - reactor pressure vessel failure and molten core concrete interactions.

1.2 Objective

The overall objective of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project is to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. Corresponding and supporting objectives are as follows:

- Incorporate the significant plant improvements and updates not reflected in earlier assessments including system improvements, training and emergency procedures, offsite emergency response, and recent security-related enhancements described in Title 10, Section 50.54(hh) of the *Code of Federal Regulations* (10 CFR 50.54(hh)) as well as plant updates in the form of power uprates and higher core burnup.

- Incorporate state-of-the-art integrated modeling of severe accident behavior that includes the insights of some 25 years of research into severe accident phenomenology and radiation health effects.

- Evaluate the potential benefits of recent security-related mitigation improvements in preventing core damage and reducing an offsite release should one occur.

- Enable NRC to communicate severe-accident-related aspects of nuclear safety to stakeholders including Federal, State, and local authorities; licensees; and the general public.

- Update quantification of offsite consequences found in earlier NRC publications such as NUREG/CR-2239, "Technical Guidance for Siting Criteria Development [1].

1.3 Approach

The basic approach for the SOARCA project was to utilize the self consistent, integrated modeling of accident progression and offsite consequences drawn from current best practices modeling to estimate offsite consequences for important classes of events. This was accomplished by modeling accident progression (reactor and containment thermal-hydraulic and fission product response), which is embodied in the MELCOR code, coupled with modeling offsite consequences (in the MACCS2 code) in a consistent manner (e.g., evacuation timing) and with improved input in important areas. Selection of the events for analysis was based on a consideration of insights from past and current PRA and from research on accident behavior and failure modes important to latent cancer and early fatality risk. Selection of events for quantification also properly included probability to focus on more likely and important contributors. Figure 1 illustrates the four main elements of SOARCA (i.e., scenario selection, mitigative measures analysis, accident progression and source term, and offsite radiological consequences).

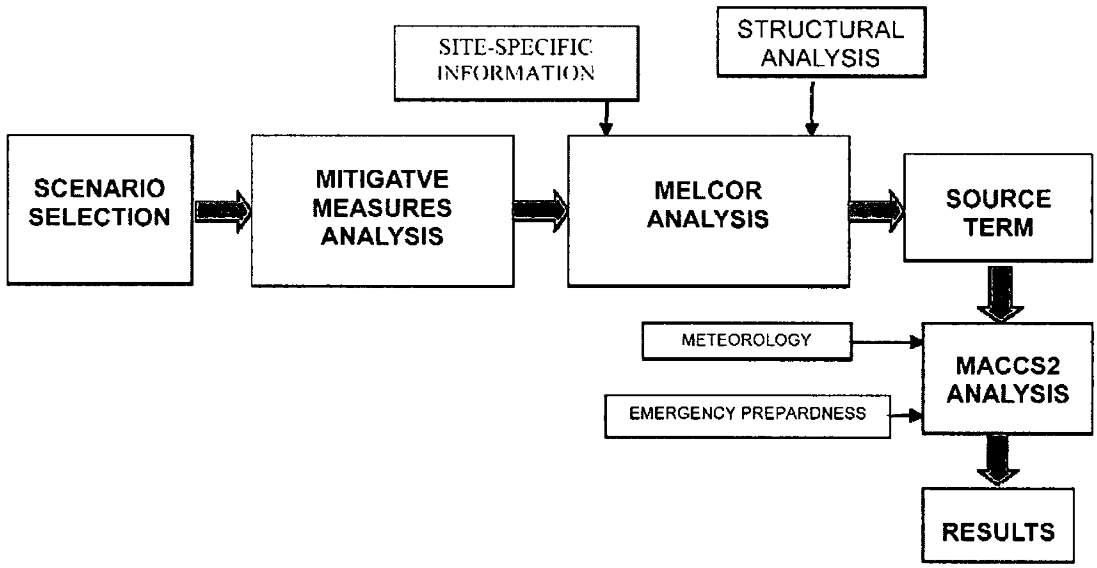

Figure 1. The State-of-the-Art Reactor Consequence Analyses Process.

It is believed that more can be learned at this juncture by focusing on a relatively few important events and quantifying the plant and offsite response rigorously and realistically than by approximate modeling of many events, including extremely rare events. This approach of focusing on relatively few but important events also allows us to efficiently and explicitly address the benefits of additional mitigation in further reducing the likelihood of core damage and offsite consequences. The offsite consequence analyses were performed on a site-specific basis (reflecting site-specific population distributions, weather, and emergency preparedness) and also included improved understanding of non-site-specific input.

Selection of events considered individual plant examinations (IPEs)[1], individual plant examinations of external events (IPEEEs), standardized plant analysis risk (SPAR) models, and NUREG-1150 risk studies. Information related to system and procedural plant improvements that have been incorporated as part of the industry's response to the NRC's security initiatives (e.g., the purchase and development of procedures for diesel-driven pumps in response to 10 CFR 50.54(hh) requirements) as well as necessary plant information was included in the scenario selection evaluation and incorporated in plant modeling.

[1] As requested by the NRC in Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities" (November 23, 1988), the utilities conducted risk analyses that considered the unique aspects of a particular nuclear power plant, identifying the specific vulnerabilities to the plant to severe accidents.

The SOARCA approach was subjected to extensive peer review by committees of experts from the severe accident progression and consequence modeling and phenomenology communities. Peer reviews were conducted prior to the execution of analyses to examine the overall modeling approach as well as review of results near completion to identify important phenomenological issues.

1.4 Historical Perspectives

The following sections describe some of the important historical studies that have preceded the SOARCA project.

1.4.1 NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," 1982

NRC contracted with Sandia National Laboratories to develop a technical guidance report for siting future reactors [1]. Guidance was requested regarding (1) criteria for population density and distribution surrounding future sites, and (2) standoff distances of plants from offsite hazards.

Because the purpose was to develop criteria for siting by evaluating aspects of population and meteorology separately, 5 types of accidents would be imposed on each plant in the 91-site study. The accidents or "siting source term events" (SST event) would be derived from the previous Reactor Safety Study (WASH-1400) [3], and each SST event would be assumed identical regardless of the reactor size or plant design.

- SST1 – Severe core damage. Loss of all safety systems and loss of containment after 1.5 hours.

- SST2 – Severe core damage. Containment systems (e.g., sprays, suppression pools) function to reduce radioactive release, but containment leakage is large after 3 hours.

- SST3 – Severe core damage. Containment systems function, but small containment leakage (1 % per day) after 1 hour.

The early fatality results for most of the 91 sites were similar due to a low population density close to the sites. Using the SST1 model with a population density of 50 persons per square mile resulted in 47 to 140 early fatalities and 730 to 860 latent cancer fatalities within 500 miles of the reactor. For the more realistic release represented by SST2 events, the mean values from typical plants were zero early fatalities and 95-140 latent cancer fatalities.

For high-population density sites, the consequences were higher although not proportionally higher, and this is a result of two factors. First, as the distance from the accident site increases, the area increases with the square of the distance and in turn may include higher population density. Secondly, it was assumed in CRAC2 (and in the present study) that long-term, offsite response actions would be taken to reduce the doses (i.e., a trade-off exists between dose and cost). For instance, the amount of land removed from public use (interdicted or condemned) was found to be sensitive to the release fraction of cesium, while the total population dose was less affected. The factor that affected the split between the interdicted land and the population dose was the criterion that was used to define interdiction (the habitability criterion). The highest consequence site using the SST1 model with a NYC population density resulted in a latent cancer increase of 0.06 percent over normal incidence. For the more realistic release represented by SST2 events, the same location resulted in a latent cancer increase of 0.004 percent increase over normal incidence.

1.4.2 NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," 1990

NUREG-1150 [2] documents the results of an extensive NRC-sponsored PRA. The study examined five plants, representative of classes of reactor and containment designs to give an understanding of risks for these particular plants. Selected insights regarding the classes of plants were also obtained in the study. The improved PRA methodology used in the NUREG-1150 study significantly enhanced the understanding of risk at nuclear power plants and is considered a significantly updated and improved revision to the Reactor Safety Study. A major improvement was the specific inclusion of an uncertainty estimate for the core damage frequency and source term portions of the study but only in a limited way for the offsite consequence portion (uncertain weather data was addressed but not other uncertain input parameters). The uncertainty estimate was based on extensive use of expert elicitation.

The five nuclear power plants analyzed in NUREG-1150 are:

- Unit 1 of the Surry Power Station, a Westinghouse-designed three-loop reactor in a subatmospheric containment building located near Williamsburg, Virginia.
- Unit 1 of the Zion Nuclear Power Plant, a Westinghouse-designed four-loop reactor in a large, dry containment building located near Chicago, Illinois.
- Unit 1 of the Sequoyah Nuclear Power Plant, a Westinghouse-designed four-loop reactor in an ice condenser containment building located near Chattanooga, Tennessee.
- Unit 2 of the Peach Bottom Atomic Power Station, a General Electric-designed BWR-4 reactor in a Mark I containment building located near Lancaster, Pennsylvania.
- Unit 1 of the Grand Gulf Nuclear Station, a General Electric-designed BWR-6 reactor in a Mark III containment building located near Vicksburg, Mississippi.

The various accident sequences that contribute to the core damage frequency from internal initiators can be grouped by common factors into categories. NUREG-1150 uses the accident categories depicted in Table 1 below: station blackout (SBO), anticipated transients without scram (ATWS), other transients (TRANS), interfacing system LOCAs (SG IF Sys), and other LOCAs. The selection of such categories is not unique but merely a convenient way to group the results.

Table 1. Summary of Core Damage Frequency from NUREG-1150

Plant Name	Internal Initiators						External Initiators
	SBO	ATWS	TRANS	SG/IF Sys	LOCA†	Core Damage Total/yr	Fire & Seismic
Surry	2.7E-5	1.6E-6	2.0E-6	3.4E-6	6.0E-6	4.0E-5	2.6E-5
Peach Bottom	2.2E-6	1.9E-6	1.4E-7	-	2.6E-7	4.5E-6	2.3E-5

† The LOCA category shown here includes LOCAs that are initiated by pipe break events. Transient-induced LOCAs are captured under the other categories shown in the table.

1.5 Scope

The central focus of the SOARCA project was to introduce the use of a detailed, best-estimate, self-consistent quantification of sequences based on current scientific knowledge and plant capabilities. The essence of the analysis methodology is the application of the integrated severe accident progression modeling tool, the MELCOR code. The analysis used an improved offsite consequence (MACCS2) code, including both improved code input and updated sequence-specific emergency response. Because the priority of this work was to bring more detailed, best-estimate, and consistent analytical modeling to bear in determining realistic outcomes of severe accident scenarios, the benefits of this state-of-the-art modeling could most efficiently be demonstrated by applying these methods to a set of the more important severe accident sequences. Thus, the project elected to limit its analysis to a set of important accident sequences considering both likelihood and potential consequences. The sequences that eventually were selected (e.g., station blackout, ISLOCA, thermally induced steam generator tube rupture) are in fact sequences that also have been considered to be important in recent and past probabilistic assessments.

The following several classes of accident events were not considered as part of the SOARCA project:

- Multi-unit accidents.
- Low power and shutdown accidents.
- Extreme seismic events that lead directly to gross containment failure simultaneous with reactor core damage.
- Spent fuel pool accidents.
- Security events.

Multi-unit accidents (events leading to reactor core damage at multiple units on the same site) could be caused by certain initiators such as an earthquake. Most PRAs developed to date do not explicitly consider multi-unit accidents because NRC policy is to apply the Commission's Safety Goals (51 FR 28044) [4] and subsidiary risk acceptance guidelines (see Regulatory Guide 1.174) on a "per reactor" basis [5]. Therefore, no multi-unit accident scenarios were selected for the SOARCA project.

Low power and shutdown accidents are potentially significant because the plant configuration is altered—the containment may be open and the reactor safety systems may be realigned. However, offsetting mitigating attributes include a potentially much smaller decay heat level and low pressure that allows for easier cooling of the reactor fuel. In this area, SOARCA has focused on the accidents that historically have received the most attention—the accidents initiated at full power.

Extreme seismic events that involve failure of the containment and lead to core damage have been excluded. We conclude that substantially more research is needed before it is feasible to undertake a realistic, best-estimate analysis of such rare events. NRC has developed plans to conduct this seismic PRA research.

Spent fuel pool accidents also contribute to overall risk associated with nuclear reactors because significant quantities of spent fuel are stored onsite in such pools. Past NRC studies including the most recent publicly available study, NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants" (February 2001) would suggest that risk from the most severe spent fuel pool accidents is low yet the consequences could be serious due to the release of a large inventory of cesium and other radioisotopes. Since that time, NRC has undertaken substantial analytical and empirical research to improve both the modeling of spent fuel pool accidents as well as research to identify significant improvements to spent fuel pool safety as part of NRC's security-related research following the terrorist attacks of September 11, 2001. Based on the results of this research, NRC concludes that the assessment of spent fuel pool risk that was assessed very conservatively in past studies such as NUREG-1738 is now much lower owing to both the new physical safety improvements required by NRC and the improved modeling capability. Therefore, when developing the SOARCA project, NRC elected to exclude spent fuel pool accidents from its scope. That exclusion does not negate the benefits of applying more detailed best-estimate methods to spent fuel pool accidents at some point in the future.

NRC did not include security events as part of SOARCA to preclude providing any specific information that may materially assist in the planning or carrying out of a terrorist attack on a nuclear power plant. However, we have stated that the security-related studies conducted after September 11, 2001, led us to conclude that previous risk studies were unnecessarily conservative in certain areas and results and that plant improvements plus improved modeling would confirm that radionuclide releases and early fatalities were substantially smaller than suggested by earlier studies.

1.6 Basis of Accident Selection

In the selection of important sequences, the SOARCA project would have ideally included those sequences found to be important to risk as demonstrated by a full-scope level 3 PRA. In practice, that was not feasible because no current full-scope level 3 PRAs were generally available (considering both internal and external events) to draw upon. However, the preponderance of level 1 PRA information combined with our insights on severe accident behavior is available on dominant core damage sequences, especially internal event sequences. This information combined with our understanding of containment loadings and failure mechanisms together with radionuclide release, transport, and deposition allow us to utilize core damage frequency (CDF) as a surrogate criterion for risk. Thus, for SOARCA, we elected to analyze sequences with a CDF greater than 10^{-6} per reactor-year. In addition, we included sequences that have an inherent potential for higher consequences (and risk) with a lower CDF (i.e., those with a frequency greater than 10^{-7} per reactor-year). Such sequences would be associated with events involving containment bypass or leading to an early failure of the containment. By adopting these criteria, we are reasonably assured that the more probable and important core melt sequences will be captured.

1.7 Mitigated and Unmitigated Cases

An important objective of the SOARCA project was to assess the impact of severe accident mitigative features and reactor operator actions to mitigating the accident. This was done by evaluating in detail the operator actions and equipment that may be available (including 10 CFR 50.54(hh) equipment) to mitigate the specific accident sequences to determine if time was available to take corrective action and whether the equipment itself would be available given the sequence.

Early in the project (2007), SOARCA staff visited the Peach Bottom Atomic Power Station and the Surry Power Station. During the visits, tabletop exercises were conducted for each scenario. Participants included plant senior reactor operators and PRA analysts. SOARCA staff provided initial and boundary conditions and elicited how plant staff would respond. Through the tabletop exercises, a timeline of operator actions was developed for each scenario. These mitigative measures analyses were qualitative, sequence-specific systems and operational analyses based on licensee-identified mitigative measures from Emergency Operation Procedures, Severe Accident Mitigation Guidelines, 10 CFR 50.44(hh) measures, assistance from the Technical Support Center, and other severe accident guidelines that are applicable to and determined to be available during a scenario whose availability, capability, and timing was utilized as an input into the MELCOR analyses. For scenarios involving a seismic initiator, operator response times were lengthened to reflect the severity of the seismic event. Based on the results of the tabletop exercises, SOARCA staff concluded that the scenarios could reasonably be mitigated, resulting in prevention of core damage or delaying or reducing radiation release.

A limitation of this approach is that a comprehensive human reliability assessment has not been performed to quantify the probabilities of plant personnel succeeding in implementing these measures. However, NRC has issued 10 CFR 50.54(hh) requiring plant licensees to possess the equipment, develop the strategies, and have trained personnel to implement these mitigative measures. The 10 CFR 50.54(hh) measures are the result of a major effort by industry and NRC in the 2004–2008 timeframe to develop means to mitigate events involving loss of large areas of the plant due to fire and explosions. These measures are new and diverse and include the following major elements:

- Procedures for manually operating turbine-driven injection (RCIC, TD-AFW).
- Portable diesel-driven pumps for injecting into RCS (BWR) and steam generators (PWR).

- Alternative means to depressurize.
- Portable power supplies for critical indication.

The assessment of mitigation measures has received comment during the project from stakeholders including Advisory Committee on Reactor Safeguards members and NRC regionally based senior reactor analysts. They noted that SOARCA did not include a quantitative human reliability analysis. However, the 10 CFR 50.54(hh) procedures and training were inspected as part of security assessments with site-specific evaluations prepared. SOARCA staff performed followup site visits in June and August 2010 to explicitly address Reactor Core Isolation Cooling (RCIC) blackstart and run for STSBO and manual operation of TD-AFW and to discuss fact check comments. The site visits included a review of RCIC blackstart and run procedures, additional tabletop exercises to address conservatism in the assumed PWR STSBO timeline, and walkdowns and detailed reviews of procedures. SOARCA staff concluded, following the site visits, a greater likelihood of implementing mitigation.

For sequences in which it was determined that mitigative measures would be taken, detailed accident progression analyses were conducted to assess the efficacy of those measures. For such sequences, accident progression and offsite consequence analyses were also performed assuming the mitigative measures were not taken to demonstrate the relative importance/significance of those measures.

For those scenarios within the scope of SOARCA, applicable mitigative measures that are potentially available (not eliminated by initial conditions) were identified. The systems and operations analyses were based on the initial conditions and anticipated subsequent failures to:

- Verify the availability of the primary system.
- Determine the availability of support systems and equipment.
- Determine time estimates for implementation.

Based on these scenario specifications, MELCOR was used to determine the effectiveness of those mitigative measures that are expected to be available at a given time.

1.8 Key Assumptions

In the development of the accident and consequence analysis for the SOARCA project, the concepts, applications, and parameters are identified in detail in the applicable report sections. Assumptions are identified throughout the report in the appropriate sections that address the analysis and in the appendices. A detailed listing of assumptions is not included here, in part because many are specific to a particular plant or accident scenario. Discussion of the application of best-estimate modeling practices for MELCOR is summarized in a separate report, NUREG/CR-7008 [6].

1.9 Uncertainty Analysis

As part of the SOARCA project, a number of sensitivity studies have been performed to examine issues associated with accident progression, mitigation and offsite consequences for the accident scenarios of interest. These sensitivity studies were performed to examine specific issues and to ensure the robustness of the conclusions documented in this report. Single sensitivity studies, however, do not form a complete picture of the uncertainty associated with the accident progression and offsite consequence modeling. Such a picture requires a more comprehensive evaluation of both epistemic (state of knowledge) and aleatory (random) modeling uncertainties.

In general terms, the "best-estimate" offsite consequence results presented in this documentation reflect the aleatory uncertainty associated with weather conditions at the time of the accident scenario considered. These best-estimate offsite consequence values represent the expected (mean) value of the probability distribution obtained from a large number of weather "trials." The impact of epistemic model parameter uncertainty will be evaluated in a follow-on uncertainty study by randomly sampling distributions for key model parameters that were considered to have a potential impact on the offsite consequences. The intended purpose of this uncertainty study is to develop insight into the overall sensitivity of the SOARCA results to input uncertainty. This initial study will leverage existing models and software, along with a representative set of uncertain parameters, to evaluate the method and feasibility of conducting an uncertainty analysis and will benefit a longer-term study by identifying areas requiring more focused effort.

Ideally, the uncertainty study described here would be conducted for each accident sequence considered by the SOARCA project. Practical considerations require that a more limited study be conducted. As a result, a detailed uncertainty study will be performed for a single-accident sequence rather than all of SOARCA sequences. The Peach Bottom Unmitigated Long Term Station Blackout (LTSBO) Scenario has been selected as the accident scenario used to develop insight into the overall sensitivity of the SOARCA analysis to the input uncertainty. The justifications for this choice are both technical and programmatic:

- The performance of the safety release valve as it impacts main steam line failure in the LTSBO scenario was an important sensitivity study identified by the peer review committee.

- Several implied and explicit commitments have been made by SNL and NRC staff to further explore this issue in the uncertainty analysis.

The LTSBO release timing and consequences are characteristic of the majority of the SOARCA scenarios (i.e., long-release timing relative to evacuation time and correspondingly small offsite consequences). This makes the choice of the LTSBO consistent with the objectives of the SOARCA project to explore the center of the risk distribution as opposed to a more outlying case such as the Surry ISLOCA or SGTR.

In addition, to make the uncertainty study more tractable, only a subset of model parameters will be considered. The model parameters included in the study, as well as the distributions used to characterize the uncertainty in the accident progression parameters, will be identified and characterized by informal elicitation from the SOARCA analysis team subject matter experts. The subject matter experts were asked define distributions for the parameters that they considered most important. They were also asked to provide a technical basis for the distribution definitions. The parameter uncertainty analysis will be based upon a mapping between uncertain inputs and the Peach Bottom LSTBO analysis results using (1) partial rank correlation coefficients (PRCCs), (2) stepwise rank regression analyses, and (3) scatter plots.

1.10 Structure of NUREG-1935 and Supporting Documents

The structure of the NUREG is in multiple volumes. This volume is the introduction to the SOARCA project and describes the approach and procedures used in the study and summarizes the project results and conclusions. Appendices A and B contain the plant-specific SOARCA results for the Peach Bottom and Surry plants, respectively.

This Page Intentionally Left Blank

2.0 ACCIDENT SCENARIO SELECTION

An accident sequence begins with the occurrence of an initiating event (e.g., a loss of offsite power, a loss-of-coolant accident [LOCA], or an earthquake) that perturbs the steady state operation of the nuclear power plant. The initiating event challenges the plant's control and safety systems, whose failure could potentially cause damage to the reactor fuel and result in the release of radioactive fission products. Because a nuclear power plant has numerous diverse and redundant safety systems, many different accident sequences are possible depending on the type of initiating event that occurs, the amount of equipment that fails, and the nature of the operator actions involved.

One way to systematically identify possible accident sequences is to develop accident sequence logic models using event tree analysis as is done in probabilistic risk assessments (PRAs). Pathways through an event tree represent accident sequences. Typically, the analysis is divided into two parts: (1) a Level 1 PRA that represents the plant's behavior from the occurrence of an initiating event until core damage occurs and (2) a Level 2 PRA that represents the plant's behavior from the onset of core damage until radiological release occurs. The development of accident sequence logic models requires detailed information about the plant and the expertise of engineers and scientists from a wide variety of technical disciplines. As a result, the construction of accident sequence logic models is a complex and time-consuming activity.

Many PRAs have already been completed by the U.S. Nuclear Regulatory Commission (NRC) and nuclear power plant licensees. However, because of the improvements in PRA technology and plant capabilities and performance, the more current PRA information available was given more importance.

2.1 Approach

Figure 2 illustrates the overall process used to identify and characterize accident scenarios for the SOARCA project. The SOARCA scenarios were selected from the results of existing PRAs. Some of these existing PRAs model accident sequences out to the point of radiological release (i.e., they are Level 2 PRAs); however, the majority of existing PRAs are limited to the onset of core damage (i.e., Level 1 PRAs). Therefore, the SOARCA scenario selection process was developed with an eye toward the type and limitations of the information contained in existing PRAs. Core-damage sequences from previous staff and licensee PRAs were identified and binned into core-damage groups. A core-damage group consists of core-damage sequences that have similar timing for important severe accident phenomena and similar containment or engineered safety feature operability. The groups were screened according to their approximate core-damage frequencies to identify the most significant groups. Finally, the accident scenario descriptions were augmented by assessing the status of containment systems (which are not typically modeled in Level 1 PRAs).

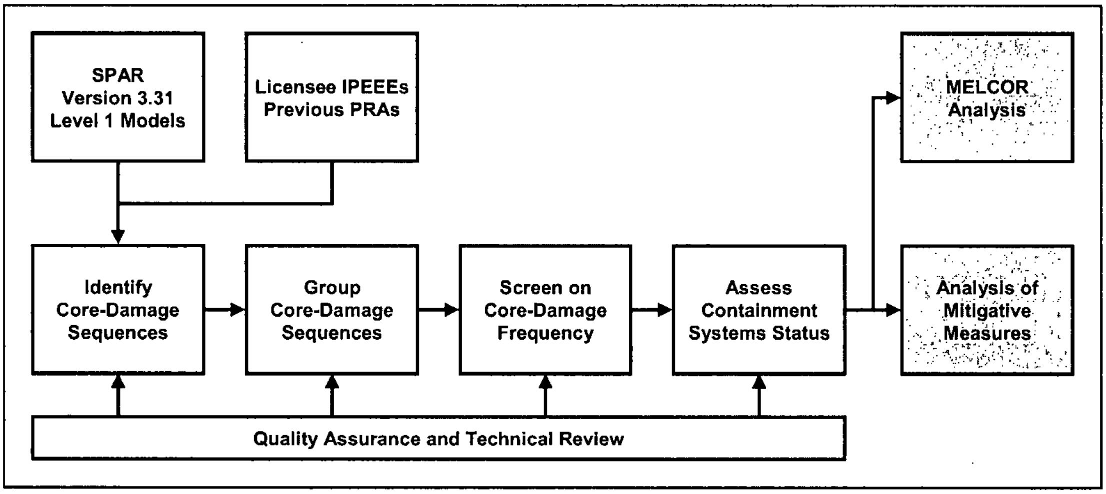

Figure 2. SOARCA Accident Scenario Selection and Analysis Process.

The analyses done using MELCOR and MACCS2 were limited based on the following core damage frequency (CDF) screening guidelines:

- 10^{-6} per reactor-year for most scenarios.
- 10^{-7} per reactor year for scenarios that are known to have the potential for higher consequences (e.g., containment bypass scenarios such as steam generator tube rupture and interfacing system loss-of-coolant accident [ISLOCA] initiators).

To accomplish this, the release characteristics were grouped so that they are representative of scenarios binned into those groups. In addition, the groups are sufficiently broad to include the potentially risk-significant but lower-frequency scenarios. As a result of limitations in available Level 2 analyses and models, the scenario selection and screening was performed using core-damage frequency (CDF) per reactor-year as the screening criterion rather than radionuclide release frequency.

The application of the screening guidelines to the available level 1 PRA information for the pilot plants resulted in the identification of two basic types of scenarios—station blackouts and bypass scenarios. This result presents certain advantages with respect to consideration of the inherent adequacy of our criteria and the adequacy of the scope of scenarios analyzed. First, station blackout scenarios are representative of a broad class of events in PRA—loss-of-heat removal events. Selection of station blackout (SBO) events in SOARCA ensures that we have covered that broader class of transients involving a loss-of-heat removal and further, by including a short-term blackout we have reasonably bounded that class of accidents (which could include other events such as loss of service water or loss of component cooling water but which develop more slowly). Also, for the PWR, the station blackout also includes, in part, the effect of a small loss of coolant by considering reactor coolant pump seal leakage. In addition, by the selection of station blackout sequences for analysis, we also include the effects of loss-of-containment heat removal (fan coolers) and loss-of-containment spray systems (which are all electrically powered) to remove airborne radionuclides. Thus, our non-bypass sequences also result in containment failure that would not be the case for all other such loss-of-heat removal transients in a typical PRA. Therefore, although we have used CDF for screening, in effect the CDF in these cases also represents the radionuclide release frequency.

Although we have not included medium or large loss-of-inventory accidents—because of their very low frequency—it should be noted that such internal events were well below our selection criteria for the boiling-water reactor and comfortably below our screening criterion of 10^{-6} for the pressurized-water reactor plant. For Peach Bottom, the medium and large LOCAs had frequencies of $2x10^{-9}$ and $1x10^{-9}$ /ry. For Surry, the medium and large LOCAs had frequencies of $6x10^{-8}$ and $7x10^{-10}$ /ry. Only a fraction of these sequences would have resulted in containment failure because a loss of containment heat removal may not have occurred. Because for Surry we have included an ISLOCA sequence, it can be argued that we also have reasonably bounded events involving a LOCA inside containment for that plant.

All the sequences identified in the SOARCA study are significant in an absolute sense. The American Society of Mechanical Engineers' "Standard for Probabilistic Risk Assessment for Nuclear Power Plants," ASME RA-Sb-2005, which was endorsed by the staff in Regulatory Guide 1.200, defines a significant sequence, in part, as one that individually contributes more than 1 percent to the core damage frequency (CDF). A CDF of 10^{-4} per reactor-year is an acceptable surrogate for the quantitative health objectives contained in the Commission's Safety Goal Policy Statement (51 FR 28044) (see Appendix D of NUREG-1860). It then follows that the SOARCA sequence selection criterion of 10^{-6} is 1 percent of an acceptable CDF goal, and the SOARCA sequences are consistent with previously issued regulatory guidance.

Another way to judge the impact of low-frequency events is to consider the increase in the latent cancer consequences that would be necessary to offset the lower frequency. Conceptually, an event with a larger radiological release could have greater latent cancer risk if the increase in the radiological release is larger than the decrease in frequency of the event. For example, assuming the accident timing remains the same and using a linear no-threshold risk assumption, a 10^{-8} per reactor-year event must have a radiological release more than 10 times the magnitude of an event with a frequency of 10^{-7} per reactor-year to pose greater latent cancer risk. Because we are including events with substantial volatile releases on the order of 10 percent, it is not feasible to achieve greater latent cancer fatality risk by increasing the magnitude of the release by more than a factor of 10.

Other than the magnitude of the radiological release, a major impact on both early and latent cancer fatality risks is derived from the timing of the offsite release. In this respect, we have examined candidate SOARCA sequences with timing in mind, both the timing of core damage and the timing of containment failure. As part of this consideration, we addressed, for the Peach Bottom plant, an additional sequence—the short-term SBO—even though it did not satisfy our selection criterion. The short-term SBO frequency is roughly an order of magnitude lower than the long-term SBO ($3x10^{-7}$ per reactor-year versus $3x10^{-6}$ per reactor-year); however, the short-term SBO has a more prompt radiological release and a slightly larger release over the same interval of time. Our initial qualitative assessment of the short-term SBO led us to conclude that it would not have greater risk significance than the long-term SBO. Although it was a more prompt release (8 hours versus 20 hours), the release was delayed beyond the time needed for successful evacuation. To demonstrate the points regarding risk versus frequency for lower frequency events, we nonetheless included a detailed analysis of the short-term SBO. For that analysis, the absolute risk is indeed smaller for the short-term SBO than for the long-term SBO. The same trends apply for the Surry sequences where the lower-frequency sequences may have greater conditional risk but absolute risk is smaller than or equivalent to other higher-frequency sequences.

Finally, we routinely considered core damage initiators and phenomenological containment failure modes in SOARCA that have been considered in the past, except for those that have been excluded by extensive research (alpha mode failure, direct containment heating, and gross failure without prior leakage). Our detailed analysis includes modeling of behavior (including radionuclide transport and release) associated with long-term containment pressurization, Mark I liner failure, induced steam generator tube rupture, hydrogen combustion, and core concrete interactions.

We also have compared the SOARCA sequences against those identified as important to risk in NUREG-1150 for the Surry and Peach Bottom plants. Adjusting for the improvements in our understanding of phenomena due to the research completed since the NUREG-1150 study was completed (roughly 18 years ago), we have found that, with one exception, SOARCA addresses the more likely and important sequences identified in that landmark study. The one exception—a sequence identified in NUREG 1150 that has not been analyzed for the SOARCA project—involved an extreme earthquake that directly results in a large breach of the reactor coolant system (large loss-of-coolant accident [LOCA]), a large breach of the containment, and an immediate loss of safety systems. We conclude that this sequence is not appropriate for consideration as part of SOARCA for a number of reasons.

Foremost, the state of quantification of such extreme and low-frequency seismic events is poor, considerable uncertainty exists in the quantification of the seismic-loading condition itself, and a detailed soil-structure interactions analysis was not performed for the plant (and its equipment) response to the seismic loads. The analysis of the plant's components to the seismic acceleration—commonly referred to as fragility analysis—is a key component, and the lack of detailed analysis in this area makes current consideration of this event incompatible with the thrust of SOARCA, which is the performance of detailed, realistic analyses. Moreover, recent experience at nuclear plants in Japan strongly suggests that nuclear plant designs possess inherently greater capability to withstand the effects of extremely large earthquakes. In addition, it would not be sufficient to perform a nuclear plant risk evaluation of this event (even if it were currently feasible) without also performing an assessment of the concomitant nonnuclear risk associated with such a large earthquake. This assessment would have to include an analysis of the impact on public health of an extremely large earthquake—larger than that generally considered in residential or commercial construction codes—to provide the perspective on the relative risk posed by operation of the plant.

Kashiwazaki-Kariwa, 2007

In summary, SOARCA addresses the more likely (though still remote) and important sequences that are understood to compose much of the severe accident risk from nuclear plants. We conclude that the general methods of SOARCA (i.e., detailed, consistent, phenomenologically based, sequence-specific accident progression analyses) are applicable to PRA methodology and should be the focus of improvements in that regard.

2.2 Scenarios Initiated by Internal Events

The scenarios generated by internal events and the availability of containment systems for these scenarios were identified using NRC's plant-specific standardized plant analysis risk (SPAR) models, licensee PRAs, and other risk information sources. The SPAR models are used to support NRC's oversight of licensed commercial nuclear power plants and have been developed and maintained under a formal quality assurance program. The Peach Bottom SPAR model has been peer reviewed against staff-endorsed industry consensus PRA standards. Both the Surry and Peach Bottom licensee PRAs have been peer reviewed against the same standards. In addition, the SPAR model accident sequence results (including the sequence minimal cut sets) are periodically compared to the results from licensee PRAs under the Mitigating System Performance Index program, which is part of the NRC's Reactor Oversight Process. As a result, both the qualitative and quantitative results from the Surry and Peach Bottom SPAR models are in reasonable agreement with the corresponding licensee PRAs.

The following process was used to determine the scenarios for further analyses in the SOARCA project:

- Candidate accident scenarios were identified in analyses using plant-specific SPAR models (Version 3.31).
  - Initial Screening. Screened out initiating events with low CDFs ($<10^{-7}$) and sequences with a CDF $<10^{-8}$. This step eliminated 4 percent of the overall CDF for Peach Bottom and 7 percent of the overall CDF for Surry.
  - Sequence Evaluation. Identified and evaluated the dominant cutsets for the remaining sequences. Determined system and equipment availabilities and accident sequence timing.
  - Scenario Grouping. Grouped sequences with similar times to core damage and equipment availabilities into scenarios.
- Containment systems availabilities for each scenario were assessed using system dependency tables that delineate the support systems required for performance of the target front-line systems and from a review of existing SPAR model system fault trees.
- Core-damage sequences from the licensee PRA model were reviewed and compared with the scenarios determined by using the SPAR models. Differences were resolved during meetings with licensee staff.
- The screening criteria (CDF $< 10^{-6}$ for most scenarios and $< 10^{-7}$ for containment bypass sequences) were applied to eliminate scenarios from further analyses.

This process provides the basic characteristics of each scenario. However, it is necessary to have more detailed information about scenario than is contained in a PRA model. To capture the additional scenario details, further analysis of system descriptions and review of the normal and emergency operating procedures (EOPs) is required. This review includes the analysis of mitigation measures beyond those treated in current PRA models. Mitigation measures treated in SOARCA include the licensee's EOPs, severe accident management guidelines, and 10 CFR 50.54(hh) mitigation measures. Section 3.0 describes the mitigation measures assessment process used to determine what mitigation measures would be available and the associated timing to implement.

2.2.1 Scenarios Initiated by External Events

External events include internal flooding and fire; seismic events; extreme wind-, tornado-, and hurricane-related events; and other similar events that may be applicable to a specific site. The external event scenarios developed for SOARCA analysis were derived from a review of past studies such as the NUREG-1150 study, individual plant examination for external event submittals, and other relevant generic information. Detailed sequence characteristics are more difficult to specify for external event scenarios because of the general lack of external event PRA models industrywide. As a result, the SOARCA external event scenarios are heuristically based as opposed to the internal event scenarios that were developed through more formal, rigorous PRA methods.

All of the external event scenarios were assumed to be seismically initiated. In general, seismically initiated scenarios present unique challenges to successfully implement onsite mitigative measures and offsite protective actions. In addition, seismically initiated scenarios were found to be important contributors to the overall external event core damage and release frequencies in previous risk assessments. They are not intended to represent what might happen during other types of internal hazard/external event accident scenarios.

No attempt was made to match the frequencies of the external event scenarios to the actual sequence frequencies in any of the input information sources because much of the available quantitative risk information on external events is dated. For example, since the input information sources were published, new seismic hazard estimates have been developed. As a result, the estimated frequencies of the external event scenarios were based on expert judgment that considered the impact of changes in seismic data and methods on the published external event PRA results. Care was taken to ensure that the external event scenario selection maintained the observed relative importance of external events CDF versus internal events CDF.

In addition, the lack of detailed external event PRA information also necessitated the use of expert judgment to establish the containment safeguards status and to provide the scenario initial and boundary conditions needed for the accident progression calculations performed by MELCOR.

2.3 Accident Scenarios Selected for Surry

Four accident sequences were selected for the Surry plant (two initiated by internal events and two initiated by external events). The following sections identify each selected accident scenario, provide its representative core-damage frequency, and summarize the accident scenario in terms of its initiating event, equipment failures, and operator errors.

2.3.1 Surry Internal Event Scenarios

Two internal event scenarios for Surry were determined to meet the criteria for further analysis.

1. Initiating Event: Spontaneous Steam Generator Tube Rupture

   Representative CDF: 5×10^{-7} per reactor-year (SPAR)

   Scenario Summary: This scenario is initiated by a spontaneous rupture in one steam generator tube. The operators fail to (1) isolate the faulted steam generator or to cooldown and depressurize the reactor and (2) initiate long-term heat removal. Core damage occurs because of refueling water storage tank (RWST) depletion and the operator failure to refill the RWST or cross-connect to another water source. Auxiliary feedwater (AFW), high-pressure injection (HPI), low-pressure injection (LPI), and containment spray are available, if needed. However, high-pressure recirculation, low-pressure recirculation, and the recirculation sprays will be unavailable as a result of lack of water in the containment sump.

   Comparison with Licensee PRA: The licensee PRA calculates a CDF 1×10^{-6} per reactor-year for this scenario. The conditional core-damage probabilities are virtually identical for the SPAR analysis (1.4×10^{-4}) and for the licensee PRA (1.5×10^{-4}). The difference in the calculated CDFs is mainly attributable to the difference in initiating event frequency. Because both the SPAR model and licensee-calculated CDFs for this scenario are above the 1×10^{-7} per reactor-year threshold for containment bypass scenarios, this scenario was retained for further analysis.

2. Initiating Event: Interfacing Systems LOCA in the LPI System

   Representative Frequency: 7×10^{-7} per reactor-year (licensee PRA)

   Scenario Summary: This scenario is initiated by failure of two check valves in series in the discharge path of the LPI system. The flow of reactor coolant system (RCS) fluid passing through the failed check valves results in pressurization of the LPI piping in the Safeguards Building and its subsequent pipe rupture. The rupture location cannot be isolated. The ability to inject via the LPI is failed by the rupture. The HPI system remains available because the pumps are located in a separate location. Core damage occurs because of RWST depletion and operator failure to refill the RWST or cross-connect to another water source.

   Comparison with Licensee PRA: The SPAR model for Surry calculates a CDF 3×10^{-8} per reactor-year for this scenario compared to the licensee PRA value of $7\text{x}10^{-7}$. The largest contributor to the difference in scenario CDFs between the SPAR model and the licensee

PRA model is the conditional probability of the low-pressure portion of the LPI piping system rupturing given that the two isolation check valves have failed open. The SPAR model assigns a value of 0.1 for this event while the Surry PRA assigns a probability of failure of 1.0. This sequence group was retained for further analysis because the licensee PRA frequency exceeds the SOARCA screening criteria and it has historically been important to PWR risk.

2.3.2 Surry External Event Scenarios

Two external event scenarios for Surry were determined to meet the criteria for further analysis.

1. Initiating Event: Seismic-Initiated Long-Term Station Blackout

   Representative Frequency: 1×10^{-5} to 2×10^{-5} per reactor-year

   Scenario Summary: This scenario is initiated by a beyond-design-basis earthquake (0.3–0.5g peak ground acceleration [PGA]). The seismic event results in loss of offsite power (LOOP) and failure of onsite emergency alternating current (AC) power resulting in an SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable including the containment systems (containment spray and fan coolers). The turbine-driven auxiliary feedwater (TDAFW) system is available initially. In the long term, loss of the TDAFW may occur because of battery depletion and loss of direct current (DC) power for sensing and control. Because of loss of pump seal cooling, a reactor coolant pump seal leakage will occur.

2. Initiating Event: Seismic-Initiated Short-Term Station Blackout

   Representative Frequency: 1×10^{-6} to 2×10^{-6} per reactor-year

   Scenario Summary: This scenario is initiated by a beyond-design-basis earthquake (0.5–1.0g PGA). The seismic event results in a LOOP and failure of onsite emergency AC power resulting in a SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable, including the containment systems (containment spray and fan coolers). The seismic event also results in a loss of DC power resulting in the unavailability of the TDAFW system. The earthquake ruptures the Emergency Condensation Storage Tank causing it to immediately empty, rendering the turbine-driven auxiliary feedwater system initially unavailable.

An additional seismic-initiated short-term station blackout scenario is considered, involving a thermally induced steam generator tube rupture (TI-SGTR). The representative frequency for this event is estimated to range between 1×10^{-7} and 8×10^{-7} per reactor-year from NUREG-1570 [7].

2.4 Accident Scenarios Selected for Peach Bottom

Two accident scenarios were selected for the Peach Bottom plant (both initiated by a seismic event). The following sections identify each selected accident scenario, provide its representative core-damage frequency, and summarize the accident scenario in terms of its initiating event, equipment failures, and operator errors.

2.4.1 Peach Bottom Internal Event Scenarios

The loss of vital AC Bus E-12 was initially estimated to have a frequency above the SOARCA screening criterion of 1x10-6/reactor-year. However, after further review of the SPAR model and comparison with the licensee's PRA, the scenario was determined to have a CDF below the screening criteria. Because the MELCOR analysis provided unique insights into the response of the plant to an internal event sequence, the MELCOR analysis was retained.

2.4.2 Peach Bottom External Event Scenarios

1. Initiating Event: Seismic-Initiated Long-Term Station Blackout

   Representative Frequency: 1×10^{-6} to 5×10^{-6} per reactor-year

   Scenario Summary: This scenario is initiated by a beyond-design-basis earthquake (0.3–0.5g PGA). The seismic event results in a LOOP, failure of onsite emergency AC power, and failure of the Conowingo Dam power line resulting in an SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable including the containment systems (containment spray). The turbine-driven injection systems—high-pressure coolant injection (HPCI) and/or reactor core isolation cooling (RCIC)—are available initially. Loss of room cooling and/or battery depletion results in eventual failure of these systems leading to core damage.

2. Initiating Event: Seismic-Initiated Short-Term Station Blackout

   Representative Frequency: 1×10^{-7} to 5×10^{-7} per reactor-year

   Scenario Summary: This scenario is initiated by a beyond-design-basis earthquake (0.5–1.0g PGA). The seismic event results in a LOOP, failure of onsite emergency AC power, and failure of the Conowingo Dam power line resulting in an SBO event where neither onsite nor offsite AC power are recoverable. All systems dependent on AC power are unavailable including the containment systems (containment spray). In addition, HPCI and RCIC are unavailable due to loss of DC power. The earthquake ruptures the condensate storage tank causing it to immediately empty into the surrounding concrete enclosure with water rising to a depth of 9.2 ft for Unit 2 and 10.9 ft for Unit 3. The earthquake causes the fire water system to fail.

   Note: The short-term SBO scenario does not meet the SOARCA screening criterion of 1×10^{-6} per reactor-year; however, the scenario was retained for analysis to assess the risk importance of a lower frequency, potentially higher consequence scenario. This type of scenario has been a risk-important severe accident scenario in past PRA studies and, at a frequency of 5×10^{-7} per reactor-year, it is only a factor of 2 below the screening criterion.

2.5 Generic Factors

The results of existing PRAs indicate that the likelihood of a nuclear power plant accident sequence that releases a significant amount of radioactivity is very small owing to the diverse and redundant barriers and numerous safety systems in the plant, the training and skills of the reactor operators, testing and maintenance activities, and the regulatory requirements and oversight of NRC. In addition, it is important to recognize that risk estimates of nuclear power plants have decreased over the years. Several reasons exist for these decreases:

- Utilities have completed plant modifications intended to remedy concerns raised in earlier PRAs.
- Plants exhibit better performance as evidenced by reductions in initiating event frequencies, improvements in equipment reliability, and higher equipment availability. Nuclear power plant equipment has become more reliable and available due to improved maintenance practices motivated by implementation of the Maintenance Rule (10 CFR 50.65) [8].
- New regulations have been created such as the ATWS Rule (10 CFR 50.62) [9] and the Station Blackout Rule (10 CFR 50.63) [10] that directly affect the likelihood of certain types of accidents. It should be noted that although the ATWS Rule and the SBO Rule were issued prior to the completion of NUREG-1150, the impact of these rules on risk was not addressed by NUREG-1150.
- PRA methodologies have improved, allowing a more realistic assessment of risk to be made. In this category, improvements in common-cause failures analysis are noteworthy.

As a result, risk estimates reflect the impacts of constantly changing plant operational, regulatory, and PRA technology environments. Any attempt to identify significant accident sequences should be viewed as a "snapshot" of the plant at the time the analysis was completed.

3.0 METHODS USED FOR MITIGATIVE MEASURES ASSESSMENT

Section 2.0 describes the probabilistic risk assessment (PRA) information sources including the U.S. Nuclear Regulatory Commission's (NRC's) SPAR models, licensees' PRA models, NUREG-1150, and expert judgment that were used to identify risk-important sequence groups leading to core damage and containment failure or bypass. This section describes the methods used to determine what mitigation measures would be available and the associated timing to implement.

3.1 Site-Specific Mitigation Strategies

In preparation for the detailed, realistic modeling of accident progression and offsite consequences, the SOARCA project staff had extensive cooperation from the licensees to develop high-fidelity plant systems models, define operator actions including the most recently developed mitigative actions, and develop models for simulation of site-specific and scenario-specific emergency planning. Moreover, in addition to input for model development, licensees provided information from their own PRA on accident scenarios. Through tabletop exercises (with senior reactor operators, PRA analysts, and other licensee staff) of the selected scenarios, licensees provided input on the timing and nature of the operator actions to mitigate the selected scenarios. The licensee input for each scenario was used to develop timelines of operator actions and equipment lineup or setup times for the implementation of the available mitigation measures. This includes mitigation measures beyond those treated in current PRA models.

Mitigation measures treated in SOARCA include emergency operating procedures, Severe Accident Management Guidelines (SAMGs), and 10 CFR 50.54(hh) mitigation measures. These mitigation measures refer to additional equipment and strategies required by NRC following the terrorist attacks of September 11, 2001, to further improve severe accident mitigation capability. NRC inspectors completed the verification of licensee implementation (i.e., equipment, procedures, and training) of 10 CFR 50.54(hh) mitigation measures in December 2008. These mitigation measures were developed for use during scenarios involving large fires and explosions. One type of 10 CFR 50.54(hh) measure is portable, self-powered equipment including generators and diesel driven pumps. Portable generators provide electrical power to indication equipment to provide critical indications such as reactor vessel water level. Portable generators also provide electrical power needed to operate safety relief valves. Portable diesel-driven pumps provide a diverse and independent means of injecting water into the reactor coolant system and steam generators. Another type of 10 CFR 50.54(hh) measure is starting and controlling without electrical control power the plant's existing turbine-driven injection systems including reactor core isolation cooling (RCIC) and turbine-driven auxiliary feedwater.

Based on tabletop exercises, the project staff judged that mitigation measures including emergency operating procedures, severe accident guidelines, and security-related mitigative measures will either avert core damage or delay or reduce the release of radiation. To quantify the benefits of the mitigation measures and to provide a basis for comparison to past analyses of unmitigated severe accident scenarios, the project staff also analyzed the scenarios assuming the events proceed as unmitigated by available onsite mitigation measures and lead ultimately to core damage or an offsite release. This NUREG refers to these as unmitigated scenarios because they are not effectively mitigated in the short term by onsite resources.

3.1.1 Sequence Groups Initiated by External Events

Scenarios identified in SOARCA included both externally and internally initiated events. The externally initiated events frequently included events for which seismic, fire, and flooding initiators were grouped together. For the externally initiated events, the timeline of operator actions was developed assuming the initiator was a seismic event because the seismic initiator was judged to be the most severe initiator in terms of timing and with respect to how much equipment would be available to mitigate. Thus, some conservatism is involved in attributing all of the event likelihood to a seismic initiator.

The PRA screening identified the following sequence groups that were initiated by external events and met the SOARCA screening criteria of $1x10^{-6}$/reactor-year for containment failure events and $1x10^{-7}$/reactor-year for containment bypass events:

- Peach Bottom long-term station blackout (SBO): $1x10^{-6}$ to $5x10^{-6}$/reactor-year.
- Surry long-term SBO: $1x10^{-5}$ to $2x10^{-5}$/reactor-year.
- Surry short-term SBO: $1x10^{-6}$ to $2x10^{-6}$/reactor-year.
- Surry short-term SBO with thermally induced steam generator tube rupture: $1x10^{-7}$ to $8x10^{-7}$/reactor year.

It is important to note that, although it is not included in the above list, the seismically induced Peach Bottom short-term SBO was also retained for analysis. With a frequency of $1x10^{-7}$ to $5x10^{-7}$/reactor-year, this scenario does not explicitly meet the SOARCA screening criterion. Nonetheless, it was retained to assess the risk importance of a lower frequency, potentially higher consequence scenario.

These sequence groups were initiated by a seismic, fire, or flooding event. The mitigation measures assessment for each of these sequence groups was performed assuming the initiator was a seismic event because it was judged to be limiting in terms of how much equipment would be available to mitigate. Fewer mitigation measures are expected to be available for a seismic event than for an internal fire or flooding event. For example, a seismic event would destroy makeup tanks while fire and flooding events would not. Based on the estimated level of plant damage, the availability of 10 CFR 50.54(hh) mitigation measures, their implementation time, and the timing and effectiveness of the emergency response, organization support (e.g., in the Technical Support Center [TSC] and Emergency Operating Facility [EOF]) was evaluated.

Seismic events considered in SOARCA result in loss of offsite and onsite AC power and, for the more severe seismic events, loss of DC power. Under these conditions, the turbine-driven systems RCIC and turbine-driven auxiliary feedwater (TD-AFW) are important mitigation measures. Boiling-water reactor (BWR) SAMGs include starting RCIC without electricity to cope with station blackout conditions. This is known as RCIC black start. The 10 CFR 50.54(hh) mitigation measures have taken this a step further and also include long-term operation of RCIC without electricity (RCIC black run) using a portable generator to supply indications such as reactor pressure vessel (RPV)-level indication to allow the operator to manually adjust RCIC flow to prevent RPV overfill and flooding of the RCIC turbine. Similar procedures have been developed for pressurized-water reactors for TD-AFW. For the Peach Bottom and Surry long-term SBO sequence groups, RCIC and TD-AFW can be used to cool the core until battery exhaustion. In addition, black start procedures can be used for the Peach Bottom short-term SBO sequence. After battery exhaustion, black run of RCIC and TD-AFW can be used to continue to cool the core. MELCOR calculations are used to demonstrate core cooling under these conditions.

Seismic PRAs for Peach Bottom and Surry do not describe general plant damage and accessibility. The damage was assumed to be widespread and accessibility to be difficult consistent with the unavailability of many plant systems. The seismic event was assumed to fail the condensate storage tank in the Peach Bottom long-term SBO, which is the primary water reservoir for RCIC. Consequently, RCIC must be initially supplied from the torus. MELCOR calculations showed that several hours would be available before torus temperature and pressure conditions precluded this. However, this would provide sufficient time to identify or arrange for another water reservoir for RCIC, such as the Peach Bottom cooling tower basin (a large low-lying reinforced concrete structure). For the Surry long-term SBO, the TD-AFW system and the emergency condensate storage tank (ECST) were not expected to fail. Consequently, the cooling water was supplied to the steam generators for reactor coolant system (RCS) heat removal. It was assumed that eventually operators would provide makeup to the ECST. For the Surry short-term SBO, the ECST was assumed to fail and an alternative reservoir was assumed to be available by 8 hours; this could be achieved by using a fire truck or portable pump to draw from the river.

Also, for the Surry short-term SBO, the low-pressure injection and containment spray safety-related piping were judged not likely to fail. This judgment was primarily based on NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants"[11], to help extrapolate the potential viability of safety-related piping after a 1.0 g event. This conclusion also considered other related studies including a German study, "Seismic PSA of the Neckarwestheim 1 Nuclear Power Plant" [12], that physically simulated ground motion equal to 1 g on an existing plant. The integrity of this piping provided a connection point for a portable, diesel-driven pump to inject into the RCS or into the containment spray systems. Licensee staff estimated that transporting the pump and connecting it to plant piping takes about 2 hours. However, for the short-term SBO, this mitigation measure was estimated to take 8 hours owing to the higher level of damage. Because the installation time was beyond the estimated time to fuel damage and vessel failure (3 hours to core damage, 7 hours to lower head failure), the containment spray system was the preferred mitigation measure. A better understanding of the effect of large seismic events on general plant conditions would be helpful in reducing uncertainty in availability and accessibility for mitigation measures. If accessibility was not significantly impaired and delay in using the portable pump was limited to 2 hours, then core damage could be averted.

The 10 CFR 50.54(hh) mitigation measures include portable equipment (such as portable power supplies to supply indication, portable diesel-driven pumps, and portable air bottles to open air-operated valves) together with procedures to implement these measures under severe accident conditions. Surry's portable equipment and fire truck are stored onsite in a one-story, multibay garage. Some of Peach Bottom's portable equipment is stored in an open bay in the water treatment building and some is stored outside under a tarp. Hence, it was believed that portable equipment could be accessed and deployed for the seismic conditions evaluated in SOARCA. Time estimates to implement individual mitigation measures were estimated by the NRC based on licensee input for each sequence group scenario presented by NRC. Also, for portable equipment at Surry, the time estimates reflect exercises run by licensee staff that provided actual times to move the equipment into place. The time estimates for staffing the technical support centers and the emergency operating facilities were estimated based on regulatory requirements and the potential for additional delays resulting from the possible effect of the seismic event on roads and bridges.

The mitigation measures assessment noted the possibility of bringing in equipment from offsite (e.g., fire trucks, pumps, and power supplies from sister plants or from contractors), but it did not quantify the types, amounts, and timing of this equipment arriving and being implemented. This equipment is judged to be more effective in mitigating an environmental release after it begins. Section 3.2 provides additional information on equipment available offsite and time estimates for transporting this equipment.

Because multi-unit accident sequences were not selected for the SOARCA project, the mitigation measures assessment for external events was performed assuming that the operators only had to mitigate an accident at one reactor even though Peach Bottom and Surry are two-unit sites.

3.1.2 Sequence Groups Initiated by Internal Events

The PRA screening identified the following sequence groups that were initiated by internal events and met the SOARCA screening criteria of $1x10^{-6}$/reactor-year for containment failure events and $1x10^{-7}$/reactor-year for containment bypass events:

- Surry interfacing systems loss-of-cooling accident (ISLOCA): $7x10^{-7}$/reactor-year (licensee PRA), $3x10^{-8}$/reactor-year (SPAR).

- Surry spontaneous steam generator tube rupture: $5x10^{-7}$/reactor-year.

These sequence groups result in core damage as a result of assumed operator errors. For the ISLOCA, the operators fail to refill the RWST or cross-connect to the unaffected unit's RWST. For the spontaneous steam generator tube rupture (SGTR), the operators fail to (1) isolate the faulted steam generator, (2) depressurize and cool down the RCS, and (3) refill the RWST or cross-connect to the unaffected unit's RWST.

The SPAR model and the licensee's PRA concluded that these two events proceed to core damage as a result of the above-postulated operator errors. However, these PRA models do not appear to have credited the significant time available for the operators to correctly respond to events. They also do not appear to credit technical assistance from the TSC and the EOF. For the ISLOCA, the realistic analysis of thermal hydraulics presented in Appendix B subsequently estimated 3 hours until the RWST is empty and 10 hours until fission product release begins, providing time for the operators to correctly respond. The ISLOCA time estimates are based on a double ended pipe rupture. These estimates only would be longer for much smaller break sizes because of a flow limiting orifice in the piping system. For the SGTR, the realistic analysis of thermal hydraulics showed from 24 to 48 hours until core damage begins. Therefore, based on realistic time estimates by which the technical assistance is received from the TSC and the EOF, it was highly likely the operators would correctly respond to the events. These time estimates included consideration of indications that the operators would have of the bypass accident, operator training on plant procedures for dealing with bypass accidents and related drills, and assistance from the TSC and EOF, which were estimated to be staffed and operational by 1 to 1.5 hours into the event.

The mitigation measures assessment for internal events also included 10 CFR 50.54(hh) mitigation measures, but these measures were subsequently shown to be redundant to the wide variety of equipment and indications available for mitigating the ISLOCA and SGTR. ISLOCA and SGTR are internal events that involve few equipment failures and are controlled by operator errors.

The PRA screening for Peach Bottom initially identified the Loss of Vital AC Bus E12 sequence group as exceeding the SOARCA screening criterion of $1x10^{-6}$/reactor-year. However, an inappropriate modeling assumption was subsequently found in the SPAR model, and the sequence group frequency was determined to be below the SOARCA screening criterion. However, by the time the issue was discovered, the mitigation measures assessment and the MELCOR analysis were complete. The MELCOR analysis described in Appendix A demonstrated that this sequence group did not result in core damage, even without crediting 10 CFR 50.54(hh) mitigation measures. Nevertheless, the mitigation measures assessment and the MELCOR analysis for this sequence are described in this report to demonstrate the benefit of a detailed review of success criteria using integrated thermal-hydraulic analysis.

3.2 Truncation of Releases

Many resources are available at the State, regional, and national level that would be available to mitigate a nuclear power plant accident. The staff reviewed available resources and emergency plans and determined that adequate mitigation measures could be brought onsite within 24 hours and connected and functioning within 48 hours.

The National Response Framework (NRF) would be implemented in response to a severe nuclear power plant accident to coordinate the national-level response. Under the NRF, the Department of Homeland Security (DHS) would be the coordinating agency and NRC would be a cooperating agency. The NRF is exercised periodically and provides access to the full resources of the Federal Government. NRC has an extensive, well trained, and exercised emergency response capability and has onsite resident inspectors. These onsite inspectors are equipped and available to provide first-hand knowledge of accident conditions. NRC would activate the incident response team at the NRC regional office and headquarters. The focus of the NRC response is to ensure that public health and safety is protected and to assist the licensee with the response by working with DHS to coordinate the national response. Concurrently, the NRC regional office would send a site team to staff positions in the reactor control room, TSC, and EOF to support the response. NRC performs an independent assessment of the actions taken or proposed by the licensee to confirm such actions will stop core damage.

Both Surry and Peach Bottom are supported by a remote EOF. The emergency response organization at the EOF has access to fleetwide emergency response personnel and equipment including the 10 CFR 50.54(hh) mitigation measures and equipment from sister plants. These assets as well as those from neighboring utilities and State preparedness programs could be brought to bear on the accident if needed. Every licensee participates in a full onsite and offsite exercises every 2 years where response to severe accidents and coordination with offsite response organizations is demonstrated and inspected by NRC and the Federal Emergency Management Agency. In addition, the Institute for Nuclear Power Operations and the Nuclear Energy Institute would activate their emergency response centers to assist the site as needed.

All of the described resources would be available to the site to mitigate the accident. Although some of these efforts would be ad hoc, knowledgeable personnel and an extensive array of equipment would be available and were considered in the conclusion that radiological releases would be truncated within 48 hours except for the Surry LTSBO sequence, which was truncated at 72 hours.

4.0 SOURCE TERM ANALYSIS

The source term is defined as the quantity, timing, and characteristics of the release of radioactive material to the environment following a postulated severe accident. The U.S. Nuclear Regulatory Commission (NRC) has defined, calculated, and used source terms for a variety of research and regulatory activities. Two uses include (1) siting and regulatory applications and (2) probabilistic risk or consequence assessments. Two source terms used for regulatory applications include TID-14844 [13] and the alternate source term [14]. In contrast to the definition above, the regulatory source terms are releases to the containment that are available for release to environment. The second use of the source term is an assessment of health consequence risks from severe accidents. Many significant examples of the latter application exist including NUREG/CR-2239 [1], NUREG-1150 [2], and SOARCA. In NUREG/CR-2239, five prescribed source terms of widely ranging severity were defined (the core melt source terms were assigned a representative frequency) and used to calculate the health consequences. In NUREG-1150, a comprehensive, plant-specific evaluation of accident sequences was performed using event tree models and then grouped into a manageable number of characteristic source terms to calculate the health consequences and risk. In the present SOARCA study, selected individual scenario source terms are evaluated using MELCOR code calculations and then evaluated for conditional and unconditional health risks.

Section 4.1 describes some background in key studies for regulatory and probabilistic applications. Figure 3 shows a timeline of key events and NRC studies in the evolution of nuclear safety technology as well as the key source terms studies cited in the timeline that preceded the SOARCA program (also discussed in Section 4.1 below). Next, Section 4.2 describes a history of the severe accident source term codes developed by NRC. The MELCOR code is the culmination of the NRC research and code development of severe accident phenomena for source term evaluations. The scope of the MELCOR code and the MELCOR modeling approach used in the SOARCA analyses are presented in Sections 4.2 and 4.3, respectively. The MELCOR modeling approach includes the development of the plant models, the best practices approaches to important but uncertain phenomena and equipment performance, recent advances in source term models, and the methods used to calculate the radionuclide inventories.

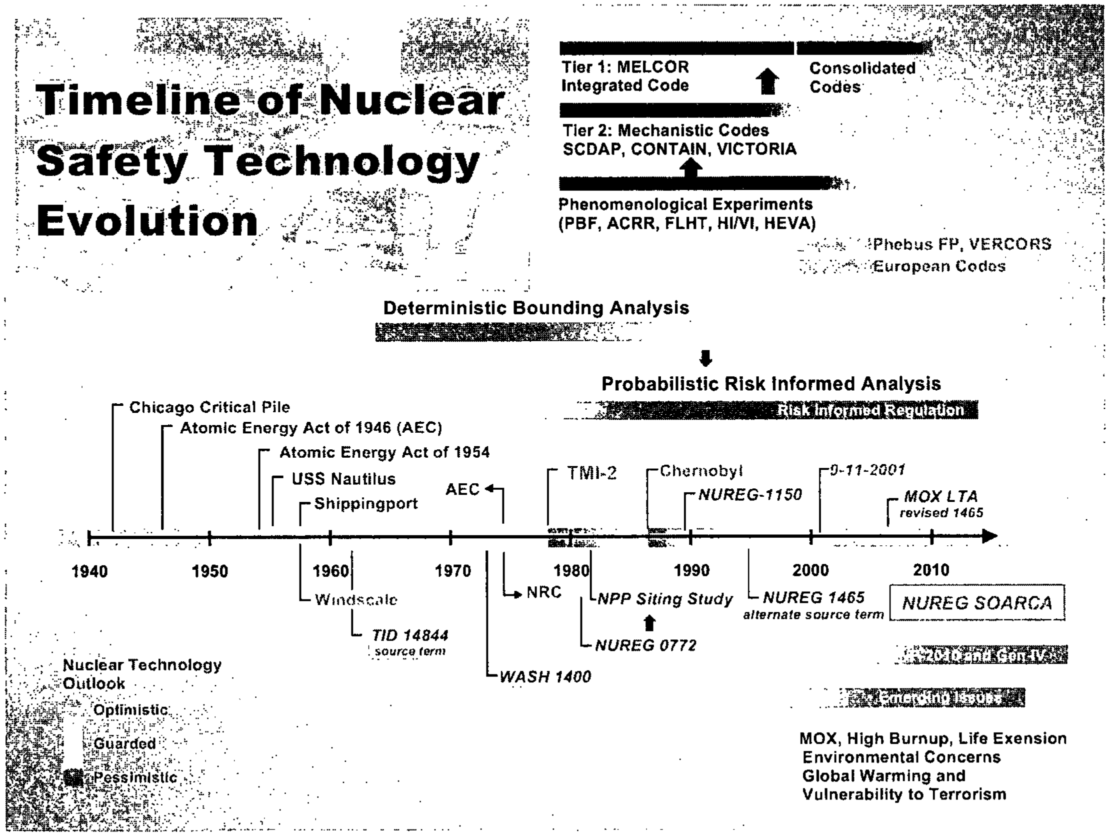

Figure 3. Timeline of Key Nuclear Power Events and Safety Studies.

4.1 Source Term Study Background

One of the earliest estimates of the source term came from the WASH-740 study in 1957 [15]. Three core damage cases were defined with increasing levels of severity. The first case was defined as a situation in which major damage to the core resulted in failure of the vessel. However, the containment remained intact and thus prevented a major release of radioactivity to the environment. This case was subsequently used to define the characteristics of the source term for reactor siting (i.e., TID-14844) [13]. In the other two cases, releases occurred offsite.

The TID-14844 source term postulated the release of all the noble gases, 50 percent of the iodine, and 1 percent of the radioactive solids to the containment. In addition, TID-14844 provided assumptions for containment leakage and for atmospheric transport. However, it was recognized that the procedures and results specified in TID-14844 were approximations (sometimes relatively poor ones) to the results that would be obtained if the effects of the all-influencing variables could be recognized and associated with fixed levels of uncertainty—an impossibility in the state-of-the-art at the time [16]. Nevertheless, TID-14844 was codified as "the maximum credible accident" in the siting regulations of 10 CFR Part 100, "Reactor Site Criteria" [17].

The next most significant source term study, the Reactor Safety Study (WASH-1400) [3], was the first systematic attempt to provide realistic estimates of public risk from potential accidents in commercial nuclear power plants. The 1975 study included analytical methods for determining both the probabilities and consequences of various accident scenarios. Event trees and fault trees were used to define important accident sequences and to quantify the reliability of engineered safety systems. A more comprehensive list of nine pressurized-water reactors (PWRs) and five boiling-water reactors (BWRs) source terms was developed. All the accidents that were believed to contribute significantly to the overall core melt frequency were grouped, or "binned," into the source term categories. The WASH-1400 source terms included characterizations of accident timing, the release duration (e.g., puff or sustained release), and the energy of the release for plume loft considerations. As an improvement over TID-14844, the radioactivity was described using eight chemical categories. The 54 most health-significant isotopes were used in health consequence calculations.

The WASH-1400 methodology used to predict the health effects from the source term was based on the newly developed Calculation of Reactor Accident Consequences (CRAC) code [18] that calculated the atmospheric dispersion and health consequences. However, an integrated tool for the calculation of the source term did not exist. The estimation of the source term used the best analytic procedures available at the time. When ample data were available, a model for the phenomenon was included as realistically as possible, but when data were lacking, consideration of the phenomenon was omitted. The resultant source terms reflected uncertainties and poor understanding of applicable phenomena. Uncertainties in accident frequencies were accounted for by adding 10 percent of the likelihood of each release category into the next larger and the next smaller category.

Subsequently, NRC documented the technical basis for source terms in NUREG-0772 [19]. NUREG-0772 assessed the assumptions, procedures, and available data for predicting fission product behavior. Four conclusions of the NUREG-0772 study were (1) a new definition of the chemical form of iodine (i.e., CsI was the dominant form), (2) the potential retention of CsI within the vessel or containment versus elemental iodine, (3) the inclusion of in-vessel retention, and (4) the role of containment engineering safety features (e.g., sprays, suppression pools, and ice condensers). However, much of the quantitative assessment in NUREG-0772 was based on scoping calculations that were only applicable to specific conditions. In particular, the examination of fission product behavior in different regions of the plant with different accidents was conducted in parallel with limited consideration of integral effects. The potential impact of the NUREG-0772 findings on reactor regulation was also examined, and the results were documented in NUREG-0771 [20].

NUREG-0771 and NUREG-0772 studies formed the basis for the designation of five accident groups as being representative of the spectrum of potential accident conditions that were documented in NUREG-0773 [21]. In 1982, the NUREG/CR-2239 siting study [1] was performed using the NUREG-0773 source terms. The five source terms were assessed to adequately span the range of possible source terms. The source terms were developed from separate effects computer code analyses that were performed in 1978. The source terms were used to calculate accident consequences at 91 U.S. reactor sites using site-specific population data and a mixture of site-specific and regionally specific meteorological data. An objective of the SOARCA study is to update this study.

In response to emerging severe accident research technology and computing power, a study was performed at Battelle Columbus Laboratories that involved the development and modification of a number of separate effects severe accident computer codes based on emerging severe accident research. The codes were coupled together to form a code suite that could calculate a complete accident sequence. The source terms for about 25 specific sequences were calculated for five operating plants using the new Source Term Code Package (STCP) code [22]. Although the STCP was a significant step forward in deterministic severe accident analysis, the code suite had some significant shortcomings. Because the code represented the linkage of many separate code modules, the data transfer and feedback effects were not always handled consistently. The technical basis for the models in the STCP was documented in NUREG-0956 [16]. The results from the STCP calculations supported the NUREG-1150 probabilistic risk assessment (PRA) [2] along with expert judgment and simplified algorithms for sequence-specific source terms.

The NUREG-1150 PRA was an effort to put the insights gained from the research on system behavior and phenomenological aspects of severe accidents into a risk perspective. An important characteristic of this study was the inclusion of the uncertainties in the calculations of core damage frequency and source term due to incomplete understanding of reactor systems and severe accident phenomena at that time. NUREG-1150 therefore used sensitivity studies, uncertainty studies, and expert judgment to characterize the likelihood of alternative events that affect the course of an accident. The elicitation of expert judgment was used to develop probability distributions for many accident progression, containment loading, structural response, and source term issues. The insights from the NUREG-1150 study have been used in several areas of reactor regulation including the development of alternative radiological source terms for evaluating design-basis accidents at nuclear reactors.

In 1995, NRC published NUREG-1465 [14], which defined an alternative accident source term for regulatory applications. The NUREG-1465 source term is considered an alternative to TID-14844, which specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving a "substantial meltdown of the core." NUREG-1465 documents the basis for more realistic estimates of the source term release into containment in terms of timing, nuclide types, quantities, and chemical form given a severe core-melt accident. This revised source term is to be applied to the design of future light-water reactors (LWRs). Current LWR licensees may voluntarily propose applications based upon it.

4.2 The MELCOR Code

The MELCOR code is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of accidents in LWR nuclear power plants as well as in nonreactor systems (e.g., spent fuel pool, dry cask). Current uses of MELCOR include estimation of fission product source terms and their sensitivities and uncertainties in a variety of applications. MELCOR is a modular code comprising three general types of packages: (1) basic physical phenomena (i.e., hydrodynamics [control volume and flow paths], heat and mass transfer to structures, gas combustion, aerosol and vapor physics); (2) reactor-specific phenomena (i.e., decay heat generation, core degradation, ex-vessel phenomena, sprays, and engineering safety systems); and (3) support functions (thermodynamics, equations of state, other material properties, data-handling utilities, and equation solvers). As a fully integrated code, MELCOR models all major systems of a reactor plant and their important coupled interactions. Figure 4 shows the MELCOR code integration of separate effects codes.

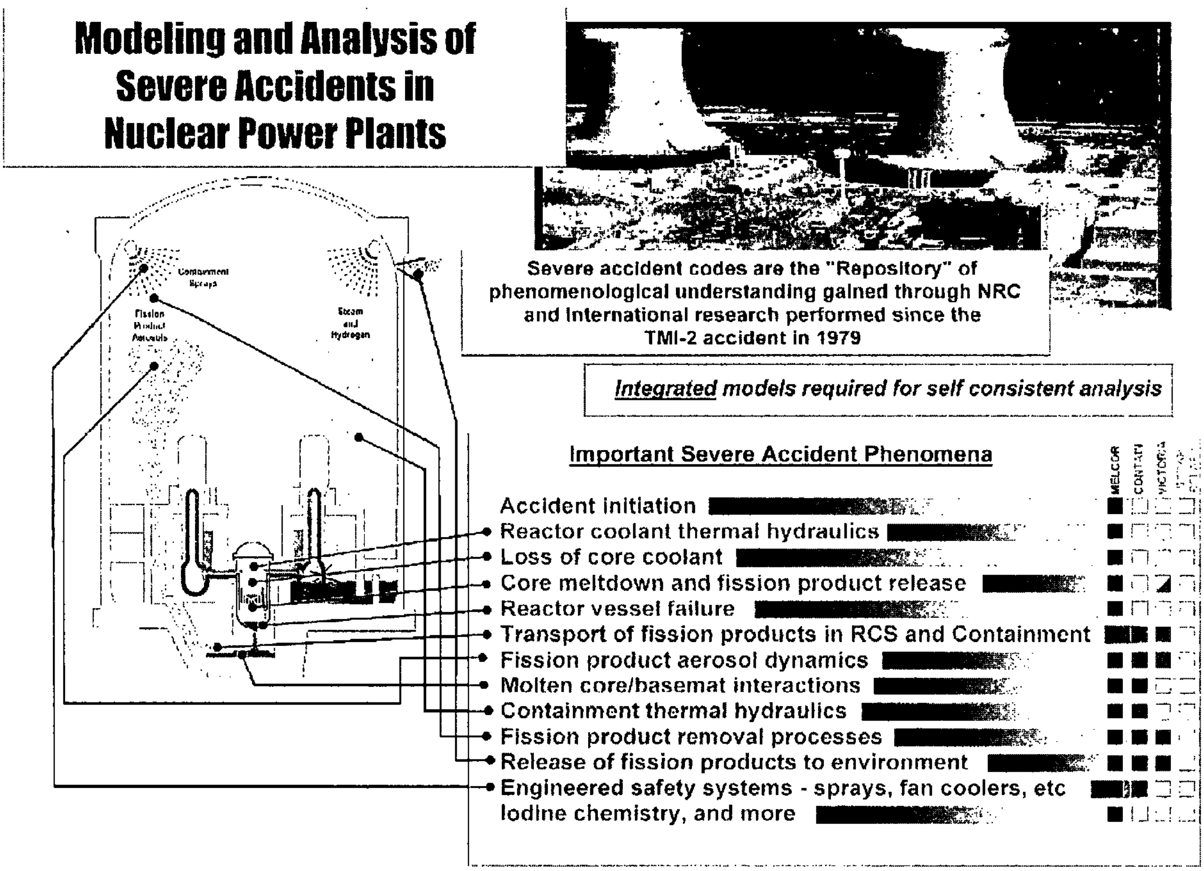

Figure 4. MELCOR Integration of Separate Effects Codes.

The scope of MELCOR includes:

- Thermal-hydraulic response of the primary reactor coolant system, reactor cavity, containment, and confinement buildings.
- Core uncovery (loss of coolant), fuel heatup, cladding oxidation, fuel degradation (loss of rod geometry), and core material melting and relocation.
- Heatup of reactor vessel lower head from relocated core materials and the thermal and mechanical loading and failure of the vessel lower head and transfer of core materials to the reactor vessel cavity.
- Core-concrete attack and ensuing aerosol generation.
- In-vessel and ex-vessel hydrogen production, transport, and combustion.
- Fission product release (aerosol and vapor), transport, and deposition.
- Behavior of radioactive aerosols in the reactor containment building including scrubbing in water pools and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling.
- The impact of engineered safety features on thermal-hydraulic and radionuclide behavior.

Most MELCOR models are mechanistic, and the use of parametric models is limited to areas of high phenomenological uncertainty where no consensus exists concerning an acceptable mechanistic approach. Current use of MELCOR often includes uncertainty analyses and sensitivity studies. To facilitate this, many of the mechanistic models have been coded with optional adjustable parameters. This does not affect the mechanistic nature of the modeling, but it does allow the analyst to easily address questions of how particular modeling parameters affect the course of a calculated transient. Parameters of this type as well as such numerical parameters as convergence criteria and iteration limits are coded in MELCOR as sensitivity coefficients, which may be modified through optional code input. It should be noted that core radioactive nuclide inventories are not utilized by MELCOR; rather, masses and decay heats of chemical element groups are used. Appropriate code calculations are performed for specific fuel and core design and are carried out to the burnup of interest to provide the initial core inventories for MELCOR severe accident analysis (see Section 4.3.1).

4.3 MELCOR Modeling Approach

Section 4.3.1 presents a high-level description of the MELCOR models that were used for the SOARCA project. Existing MELCOR models for Surry and Peach Bottom were updated to current state-of-the-art modeling practices as well as the latest version of the MELCOR code. More detailed information describing the plant models is provided in the plant-specific analysis reports (i.e., Appendix A and Appendix B for Peach Bottom and Surry, respectively).

The modeling and prediction of accident progression and radiological release in a severe accident requires the integration of a number of phenomenological models to address a range of thermal-hydraulic, materials, structural and fission product behavior as well as models for component (e.g., safety relief valve) behavior. Section 4.3.2 describes the procedure to define the best practices approach to modeling important and uncertain phenomena. NUREG/CR-7008 [6] provides a more detailed description of the best practices modeling approach. At the beginning of the SOARCA project, an independent review of MELCOR best practices modeling was conducted to provide greater assurance of the technical soundness of the analytical modeling. NRC and Sandia National Laboratories (SNL) used that review to identify and incorporate subsequent modeling insights and improvements prior to the start of plant analyses. Moreover, during the SOARCA peer review, members of the peer review committee recommended additional sensitivity analyses to explore specific modeling issues that were viewed to be both uncertain and potentially important to risk. These analyses, discussed in detail in Appendices A and B, help confirm that the MELCOR best practices modeling is sound and not subject to large uncertainty that would affect the prediction of individual risk.

Despite the fact that significant severe accident research has been conducted to resolve a number of issues previously thought to be important or highly uncertain and despite the fact that MELCOR sensitivity studies have been performed to address individual issues, uncertainty remains in the prediction of accident progression and the radionuclide release associated with severe accidents as well as in the overall prediction of individual health risk. Although SOARCA has initially and properly focused on modeling and predictions using our best understanding (without incorporation of large and arbitrary conservatisms), it is recognized that a quantitative integrated uncertainty evaluation is useful to more fully understand and characterize the uncertainty in these analyses. The uncertainty analysis (discussed in Section 1.9) performed as part of the overall SOARCA project will also be based on our improved knowledge acquired over years of research. The overall metric for the uncertainty analysis will be the conditional individual health risk predicted for the unmitigated severe accident scenario(s) selected. The scope will include the analysis of accident progression, radionuclide release to the environment, emergency planning (e.g., evacuation), plume dispersal (and deposition), and health effects modeling. Thus, the uncertainty analysis will address uncertainty in both MELCOR and MACCS2 analyses. No effort will be made to address the uncertainty in the characterization of the core damage frequency associated with the severe accident scenario, what would normally be considered to be the uncertainty in the level 1 PRA issues. This is considered to be beyond the scope of SOARCA, which has focused on the better quantification of those analyses associated with the quantification of risk given a severe accident. The principal objective of the selected uncertainty analysis is to provide a mean value of conditional risk reflecting the integrated consideration of risk that is readily comparable to the value cited as the SOARCA best-estimate value and to understand the dominant contributors to the uncertainty.

Section 4.3.3 summarizes some recent changes to the radionuclide release and cesium speciation modeling, which is important to the source term results. Finally, Section 4.3.4 describes the methodology to calculate the radionuclide inventory in.

4.3.1 Plant Models

The MELCOR models used in the SOARCA source term calculations represented the state-of-the-art. As part of the SOARCA program, the MELCOR models were updated to the most recent version of the MELCOR code.[2] The scope of the models included:

- Detailed 5-ring reactor vessel models.
- Representation of the primary reactor coolant systems (and secondary steam generator through the main steam isolation valve for Surry).
- Representation of the primary containment.
- Representation of the Peach Bottom reactor building and the Surry auxiliary building, which were radionuclide pathways in some scenarios.
- Representation of the emergency core-cooling systems (and the auxiliary feedwater system for Surry).
- Representations of the emergency, portable water-injection systems.

Through the best practices updates to each deck, the following new models were specified for both plants for these important but uncertain phenomena or equipment responses:

- Safety relief valve failure modeling addressing stochastic and high-temperature failure modes.
- An additional thermo-mechanical fuel collapse model for heavily oxidized fuel following molten zircaloy breakout.
- Enhanced lower plenum coolant debris heat transfer that recognizes breakup and multidimensional cooling effects not present in the 1-dimensional counter-current flooding model in older versions of MELCOR (e.g., [23]).
- Updated, plant-specific chemical element masses and decay heats (see Section 4.3.4).
- A new Oak Ridge National Laboratory Booth chemical element release model and new Cs speciation model (see Section 4.3.3).

[2] MELCOR Version 1.8.6 was used for all SOARCA calculations. MELCOR Version 2.0 was released during the initial phase of the SOARCA program. Version 2.0 is based on identical physics models as Version 1.8.6 but has been modernized to use FORTRAN 90, new input format, and to enable automated source term information for preparing MACCS2 input.

- Vessel failure based on gross failure[3] [24] using the improved one-dimensional creep rupture model with the new hemispherical head model and radial heat transfer between lower head conduction node segments.

- Enhanced ex-vessel core debris heat transfer that recognizes multidimensional effects and rates measured in MACE tests [25].

A summary of recent enhancements to the MELCOR Peach Bottom and Surry models for the SOARCA program are presented in Sections 4.3.1.1 and 4.3.1.2, respectively.

4.3.1.1 Peach Bottom MELCOR Model

The Peach Bottom MELCOR model was originally developed for code version 1.8.0 at Brookhaven National Laboratory. The model was subsequently adopted by J. Carbajo at Oak Ridge National Laboratory to study differences in fission product source term behavior predicted by MELCOR 1.8.1 and those generated for use in NUREG-1150 using the Source Term Code Package (STCP) [26]. Starting in 2001, Sandia National Laboratories made considerable refinements to the BWR/4 core nodalization to support the developmental assessment and release of MELCOR 1.8.5. These refinements concentrated on the spatial nodalization of the reactor core (both in terms of fuel/structural material and hydrodynamic volumes) used to calculate in-vessel melt progression.

Subsequent work in support of several NRC research programs has motivated further refinement and expansion of the BWR/4 model in four broad areas. The first area involved the addition of models to represent a wide spectrum of plant design features, such as safety systems, to broaden the capabilities of MELCOR simulations to a wider range of severe accident sequences. These enhancements include:

- Modifications of modeling features needed to achieve steady-state reactor conditions (recirculation loops, jet pumps, steam separators, steam dryers, feedwater flow, control rod drive hydraulic system, main steam lines, turbine/hotwell, core power profile).

- New models and control logic to represent coolant injection systems (reactor core isolation cooling, high-pressure coolant injection, residual heat removal, low-pressure core spray) and supporting water resources (e.g., condensate storage tank with switchover).

- New models to simulate reactor vessel pressure management (safety relief valves, safety valves, automatic depressurization system, and logic for manual actions to effect a controlled depressurization if torus water temperatures exceed the heat capacity temperature limit).

[3] A more complete discussion of this model is presented in NUREG/CR-7008 [6] and the MELCOR manual [17]. A penetration failure model was not used because the timing differences between gross lower head failure and penetration failure with the available penetration model are not significant to the overall accident progression (i.e., minutes difference). Also, Sandia lower head failure tests showed gross creep rupture of the lower head was measured to be the most likely mechanism for vessel failure [24].

The second area focused on the spatial representation of primary and secondary containment. The drywell portion of primary containment has been subdivided to distinguish thermodynamic conditions internal to the pedestal from those within the drywell itself. Also, refinements have been added to the spatial representation and flow paths within the reactor building (i.e., secondary containment). An updated containment failure model is included that accounts for leakage around the drywell head flange, leakage due to elevated drywell temperature, and leakage due to drywell melt-through (see Appendix A, Section 4.5). The third area has focused on bringing the model up to current "best practice" standards for MELCOR 1.8.6 (see Section 4.3.2). The fourth area of model improvements included a new radionuclide inventory and decay heat based on the recent plant operating history (see Section 4.3.4).

Although not new for SOARCA, the MELCOR Peach Bottom model includes a multiregion ex-vessel debris spreading model. The debris spreads according to its temperature relative to the solidus and liquidus temperatures of the concrete and the debris height. If the debris spreads against the drywell liner steel wall, the liner will fail if the debris temperature is above the carbon steel melting temperature.

Appendix A more fully describes the MELCOR Peach Bottom model. Figure 5 shows the MELCOR nodalization diagrams for Peach Bottom.

4.3.1.2 Surry MELCOR Model

The Surry MELCOR model applied in this study was originally generated at Idaho National Engineering Laboratory (INEL) in 1988. The model was periodically updated by Sandia National Laboratories (1990 to present) for the purposes of testing new models, advancing the state-of-the-art in modeling of PWR accident progression, and providing support to decisionmakers at NRC for analyses of various issues that may affect operational safety. Significant changes were made during the last 20 years in the approach to modeling core behavior and core melt progression as well as the nodalization and treatment of coolant flow within the reactor coolant system and reactor vessel. In 2002, the reactor vessel and reactor coolant system nodalization were updated using the SCDAP/RELAP5 Surry model to include a five-ring vessel nodalization and counter-current hot leg representation for natural circulation flow [27]. The current MELCOR Surry model is a culmination of these efforts and represents the state-of-the-art in modeling of potential PWR severe accidents.

In preparation for the SOARCA analyses described in this report, the model was further refined and expanded in three areas. The first area is an upgrade to MELCOR Version 1.8.6 core modeling. These enhancements include:

- A hemispherical lower head model that replaces the flat bottom-cylindrical lower head model.
- New models for the core former and shroud structures that are fully integrated into the material degradation modeling including separate modeling of debris in the bypass region between the core barrel and the core shroud.

- Models for simulating the formation of molten pools in both the core and lower plenum, crust formation, convection in molten pools, stratification of molten pools into metallic and oxide layers, and partitioning of radionuclides between stratified molten pools.

- A reflood quench model that separately tracks the component quench front and the quenched and unquenched temperatures.

- A control rod silver aerosol release model.

- Addition of the new Oak Ridge National Laboratory Booth radionuclide release model for modern high-burnup fuel.

The second area focused on the addition of user-specified models to represent a wide spectrum of plant design features and safety systems to broaden the capabilities of MELCOR to a wider range of severe accident sequences. These enhancements included:

- Update of the containment leakage model to include nominal leakage and leakage due to containment overpressure (see Appendix B, Section 4.4).

- Update of core degradation modeling practices.

- Modeling of individual primary and secondary relief valves with failure logic for rated and degraded conditions.

- Update of the containment flooding characteristics.

- Heat loss from the reactor to the containment.

- Separate motor and turbine-driven auxiliary feedwater models with control logic for plant automatic and operator cooldown responses.

- New turbine-driven auxiliary feedwater models for steam flow, flooding failure, and performance degradation at low pressure.

- Nitrogen discharge model for accumulators.

- Update of the fission product inventory, the axial and radial peaking factors, and an extensive fission product tracking control system.

- Improvements to the natural circulation in the hot leg and steam generator and the potential for creep rupture.

Appendix B more fully describes the MELCOR Surry model. Figure 6 shows the MELCOR nodalization diagrams for Surry.

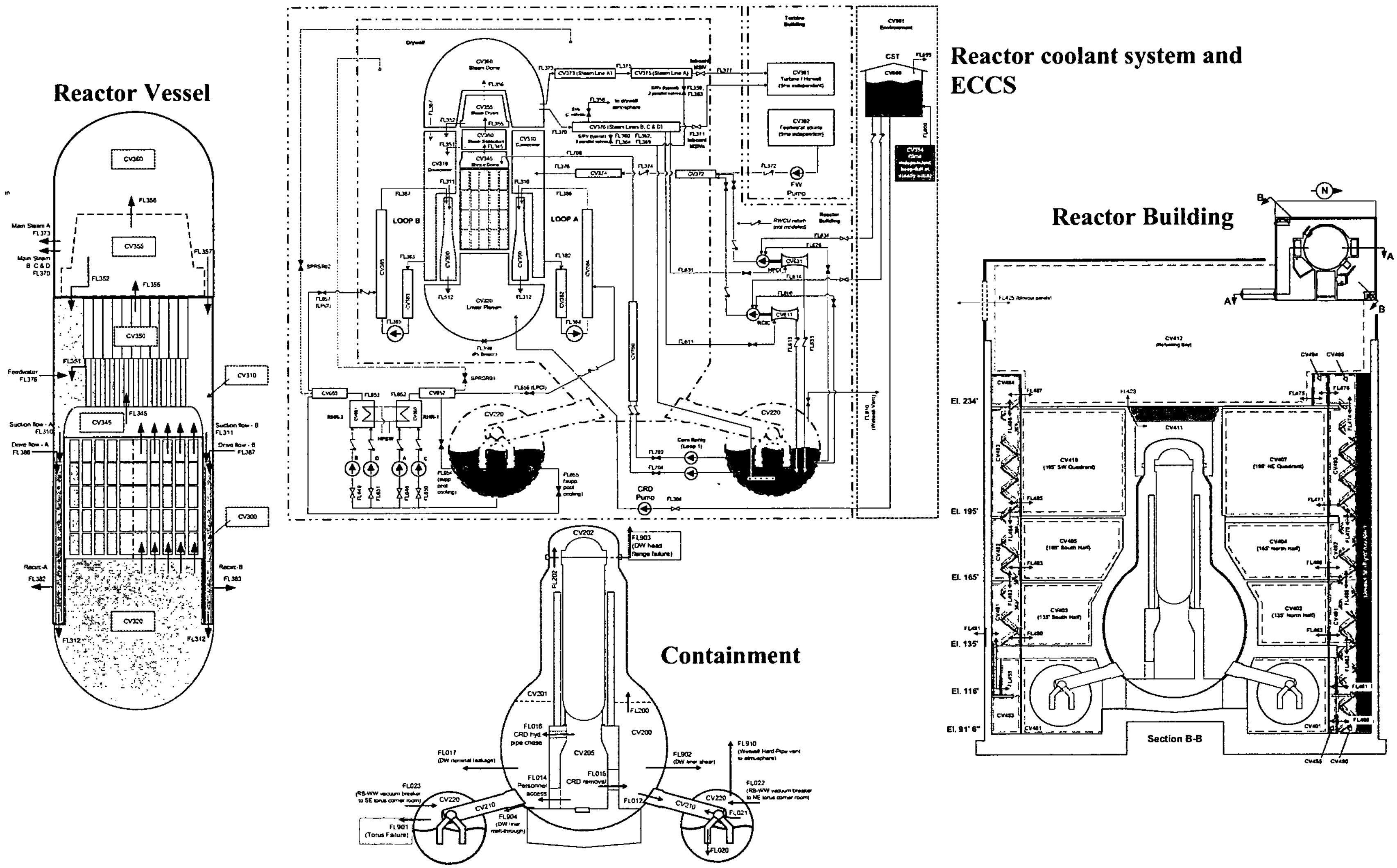

Figure 5. The Peach Bottom MELCOR Nodalization.

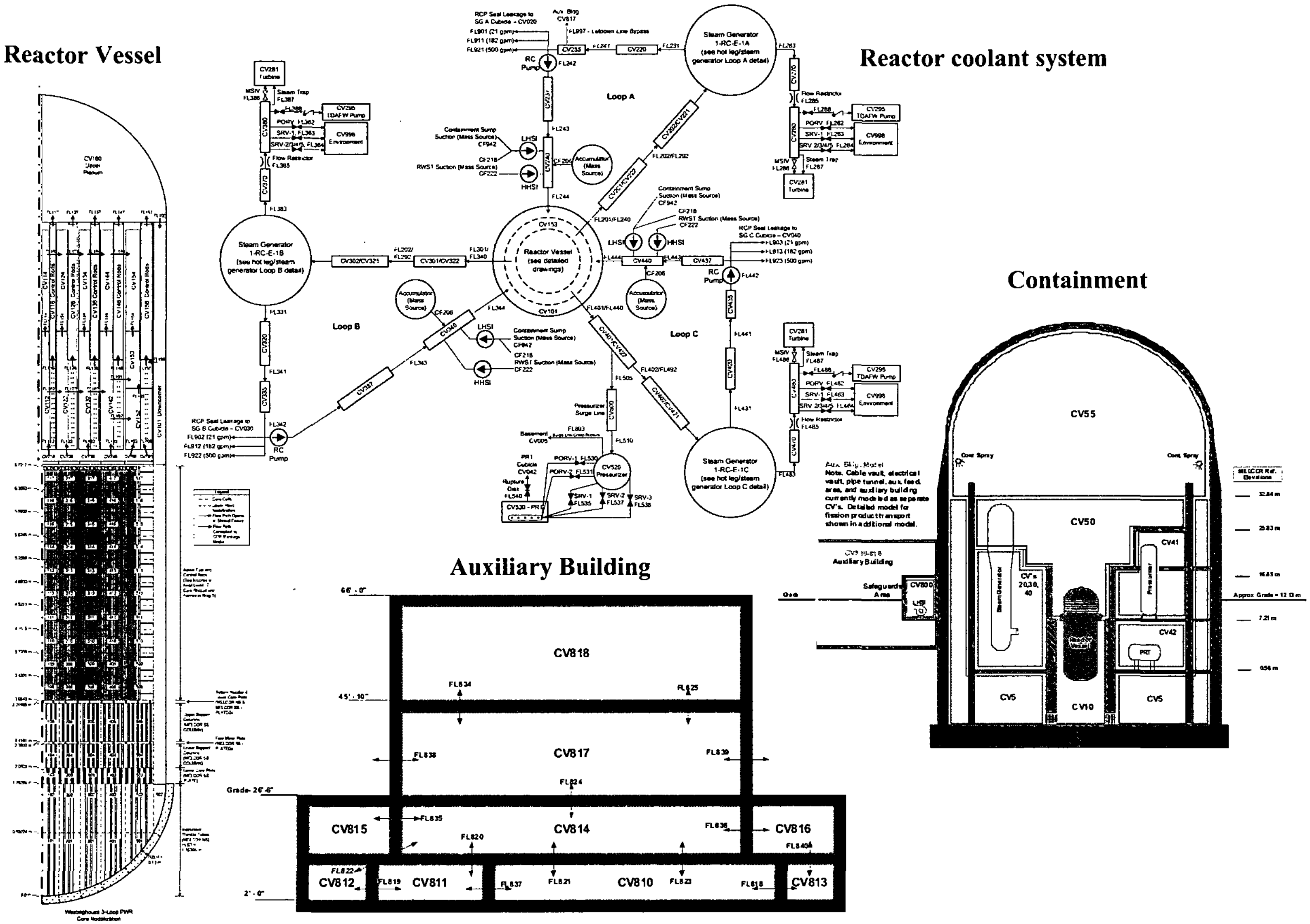

Figure 6. The Surry MELCOR Nodalization.

4.3.2 Best Modeling Practices

The accident progression analysts developed a list of key uncertain phenomena that can have a significant effect on the progression of the accident. Each issue was outlined, and a recommended modeling approach or base case values were identified in plant-specific reports for Peach Bottom (Appendix A) and Surry (Appendix B). A discussion of the specific modeling practices are described in NUREG/CR-7008 [6].

Several early containment failure modes of historical interest were excluded from the SOARCA project due their assessed low likelihood of occurrence. These include:

- Alpha mode containment failure, which is an in-vessel steam explosion during melt relocation that simultaneously fails the vessel and the containment. A group of leading experts in this field, referred to as the Steam Explosion Review Group, concluded in a position paper published by the Nuclear Energy Agency Committee on the Safety of Nuclear Installations [28] that the alpha-mode failure issue for Western-style reactor containment buildings can be considered resolved from a risk perspective, having little or no significance to the overall risk from a nuclear power plant.

- Direct containment heating (DCH), which causes containment failure in PWR containments. NRC research has shown an early failure of the PWR reactor coolant system due to high temperature natural circulation will likely depressurize the reactor coolant system prior to vessel failure. Importantly, extensive NRC testing and analyses have also shown that in the unlikely event a high-pressure vessel failure occurs, early containment failure due to DCH is very unlikely, with some variation depending on plant design [29]. In the case of Surry, it was concluded that no feasible likelihood exists of failing the containment.

- Early containment failure due to drywell liner melt-through in a wet cavity in Mark I containments (e.g., Peach Bottom). Through a detailed assessment of the issue, it was concluded that, in the presence of water, the probability of early containment failure by melt-attack of the liner is so low as to be considered physically unreasonable [30].

Two independent expert panels were assembled to review the proposed modeling approach for SOARCA analyses. This review was conducted during a public meeting sponsored by NRC on August 21–22, 2006, in Albuquerque, New Mexico. The panels examined the best modeling practices for the application of the severe nuclear reactor accident analysis code MELCOR for realistic evaluation of accident progression, source term, and offsite consequences. The panels also reviewed a set of code enhancements as well as consideration of the SOARCA project in general.

4.3.3 Radionuclide Modeling

The radionuclide modeling was updated in the Peach Bottom and Surry models to apply a more mechanistic radionuclide release model (i.e., the ORNL-Booth model) [31] based on assessments to recent radionuclide release tests. These assessments identified an alternative set of Booth diffusion parameters recommended by ORNL (ORNL-Booth) [32] that produced significantly improved release signatures for Cs and other fission product groups. Some adjustments to the scaling factors in the ORNL-Booth model were made for selected fission product groups including UO_2, Mo, and Ru to gain better comparisons with the FPT-1 data [33]. The adjusted model, referred to as "Modified ORNL-Booth," was subsequently compared to original ORNL VI fission product release experiments and to more recently performed French VERCORS tests [34], and the comparisons was as favorable or better than the original CORSOR-M MELCOR default release model. These modified ORNL-Booth parameters were implemented into the MELCOR code as new defaults for the SOARCA project.

Although significant improvements in release behavior were obtained for the analysis of the FPT-1 test with the ORNL-Booth parameters, some additional modification to the MELCOR release model was pursued. Evidence from the Phebus experiments increasingly indicates that the dominant chemical form of released Cs is that of Cs_2MoO_4. This is based on deposition patterns in the Phebus experiment where Cs is judged to be in aerosol form at 700C, which explains deposits in the hot upper plenum of the Phebus test section and deposition patterns in the cooler steam generator tubes. In recognition of response, a Cs_2MoO_4 radionuclide class was defined with the vapor pressure Cs_2MoO_4 and the release coefficients developed for Cs. The Mo vapor pressure is so exceedingly low that the net release is limited by the vapor pressure transport term. Because there is significantly more Mo than Cs in the radionuclide inventory, only a portion of the Mo was added to the new Cs_2MoO_4 radionuclide class.

The radionuclide input was reconfigured to (1) represent the dominant form of Cs as Cs_2MoO_4, (2) represent the dominant form of I as CsI, and (3) represent the gap inventories consistent with the NUREG-1465 recommendations [14]. The MELCOR radionuclide transport, deposition, condensation/evaporation, and scrubbing models were all activated. The model for chemisorption of Cs to stainless steel was activated. In addition, the hygroscopic coupling between the steam/fog condensation/evaporation thermal-hydraulic solutions to the airborne aerosol size and mass was also activated [31].

4.3.4 Radionuclide Inventory

One important input to MELCOR is the initial mass of the radionuclides in the fuel and their associated decay heat [31]. These values are important to the timing of initial core damage and the location and concentration of the radionuclides in the fuel. The radioisotopes in a nuclear reactor come from three primary sources: (1) "fission products," which are the result of fissions in either fissile or fissionable material in the reactor core; (2) actinides, which are the product of neutron capture in the initial heavy metal isotopes in the fuel; and (3) radioactive decay of these fission products and actinides. Integrated computer models such as the TRITON sequence in SCALE exist to capture all of these interrelated physical processes, but they are intended primarily as reactor physics tools [35]. As such, their standard output does not provide the type of information needed for SOARCA. Therefore, this report describes a method for deriving the needed information. It is important to note upfront that no changes to the physics codes were needed. The method described here merely extracts additional output from the TRITON sequence and combines it in a way that makes it useful for the SOARCA project.

4.3.4.1 Methods

Reactor physics codes implicitly account for both of the physical parameters of interest for SOARCA (i.e., decay heat power and radionuclide inventories) but they do not provide a mechanism to easily extract and combine these results. This section will describe the tools used to calculate the radioisotopic inventory and a new code developed to properly combine these results for use in the SOARCA calculations. The results were combined in a manner so as to capture actual plant operating data.

The TRITON sequence from SCALE 5.1 was used to develop input data for MELCOR. TRITON provides the capability to perform detailed 2-dimensional calculations of reactor fuel including the ability to deplete fuel to a user-defined level of accuracy. TRITON accurately models curvilinear surfaces such as cylindrical fuel rods and allows the fuel to be burned down to the sub pin-cell level. There is no requirement to perform any homogenization of the 2-dimensional geometry. TRITON allows for accurate depletion of highly self-shielded fuel such as poison pins. For more information, refer to the SCALE documentation [36].

The BLEND3 code was developed from previous work performed by ORNL and its capabilities were extended for this study. BLEND3 uses the reactor-specific fuel loading from three different cycles, the nodal exposure, and the assembly-specific power data from the licensee to derive node-averaged radioisotopic inventories. TRITON uses generic fuel assembly data and ties it to specific reactor operating conditions. Then, BLEND3 performs the following tasks. First, for a given node, BLEND3 identifies which specific power ORIGEN output files are assigned to the specified input power. Second, for three different cycles of fuel, BLEND3 interpolates a radioisotopic inventory from the relevant ORIGEN output files. Finally, using the input volume fractions for the three different cycles of fuel, BLEND3 creates a new, volumetrically averaged ORIGEN output file for the node for the specified input conditions.

The PRISM module from SCALE 5.1 was then used to drive ORIGEN decay calculations using the newly created averaged ORIGEN output files as input. PRISM is a SCALE utility module that allows the user to automate the execution of a series of SCALE calculations.

4.3.4.2 Peach Bottom Model

The Peach Bottom model is based on the Global Nuclear Fuel (GNF) 10x10 (GE-14C) fuel assembly. The GNF 10x10 is representative of a limiting fuel type actually being used in commercial BWRs. Figure 7 illustrates the GNF 10x10 model.

Twenty-seven different TRITON runs were performed to model three different cycles of fuel at nine different specific power histories. The specific power histories ranged from 2 MW/MTU to 45 MW/MTU to cover all expected BWR operational conditions. For times before the cycle of interest, an average specific power of 25.5 MW/MTU was used. For example, for second cycle fuel, the fuel was burned for its first cycle using 25.5 MW/MTU, allowed to decay for an assumed 30-day refueling outage, and then 9 different TRITON calculations were performed with specific powers ranging from 2 to 45 MW/MTU). The BLEND3 code was then applied to each of the 50 nodes in the MELCOR model using the average specific powers and volume fractions. Once new libraries for each of the 50 nodes in the model were generated, the final step in the procedure was to deplete each node for 48 hours. The decay heats, masses, and specific activities as a function of time were processed and applied as input data to MELCOR to define decay heat and the radionuclide inventory. For the SOARCA application, in keeping with the intent of using best-estimate approaches, the Peach Bottom fuel analysis of decay power and radionuclide inventories was based on the assumption that the accident occurs at a point midway in a fuel cycle.

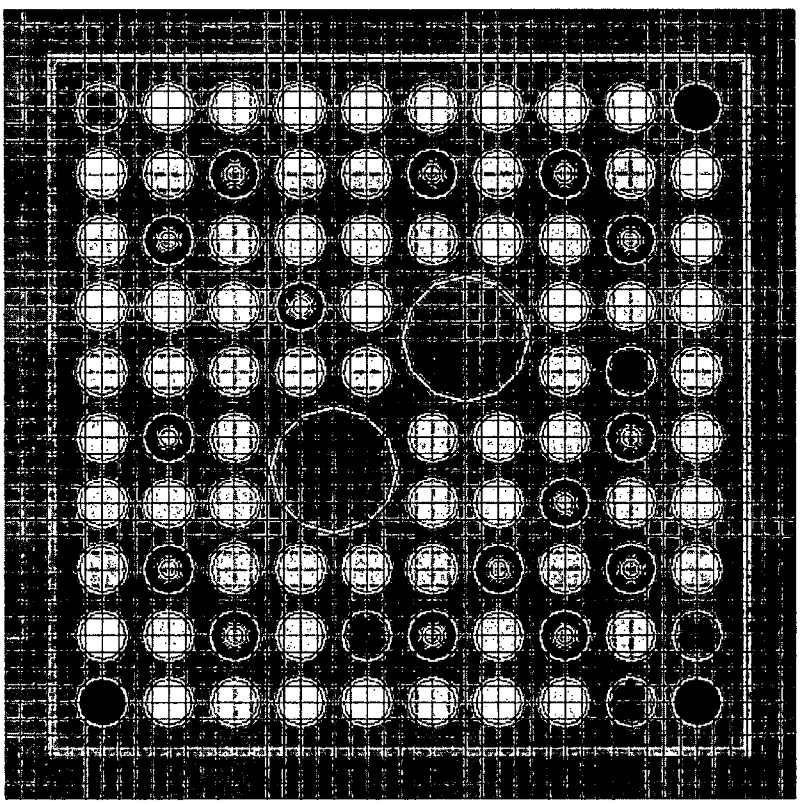

Figure 7. Schematic of Modeling Detail for BWR GNF 10x10 Assembly.

4.3.4.3 Surry Model

Previously, detailed input was developed for Surry in a separate NRC program on the source term from high-burnup uranium fuel at the end of the fuel cycle. This previous study used the same methodology as the Peach Bottom model (Section 4.3.4.2). The actual mid-cycle decay power is lower.

4.3.4.4 Evaluation of the Results

Very few measurements of decay heat are in existence, and those that do exist are not directly relevant to this study. Therefore, the discussion of the decay heat predictions will be limited to a comparison to previously published work. The best-known source of decay heat predictions is summarized in Regulatory Guide (RG) 3.54, and results from the guide will be used to assess the predictions in the current study [37]. Decay heat for two decay times will be used as a check on the consistency of the results presented in this study. By interpolation of tables in RG 3.54 for a specific power of 27 MW/MTU, decay powers at 1 and 2 years following shutdown of 9.3 W/kgU and 5.1 W/kgU, respectively, are calculated. Using the results from the Peach Bottom calculations, the corresponding decay powers are 8.92 W/kgU and 4.734 W/kgU. The maximum difference between results is about 8 percent, which is considered acceptable given the best-estimate nature of the SOARCA study compared to the methods used to generate the tables in RG 3.54.

A quantitative discussion of the radioisotopic predictions presented in this study would be of limited use given the cycle-specific nature of this work. However, it is of benefit to discuss the relevant SCALE assessment. Specifically, the TRITON module has been assessed by M. D. DeHart and S. M. Bowman [38], S. M. Bowman and D. F. Gill [39], and Germina Ilas and Ian C. Gauld [40]. These assessment reports use data from Calvert Cliffs, Obrigheim, San Onofre, and Trino Vercelles PWRs. The third report summarized comparisons to decay heat measurements from four different BWR assemblies.

5.0 OFFSITE CONSEQUENCE ANALYSES

MACCS2 [41] has been developed by Sandia National Laboratories (SNL) over the past 2 decades. It is a consequence analysis code for evaluating the impacts of atmospheric releases of radioactive aerosols and vapors on human health and on the environment. It includes all of the relevant dose pathways: cloudshine, inhalation, groundshine, and ingestion. Because it is primarily a probabilistic risk assessment tool, it accounts for the uncertainty in weather that is inherent to an accident that could occur at any point in the future.

Over the past decade, Sandia National Laboratories has developed WinMACCS for the U.S. Nuclear Regulatory Commission (NRC). WinMACCS is a user-friendly front end to MACCS2 that facilitates selection of input parameters and sampling of uncertain input parameters and performs post-processing of results. WinMACCS version 3.5.0 is used in the final SOARCA calculations. MACCS2 is still used as the computational engine underlying WinMACCS.

Version 2.4.0.5 of the MACCS2 code was used for the SOARCA offsite consequence predictions. This version includes a number of improvements to the original MACCS2 code, which can be categorized as follows:

- Atmospheric transport and dispersion modeling improvements (e.g., morning and afternoon mixing heights, alternative Briggs plume rise model, and alternative long-range plume spreading model).
- Capability to describe wind directions in 64 compass directions (instead of 16).
- Increased limits on several input parameters (e.g., a limit of 200 plume segments instead of the old limit of 4).
- Up to 20 emergency-phase cohorts (instead of the original limit of 3) to describe variations in emergency response by segments of the population.
- Enhancements in treatment of evacuation speed and direction to better reflect the spatial and temporal response of individual cohorts.
- Capability to run on a cluster of computers instead of an individual processor.
- Addition of several options for latent cancer fatality dose response (i.e., user-input yearly truncation value, user-input yearly truncation value with a lifetime restriction, and a piece-wise linear model; the original linear no-threshold model remains available).

A peer review of the MACCS2 code and modeling choices was conducted prior to the commencement of specific work on Surry and Peach Bottom. This peer review drove much of the development that has been undertaken specifically to support the SOARCA work [42].

Specific aspects of the consequence modeling in SOARCA that depart from previous studies such as NURG-1150 [2] are described in the subsequent subsections.

5.1 Weather Sampling

The weather-sampling strategy adopted for SOARCA uses the nonuniform weather-binning approach in MACCS2. This approach, which allows the user to specify a different number of random samples to be chosen from each bin, has been available since MACCS2 was first released [41] but was not commonly used in the past. Weather binning is an approach used in MACCS2 to categorize similar sets of weather data based on wind speed, stability class, and the occurrence of precipitation. This sampling strategy was chosen as a means of improving the statistical representation of the weather. This point is discussed further in the subsequent paragraphs.

The weather bins are defined in a standard way that has origins in the NUREG-1150 [2] analyses. A set of 16 weather bins differentiates stability classes and wind speeds. An additional 20 weather bins include all weather trials in which rain occurs before the initial plume segment travels a distance of 32 km (20 mi). The bins differentiate rain intensity and the distance the plume travels before rain begins. The parameters used to define the rain bins are the same as those used in NUREG-1150 and documented in the MACCS2 User's Manual [41]. Because the strategy provides for weighting of the particular trials chosen (based on the number of samples in the bin and the number of samples requested), the particular choice of a binning strategy is not important (provided a sufficient number of samples is chosen). However, a well-chosen binning strategy will reduce the number of samples required for adequate statistical precision. The binning strategy used in NUREG-1150 and for SOARCA ensures that the rain cases, which are only a fraction of the full year's data, are adequately sampled with the weighting factors used in the code accounting for the prevalence in the weather record.

For the nonuniform weather sampling strategy approach for SOARCA, the number of trials selected from each bin is the maximum of 12 trials and 10 percent of the number of trials in the bin. Some bins contain fewer than 12 trials. In those cases, all of the trials within the bin are used for sampling. This strategy results in roughly 1,000 weather trials for both Peach Bottom and Surry.

Previous calculations, such as NUREG-1150, used about 125 weather trials including an additional strategy—rotation—to account for the probability that the wind might have been blowing in a different direction when the release began. This strategy uses wind-rose data constructed from the annual weather file to determine the probability that the wind might have been in any of the compass directions. The strategy used at the time of NUREG-1150 leveraged the weather data to get 125 x 16 = 1750 results for the computational price of 125, but at a cost that the individual results are not truly independent. For the strategy chosen here, the trials are independent.

MACCS2 does not allow the rotation option to be used in concert with the network evacuation option; therefore, rotation could not be used for SOARCA. The strategy adopted for SOARCA was chosen as a compromise between obtaining adequate statistical significance and keeping central processing unit (CPU) time at a reasonable level.

5.2 Weather Data

Meteorological data used in the SOARCA project consisted of 1 year of hourly meteorological data for each site (8,760 data points per site for each meteorological parameter). This was primarily accomplished via a cooperative effort with the licensee using onsite meteorological tower observations. Each licensee provided 2 years of weather data. Hourly precipitation data were measured directly and reported by the licensees. Stability class data were derived from temperature measurements at two elevations on the site meteorological towers. The specific year of data chosen for each reactor was based on data recovery (greater than 99 percent being desirable) and proximity to the target year for SOARCA, which is 2005. Different trends (e.g., wind-rose pattern and hours of precipitation) between the years were estimated to have a relatively minor (<25 percent) effect on the final results. Specific details of the weather data are discussed in the next subsection.

For the weather record years and the particular data used in SOARCA, the recovery of data was in excess of 90 percent. The missing data were bridged over using the hourly records before and after by employing "Procedures for Substituting Values for Missing National Weather Service Meteorological Data for Use in Regulatory Air Quality Models" [43]. The meteorological data parameters were formatted for the MACCS2 (MELCOR Accident Consequence Code System, version 2) computer code.

NRC staff performed quality assurance evaluations of all meteorological data presented using the methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data" [44]. Further review was performed using computer spreadsheets. NRC staff ensured a joint data recovery rate in the 90^{th} percentile, which is in accordance with Regulatory Guide 1.23 [45] for the wind speed, wind direction, and atmospheric stability parameters. In addition, atmospheric stability was evaluated to determine if the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day). The mixing height data were retrieved from the U.S. Environmental Protection Agency's SCRAM database[4] (using years 1984-1992). Data needed for MACCS2 includes 10-meter wind speed, 10-meter wind direction in 64 compass directions, stability class (via Pasquill-Gifford scale and using representative values of 1-6 for stability classes A-F/G), hourly precipitation, and diurnal (morning and afternoon) seasonal mixing heights.

[4] EPA SCRAM Web site: http://www.epa.gov/scram001/mixingheightdata.htm

5.2.1 Summary of Weather Data

Table 2 presents a summary of the meteorological statistical data that shows the annual average ground-level wind speeds were generally low, ranging from 2.02 to 2.27 m/s at Surry and 2.12 to 2.17 m/s at Peach Bottom. The atmospheric stability frequencies were found to be consistent with expected meteorological conditions. The neutral and slightly stable conditions predominated during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day.

Figure 8 and Figure 9 show the wind direction (direction the wind blows toward) and atmospheric stability (unstable[5], neutral[6], and stable[7]) data for the years that were actually used in the consequence analyses (i.e., 2006 for Peach Bottom and 2004 for Surry). The Pasquill-Gifford stability classes were used in the MACCS2 calculations. Parsing of these classes into unstable (A-C), neutral (D), and stable (E-F) conditions for Figure 8 and Figure 9 was only used for comparisons with expected weather patterns.

Table 2. Statistical Summary of Raw Meteorological Data for SOARCA Nuclear Sites

Parameter		Peach Bottom		Surry	
		Year 2005	Year 2006	Year 2001	Year 2004†
Avg. Wind Speed (m/s)		2.17	2.12	2.02	2.27
Yearly Precipitation (hr)		588 (6.7%)	593 (6.8%)	388 (4.4%)	521 (5.9%)
Atmospheric Stability (%)	Unstable	21.43	20.56	7.09	3.94
	Neutral	63.97	62.34	69.67	77.59
	Stable	14.60	17.10	23.24	18.47
Joint Data Recovery (%)		97.53	99.25	99.58	99.24

† Year 2004, as used in the Surry meteorological analysis, is a leap year (8784 total hourly data points versus 8760 hourly data points for a regular annual period).

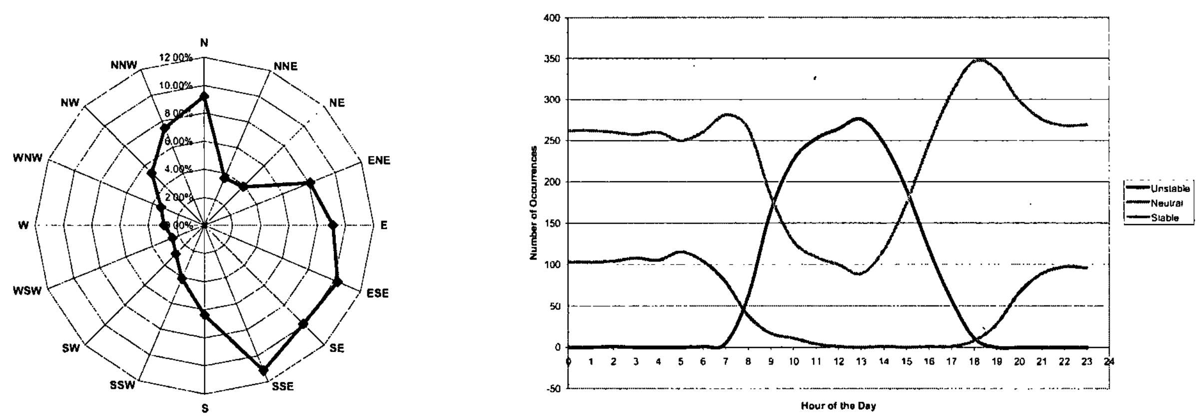

Figure 8. Peach Bottom – Year 2006 – Wind Rose and Atmospheric Stability Chart.

5 Pasquill-Gifford stability classes A, B, C
6 Pasquill-Gifford stability class D
7 Pasquill-Gifford stability class E,F, and G

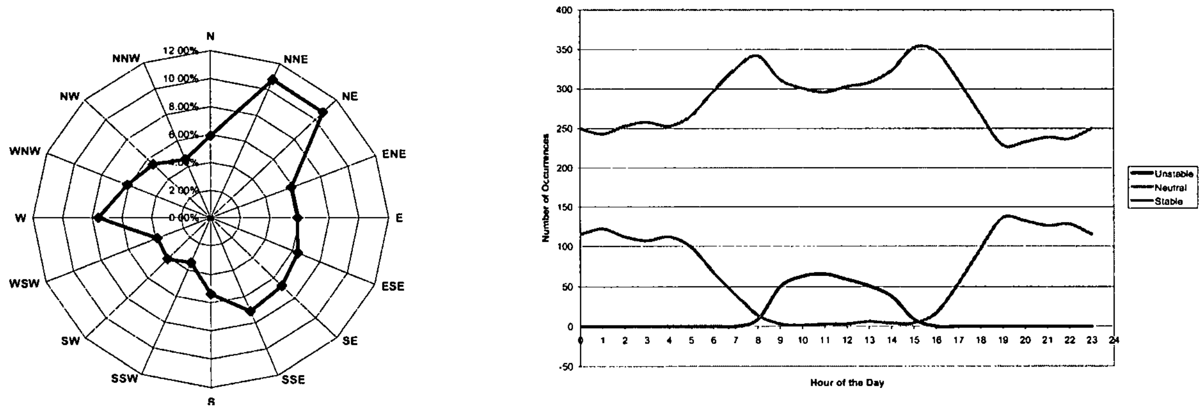

Figure 9. Surry – Year 2004 – Wind Rose and Atmospheric Stability Chart.

5.3 Emergency Response Modeling

An objective of the SOARCA project was to model emergency response in a realistic and practical manner using site-specific emergency planning information. The analysis included modeling of the timing of onsite and offsite decisions and implementation of protective actions applied to multiple population segments (called cohorts). Advancements in consequence modeling—specifically the development of WinMACCS—facilitated detailed integration of protective actions into consequence analysis providing an evolutionary advancement over previous studies. WinMACCS allows temporal and spatial elements of sheltering and evacuation to be modeled.

Emergency response programs for nuclear power plants (NPPs) are designed to protect public health and safety in the unlikely event of a radiological accident. These emergency response programs are developed, tested, and evaluated and are in place as an element of defense in depth.

Detailed emergency response planning is in place within the 10-mile Emergency Planning Zone (EPZ) with consideration that such planning provides a substantial base for expansion of response efforts in the event that this proves necessary [46]. Site-specific information was obtained from the Offsite Response Organizations (OROs) to support development of timelines by which protective actions would most likely be implemented including early actions such as evacuation of schools following declaration of a site area emergency. Integrating the response plan elements and a best estimate of the protective actions that would be implemented by each cohort was undertaken for the SOARCA project to improve the overall fidelity of the consequence analyses.

Figure 10 shows the 10- and 20-mile radial distances around the Peach Bottom site.

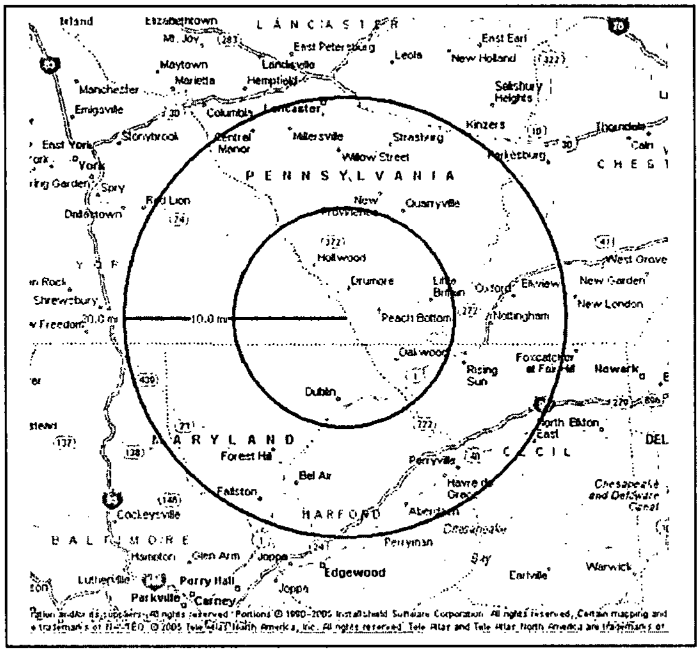

Figure 10. 10- and 20-Mile Radial Distances around the Peach Bottom Site.

The SOARCA project provided an opportunity to assess response by populations within the 10-mile EPZs and to assess possible variations of emergency response for the two sites studied. These variations include evacuation and sheltering of population groups outside the 10-mile EPZ to a distance of 20 miles from an NPP. The areas beyond the EPZ are not expected to require evacuation or shelter. If they did need to employ such actions, however, the response would be limited to areas based on plume projections. That is, it was assumed that OROs would be aware of source terms and resultant doses that could require actions beyond 10 miles and would identify the need for such actions and direct that they be implemented in an ad hoc manner. For dose calculation purposes, it was assumed that evacuees traveled to a point 30 miles from the site. This was done to be consistent with previous calculations (e.g., NUREG-1150) where evacuees had to move 10 miles beyond the evacuation zone before they disappeared from the calculations. In the past, the evacuation zone was chosen to represent the EPZ and did not consider shadow evacuation. Here, we treat shadow evacuation beyond the 10-mile EPZ out to a 20-mile radius. Thus, evacuating to a 30-mile radius results in the outermost evacuees traveling 10 miles beyond their initial location before disappearing from the calculation.

The initiating event for many of the accident scenarios considered by SOARCA is a large earthquake close to the plant site. For this event, it was assumed that severe damage would be generally localized (e.g., 30-40 km from the site). Because characterization of emergency response is based on the timing of actions by onsite and offsite response organizations to protect public health and safety (generally by instructing the public to evacuate or shelter), potential effects of an earthquake need to be considered. However, considerable uncertainty exists in characterizing the impacts of an earthquake, and the SOARCA project therefore addressed the earthquake effects in a separate calculation. A consequence analysis was performed for the accident sequences for each site, and a single seismic sensitivity analysis was performed for the more severe seismic accident postulated for the site.

The approach used for SOARCA considered the effects of a seismic event as a separate calculation, but the scope of SOARCA did not include a detailed analysis of the effects of a large-magnitude earthquake on local infrastructure. Rather, NRC performed a limited seismic analysis specific to each site. The seismic analysis indicated that bridges close to each site are unlikely to survive the earthquake and are assumed to be impassible during emergency response. Long-span bridges are those whose lengths are such that they experience multiple modes of excitation and noncoherent motion input during a seismic event. Some smaller bridges and road crossings also were assumed to fail, and some roadways were identified as failing where underlying soils could slide off into adjacent waterways. Residential and commercial structures would be damaged but generally would survive the earthquake. The local electrical grid is assumed to be out of service due to the failure of overhead power lines, switchyard equipment, or other failures. A limited backup power system is in place for the sirens at Peach Bottom, causing some sirens to be unavailable; however, backup power is available for the sirens at Surry and those sirens are assumed to be operable. Offsite response organizations would have to perform route alerting to notify the population of the need to take protective actions in areas where sirens are not functional. This consists of emergency responders driving through neighborhoods using loudspeakers or going door-to-door to notify residents of the emergency. Route alerting is a routine and effective method of informing the public [47]. Response parameters that may be affected by an earthquake (e.g., mobilization of the public, evacuation speed, shielding, etc.) were developed to reflect the potential impact.

5.3.1 Base Case Analyses of Emergency Response

For each accident sequence that resulted in a radioactive release to the environment that would invoke protective actions, a base case was developed and modeled using WinMACCS. The base case represents the protective action planning in place for EPZs [46]. Initial protective actions at Surry, for which guidance is provided in Supplement 3 to NUREG-0654/FEMA-REP-1, Rev.1 [46], would likely include evacuation of the 2-mile zone around the NPP and evacuation of a 5-mile downwind keyhole as shown in Figure 11. Pennsylvania implements a 360 degree 10-mile evacuation. For consistency in approach, the analyses include evacuation of the public residing within the 10-mile EPZ, a 20-percent shadow evacuation of the public residing in the 10- to 20-mile zone outside the EPZ [48], and sheltering of the remaining public within the 10- to 20-mile zone outside the EPZ for both sites. The population beyond 20 miles was not assumed to evacuate although this segment of the population is relocated if projected doses exceed U.S. Environmental Protection Agency (EPA) guidelines.

Population subgroups, the cohorts, were defined to provide greater fidelity in the treatment of emergency response than was possible for previous studies. For each site, six cohort groups were established. The makeup of the cohort groups varied by site depending on the population distributions and emergency management actions. As a general assumption, the accident scenario was assumed to occur during school hours, and one cohort was established for

schoolchildren within the EPZ. Other cohorts included the general public within the EPZ, general public in the 10- to 20-mile zone, special facilities within the EPZ, shadow evacuees, and a nonevacuating cohort. The nonevacuating cohort represents a small fraction (0.5 percent in this case) of the population who may choose not to evacuate when asked to do so [49].
The SOARCA project used the Evacuation Time Estimates (ETEs) provided by the licensees to derive speeds for evacuating cohorts. For NPPs, Appendix E of 10 CFR 50.47 requires that an analysis of the time required to evacuate be provided for various sectors and distances within the EPZ for transient and permanent residents. Developed by licensees to support this requirement, an ETE is a tool that provides emergency managers information on how long it may take to evacuate a portion or all of the EPZ. Using this information, emergency managers can decide if evacuation is the most appropriate protective action for a specific accident. The site-specific ETEs were used to establish the evacuation-related input parameters for MACCS2.

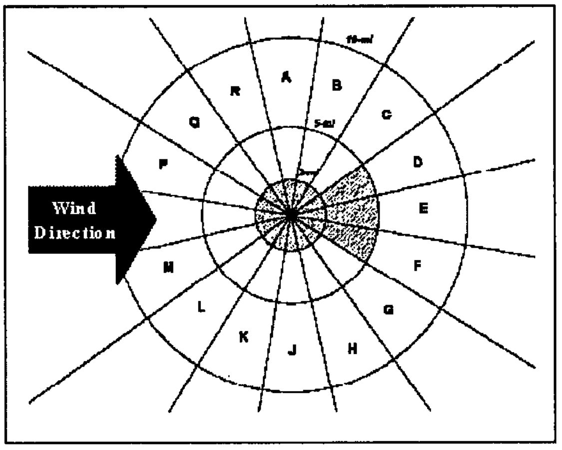

Figure 11. Standard Keyhole Evacuation.

Using WinMACCS, the emergency-planning protective-action-related parameters were integrated into the consequence modeling. WinMACCS allows for the movement of multiple cohorts and accommodates speed and direction variations for each evacuating cohort. To develop the input, the evacuation routes were reviewed to determine the likely directions that evacuees would travel. A newly developed option in WinMACCS and MACCS2 allowed modification of the speeds for grid elements where road conditions would suggest speedup or slowdown. In addition, for cases where the hourly meteorological data being used included precipitation, the speeds of all evacuating cohorts were reduced. The evacuation area was mapped onto a grid with 64 compass sectors and 15 radii that was used as the basis for the network evacuation model. The same evacuation network was used for all accident sequences at each site. Response timing and evacuation speed parameters were developed specifically for each accident sequence. For dose-modeling purposes, all evacuees were assumed to travel to a point 30 miles from the site. This distance accounts for the fact that the assembly sites are at some distance from the plant. Whether or not doses are actually received by the evacuees during any part of their travel is a complicated function of the direction of travel and the times and the directions of the plumes as they are released from the plant. Each plume disperses in a straight line in its own downwind direction.

5.3.2 Sensitivity Analyses of Emergency Response

After completion of the base-case analysis, three variations were conducted as sensitivity analyses including:

- Evacuation to a distance of 16 miles from the plant. For this analysis, complete evacuation of 16 miles around the plant was assessed. The members of the public in the 16- to 20-mile zone were assumed to shelter.

- Evacuation to a distance of 20 miles from the plant. For this analysis, an ETE was developed for the 20-mile area to provide realistic modeling parameters for the movement of the public.

- Delay in implementation of protective actions. This analysis included an assumption that there could be a 30-minute delay in the implementation of protective actions by the public. This sensitivity study assumes a delay could occur that could be due to delay in notification to offsite authorities, notification from offsite authorities to the public, receipt of the warning by the public, or other reasons. The analysis considered that cohorts take 30 minutes longer to start the implementation of protective actions than they do in the base-case analysis.

For assessment of movement of the public residing between 10 to 20 miles (shadow evacuees outside the EPZ), additional ETEs were developed for each site using the Oak Ridge Evacuation Modeling System (OREMS). OREMS was used to develop ETEs for the general public within the 10- to 20-mile zone for all sites. The level of detail in developing these ETEs was significant and included the general public and special facility population groups (e.g., hospitals, nursing homes, and prisons).

The development of the parameters for the sensitivity analyses showed that for the analyses that considered a larger evacuation area, the travel speeds were typically slower than the baseline analyses. This was because of the additional vehicle load on the roadway network. For the third sensitivity analysis where a delay in implementation was assessed, the speeds remained unchanged.

5.4 Source Term Evaluation

Source term evaluation for each of the accident sequences was performed using MELMACCS [50]. MELMACCS reads a MELCOR plot file and extracts information useful for source term definition for MACCS2. A number of user options have to be selected when using MELMACCS. The following paragraphs describe the specific choices made for SOARCA.

The first set of choices is related to the chemical groups or classes to be included in the analysis. Here, the standard set of fission product groups (i.e., the Xe, Cs, Ba, I, Te, Ru, Mo, Ce, and La groups) are all included in the analyses. A related quantity defining the burnup to be assumed when calculating the fission product inventory depends on the plant type. In an effort to provide a best-estimate fission product inventory for Peach Bottom, an ORIGEN calculation was performed for SOARCA to estimate the inventory at mid-cycle for which peak-rod burnup is estimated to be 49 MWd/kg. These data were used in MELMACCS to specify the inventory for MACCS2 and the MACCS2 input is, therefore, consistent with the MELCOR calculation. An analogous calculation was not performed for Surry; instead, a previously available fission product inventory based on the regulatory limit of burnup (65 MWd/kg for the peak fuel rod) was used. This inventory should be conservative in the sense of being overestimated, at least for most of the fission products that do not reach secular equilibrium by mid-cycle.

A set of parameters define the ground elevation (grade) in the MELCOR reference frame, the height of the building from which release occurs, and the initial plume dimensions. The MELCOR analyses used in SOARCA use reactor shutdown as the reference time, so the time of accident initiation is always set to zero in the MELMACCS input.

Aerosol deposition velocities are calculated by MELMACCS based on the geometric mean diameter of each aerosol bin as defined in the MELCOR analysis. The deposition velocities are based on expert elicitation data using the median value of the combined distribution from the experts [51]. Typical values for surface roughness and mean wind speed, 0.1 m and 2.2 m/s, respectively, are additional parameters used to determine the deposition velocities in MELMACCS. Mean wind speeds were determined from the specific weather files used in the consequence analyses. Table 3 displays the deposition velocities used in SOARCA analyses.

Table 3. Deposition Velocities Used in the SOARCA Analyses

Bin #	Median Diameter (μm)	Deposition Velocity (m/s)
1	0.15	5.35×10^{-4}
2	0.29	4.91×10^{-4}
3	0.53	6.43×10^{-4}
4	0.99	1.08×10^{-3}
5	1.8	2.12×10^{-3}
6	3.4	4.34×10^{-3}
7	6.4	8.37×10^{-3}
8	12	1.37×10^{-2}
9	22	1.70×10^{-2}
10	41	$1.70 \cdot \times 10^{-2}$

MELCOR results include the relative quantities of aerosols contained in each size bin listed in the table. MACCS2 uses this information, plus the deposition velocities in the table, to determine the rate of depletion of aerosols from the plume. Generally, the larger aerosols deposit more quickly and so are depleted more rapidly from the plume. The peak in the aerosol size distribution is usually a few micrometers, which corresponds to a deposition velocity of a few millimeters per second.

Finally, significant releases were broken up into 1-hour plume segments. MACCS2 allows plume segments to travel in only one compass direction based on weather data. More plume segments can better represent plume transport and dispersion due to possible changes in the weather (such as the wind direction) during the release. Longer plume segments were sometimes used for trivial releases such as those where the segment content is a very small fraction of the total release. Finer resolution of these releases was not necessary to maintain the fidelity of the calculation. The MELCOR analyses provided the amount of each chemical element group in each aerosol bin for each plume segment.

5.5 Site-Specific Parameters

Weather data for each site are taken from meteorological archives provided by each plant (see Section 5.2). The raw data were processed into 64 compass sectors to use the angular resolution capabilities in WinMACCS 3.4 5.0 and MACCS2 2.4.0.5.

Site files were initially created by SECPOP2000 [52] for 16 compass sectors, which is the only angular resolution supported by that code. WinMACCS was then used to interpolate these site files onto the 64 compass-sector grid that was used for the consequence analyses. The granularity of the population data for 16 compass directions is maintained for the 64 compass direction data. The SECPOP2000 population data also were scaled by a factor of 1.0533 to account for U.S. average population growth between the years 2000 and 2005.

Consequence analyses were performed using the standard approach of evaluating accidents in the following two phases:

- Emergency phase. This phase is the period of time beginning with the initiating event and continues for about 1 week. The release from the plant and plume transport through the MACCS2 grid occurs during this phase. Emergency response (i.e., evacuation and relocation of the population to reduce exposures and doses) also occurs during this phase. The length of this phase has been chosen to ensure that all plumes can exit the calculational mesh during the period because certain assumptions about doses (e.g., that all late phase doses are small enough to warrant applying the dose and dose rate effectiveness factor) could be questionable if the early phase was made too short and because the start of the late phase is the start of recovery actions that probably cannot start much earlier in time.

- Long-term phase. This phase is the period following the emergency phase and continues for 50 years. Three actions take place during the long-term phase. Land that is contaminated above the level that is allowable for habitation is decontaminated and potentially interdicted for an additional period. During this time, the land is not available for human habitation. Land that cannot be restored to habitability is condemned, in which case the residents do not return during the long-term phase. The length of the late phase has been chosen to provide a reasonable time period for exposure of the average person. For some small children, the time period is too short but for other older residents, it is too long.

Shielding factors applied to evacuation, normal activity, and sheltering for each relevant dose pathway (i.e., inhalation, deposition onto skin, cloudshine, and groundshine) were evaluated for each site based on values used in NUREG-1150 [2] and NUREG 6953, Vol. 1 [53]. A review of the discussion of shielding in the NUREG-1150 documentation suggests that the factors the authors considered were adequate for SOARCA purposes. One departure from the NUREG-1150 values is for normal activity. Each of the normal activity values was reevaluated assuming that the average person spends 19 percent of the day outdoors and 81 percent of the day indoors [46]. The value for each of the pathways was evaluated as a linear combination of 19 percent of the value for evacuation and 81 percent of the value for sheltering.

Site-specific values are used to determine long-term habitability. Most States adhere to EPA guidelines that allow a dose of 2 rem in the first year and 500 mrem per year thereafter. The EPA recommendation has traditionally been implemented in MACCS2 as 4 rem during the first 5 years (2 rem + 4 x 0.5 rem) of exposure, and that convention is adopted here. Explicit use of the EPA recommendation cannot be implemented in MACCS2 because MACCS2 only accepts one dose and one time period. Some States, like Pennsylvania, have a stricter habitability criterion (i.e., 0.5 rem/yr beginning in the first year). Thus, the habitability or return criterion is site specific and is discussed further in Appendix A and Appendix B.

Other site-specific parameters include farmland and nonfarmland values. These values also are scaled from NUREG-1150 values using the Consumer Price Index (CPI) as the basis for price escalation. A scaling factor of 1.09 was used to account for inflation between the years 2002 and 2005. Land values have an influence on the decision to decontaminate, interdict, or condemn land. If the cost of decontamination were assessed to be higher than the land value, the land is assumed to be condemned. Because the public would not be allowed to return to condemned land and, therefore, no dose would be received, the land values did have an effect on the predicted long-term health consequences.

5.6 Non-Site-Specific Parameters

Ingestion of contaminated food and water is not treated in the SOARCA analyses. The reasoning is that adequate supplies of food and water are available in the United States and can be distributed to areas affected by a reactor accident. Some farm areas would be taken out of production at least for a period of time, while other areas would be put into production to compensate and maintain a level food supply without needing to resort to consumption of contaminated food. Likewise, bottled or filtered water from uncontaminated areas would be distributed to affected areas so that no one would need to consume contaminated water.

Some States have distributed potassium iodide (KI) tablets to people who live near commercial NPPs. KI has been distributed within the EPZ at the Peach Bottom and Surry sites. The purpose of the KI is to saturate the thyroid gland with iodine so that further uptake of iodine by the thyroid is diminished. If taken at the right time, the KI can nearly eliminate doses to the thyroid gland from inhaled radioiodine. Ingestion of KI is modeled for half of the residents near plants where KI has been distributed by the State or local government. A further assumption is that most residents do not take KI at the optimal time (shortly before to immediately after plume arrival) so the efficacy is only 70 percent (i.e., the thyroid dose from inhaled radioiodine is reduced by 70 percent).

Much of the non-site-specific data used for consequence analysis in SOARCA is taken from a set of reports that document a joint NRC/Commission of the European Communities (CEC) expert elicitation study [51]. The data taken from this study include atmospheric dispersion parameters, dry deposition velocities, wet deposition parameters, and acute health-effect parameters. In all cases, the median values extracted from the elicitation study [51] are used for point-value consequence analyses in SOARCA.

In general, evacuation was modeled within a 10-mile EPZ at both sites, but the sensitivity of the offsite consequences to important variations on the 10-mile EPZ was also considered (see Section 5.3.2). For dose calculations, all evacuees are assumed to travel to a distance of 30 miles from the site. Outside of the EPZ and for the nonevacuating cohort within the EPZ, the population was assumed to relocate if the projected dose during the emergency phase exceeded a set of two upper bounds. These bounds were based on a range of dose levels published by the EPA, which is 1 to 5 rem. In SOARCA, the upper limit of this range (5 rem) was used to trigger hot spot relocation for both Surry and Peach Bottom, and the lower limit of this range (1 rem) was used to trigger normal relocation for Surry while 0.5 rem was used for Peach Bottom to be consistent with the Pennsylvania habitability criterion.

In MACCS2, hot-spot relocation is performed first and normal relocation second. The choices of times associated with normal and hot-spot relocation depended on the specific accident scenario because the first priority of emergency responders is generally to evacuate those within the EPZ. Consequently, it was assumed that hot-spot relocation would begin after evacuation was complete. The scenario-specific time for completion of the relocation includes the time for response personnel to identify the involved area, for them to notify the residents within that area that relocation is necessary, and for the residents to remove themselves from the area. Because the timing of relocation is keyed to plume arrival, there is always a period of exposure prior to initiation of relocation. The specific choices for the parameters controlling the exposure period are discussed in the appendices.

The dose conversion factors (DCFs) used in the SOARCA analyses are based on Federal Guidance Report (FGR)-13 [54]. This guidance report also recommended changes to the biological effectiveness factors (BEFs) for alpha radiation for two of the organs used to estimate latent cancer health effects to be consistent with the way the risk factors for cancers associated with those organs were evaluated. The two organs are bone marrow and breast; for these organs, the BEFs for alpha radiation were changed from the standard value of 20 to 1 and 10, respectively. Doses to these organs are used to evaluate occurrences of leukemia and breast cancer, respectively. The choice of BEFs for these tissues is dictated by [55] from EPA. Keith Eckerman [56] also recommended using dose to the pancreas as a surrogate for dose to soft tissue to estimate residual cancers. The reason for the choice of the pancreas is that it is not a tissue in which inhaled material is deposited. Because MACCS2 does not currently read the data for the pancreas from the dose conversion factor file, a workaround was created. Values of the dose coefficients for the pancreas were copied into the organ called bladder wall. Thus, residual cancers are associated with the organ called bladder wall, which actually contains data for the pancreas. The inhalation factors in FGR-13 were processed to account for a distribution of particle sizes. A mass median aerodynamic particle diameter of 1 micron was assumed with a log-normal form for the distribution and with a geometric standard deviation of about 2.5. MACCS2 does not currently allow the dose coefficient to change as the particle size distribution changes during the MACCS2 calculation. The particle size distribution might change because of changes in the emitted particle sizes for different plumes or changes as the plumes progress downwind and deposit material.

A dose and dose rate effectiveness factor (DDREF) was applied to all doses in the late phase of the offsite consequence calculation and to those doses in the early phase that were less than 20 rem to the whole body. This factor, which appears in the denominator, accounts for the fact that protracted low doses are perceived to be less effective in causing cancer than acute doses. The DDREF for all cancers except for breast was 2.0 and for the breast was 1.0.

Keith Eckerman [56] also recommended risk factors for latent health effects that come from the National Research Council's Committee on the Biological Effects of Ionizing Radiations (BEIR) V report [57] and are consistent with the modified DCF file described in the preceding paragraph. These risk factors include seven organ-specific cancers plus residual cancers that are not accounted for directly. In 2009, the National Research Council released the BEIR VII report, an additional study of the biological effects of ionizing radiation. No one-to-one correspondence exists between the cancers reported in BEIR VII compared with the earlier BEIR V report. Therefore, the dose coefficients of tissues of the body in FRG-13 may or may not be consistent with the BEIR VII cancer sites. Thus, the SOARCA staff made the decision to await EPA's review of BEIR VII and subsequent update of FGR-13 before implementing BEIR VII risk coefficients.

Decontamination parameters are based on values from NUREG-1150. Two levels of decontamination are considered just as in NUREG-1150. The cost parameters associated with decontamination are adjusted to account for inflation using the CPI. Costs associated with a reactor accident are not considered in this report; however, these parameters do affect decisions on whether contaminated areas can be restored to habitability and therefore affect predicted doses and risk of health effects.

5.7 Estimating Latent Cancer Fatality Health Effects

Experts generally agree that it is difficult to characterize cancer risk because of the low statistical precision associated with relatively small numbers of excess cases at low doses for some organs. This limits the ability to estimate trends in risk. From an epidemiological standpoint, in most if not all cases, the number of latent cancer fatalities (LCFs) attributable to radiation exposure from accidental releases from a severe accident would not be statistically detectable above the normal rate of cancer fatalities in the exposed population (i.e., the excess cancer fatalities predicted are too few to allow the detection of a statistically significant difference in the cancer fatalities expected from other causes among the same population). For example, in 2006, the World Health Organization (WHO) estimated that 16,000 European cancer deaths would be attributable to radiation released from the 1986 Chernobyl NPP accident, but these predicted numbers are small relative to the several hundred million cancer cases that are expected in Europe through 2065 from other causes. Moreover, WHO concluded that "it is unlikely that the cancer burden from the largest radiological accident to date could be detected by monitoring national cancer statistics."

New findings have been published from analyses of fractionated or chronic low-dose exposure to low, linear energy transfer radiation. In particular, these recent findings included a study of nuclear workers in 15 countries, studies of persons living in the vicinity of the Techa River in the Russian Federation who were exposed to radioactive waste discharges from the Mayak Production Association, a study of persons exposed to fallout from the Semipalatinsk nuclear test site in Kazakhstan, and studies in regions with high natural background levels of radiation. Cancer risk estimates in these studies are generally compatible with those derived from the Japanese atomic bomb data. Most recent results from analyzing these data are consistent with a linear or linear-quadratic dose-response relationship of all solid cancers together and with a linear-quadratic dose-response relationship for leukemia. A linear-quadratic form for a dose model has a dependence on the square of the dose as well as on the dose itself.

In the absence of additional information, the International Commission on Radiological Protection (ICRP), the National Academy of Sciences, and the United Nations Scientific Committee on the Effects of Atomic Radiation have each indicated that the current scientific evidence is consistent with the hypothesis that a linear, no threshold dose response relationship exists between exposure to ionizing radiation and the development of cancer in humans.

Conversely, in "Dose-effect relationships and estimation of the carcinogenic effects of low doses of ionizing radiation," March 30, 2005 [58], p. 1, the French National Academy of Medicine advocates the following:

> A linear no-threshold relationship (LNT) describes well the relation between the dose and the carcinogenic effect in this dose range (0.2 to 3 Sv) [to the whole body] where it could be tested. However, the use of this relationship to assess by extrapolation the risk of low and very low doses deserves great caution. Recent radiobiological data undermine the validity of estimations based on LNT in the range of doses lower than a few dozen mSv which leads to the questioning of the hypotheses on which LNT is implicitly based.

Although the French National Academy of Medicine raises doubts regarding the validity of using LNT to evaluate the carcinogenic risk of low doses (less than 100 millisieverts (mSv) (10 rem)) and even more so for very low doses (less than 10 mSv (1 rem)), it did not articulate what exact value should be ascribed to a dose threshold.

Ultimately, external and internal exposures to individual members of the public are converted from collective organ dose to LCFs using MACCS2. The LNT model raises the concern that the summation of trivial exposures may inappropriately attribute LCFs to individuals far from the site of the accident. Although the possibility of LCFs from very low doses cannot be ruled out, organizations such as ICRP and the Health Physics Society (HPS) consider it to be an inappropriate use of these exposures. While the National Council on Radiation Protection and Measurements (NCRP) supports the LNT model, it recommends binning exposures into ranges and considering those ranges separately. Moreover, in situations involving trivial exposures to large populations, ICRP and NCRP have noted that the most likely number of excess health effects is zero when the collective dose to such populations is equivalent to the reciprocal of the risk coefficient (about 20 person-Sv (2,000 person-rem)). Nevertheless, issues remain related to assessing public exposure, estimating offsite consequences, and communicating these assessments to the public. Several organizations such as ICRP have addressed this issue. In its most recent recommendations (ICRP Report 103, "The 2007 Recommendations of the International Commission on Radiological Protection," approved March 2007), ICRP stated the following [59]:

> Collective effective dose is an instrument for optimization, for comparing radiological technologies and protection procedures. Collective effective dose is not intended as a tool for epidemiological studies, and it is inappropriate to use it in risk projections. This is because the assumptions implicit in the calculation of collective effective dose (e.g., when applying the LNT model) conceal large biological and statistical uncertainties. Specifically, the computation of cancer deaths based on collective effective doses involving trivial exposures to large populations is not reasonable and should be avoided. Such computations based on collective effective dose were never intended, are biologically and statistically very uncertain, presuppose a number of caveats that tend not to be repeated when estimates are quoted out of context, and are an incorrect use of this protection quantity.

Although ICRP provided qualitative guidance regarding situations where collective dose should not be used, it did not provide guidance regarding when these concepts actually are, and are not, appropriate, nor did it clearly articulate the boundaries within which the calculations are valid as well as the dose ranges for which epidemiological and cellular or molecular data provide information on the health effects associated with radiation exposure. ICRP did note, however, that when ranges of exposures are large, collective dose may aggregate information inappropriately and could be misleading for selecting protective actions.

The National Academy of Sciences reported the following [57]:

> The magnitude of estimated risk for total cancer mortality or leukemia has not changed greatly from estimates in past reports such as Biological Effects of Ionizing Radiation (BEIR) and recent reports of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and ICRP. New data and analyses have reduced sampling uncertainty, but uncertainties related to estimating risk for exposure to low doses and dose rates and to transporting risks from Japanese A-bomb survivors to the U.S. population remain large.

The National Academy of Sciences goes on to conclude that "current scientific evidence is consistent with the hypothesis that there is a linear, no-threshold dose-response relationship between exposure to ionizing radiation and the development of cancer in humans."

Many groups acknowledge the uncertainties associated with estimating risk for exposure to low radiation doses. One important question that remains is what offsite health consequences are attributable to very low radiation exposure. In its most recent recommendations (ICRP Report 103) described above, ICRP warned that the computation of cancer deaths based on collective effective doses involving trivial exposures is not reasonable and should be avoided. However, the report did not explicitly provide a quantitative range for which exposures should not be considered. However, in ICRP Report 104, "Scope of Radiological Protection Control Measures" [60], ICRP concludes that the radiation dose that is of no significance to individuals should be in the range of 20–100 microsieverts (μSv) (2–10 millirem [mrem]) per year whole body dose. The International Atomic Energy Agency (IAEA) has stated that an individual dose is likely to be regarded as trivial if it is of the order of some several millirem per year.

Alternatively, HPS developed a position paper, "Radiation Risk in Perspective," revised August 2004 [61], to specifically address quantitative estimation of health risks. This position paper concludes that quantitative estimates of risk should be limited to individuals receiving a whole body dose greater than 0.05 Sv (5 rem) in 1 year or a lifetime dose greater than 0.1 Sv (10 rem) in addition to natural background radiation. HPS also concluded that risk estimates should not be conducted below these doses. The position paper further states that low dose expressions of risk should only be qualitative, discuss a range of possible outcomes, and emphasize the inability to detect any increased health detriment.

The LNT model provides a viewpoint that is consistent with the NRC regulatory approach, and past analyses using the MACCS2 code have assumed an LNT dose response model. In addition, some of these past analyses (e.g., NUREG-1150) calculated LCFs to 1,000 miles with forced deposition to account for all non-inert radionuclides in the dose calculation. NRC is neither changing nor contemplating changing radiation protection standards and policy as a result of an approach taken in the SOARCA study to characterize offsite health consequences for low-probability events. On the other hand, NRC can use different approaches for different applications. Therefore, the SOARCA analyses consider a range of dose truncation values ranging from LNT on one hand to the Health Physics Society recommendation (5 rem/yr and 10 rem lifetime) on the other hand.

Two intermediate dose-truncation levels are also considered. One is the 10 mrem/yr dose truncation value suggested in ICRP Report 104; the other is U.S.-average background radiation of 620 mrem/yr. Results for these four dose-truncation levels are reported without bias for each of the accident scenarios considered in the SOARCA study. For example, SOARCA is intended to be a best-estimate study while many of the licensing regulations are based on prescriptive assumptions such as a 2-hour dose at a site boundary.

5.8 Risk Metrics Reported

The statistic that is chosen to convey the likelihood of LCFs resulting from an accident at an NPP is the mean, population-weighted individual risk. This value is more meaningful than the predicted number of LCFs in the sense that it is easier to compare with cancer fatality rates for a population due to other causes. Individual risks can be presented as conditional risks (i.e., as if the accident had taken place) or as unconditional risks (i.e., accounting for the likelihood of the accident occurring per year of reactor operation). The latter definition of risk is the more common and useful one because it conveys the full meaning of risk, which is probability (or frequency) times consequence. Finally, the individual risk within 10 miles can be compared directly with results from NUREG-1150.

The term "population-weighted" in the preceding paragraph carries the meaning of the effect of population distribution along with wind rose probabilities on the predicted risk. This statistic is simply the number of predicted fatalities divided by the population within a specified region. The use of the word "mean" is intended to convey that the results are weighted averages over the annual weather trials used in the analysis. The work presented in this report only considers uncertainty in the weather. Subsequent work is intended to explore the effect of uncertainties in other input parameters on the predictions.

Mean, population-weighted individual risks are presented from 0 to 50 miles. The 0 to 10 mile range represents the population within the EPZ. The range from 0 to 50 miles is generally used in severe accident mitigation alternative and severe accident mitigation design alternative analyses. Risks to populations beyond 50 miles diminish with increasing distance, so the risk within 50 miles represents an upper bound for risk to populations at larger distances.

In addition to an average over weather and over a region, the word "mean" in the reported risk metrics also carries the concept of an average individual. In the analyses reported here, the population is represented as set of cohorts. Each cohort behaves somewhat differently. For example, school children are assumed to evacuate earlier than the general public. Because of this, one cohort may have a different risk than another. The risks presented in this report are averaged over all individuals who live within a geographic area. Thus, the risk metric chosen for this work—the mean, population-weighted individual risk—represents an averaging process at three different levels: weather, space, and variations in individual behavior.

6.0 RESULTS AND CONCLUSIONS

To assess the benefits of the various mitigative measures and to provide a basis for comparison to past analyses of unmitigated severe accident scenarios, the SOARCA project analyzed the selected scenarios assuming 10 CFR 50.54(hh) procedures were successful (mitigated) as well assuming they were unsuccessful (unmitigated). The following sections summarize the results of the Peach Bottom and Surry unmitigated accident progression and offsite consequence analyses. The results presented here focus on the release of radionuclides to the environment and the associated offsite health consequences. Greater detail regarding these results and the results for mitigated scenarios is provided in the appendices to this NUREG.

6.1 Accident Progression and Radionuclide Release

An important result of the MELCOR accident progression analyses was the insight that accident progression in severe accidents proceeds much more slowly than earlier treatments indicated. The reasons for this are principally twofold: (1) research and development of better phenomenological modeling has produced a much more protracted and delayed core degradation transient with substantial delays of reactor vessel failure and (2) all aspects of accident scenarios receive more realistic treatment, which includes more complete modeling of plant systems and often yields delays in core damage and radiological release. In general, bounding approaches have been used in past simplified treatments using qualitative logical models. In SOARCA, where specific self-consistent scenarios are analyzed in an integral fashion using MELCOR, the result is that accident conditions or attributes that contribute to a more severe response in one area may produce an ameliorating effect in another area.

In the most likely accident considered in SOARCA (assuming no 10 CFR 50.54(hh) mitigation)—the long-term station blackout (SBO)—core damage was delayed for 10 to 16 hours and reactor vessel failure was delayed for about 20 hours. About 20 hours (boiling-water reactor [BWR]) or 45 hours (pressurized-water reactor [PWR]) were available before the onset of offsite radiological release due to containment failure. In the most widely referenced siting study scenario (identified as the SST1 release), it was assumed that a major release occurs in 1½ hours. Even in the case of the unmitigated short-term SBO (where core damage may begin in 1 to 3 hours), reactor vessel failure is delayed for roughly 8 hours allowing time for restoration of cooling and prevention of vessel failure. In these cases, containment failure and radiological release is delayed for 8 hours (BWR) or 24 hours (PWR). The two bypass events analyzed exhibit divergent behavior. Substantial delays occur for the interfacing systems loss-of-coolant accident (LOCA). Core damage was delayed for 9 hours and reactor vessel failure was delayed for 15 hours. Radiological releases commenced after 10 hours. The thermally induced steam generator tube rupture proceeds most rapidly of all the scenarios analyzed, with the start of core damage after 3 hours and reactor vessel failure in 7.5 hours. Radiological releases to the environment occur after 3.5 hours. However, the radiological release for the thermally induced steam generator tube rupture is shown by analyses to be substantially reduced. Table 4 and Table 5 provide key accident progression timing results for SOARCA scenarios. Table 4 shows the same times for lower head failure and start of the release to the environment because drywell shell melt-through occurs about 15 minutes after lower head failure.

Table 4. Peach Bottom Accident Progression Timing Results

Scenario	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)
Long-term SBO	10	20	20
Short-term SBO	1	8	8
Short-term SBO with RCIC Blackstart	5	13.2	13.4

Table 5. Surry Accident Progression Timing Results

Scenario	Time to start of core damage (hours)	Time to lower head failure (hours)	Time to start of release to environment (hours)
Long-term SBO	16	21	45
Short-term SBO	3	7	25
Thermally induced steam generator tube rupture	3	7.5	3.5
Interfacing systems LOCA	9	15	10

The SOARCA study also demonstrated that the magnitude of the radionuclide release is likely to be much smaller than the SST1 source term again as a result of (1) extensive research and improved modeling and (2) integrated and more complete plant simulation. Some releases of important radionuclides such as iodine and cesium are predicted to be about 10 percent but are more generally in the range of 0.5 to 2 percent. By contrast, the SST1 source term assumed an iodine release of 45 percent and a cesium release of 67 percent. Figure 12 and Figure 13 provide the radionuclide release results for iodine and cesium.

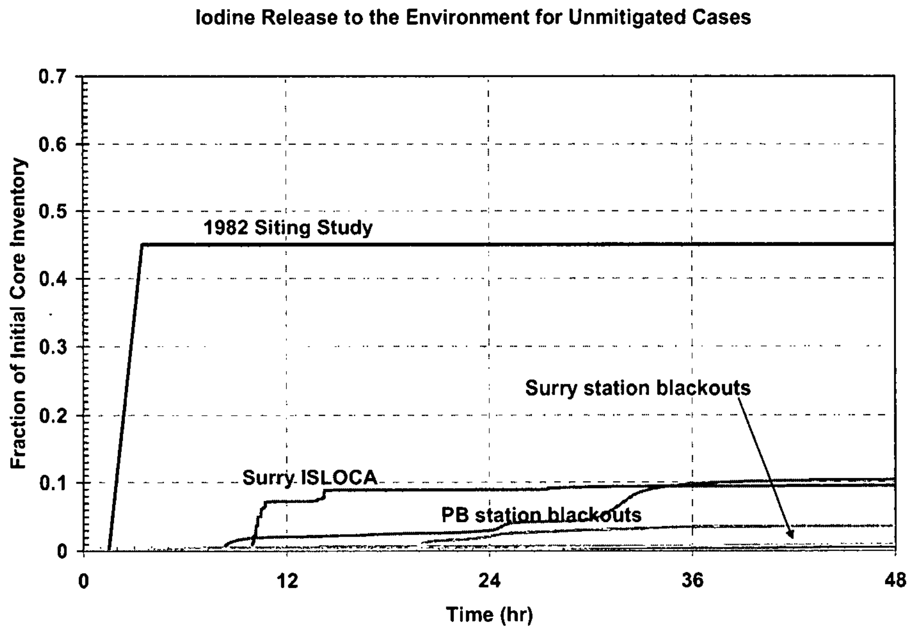

Figure 12. Iodine Releases to the Environment for SOARCA Unmitigated Scenarios.

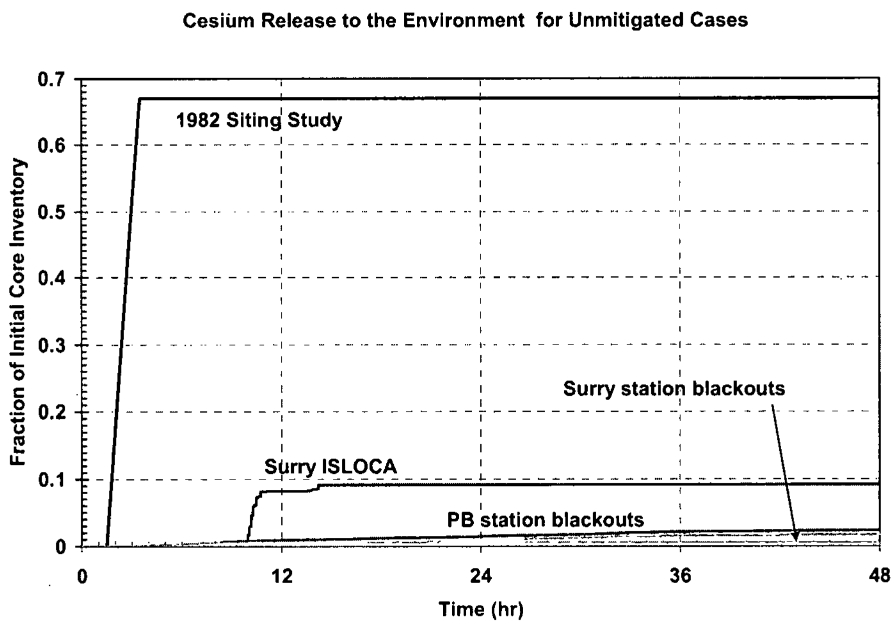

Figure 13. Cesium Releases to the Environment for SOARCA Unmitigated Scenarios.

Sequences involving large early releases have influenced the results of past probabilistic risk assessments (PRAs) and consequence studies. For example, the siting study results were controlled by an internally initiated event with a large early release that was assigned a representative frequency of 1×10^{-5}/year. However, in the SOARCA study, no sequences with a frequency above 1×10^{-7}/year resulted in a large early release even considering external events and unsuccessful mitigation. This is a result of research conducted over the last 2 decades that has shown that phenomena earlier believed to lead to a large early release are of extremely low probability or physically unfeasible. This research was focused on phenomena that have been previously assumed to be prime contributors to severe accident risk including direct containment heating and alpha mode failure.

The PWR SBO with a thermally induced steam generator tube rupture has been believed to result in a large, relatively early release potentially leading to higher offsite consequences. However, a MELCOR analysis performed for SOARCA showed that the release was small owing to thermally induced failures of other reactor coolant system components after the tube rupture. Also, the release was somewhat delayed; for the short-term SBO where loss of injection occurred at the start of the accident, the tube rupture and release began about 3.5 hours into the event. Moreover, core damage, tube rupture, and radiological release could be delayed for many hours if auxiliary feedwater were available even for a relatively short time period.

6.2 Offsite Radiological Consequences

The result of the accident progression and source-term analysis is that releases are delayed and of a diminished magnitude. Because of this and the realistic simulation of emergency response, essentially no early fatalities were predicted as close-in populations were evacuated before or shortly after plume arrival.

Latent health effects calculated using any of the dose-response models (in combination with the frequency of release) referenced in this study are small in comparison to the NRC Safety Goal. Much of the latent cancer risk was in fact derived from the relatively small doses received by populations returning to their homes in accordance with emergency planning guidelines. Because much of the health risk is due to the eventual return of the population, it is therefore controllable; however, there are implied economic costs. For example, for the Peach Bottom long-term SBO, for individuals living within the emergency planning zone (EPZ), 99 percent of the latent cancer risk derives from the long-term dose received by the population returning to their homes and being exposed to small radiation doses. Similarly, about 70 percent of the latent cancer risk to individuals within 50 miles is from returning home. The percentage is larger for the EPZ, due to its evacuation prior to the start of the release. Here, the prediction of latent cancer risk, though very small, is strongly influenced by the relationship between low-dose health effects modeling and criteria for allowing return of populations.

Estimates of conditional (i.e., assuming the accident has occurred) individual latent cancer risk range from roughly 10^{-3} to 10^{-4}, using the linear no-threshold relationship (LNT) dose response model (other dose models result in lower or much lower conditional risk). If one also accounts for the probability of the severe accident itself, without successful mitigation (denoted as the unconditional risk below), the risk to an individual for an important severe accident scenario is on the order of 10^{-9} to 10^{-10} per reactor year. These risk estimates are a million times smaller than the U.S. average risk of a cancer fatality of $2x10^{-3}$ per year. Table 6 and Table 7 provide the risk estimates for individual SOARCA scenarios without successful mitigation using the LNT dose response model. The risk estimates are based on an assumed truncation of the release at 48 hours (72 hours for the Surry long-term SBO) as a result of continually escalating mitigation actions. The core damage frequencies shown in Table 6 and Table 7 assume the probability of 10 CFR 50.54(hh) mitigation and other mitigating efforts through implementation of Severe Accident Management Guidelines and emergency operating procedures is zero.

Table 6. Peach Bottom Results for Scenarios without Successful Mitigation and Assuming LNT Dose Response Model

Scenario	**Core damage frequency (per reactor-year)**	**Conditional risk of latent cancer fatality for an individual located within 10 miles**	**Unconditional risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)**
Long-term SBO	$3x10^{-6}$	$2x10^{-4}$	$6x10^{-10}$
Short-term SBO	$3x10^{-7}$	$2x10^{-4}$	$7x10^{-11}$
Short-term SBO with RCIC Blackstart	$3x10^{-7}$	$2x10^{-4}$	$7x10^{-11}$

Table 7. Surry Results for Scenarios Without Successful Mitigation and Assuming LNT Dose Response Model

Scenario	Core damage frequency (per reactor-year)	Conditional risk of latent cancer fatality for an individual located within 10 miles	Unconditional risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)
Long-term SBO	$2x10^{-5}$	$5x10^{-5}$	$7x10^{-10}$
Short-term SBO	$2x10^{-6}$	$9x10^{-5}$	$1x10^{-10}$
Thermally induced steam generator tube rupture	$4x10^{-7}$	$3x10^{-4}$	$1x10^{-10}$
Interfacing systems LOCA	$3x10^{-8}$	$8x10^{-4}$	$2x10^{-11}$

To provide additional information on the potential range of health consequences, the SOARCA project has developed latent cancer risk estimates assuming the LNT (endorsed by NCRP) and a range of truncation doses below which the cancer risk is not quantified. Dose truncation values used for SOARCA included 10 mrem/year (representing ICRP), 620 mrem/year (representative background radiation), and 5 rem/year with a 10 rem lifetime cap (endorsed by the Health Physics Society [HPS]). Table 8 and Table 9 show the results of sensitivity calculations for dose truncation values for background and the HPS position compared with LNT results. Using these truncation values makes the already small latent cancer fatality risk estimates even smaller and, in some cases, by orders of magnitude.

For Peach Bottom scenarios, the background results in **Table 8** are the same as the HPS results because both truncation values exceed the plant-specific population return criterion of 0.5 rem/year. For Surry scenarios except interfacing system LOCA (ISLOCA), the background results in Table 8 differ from the HPS results because the background truncation value clearly falls below the plant-specific population return criterion of 4 rem over 5 years, which is intended to represent EPA's (adopted in Virginia) criterion of 2 rem in the first year and 0.5 rem/year in subsequent years; however, the HPS truncation value does not. The ISLOCA results are the same to one significant digit within a radius of 10 miles for both truncation values because most of the emergency-phase doses exceed both of these criteria while, on the other hand, long-term doses make an insignificant contribution to the overall doses. Using the 10 mrem/year truncation value made a relatively small change in the latent cancer risks compared with the LNT model and, therefore, these results were not included in **Table 8** and Table 9. The results in **Table 8** and Table 9 assume the release is truncated at 48 hours (72 hours for Surry long-term SBO) and the probability of 10 CFR 50.54(hh) mitigation is zero.

SOARCA analysis included predictions of individual latent cancer fatality risk for several distance intervals including 0 to 10 miles and 0 to 50 miles. The analysis indicated that individual latent cancer risk estimates generally decrease with increasing distance in large part due to plume dispersion and fission product deposition closer to the site.

Table 8. Peach Bottom Results for Scenarios without Successful Mitigation for LNT and Alternative Dose Response Models

Scenario	Unconditional risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)		
	Linear No Threshold	Background	Health Physics Society
Long-term SBO	$6x10^{-10}$	$5x10^{-12}$	$5x10^{-12}$
Short-term SBO	$7x10^{-11}$	$3x10^{-12}$	$3x10^{-12}$
Short-term SBO with RCIC Blackstart	$7x10^{-11}$	$3x10^{-12}$	$3x10^{-12}$

Table 9. Surry Results for Scenarios without Successful Mitigation for LNT and Alternative Dose Response Models

Scenario	Unconditional risk of latent cancer fatality for an individual located within 10 miles (per reactor-year)		
	Linear No Threshold	Background	Health Physics Society
Long-term SBO	$7x10^{-10}$	$6x10^{-12}$	$2x10^{-14}$
Short-term SBO	$1x10^{-10}$	$5x10^{-12}$	$2x10^{-14}$
Thermally induced steam generator tube rupture	$1x10^{-10}$	$3x10^{-11}$	$5x10^{-12}$
Interfacing systems LOCA	$2x10^{-11}$	$8x10^{-12}$	$8x10^{-12}$

Because the SBO scenarios were seismically induced, several sensitivity studies were also performed to evaluate the potential impact of the seismic event on the evacuation. Although road network infrastructure may be damaged during an earthquake, resulting in reduced evacuation speeds, other effects such as wider deployment of emergency responders and a larger shadow evacuation may improve evacuation timing. The sensitivity studies conducted for both Surry and Peach Bottom indicated changes to the evacuation resulting from the earthquake would change the latent cancer fatalities by less than 10 percent and may actually cause the consequences from radionuclide release to decrease as in the case of the Peach Bottom plant.

6.3 Comparison to NUREG/CR-2239 (the 1982 Siting Study)

The SOARCA offsite early-fatality risk estimates are dramatically smaller than reported in NUREG/CR-2239 [1]. This 1982 siting study predicted 92 early fatalities for Peach Bottom and 45 early fatalities for Surry for the SST1 source term. In contrast, SOARCA predicted that the early fatality risk was essentially zero for both sites.

For latent cancer fatality results, the exact basis for NUREG/CR-2239 estimates could not be recovered, but literature searches and sensitivity analyses with MACCS2 suggested that these estimates are for the population within 500 miles of the site. Moreover, an attempt to reproduce the results of NUREG/CR-2239 led to agreement within about a factor of 2. Given the uncertainty in the basis for these results, an additional set of calculations was performed to enable the current, state-of-the-art results to be compared with the older NUREG/CR-2239 siting study. For this set of calculations, the most severe source terms predicted by the SOARCA analyses (cf. Figure 12 and Figure 13) were replaced by the largest source term from the 1982 siting study— the so-called SST1 source term. No other modeling or parameter changes were made, including the timing of public evacuation. Thus, this comparison does not attempt to replicate NUREG/CR-2239; it simply evaluates the largest source term, SST1, from that study and compares the results with those from the current work.

Table 10 and Table 11 summarize the comparison to the NUREG/CR-2239 source term results for the Peach Bottom and Surry sites, respectively, assuming an LNT dose-response function. Although the SST1 source term is identical in both comparisons, the risks associated with this source term shown in the tables are different due to the difference in evacuation modeling and other offsite consequence parameters for the two sites.

Table 10. Conditional (i.e., assuming accident occurs), Mean, LNT, Latent-Cancer-Fatality Risks for Residents within the Specified Radii of the Peach Bottom Site

Radius of Circular Area (mi)	SST1	Unmitigated STSBO
10	6.0×10^{-3}	2.3×10^{-4}
20	2.0×10^{-3}	4.0×10^{-4}
50	4.7×10^{-4}	1.6×10^{-4}

(Risks are based on the SST1 Source Term from the 1982 Siting Study and the unmitigated STSBO sequence.)

Table 11. Conditional, Mean, LNT, Latent-Cancer-Fatality Risks for Residents within the Specified Radii of the Surry Site

Radius of Circular Area (mi)	SST1	Unmitigated ISLOCA	Unmitigated STSBO with TISGTR
10	$1.0x10^{-2}$	$7.9x10^{-4}$	$3.2x10^{-4}$
20	$5.1x10^{-3}$	$6.6x10^{-4}$	$1.9x10^{-4}$
50	$1.5x10^{-3}$	$3.5x10^{-4}$	$6.5x10^{-5}$

(Risks are for the SST1 source term from the 1982 Siting Study, the unmitigated ISLOCA, and the unmitigated STSBO with TISGTR sequences.)

This comparison demonstrates that, with offsite consequence modeling parameters equivalent to those used in the SOARCA analyses, the SST1 source term is early enough and large enough to produce latent cancer fatality risks 3 to 30 times greater than the most severe consequences predicted by the SOARCA analyses.

6.4 Conclusions

The results of the SOARCA project represent a major change in our perception of severe reactor accidents and their consequences. Specific conclusions of the project are as follows:

- Mitigation is likely for all of the selected important scenarios due to time available for operator actions and redundancy and diversity of equipment. Mitigation also resulted in no core damage for all scenarios except for the Surry STSBO and the Surry (TI-SGTR).
  - For the Surry STSBO with mitigation, there was no containment failure within 48 hrs.
  - For the Surry TI-SGTR, the predicted individual latent cancer risk for the EPZ was small, $1x10^{-10}$ per reactor-yr, assuming LNT.
- For all core damage scenarios, the best-estimate MELCOR analyses indicated that accidents would progress more slowly and with smaller releases than past treatments (e.g., NUREG/CR-2239) generally indicated. Large, early releases were not predicted.
- Individual early fatality risk is essentially zero.
- Individual latent cancer risk from the selected specific, important scenarios is thousands of times lower than the NRC Safety Goal and millions of times lower than all other cancer risks, even assuming the LNT dose response model.

- Using a dose response model that truncates annual doses below normal background levels results in a further reduction to the latent cancer risk (by a factor of 100 for smaller releases and a factor of 3 for larger releases).

- Latent cancer fatality predictions are generally dominated by long-term exposure to small annual doses (~500 mrem) in conjunction with return criteria for calculations using the LNT assumption.

- Bypass events do not pose a higher latent cancer risk, and a higher conditional risk is offset by lower frequency.

- Explicit consideration of seismic impacts on emergency response (e.g., loss of bridges, traffic signals, and delayed notification) did not significantly impact risk predictions.

- The dominance of external events suggests the need for a corresponding PRA focus and seismic research.

7.0 REFERENCES

[1] NUREG/CR-2239, *Technical Guidance for Siting Criteria Development.* 1982, Sandia National Laboratories: Albuquerque, NM.

[2] NUREG-1150, V., *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants.* 1990, U.S. Nuclear Regulatory Commission: Washington, D.C.

[3] WASH-1400, *WASH-1400: Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants.* 1975, U.S. Nuclear Regulatory Commission: Washington, D.C.

[4] *Safety Goals for the Operation of Nuclear Power Plants*, in *Federal Register, 51 FR 28044.* 1986.

[5] RG-1.174, *An approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the licensing basis.* 2002, U.S. Nuclear Regulatory Commission: Washington, D.C.

[6] NUREG/CR-7008, "Best Practices for Simulation of Severe Accident Progression at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, D.C., 2010 (To be published)

[7] NUREG-1570, "Risk Assessment of Severe-Accident-Induced Steam Generator Tube Rupture," U.S. Nuclear Regulatory Commission, Washington, D.C., 1998.

[8] *Requirements for monitoring the effectiveness of maintenance at nuclear power plants.*, in *U.S. Code of Federal Regulations, 10 CFR Part 50.65.* 1999.

[9] *Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.*, in *U.S. Code of Federal Regulations, 10 CFR Part 50.62.* 1984.

[10] *Loss of all alternating current power*, in *U.S. Code of Federal Regulations, 10 CFR Part 50.63.* 1988.

[11] NUREG/CR-4334, *NUREG/CR-4334: An Approach to the Quantification of Seismic Margins in Nuclear Power Plants.* 1985, Lawrence Livermore National Laboratory: Livermore, CA.

[12] Klapp, Ulrich, et al., *Seismic PSA of the Neckarwestheim 1 Nuclear Power Plant.* in Transactions, SMiRT 19, 19th International Conference on Structural Mechanics in Reactor Technology, August 2007, Toronto, Canada: International Association for Structural Mechanics in Reactor Technology.

[13] DiNunno, J.J. and e. al., *TID-14844: Calculation of Distance Factors for Power and Test Reactor Sites.* 1962, U.S. Atomic Energy Commission: Washtington, D.C.

[14] Soffer, L., et al., *NUREG-1465: Accident Source Terms for Light-Water Nuclear Power Plants.* 1995, U.S. Nuclear Regulatory Commission: Washington, D.C.

[15] WASH-740, *WASH-740: Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants*. 1957, Atomic Energy Commission: Washington, D.C.

[16] Silberberg, M., et al., *NUREG-0956: Reassessment of the Technical Bases for Estimating Source Terms*. 1986, U.S. Nuclear Regulatory Commission: Washington, D.C.

[17] *Determination of exclusion area, low population zone, and population center distance*, in *U.S. Code of Federal Regulations, 10 CFR Part 100.11*. 1962.

[18] Wall, I.B., et al., *NUREG-0340: Overview of the Reactor Safety Study Consequence Model*. 1977, U. S. Nuclear Regulatory Commission: Washington, D.C.

[19] NUREG-0772, *NUREG-0772: Technical Bases for Estimating Fission Product Behavior During LWR Accidents*. 1981, U.S. Nuclear Regulatory Commission: Washington, D.C.

[20] NUREG-0771, *NUREG-0771, Draft for Comment: Regulatory Impact of Nuclear Reactor Accident Source Term Assumption*. 1981, U.S. Nuclear Regulatory Commission: Washington, D.C.

[21] Blond, R., et al., *NUREG-0773: The Development of Severe Accident Source Terms: 1957-1981*. 1982, U.S. Nuclear Regulatory Commission: Washington, D.C.

[22] Gieske, J.A., et al., *BMI-2104: Radionuclide Release Under Specific LWR Accident Conditions*. 1985, Battelle Memorial Insitute: Columbus, OH.

[23] Magallon, D., I. Huhtiniemi, and H. Hohmann, *Lessons learned from FARO/TERMOS corium melt quenching experiments*. Nuclear Engineering and Design, 1999. **189**: p. 223-238.

[24] Chu, T.Y., et al., *NUREG/CR-5582, SAND98-2047: Lower Head Failure Experiments and Analyses*. 1998, Sandia National Laboratories: Albuquerque, NM.

[25] Farmer, M.T., S. Lomperski, and S. Basu. *Results of Reactor Material Experiments Investigating 2-D Core-Concrete Interaction and Debris Coolability*. in *International Conference on Advanced Power Plants, ICAPP'04*. 2004. Pittsburgh, Pennsylvania.

[26] Carbajo, J.J., *NUREG/CR-5942, ORNL/TM-12229: Severe Accident Source Term Characteristics for Selected Peach Bottom Sequences Predicted by the MELCOR code*. 1993, Oak Ridge National Laboratory: Oak Ridge, TN.

[27] Bayless, P.B., *NUREG/CR-5214, EGG-2547: Analysis of Natural Circulation During A Surry Station Blackout Using SCDAP/RELAP5*. 1988, Idaho National Engineering Laboratory: Idaho Falls, ID.

[28] NEA/CSNI, *NEA/CSNI/R(99)24: Technical Opinion Paper on Fuel-Coolant Interaction*. 2000, Nuclear Energy Agency Committee on the Safety of Nuclear Installations, Organization for Economic Cooperation and Development: Paris, France.

[29] Pilch, M.M. and T.G. Theofanous, *The probability of containment failure by direct containment heating in Zion*. Nuclear Engineering and Design, 1996. **164**: p. 1-36.

[30] Theofanous, T.G., et al., *NUREG/CR-6025: The Probability of Mark-I Containment Failure by Melt-Attack of the Liner*. 1993, U.S. Nuclear Regulatory Commission: Washington, D.C.

[31] Gauntt, R.O., et al., *NUREG/CR-6119, Vol., Rev. 3, MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide, Version 1.8.6*. 2005, Sandia National Laboratories: Albuquerque, NM.

[32] Lorenz, R.A. and M.F. Osborne, *NUREG/CR-6261: A Summary of ORNL Fission Product Release Tests with Recommended Release Rates and Diffusion Coefficients*. 1995, U.S. Nuclear Regulatory Commission: Washington, D.C.

[33] Clement, B. and T. Haste, *Comparison report on International Standard Problem ISP-46 (Phebus FPT-1)*. 2003, Note Technique SEMAR 03/021, Draft Final Report.

[34] Ducros, G., et al., *Fission product release under severe accidental conditions: general presentation of the program and synthesis of VERCORS 1-6 results*. Nuclear Engineering and Design, 2001. **208**: p. 191-203.

[35] Greene, N.M., L.M. Petrie, and R.M. Westfall, *ORNL/TM-2005/39, Version 5, Vols. I-III: NITAWL-III: Scale System Module for Performing Resonance Shielding and Working Library Production, SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*. 2005, Oak Ridge National Laboratory: Oak Ridge, TN.

[36] ORNL/TM-2005/39, *ORNL/TM-2005/39, Version 5.1., Vols. I-III: SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*. 2006, Oak Ridge National Laboratory: Oak Ridge, TN.

[37] RG-3.54, R., *Spent Fuel Heat Generation in an Independent Spent Fuel Installation*. 1999, U.S. Nuclear Regulatory Commission: Washington, D.C.

[38] DeHart, M.D. and S.M. Bowman. *Improved radiochemical assay analyses using TRITON depletion sequences in SCALE*. in *Proceedings of International Atomic Energy Agency Technical Meeting "Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition*. 2005. London, United Kingdom: International Atomic Energy Agency.

[39] Bowman, S.M. and D.F. Gill. *Validation of Standardized Computer Analyses for Licensing Evaluation/TRITON Two-Dimensional and Three-Dimensional Models for Light Water Reactor Fuel*. in *PHYSOR-2006, American Nuclear Society Topical Meeting on Reactor Physics: Advances in Nuclear Analysis and Simulation*. 2006. Vancouver, British Columbia, Canada: American Nuclear Society.

[40] Germina, I. and I.C. Gauld, *Analysis of Decay Heat Measurements for BWR Fuel Assemblies*. Transactions of the American Nuclear Society, 2006. **94**: p. 385-387.

[41] Chanin, D. and M.L. Young, *NUREG/CR-6613, SAND97-0594: Code Manual for MACCS2 User's Guide*. 1997, Sandia National Laboratories: Albuquerque, NM.

[42] Laur, M.N., "Meeting with Sandia National Laboratories and an Expert Panel on MELCOR/MACCS Codes in Support of the State of the Art Reactor Consequence Analysis Project," Memo to J.T. Yerokun, Agency Document Access and Management System Accession Number ML062500078, U.S. Nuclear Regulatory Commission, Washington, D.C., September, 2006.

[43] Dennis Atkinson and Russell F. Lee, "Procedures for Substituting Values for Missing NWS Meteorological Data for Use in Regulatory Air Quality Models," July 7, 1992, http://www.rflee.com/RFL_Pages/missdata.pdf

[44] NUREG-0917, *NUREG-0917: Nuclear Regulatory Commission Staff Computer Programs for Use With Meteorological Data.* 1982, U.S. Nuclear Regulatory Commission: Washington, D.C.

[45] RG-1.23, *Regulatory Guide 1.23, Rev. 1: Meteorological Monitoring Programs for Nuclear Power Plants.* 2007, U.S. Nuclear Regulatory Commission: Washington, DC.

[46] NUREG-0654, *NUREG-0654/FEMA-REP-1, Rev. 1: Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants.* 1979, U.S. Nuclear Regulatory Commission: Washington, DC.

[47] NUREG/CR-6864, *NUREG/CR-6864: Identification and Analysis of Factors Affecting Emergency Evacuations.* 2005, Sandia National Laboratories: Albuquerque, NM.

[48] NUREG/CR-6953, Vol. 2: Review of NUREG-0654, Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents": Focus Groups and Telephone Survey. October 2008

[49] NUREG/CR-6864, Vol. 1: "Identification and Analysis of Factors Affecting Emergency Evacuations: Main Report." January 2005

[50] McFadden, K.L., N.E. Bixler, and R.O. Gauntt, *MELMACCS System Documentation (MELCOR to MACCS2 interface definition).* 2005, Sandia National Laboratories: Albuquerque, NM.

[51] Bixler, N.E., E. Clause, C. W. Morrow, and J. A. Mitchell, *Evaluation of Distributions Representing Important Non-Site Specific Parameters in Off-Site Consequence Analyses*, to be published as a NUREG Report, Sandia National Laboratories: Albuquerque, NM, 2010.

[52] Bixler, N.E., et al., *NUREG/ER-6525, Rev. 1, SAND2003-1648P: SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program.* 2003, Sandia National Laboratories: Albuquerque, NM.

[53] NUREG-6953, "Review of NUREG-0654, Supplement 3, 'Criteria for Protective Action Recommendations for Severe Accidents' - Focus Group and Telephone Survey," U.S. Nuclear Regulatory Commission, Washington, D.C., 2008

[54] EPA (2202) Federal Guidance Report No. 13 CD Supplement, E.-C.-.-., Rev. 1. 2002, Prepared by Oak Ridge National Laboratory, Oak Ridge, TN for Office of Air and Radiation, U. S. Environmental Protection Agency: Washington, DC.

[55] EPA (1994). *Estimating Radiogenic Cancer Risk.* EPA 402-R-93-076 (U.S. Environmental Protection Agency, Washington, D.C.)

[56] Eckerman, K. F., *Risk Coefficients for SOARCA Project*, memo, Oak Ridge National Laboratory, May 2008.

[57] NAS, *Health Effects of Exposure to Low Levels of Ionizing Radiaiton: BEIR V.* 1990, National Academy of Sciences, National Research Council, National Academy Press: Washington, DC.

[58] Aurengo, A., et al., *French National Academy of Medicine report: Dose-effect relationships and estimation of the carcinogenic effects of low doses of ionizing radiation.* 2005, French Academy of Sciences, French National Academy of Medicine.

[59] ICRP, *The 2007 Recommendations of the International Commission on Radiological Protection.* Annals of the ICRP, 2007. **37**(Nos. 2-4).

[60] ICRP, *Scope of Radiological Protection Control Measures.* Annals of the ICRP, 2007. **37**(Nos. 5).

[61] HPS, *PS010-1: Position Statement of the Health Physics Society -- Radiation Risk in Perspective.* 2004, Health Physics Society: McLean, VA.