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Evaluation of Severe Accident Risks: Sequoyah, Unit 1

Main Report

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Sandia National Laboratories
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ABSTRACT

In support of the Nuclear Regulatory Commission's (NRC's) assessment of the risk from severe accidents at commercial nuclear power plants in the U.S. reported in NUREG-1150, the Severe Accident Risk Reduction Program (SARRP) has completed a revised calculation of the risk to the general public from the operation of the Sequoyah Power Station, Unit 1. This power plant, located in southeastern Tennessee, is operated by the Tennessee Valley Authority (TVA).

The emphasis in this risk analysis was not on determining a "so-called" point estimate of risk. Rather, it was to determine the distribution of risk, and to discover the uncertainties that account for the breadth of this distribution.

The offsite risk from internal initiating events was found to be quite low with respect to the safety goals. The containment appears quite likely to successfully withstand the loads that might be placed upon it if the core melts and the reactor vessel fails. A good portion of the risk, in this analysis, comes from initiating events which bypass the containment, such as interfacing system pipe breaks and steam generator tube ruptures. These events are estimated to have a relatively low frequency of occurrence, but their consequences are relatively large. Other events that contribute to offsite risk involve early containment failures, that is, failures that occur during degradation of the core or failures that occur near the time of vessel breach. Early containment failures are largely attributable to station blackout accidents. Considerable uncertainty is associated with the risk estimates produced in this analysis. The offsite risk from external initiating events was not included in the scope of this analysis.

CONTENTS

SUMMARY........ S.1

1. INTRODUCTION........ 1.1

1.1 Background and Objectives of NUREG-1150........ 1.1

1.2 Overview of Sequoyah Power Station, Unit 1........ 1.3

1.3 Changes Since the Draft Report........ 1.5

1.4 Structure of the Analysis........ 1.8

1.5 Organization of This Report........ 1.17

1.6 References........ 1.18

2. ANALYSIS OF THE ACCIDENT PROGRESSION........ 2.1

2.1 Sequoyah Features Important to the Accident Progression.... 2.1
2.1.1 The Sequoyah Containment Structure........ 2.2
2.1.2 The Ice Condenser........ 2.2
2.1.3 The Containment Spray System........ 2.2
2.1.4 The ARFS........ 2.3
2.1.5 The Hydrogen Ignition System........ 2.3
2.1.6 The Compartmental Structure of the Containment...... 2.3
2.1.7 Sump and Cavity Arrangement........ 2.3

2.2 Interface with the Core-Damage Frequency Analysis........ 2.4
2.2.1 Definition of Plant Damage States........ 2.4
2.2.2 PDS Frequencies........ 2.6
2.2.3 High-Level Grouping of PDSs........ 2.12
2.2.4 Variables Sampled in the Accident Frequency Analysis........ 2.12

2.3 Description of the APET........ 2.21
2.3.1 Overview of the APET........ 2.21
2.3.2 Overview of the APET Quantification........ 2.23
2.3.3 Variables Sampled for the Accident Progression Analysis........ 2.31

2.4 Description of the Accident Progression Bins........ 2.53
2.4.1 Description of the Bin Characteristics........ 2.53
2.4.2 Rebinning........ 2.61
2.4.3 Summary Bins for Presentation........ 2.66

2.5 Results of the Accident Progression Analysis........ 2.69
2.5.1 Results for Internal Initiators........ 2.70
2.5.1.1 Results for PDS Group 1 - Slow SBO........ 2.70
2.5.1.2 Results for PDS Group 2 - Fast SBO........ 2.72

2.5.1.3 Results for PDS Group 3 - LOCAs................ 2.75
2.5.1.4 Results for PDS Group 4 - Event V.............. 2.77
2.5.1.5 Results for PDS Group 5 - Transients........... 2.77
2.5.1.6 Results for PDS Group 6 - ATWS................. 2.80
2.5.1.7 Results for PDS Group 7 - SGTRs................ 2.83
2.5.1.8 Core Damage Arrest and Avoidance of VB......... 2.85
2.5.1.9 Early Containment Failure...................... 2.86
2.5.1.10 Summary....................................... 2.88

2.5.2 Sensitivity Analysis for Internal Initiators........ 2.92

2.6 Insights from the Accident Progression Analysis............2.103

2.7 References...2.106

3. RADIOLOGICAL SOURCE TERM ANALYSIS.............................. 3.1

3.1 Sequoyah Features Important to the Source Term Analysis.. 3.1

3.2 Description of the SEQSOR Code............................ 3.3
3.2.1 Overview of the Parametric Model................... 3.3
3.2.2 Description of SEQSOR.............................. 3.4
3.2.3 Variables Sampled in the Source Term Analysis....... 3.12

3.3 Results of the Source Term Analysis....................... 3.16
3.3.1 Results for Internal Initiators.................... 3.16
3.3.1.1 Results for PDS Group 1: Slow SBO............. 3.16
3.3.1.2 Results for PDS Group 2: Fast SBO............. 3.21
3.3.1.3 Results for PDS Group 3: LOCAs................ 3.24
3.3.1.4 Results for PDS Group 4: Event V.............. 3.27
3.3.1.5 Results for PDS Group 5: Transients........... 3.30
3.3.1.6 Results for PDS Group 6: ATWS................. 3.33
3.3.1.7 Results for PDS Group 7: SGTRs................ 3.36
3.3.1.8 Results for Generalized Accident Progression Bins............................. 3.39
3.3.1.9 Summary....................................... 3.40

3.4 Partitioning of the Source Terms for the Consequence Analysis.. 3.52
3.4.1 Results for Internal Initiators.................... 3.52

3.5 Insights from the Source Term Analysis.................... 3.64

3.6 References.. 3.65

4. CONSEQUENCE ANALYSIS... 4.1

4.1 Description of the Consequence Analysis................... 4.1

4.2 MACCS Input for Sequoyah.................................. 4.2

4.3 Results of the MACCS Consequence Calculations.............. 4.5
4.3.1 Results for Internal Initiators.................... 4.5

4.4 References.. 4.4

5. RISK RESULTS FOR SEQUOYAH.................................. 5.1

5.1 Results for Internal Initiators........................... 5.1
5.1.1 Risk Results...................................... 5.1
5.1.2 Contributors to Risk.............................. 5.10
5.1.3 Contributors to Uncertainty....................... 5.18

5.2 References.. 5.2

6. INSIGHTS AND CONCLUSIONS.................................. 6.1

FIGURES

1. Back-End Documentation for NUREG-1150........................ xiii

S.1 Overview of Integrated Plant Analysis in NUREG-1150........ S.5
S.2 Mean Probability of APBs for the Summary PDSs.............. S.9
S.3 Probability of Core Damage Arrest.......................... S.10
S.4 Probability of Early Containment Failure................... S.10
S.5 Exceedance Frequencies for Release Fractions............... S.16
S.6 Consequences Conditional on Source Terms................... S.19
S.7 Exceedance Frequencies for Risk............................ S.21
S.8 Distributions of Annual Risk............................... S.26
S.9 Fractional PDS Contributions to Annual Risk................ S.26
S.10 Fractional APB Contributions to Annual Risk............... S.27

1-1 Section of the Sequoyah Containment........................ 1.4
1-2 Overview of Integrated Plant Analysis in NUREG-1150........ 1.9
1-3 Example Risk CCDF.. 1.15

2.5-1 Probability of Core Damage Arrest.................. 2.87

2.5-2 Probability of Early Containment Failure............ 2.89

2.5-3 Mean Probability of Summary APBs for
Summary PDSs.. 2.90

2.5-4 Core Damage Frequency Distributions
for the PDS Groups.................................. 2.91

2.5-5 Distribution of Frequencies for Summary APBs........ 2.93

3.2-1 Blood Flow Diagram for SEQSOR....................... 3.7

3.3-1 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 1: Slow SBO) 3.19

3.3-2 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 2: Fast SBO) 3.23

3.3-3 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 3: Loss-of-Coolant Accidents 3.26

3.3-4 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 4: Event V) 3.29

3.3-5 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 5: Transients) 3.32

3.3-6 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 6: ATWS) 3.35

3.3-7 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 7: SGTRs) 3.38

3.3-8 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (CF During CD) 3.41

3.3-9 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (Alpha Mode) 3.42

3.3-10 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (CF at VB with the RCS at High Pressure) 3.43

3.3-11 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (CF at VB with the RCS at Low Pressure) 3.44

3.3-12 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (Late CF) 3.45

3.3-13 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (Event V, Dry) 3.46

3.3-14 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (Event V, Wet) 3.47

3.3-15 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators, "G" SGTRs (Secondary SRVs Reclosing) 3.48

3.3-16 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators, "H" SGTRs (Secondary SRVs Stuck Open) 3.49

3.3-17 Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (All Internal Initiators) 3.50

3.4-1 Distribution of Nonzero Early and Chronic Health Effect Weights for Internal Initiators 3.54

4.3-1 Consequences Conditional on Source Terms 4.8

5.1-1 Exceedance Frequencies for Risk (Sequoyah, All Internal Initiators) 5.2

5.1-2 Distributions of Annual Risk (Sequoyah, All Internal Initiators) 5.7

5.1-3 Fractional PDS Contributions to Annual Risk, Sequoyah (Internal Initiators) 5.13

5.1-4 Fractional APB Contributions to Annual Risk, Sequoyah (Internal Initiators) 5.15

TABLES

1. NUREG-1150 Analysis Documentation xiv

S.1 Design Features Relevant to Severe Accidents S.3
S.2 Sequoyah Core Damage Frequencies S.7
S.3 Two Methods of Calculating Contributions to Mean Risk S.25

2.2-1 PWR Plant Damage State Characteristics 2.17

2.2-2 Plant Damage States for Sequoyah 2.18

2.2-3 PDS Frequencies Comparison, Sequoyah 2.10

2.2-4 Relationship Between PDS Groups and Summary Groups 2.13

2.2-5 Variables Sampled in the Accident Frequency Analysis for Internal Initiators 2.15

2.3-1 Questions in the Sequoyah APET 2.25

2.3-2 Variables Sampled in the Accident Progression Analysis for Internal Initiators 2.33

2.5-1 Results of Accident Progression Analysis for Sequoyah Internal Initiators Group 1: Slow SBO 2.71

2.5-2 Results of Accident Progression Analysis for Sequoyah Internal Initiators Group 2: Fast SBO 2.74

2.5-3 Results of Accident Progression Analysis for Sequoyah Internal Initiators Group 3: LOCAs 2.76

2.5-4 Results of Accident Progression Analysis for Sequoyah Internal Initiators Group 4: Event V 2.78

2.5-5 Results of Accident Progression Analysis for Sequoyah Internal Initiators Group 5: Transients 2.79

2.5-6 Results of Accident Progression Analysis for Sequoyah Internal Initiators Group 6: ATWS 2.82

2.5-7 Results of Accident Progression Analysis for Sequoyah Internal Initiators Group 7: SGTRs 2.84

2.5-8 Comparison of APET Results With and Without T-I Hot Leg Breaks and SGTRs PDS Group 1: Slow SBO 2.99

2.5-9 Comparison of APET Results With and Without T-I Hot Leg Breaks and SGTRs PDS Group 2: Fast SBO 2.99

2.5-10 Comparison of APET Results With and Without T-I Hot Leg Breaks and SGTRs PDS Group 5: Transients 2.101

2.5-11 Comparison of APET Results With and Without T-I Hot Leg Breaks and SGTRs PDS Group 6: ATWS 2.102

3.2-1 Isotopes in Each Radionuclide Release Class 3.3

3.2-2 Variables Sampled in the Source Term Analysis 3.13

3.3-1 Mean Source Terms for Sequoyah Internal Initiators (PDS Group 1: Slow SBO) 3.20

3.3-2 Mean Source Terms for Sequoyah Internal Initiators (PDS Group 2: Fast SBO) 3.22

3.3-3 Mean Source Terms for Sequoyah Internal Initiators (PDS Group 3: Loss-of-Coolant Accidents)........... 3.25

3.3-4 Mean Source Terms for Sequoyah Internal Initiators (PDS Group 4: Event V)............................ 3.28

3.3-5 Mean Source Terms for Sequoyah Internal Initiators (PDS Group 5: Transients)......................... 3.31

3.3-6 Mean Source Terms for Sequoyah Internal Initiators (PDS Group 6: Anticipated Transient Without Scram)................................... 3.34

3.3-7 Mean Source Terms for Sequoyah Internal Initiators (PDS Group 7: SGTRs).............................. 3.37

3.4-1 Summary of Early and Chronic Health Effect Weights for Internal Initiators........................... 3.54

3.4-2 Distribution of Source Terms with Nonzero Early Fatality and Chronic Fatality Weights for Internal Initiators............................... 3.57

3.4-3 Distribution of Source Terms with Zero Early Fatality Weight and Nonzero Chronic Fatality Weights for Internal Initiators................... 3.59

3.4-4 Mean Source Terms Resulting from Partitioning for Internal Initiators - Sequoyah................ 3.60

4.1-1 Definition of Consequence Analysis Results.......... 4.3

4.2-1 Site Specific Input Data for Sequoyah MACCS Calculations................................ 4.4

4.2-2 Shielding Factors for Sequoyah MACCS Calculations...................................... 4.4

4.3-1 Mean Consequence Results for Internal Initiators (Population Doses in Sv)............... 4.6

5.1-1 Distributions for Annual Risk at Sequoyah Due to Internal Initiators........................ 5.9

5.1-2 Fractional PDS Contributions to Annual Risk at Sequoyah Due to Internal Initiators............. 5.12

5.1-3 Fractional APB Contributions (%) to Annual Risk at Sequoyah Due to Internal Initiators............. 5.14

5.1-4 Summary of Regression Analyses for Annual Risk at Sequoyah for Internal Initiators................ 5.21

5.1-5 Summary of Regression Analyses for Annual Risk at Sequoyah for PDS Group 1: Slow SBO 5.22

5.1-6 Summary of Regression Analyses for Annual Risk at Sequoyah for PDS Group 2: Fast SBO 5.23

5.1-7 Summary of Regression Analyses for Annual Risk at Sequoyah for PDS Group 3: LOCAs 5.24

5.1-8 Summary of Regression Analyses for Annual Risk at Sequoyah for PDS Group 4: Event V 5.25

5.1-9 Summary of Regression Analyses for Annual Risk at Sequoyah for PDS Group 5: Transients 5.26

5.1-10 Summary of Regression Analyses for Annual Risk at Sequoyah for PDS Group 6: ATWS 5.27

5.1-11 Summary of Regression Analyses for Annual Risk at Sequoyah for PDS Group 7: SGTRs 5.28

FOREWORD

This is one of numerous documents that support the preparation of the final NUREG-1150 document by the NRC Office of Nuclear Regulatory Research. Figure 1 illustrates the documentation of the accident progression, source term, consequence, and risk analyses. The direct supporting documents for the first draft of NUREG-1150 and for the revised draft of NUREG-1150 are given in Table 1. They were produced by the three interfacing programs that performed the work - the Accident Sequence Evaluation Program (ASEP) at Sandia National Laboratories, the (SARRP), and the PRA Phenomenology and Risk Uncertainty Evaluation Program (PRUEP). The Zion volumes were written by Brookhaven National Laboratory and Idaho National Engineering Laboratory.

The Accident Frequency Analysis, and its constituent analyses, such as the Systems Analysis and the Initiating Event Analysis, are reported in NUREG/CR-4550. Originally, NUREG/CR-4550 was published without the designation "Draft for Comment." Thus, the current revision of NUREG/CR-4550 is designated Revision 1. The label Revision 1 is used consistently on all volumes, including Volume 2 which was not part of the original documentation. NUREG/CR-4551 was originally published as a "Draft for Comment." While the current version could have been issued without a revision indication, all volumes of NUREG/CR-4551 have been designated Revision 1 for consistency with NUREG/CR-4550.

The material contained in NUREG/CR-4700 in the original documentation is now contained in NUREG/CR-4551; NUREG/CR-4700 is not being revised. The contents of the volumes in both NUREG/CR-4550 and NUREG/CR-4551 have been altered. In both documents now, Volume 1 describes the methods used in the analyses, Volume 2 presents the elicitation of expert judgment, Volume 3 concerns the analyses for Surry, Volume 4 concerns the analyses for Peach Bottom, and so on. The Sequoyah analysis is contained in Volume 5 of NUREG/CR-4551. Note that the Sequoyah plant was also treated in Volume 2 of the original Draft for Comment version of NUREG/CR-4551 and NUREG/CR-4700.

In addition to NUREG/CR-4550 and NUREG/CR-4551, there are several other reports published in association with NUREG-1150 that explain the methods used, document the computer codes that implement these methods, or present the results of calculations performed to obtain information specifically for this project. These reports include:

NUREG/CR-5032, SAND87-2428, "Modeling Time to Recovery and Initiating Event Frequency for Loss of Off-site Power Incidents at Nuclear Power Plants," R. L. Iman and S. C. Hora, Sandia National Laboratories, Albuquerque, NM, January 1988.

NUREG/CR-4840, SAND88-3102, "Procedures for External Core Damage Frequency Analysis for NUREG-1150," M. P. Bohn and J. A. Lambright, Sandia National Laboratories, Albuquerque, NM, December 1988.

NUREG/CR-5174, SAND88-1607, J. M. Griesmeyer and L. N. Smith, "A Reference Manual for the Event Progression and Analysis Code (EVNTRE)," Sandia National Laboratories, Albuquerque, NM, September 1989.

NUREG/CR-5380, SAND88-2988, S. J. Higgins, "A User's Manual for the Post Processing Program PSTEVNT," Sandia National Laboratories, Albuquerque, NM, November 1989.

NUREG/CR-4624, BMI-2139, R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios," Volumes I-V, Battelle's Columbus Division, Columbus, OH, 1986.

NUREG/CR-5062, BMI-2160, M. T. Leonard et al., "Supplemental Radionuclide Release Calculations for Selected Severe Accident Scenarios," Battelle Columbus Division, Columbus, OH, 1988.

NUREG/CR-5331, SAND89-0072, S. E. Dingman et al., "MELCOR Analyses for Accident Progression Issues," Sandia National Laboratories, Albuquerque, NM, November 1990.

NUREG/CR-5253, SAND88-2940, R. L. Iman, J. C. Helton, and J. D. Johnson, "PARTITION: A Program for Defining the Source Term/Consequence Analysis Interfaces in the NUREG-1150 Probabilistic Risk Assessments User's Guide," Sandia National Laboratories, Albuquerque, NM, May 1990.

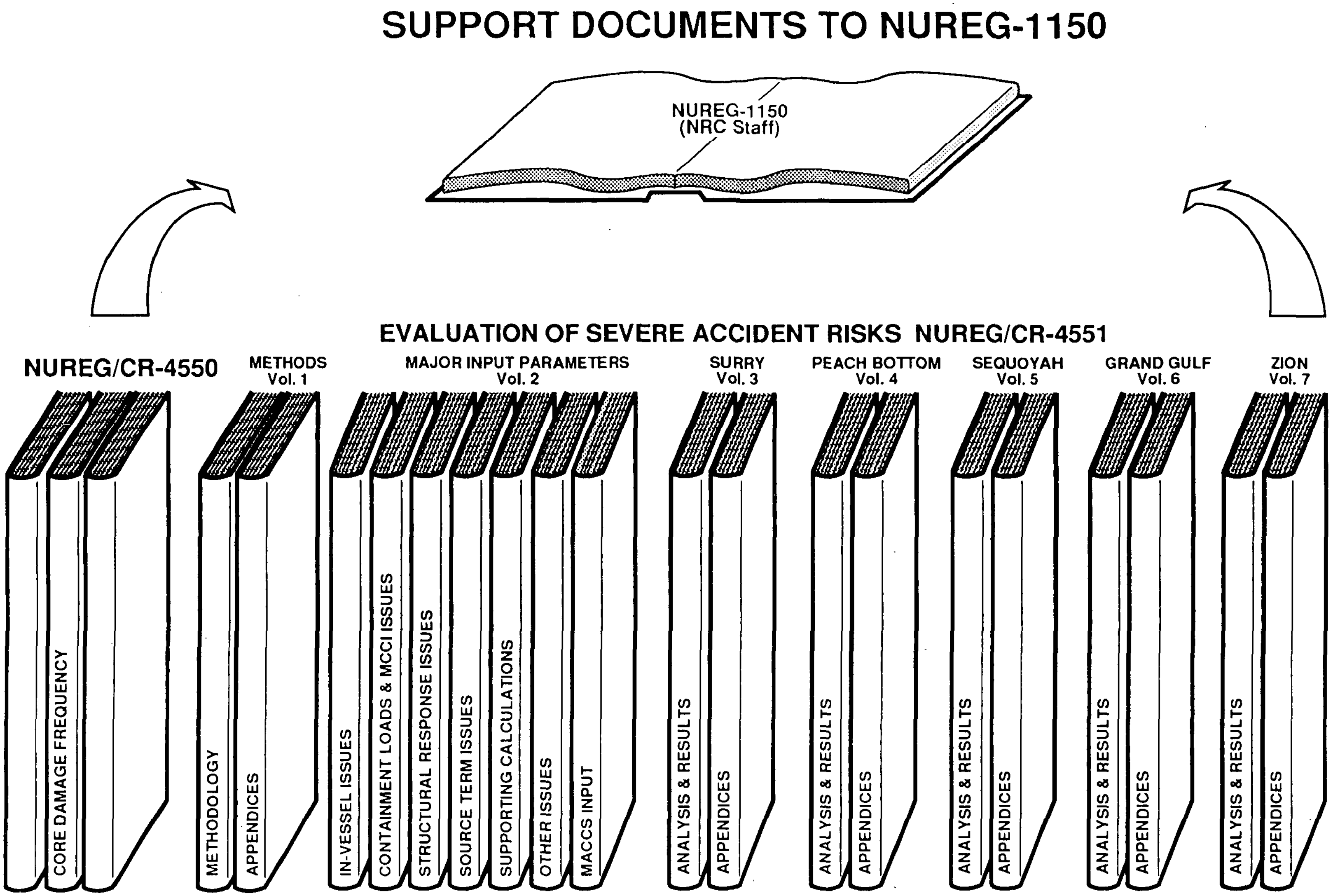

Figure 1. Back-End Documentation for NUREG-1150.

Table 1. NUREG-1150 Analysis Documentation

Original Documentation

NUREG/CR-4550

Analysis of Core Damage Frequency From Internal Events

Vol.	
1	Methodology
2	Summary (Not Published)
3	Surry Unit 1
4	Peach Bottom Unit 2
5	Sequoyah Unit 1
6	Grand Gulf Unit 1
7	Zion Unit 1

NUREG/CR-4551

Evaluation of Severe Accident Risks and the Potential for Risk Reduction

Vol.	
1	Surry Unit 1
2	Sequoyah Unit 1
3	Peach Bottom Unit 2
4	Grand Gulf Unit 1

NUREG/CR-4700

Containment Event Analysis for Potential Severe Accidents

Vol.	
1	Surry Unit 1
2	Sequoyah Unit 1
3	Peach Bottom Unit 2
4	Grand Gulf Unit 1

Revised Documentation

NUREG/CR-4550, Rev. 1, Analysis of Core Damage Frequency

Vol.	Part	
1		Methodology
2	Part 1	Expert Judgment Elicit. Expert Panel
	Part 2	Expert Judgment Elicit. Project Staff
3	Part 1	Surry Unit 1 Internal Events
	Part 2	Surry Unit 1 Internal Events App.
	Part 3	Surry External Events
4	Part 1	Peach Bottom Unit 2 Internal Events
	Part 2	Peach Bottom Unit 2 Int. Events App.
	Part 3	Peach Bottom Unit 2 External Events
5	Part 1	Sequoyah Unit 1 Internal Events
	Part 2	Sequoyah Unit 1 Internal Events App.
6	Part 1	Grand Gulf Unit 1 Internal Events
	Part 2	Grand Gulf Unit 1 Internal Events App.
7		Zion Unit 1 Internal Events

NUREG/CR-4551, Rev. 1, Eval. of Severe Accident Risks

Vol.	Part	
1		Part 1, Methodology; Part 2, Appendices
2	Part 1	In-Vessel Issues
	Part 2	Containment Loads and MCCI Issues
	Part 3	Structural Issues
	Part 4	Source Term Issues
	Part 5	Supporting Calculations
	Part 6	Other Issues
	Part 7	MACCS Input
3	Part 1	Surry Analysis and Results
	Part 2	Surry Appendices
4	Part 1	Peach Bottom Analysis and Results
	Part 2	Peach Bottom Appendices
5	Part 1	Sequoyah Analysis and Results
	Part 2	Sequoyah Appendices
6	Part 1	Grand Gulf Analysis and Results
	Part 2	Grand Gulf Appendices
7	Part 1	Zion Analysis and Results
	Part 2	Appendices

ACRONYMS AND INITIALISMS

ADV	atmospheric dump valves
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AOV	air-operated valve
APB	accident progression bin
APET	accident progression event tree
ARF	air return fan
ARFS	air return fan system
ASEP	accident sequence evaluation
ATWS	anticipated transient without scram
BMT	basemat meltthrough
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
CCF	common cause failure
CCI	core-concrete interaction
CCDF	complementary cumulative distribution function
CCP	centrifugal charging pump
CCW	component cooling water
CDF	cumulative distribution function
CF	containment failure
CH	chronic health effect weight
CFW	chronic fatality weight
CHR	containment heat removal
CSS	containment spray system
CST	condensate storage tank
DCH	direct containment heating
DF	decontamination factor
DG	diesel generator
EACPS	emergency ac power system
ECCS	emergency core cooling system
EF	early fatality
EFW	early fatality weight
EH	early health effect weight
EOP	emergency operating procedures
EPRI	Electric Power Research Institute
ESW	emergency service water
EVSE	ex-vessel steam explosion
FSAR	final safety analysis report
HIS	hydrogen ignition system
HPI	high pressure injection
HPIS	high pressure injection system
HPRS	high pressure recirculation system
HPME	high pressure melt ejection
HRA	human reliability analysis

IC	ice condenser
ICS	ice condenser system
ICIR	in-core instrumentation room
INEL	Idaho National Engineering Laboratory
IVSE	in-vessel steam explosion
LC	lower compartment (of containment)
LCF	latent cancer fatalities
LHS	Latin Hypercube Sampling
LOCA	loss-of-coolant accident
LOSP	loss of offsite power
LP	lower plenum (of ice condenser)
LPI	low pressure injection
LPIS	low pressure injection system
LPRS	low pressure recirculation system
LWR	light water reactor
MCDF	mean core damage frequency
MDP	motor-driven pump
MFWS	main feedwater system
MOV	motor-operated valve
MSIV	main steam isolation valve
MSL	main steam line
NRC	Nuclear Regulatory Commission
PDS	plant damage state
PORV	power-operated relief valve
PRA	probabilistic risk analysis
PWR	pressurized water reactor
PZR	pressurizer
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RWST	refueling water storage tank
SBO	station blackout
SERG	steam explosion review group
SG	steam generator
SGTR	steam generator tube rupture
SIS	safety injection system
SLC	standby liquid control
SNL	Sandia National Laboratories
SOV	solenoid-operated valve
SRV	safety relief valve
STCP	source term code package
STD	steam-turbine-driven
STSG	source term subgroup

TAF	top of active fuel
TDP	turbine-driven pump
T-I	temperature-induced
TMCD	total mean core damage
UC	upper compartment (of containment)
UP	upper plenum (of ice condenser)
UTAF	uncovering of TAF
VB	vessel breach

ACKNOWLEDGMENTS

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SUMMARY

S.1 Introduction

The United States Nuclear Regulatory Commission (NRC) has recently completed a major study to provide a current characterization of severe accident risks from light water reactors (LWRs). This characterization is derived from integrated risk analyses of five plants. The summary of this study, NUREG-1150,[1] has been issued as a second draft for comment.

The risk assessments on which NUREG-1150 is based can generally be characterized as consisting of four analysis steps, an integration step, and an uncertainty analysis step:

1. Accident frequency analysis: the determination of the likelihood and nature of accidents that result in the onset of core damage.

2. Accident progression analysis: an investigation of the core damage process, both within the reactor vessel before it fails and in the containment afterwards, and the resultant impact on the containment.

3. Source term analysis: an estimation of the radionuclide transport within the reactor coolant system (RCS) and the containment, and the magnitude of the subsequent releases to the environment.

4. Consequence analysis: the calculation of the offsite consequences, primarily in terms of health effects in the general population.

5. Risk integration: the assembly of the outputs of the previous tasks into an overall expression of risk.

6. Uncertainty analysis: the propagation of the uncertainties in the initiating events, failure events, accident progression branching ratios and parameters, and source term parameters through the first three analyses above, and the determination of which of these uncertainties contributes the most to the uncertainty in risk.

This volume presents the details of the last five of the six steps listed above for the Sequoyah Nuclear Station, Unit 1. The first step is described in NUREG/CR-4550.[2]

S.2 Overview of Sequoyah Nuclear Station, Unit 1

The Sequoyah Power Station, Unit 1 is operated by the Tennessee Valley Authority (TVA) and is located on the west shore of the Chickamauga Lake in southeastern Tennessee, about 10 miles northeast of Chattanooga, Tennessee. There are two units located on the site; Unit 2 is essentially identical to Unit 1.

The nuclear reactor of Sequoyah Unit 1 is a 1148 MWe (3411 MWt) pressurized water reactor (PWR) designed and built by Westinghouse. The reactor coolant system (RCS) has four U-tube steam generators (SGs) and four reactor coolant pumps (RCPs). The containment and the balance of the plant were designed and built by the utility, TVA. Unit 1 began commercial operation in 1981.

Table S.1 summarizes the design features of the plant relevant to severe accidents. Of particular interest is the ice condenser designed to be a passive pressure-suppression system. The containment is a free-standing steel structure, with a fairly low design pressure (11 psig). The ability to crosstie the 6.9 kV emergency buses at Unit 1 and Unit 2 helps to reduce the frequency of station blackout (SBO) at Unit 1. The process for switching the emergency core cooling system from injection mode to recirculation mode is only partially automated and requires that a series of operator actions be accomplished in a relatively short time. Operator error in this process, as well as common-cause failures account for a relatively high frequency for loss-of-coolant (LOCA) accidents at Sequoyah.

S.3 Description of the Integrated Risk Analysis

Risk is determined by combining the results of four constituent analyses: accident frequency, accident progression, source term, and consequence analyses. Uncertainty in risk is determined by assigning distributions to important variables, generating a sample from these variables, and propagating each observation of the sample through the entire analysis. The sample for Sequoyah consisted of 200 observations involving variables from the first three constituent analyses. The risk analysis synthesizes the results of the four constituent analyses to produce measures of offsite risk and the uncertainty in that risk. This process is depicted in Figure S.1. The boxes in this figure show the computer codes used. The interfaces between constituent analyses are shown between the boxes. A mathematical summary of the process, using a matrix representation, is given in Section 1.4 of this volume.

The accident frequency analysis uses event tree and fault tree techniques to investigate the manner in which various initiating events can lead to core damage and the frequency of various types of accidents. Experimental data, past observational data, and modeling results are combined to produce frequency estimates for the minimal cut sets that lead to core damage. A minimal cut set is a unique combination of initiating event and individual hardware or operator failures. The minimal cut sets are grouped into plant damage states (PDSs), where all minimal cut sets in a PDS provide a similar set of initial conditions for the subsequent accident progression analysis. Thus, the PDSs form the interface between the accident frequency analysis and the accident progression analysis. The outcome of the accident frequency analysis is a frequency for each PDS or group of PDSs for each observation in the sample.

The accident progression analysis uses large, complex event trees to determine the possible ways in which an accident might evolve from each PDS. The definition of each plant damage state provides enough information to define the initial conditions for the accident progression event tree (APET) analysis. Past observations, experimental data, mechanistic code calculations, and expert judgment were used in the development of the model for accident progression that is embodied in the APET and in the selection of the branch probabilities and parameter values used in the APET. Due to the large number of questions in the Sequoyah APET and the fact that many of these questions have more than two outcomes, there are far too many paths through the APET to permit their individual consideration in subsequent source term and consequence analysis. Therefore, the paths through the trees are grouped into accident progression bins (APBs), where each bin is a group of paths through the event tree that define a similar set of conditions for source term analysis. The properties of each accident progression bin define the initial conditions for the estimation of a source term. The result of the accident progression analysis is a probability for each APB, conditional on the occurrence of a PDS, for each observation in the sample.

Table S.1
Design Features Relevant to Severe Accidents
Sequoyah Unit 1

Emergency Core Cooling (ECCS)	Safety Injection System (SIS) Two motor-driven pumps (MDPs) Suction from refueling water storage tank (RWST) or low pressure recirculation system (LPRS) Provides high head injection
	Charging System Two centrifugal charging pumps (CCPs) Suction from RWST or LPRS Provides feed and bleed cooling, RCP seal flow, and high head injection
	Low Pressure Injection System (LPIS) Two MDPs Suction from RWST or containment sump Provides suction to the SIS and charging system
	Accumulators Four accumulators containing borated water Pressurized to 660 psig
Emergency Core Heat Removal	Auxiliary Feedwater System (AFWS) Two MDPs and one turbine-driven pump (TDP)
	Feed and Bleed Utilizes Charging System and PORVs
Reactivity Control	Reactor Protection System (automatic scram)
	Manual scram

Table S.1 (continued)

Emergency Electrical Power	AC Electrical Power Two diesel generators (DGs) for each units Each DG dedicated to 6.9 kV bus (can be crosstied)
	DC Electrical Power Station batteries designed to last 2 hours Each DC battery board has normal and alternate power supply
Containment Structure	Ice condenser containment Free-standing steel structure Design pressure is 10.8 psig Free volume is ~ 1.25 million ft^3
Containment Heat Removal	Containment Spray System (CSS) Provides long-term emergency heat removal Two centrifugal pumps
Support Systems	Component Cooling Water (CCW) Five pumps and three heat exchangers for 2 Units Provides cooling for RCP seals and emergency equipment
	Service Water System (SWS) Eight self-cooled pumps for 2 Units
Sump and Reactor Cavity	No connection between sump and cavity at a low level in the containment
	Overflow from sump can fill the cavity if the RWST Contents are injected into containment and a significant amount of ice melts
Containment Systems	Hydrogen Igniter System (HIS) Prevents buildup of large quantities of hydrogen in the containment--requires ac power
	Air Return Fan System (ARFS) Mixes containment atmosphere--requires ac power
	Ice Condenser System (ICS) Provides passive pressure-suppression capability Contains 2.5 x 10^6 lb of borated ice

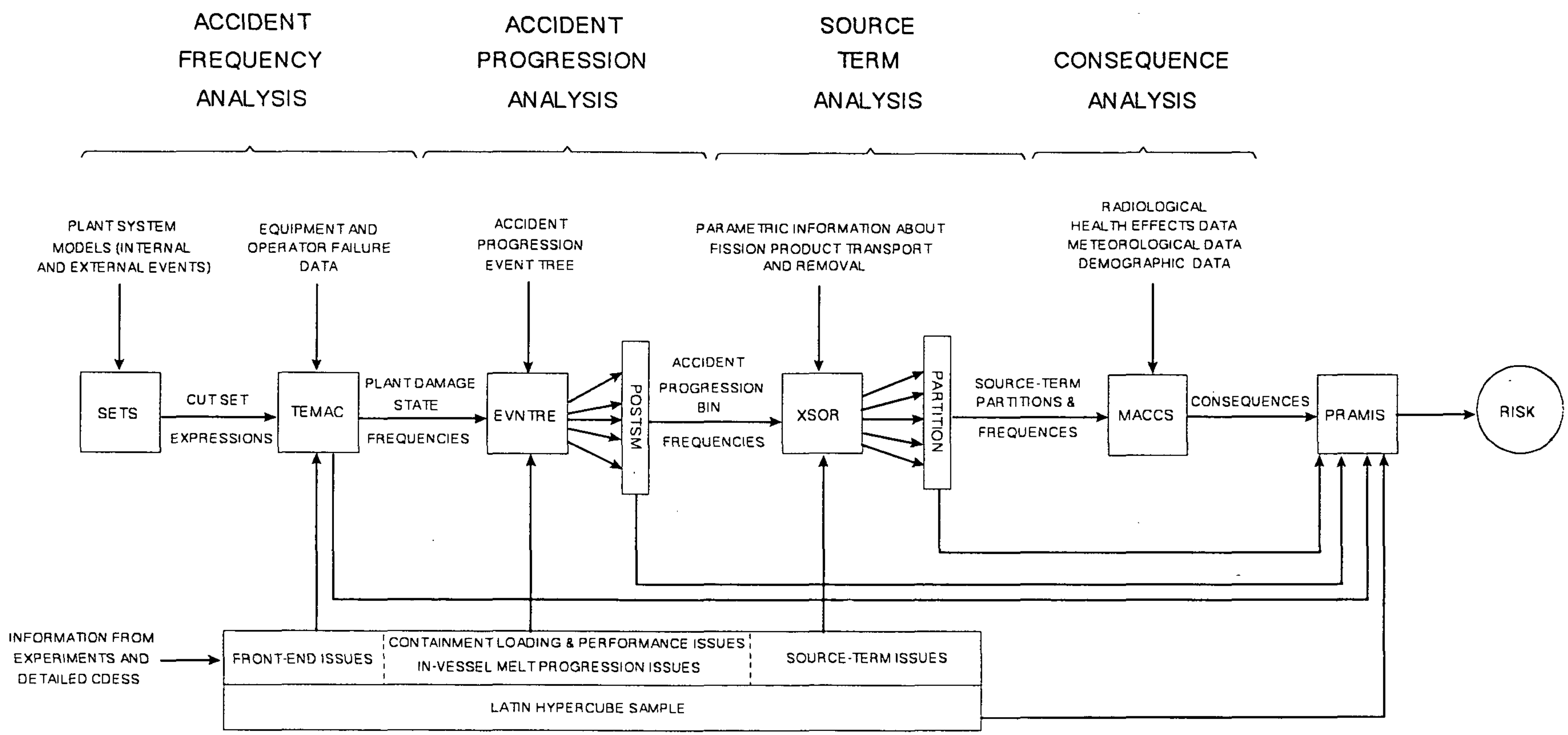


Figure S.1. Overview of Integrated Plant Analysis in NUREG-1150

A source term is calculated for each APB with a non-zero conditional probability for each observation in the sample by SEQSOR, a fast-running parametric computer code. SEQSOR is not a detailed mechanistic model; it is not designed to model the fission product transport, physics, and chemistry from first principles. Instead, SEQSOR integrates the results of many detailed codes and the conclusions of many experts. Most of the parameters used in calculating fission product release fractions in SEQSOR are sampled from distributions provided by an expert panel. Because of the large number of APBs, use of a fast-executing code like SEQSOR is necessary.

The number of APBs for which source terms are calculated is so large that it is not computationally practical to perform a consequence calculation for every source term. As a result, the source terms had to be combined into source term groups. Each source term group is a collection of source terms that result in similar consequences. The process of determining which APBs go to which source term group is called partitioning. This process considers the potential of each source term group to cause early fatalities and latent cancer fatalities. The result of the source term calculation and subsequent partitioning is that each APB for each observation is assigned to a source term group.

A consequence analysis is performed for each source term group, generating both mean consequences and distributions of consequences. Since each APB is assigned to a source term group, the consequences are known for each APB of each observation in the sample. The frequency of each PDS for each observation is known from the accident frequency analysis, and the conditional probability of each APB is determined for each PDS group for each observation in the accident progression analysis. Thus, for each APB of each observation in the sample, both frequency and consequences are determined. The risk analysis assembles and analyzes all these separate estimates of offsite risk.

S.4 Results of the Accident Frequency Analysis

The accident frequency analysis for Sequoyah is documented elsewhere.[2] This section only summarizes the results of the accident frequency analyses since they form the starting point for the analyses that are covered in this volume. Table S.2 lists four summary measures of the core damage frequency distributions for Sequoyah for the seven internally initiated PDSs. The four summary measures are the mean, and the 5th, 50th (median) and 95th percentiles.

The 26 internally initiated PDSs which had mean frequencies above 1.0E-7/R-yr are placed into the seven PDS groups listed in Table S.2. These 26 PDSs account for over 99% of the total mean core damage frequency (MCDF) of 5.6E-5/R-yr. In both SBO groups, offsite power is lost and the diesel generators fail to start and run. In the slow SBO group, the steam-turbine-driven (STD) auxiliary feedwater system (AFWS) operates until the batteries are depleted; in the fast SBO group the STD AFWS fails. In both SBO groups, core degradtion may be arrested before the vessel fails if offsite power is recovered in time. The LOCA PDS group consists of accidents initiated by breaks of all four sizes (A, S_1, S_2, and S_3). In some of the PDSs in this group, the low pressure injection system (LPIS) is operating at the onset of core damage, so the arrest of core degradation before the vessel lower head fails is possible for these PDSs.

Table S.2
Sequoyah Core Damage Frequencies
Internal Initiators

PDS	Core Damage Frequency (1/R-yr) 5%	Median	Mean	95%	% Mean TCD Frequency
1 Slow SBO	1.4E-07	1.6E-06	4.6E-06	1.6E-05	9
2 Fast SBO	5.5E-07	3.8E-06	9.3E-06	3.5E-05	17
3 LOCAs	6.6E-06	2.0E-05	3.5E-05	1.1E-04	63
4 Event V	1.5E-11	2.0E-08	6.5E-07	3.4E-06	1
5 Transient	2.2E-07	1.2E-06	2.3E-06	8.2E-06	4
6 ATWS	4.2E-08	5.0E-07	2.1E-06	8.5E-06	3
7 SGTR	2.2E-08	3.8E-07	1.7E-06	9.4E-06	3
Total	1.5E-05	3.9E-05	5.6E-05	1.6E-04	

Event V is initiated by the failure of two check valves that isolate LPIS piping from the RCS. The check valve failures expose the low pressure piping to full primary system pressure, and it ruptures. The break is outside containment, so the break fails both the RCS and the injection system and bypasses the containment. The transient group consists of two PDSs that have failure of both the AFWS and Feed and Bleed cooling function. Core damage arrest is possible for one of the PDSs if the RCS pressure can be reduced since both LPIS and high pressure injection system (HPIS) are operable. The ATWS group contains three PDSs in which the nuclear reaction is not brought under control at the start of the accident. The two PDSs that comprise the steam generator tube rupture (SGTR) group include one PDS in which the safety relief valves (SRVs) in the secondary system stick open ("H" SGTR), and one PDS in which these SRVs reclose after opening ("G" SGTR).

S.5 Accident Progression Analysis

S.5.1 Description of the Accident Progression Analysis

The accident progression analysis is performed by means of a large and detailed event tree, the APET. This event tree forms a high level model of the accident progression, including the response of the containment to the loads placed upon it. The APET is not meant to be a substitute for detailed, mechanistic computer simulation codes. Rather, it is a framework for integrating the results of these codes together with experimental results and expert judgment. The detailed, mechanistic codes require too much computer time to be run for all the possible accident progression paths. Furthermore, no single available code treats all the important phenomena in a complete and thorough manner that is acceptable to all those knowledgeable in the field. Therefore, the results from these codes, as interpreted by experts, are summarized in an event tree. The resulting APET can be evaluated quickly by computer, so that the full diversity of possible accident progressions can be considered and the uncertainty in the many phenomena involved can be included.

The APET treats the progression of the accident from the onset of core damage through the core-concrete interaction (CCI). It accounts for various events that may lead to the release of fission products due to the accident. The Sequoyah APET consists of 111 questions, most of which have more than two branches. Five time periods are considered in the tree. The recovery of offsite power is considered both before vessel failure as well as after vessel failure. The possibility of arresting the core degradation process before failure of the vessel is explicitly considered. Core damage arrest may occur following the recovery of offsite power or when depressurization of the RCS allows injection by an operating system (HPIS or LPIS) that previously could not function. Containment failure is considered during the time of core degradation (due to hydrogen combustion or detonation), at vessel breach (VB) (due to vessel blowdown, hydrogen combustion, direct containment heating, and steam explosions), after vessel failure (due to hydrogen combustion), and after several days (due to basemat melt-through or eventual overpressure if containment cooling is not restored). Five mechanisms, four of them inadvertent, for depressurizing the vessel before failure are included in the APET.

The APET is so large and complex that it cannot be presented graphically and must be evaluated by computer. A computer code, EVNTRE, has been written for this purpose. In addition to evaluating the APET, EVNTRE sorts the myriad possible paths through the tree into a manageable number of outcomes, denoted APBs.

S.5.2 Results of the Accident Progression Analysis

Results of the accident progression analysis for internal initiators at Sequoyah are summarized in Figures S.2, S.3, and S.4. Figure S.2 shows the mean distribution among the summary APBs for the summary PDS groups. Technically, this figure displays the mean probability of a summary APB conditional on the occurrence of a PDS group. Since only mean values are shown, Figure S.2 gives no indication of the range of values encountered. The distributions of the expected conditional probability for core damage arrest for a given PDS group are shown in Figure S.3. Similarly, the distributions of the expected conditional probability for early containment failure for a given PDS group are displayed in Figure S.4. Early containment failure means one that occurs any time before VB, at VB, or within a few minutes after VB.

Figure S.2 indicates the mean probability of the possible outcomes of the accident progression analysis. The width of each box in the figure indicates how likely each accident progression outcome is for each type of accident. Except for the Bypass initiators, either no failure of the vessel (safe stable state) or no failure of the containment are by far the most likely outcomes for internal initiators.

PLANT DAMAGE STATE
(Mean Core Damage Frequency)

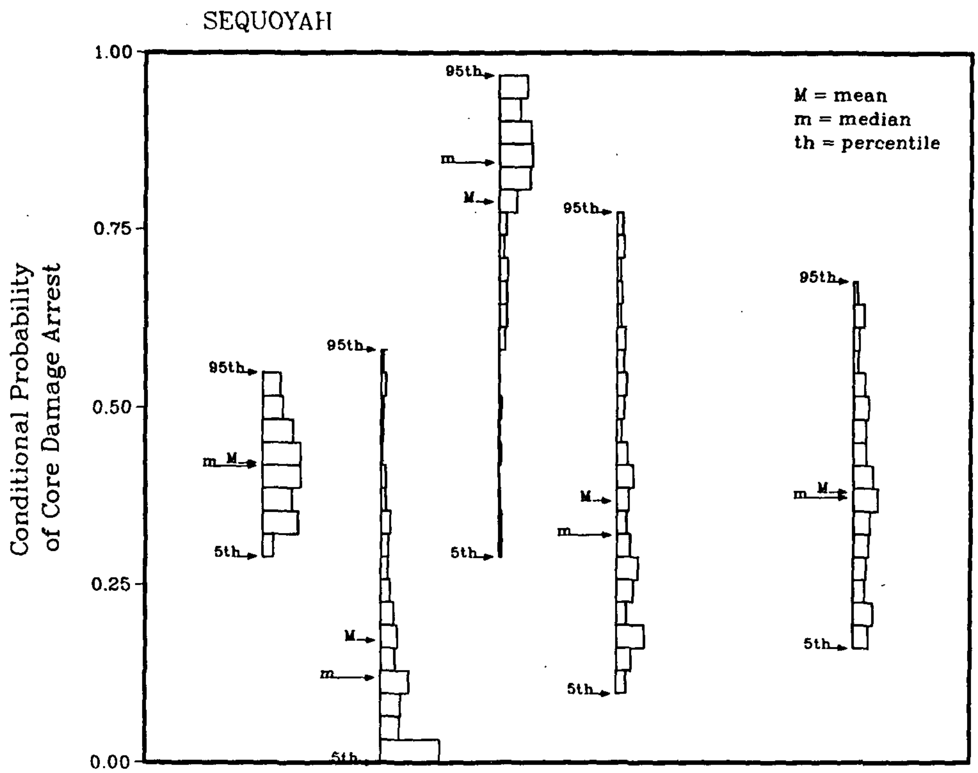
ACCIDENT PROGRESSION BIN	LOSP (1.38E-05)	ATWS (2.07E-06)	Transients (2.32E-06)	LOCAs (3.52E-05)	Bypass (2.39E-06)	Frequency Weighted Average (5.58E-05)
VB, early CF (during CD)	0.014	0.003		0.002		0.005
VB, alpha, early CF (at VB)	0.002	0.003		0.002		0.002
VB > 200 psi, early CF (at VB)	0.064	0.023	0.014	0.031		0.035
VB < 200 psi, early CF (at VB)	0.054	0.020	0.004	0.014		0.023
VB, late CF	0.153	0.001		0.001		0.038
VB, BMT, very late CF	0.065	0.151	0.039	0.260		0.171
Bypass	0.001	0.134	0.006		0.996	0.056
VB, No CF	0.200	0.471	0.137	0.301		0.269
No VB, early CF (during CD)	0.038	0.001	0.005	0.002		0.011
No VB	0.384	0.171	0.785	0.367		0.371

BMT = Basemat Meltthrough
CF = Containment Failure
VB = Vessel Breach
CD = Core Degradation

Sequoyah

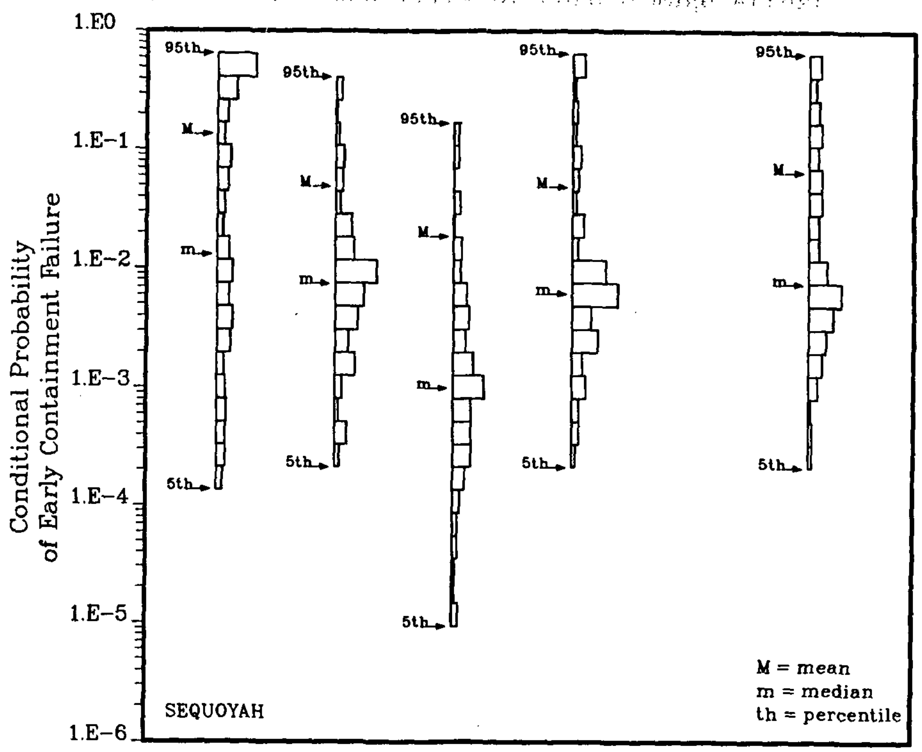
Figure S.2. Mean Probability of APBs for the Summary PDSs

If core damage is not arrested and the accident proceeds to failure of the vessel, Figure S.2 shows that no failure of the containment is the most likely outcome for all types of accidents. If containment failure occurs, early failure (at or before VB) is predicted have a mean probability of about 0.06 and late failure is more likely than early failure, with a mean probability of about 0.20. Late failure may be due to hydrogen ignition some hours after VB, basemat meltthrough (BMT), or eventual overpressure after several days if containment heat removal (CHR) is not restored. Of these three late failure modes, eventual overpressure is the most likely



PDS Group	LOSP	ATWS	Transients	LOCA	Bypass	All
Core Damage Freq.	1.4E-05	2.1E-06	2.3E-06	3.5E-05	2.4E-06	5.6E-05

Figure S.3. Probability of Core Damage Arrest



PDS Group	LOSP	ATWS	Transients	LOCAs	Bypass	F.W.A
Core Damage Freq.	1.4E-05	2.1E-06	2.3E-06	3.5E-05	2.4E-06	5.6E-05

Figure S.4. Probability of Early Containment Failure for internal initiators, because roughly 63% of the total mean core damage frequency is attributed to the LOCA PDS group, in which there is a high probability that the long-term heat removal by the containment spray system fails. The results of this analysis indicate that there is a high likelihood that the reactor cavity will contain water at VB. The presence of a large amount of water inhibits the dispersal of debris from the cavity, thus lowering the threat from direct containment heating at VB. The presence of water also contributes to the probability that core debris released from the vessel will be cooled. If CCI does initiate, the release will be scrubbed by the overlaying pool of water. On the other hand, water in the cavity can increase the possibility of ex-vessel steam explosions which can indirectly threaten the integrity of the containment. Containment failure by ex-vessel steam explosion was investigated in this study and was found to be a minor threat. An ex-vessel steam explosion can also contribute to the radionuclide release at VB.

<u>Core Damage Arrest</u>. It is possible to arrest the core damage process, avoid VB, and achieve a safe, stable state (as at TMI-2) if coolant injection is restored before the core degradation process has gone too far. Recovery of injection is due to one of two events. In the loss of offsite power (LOSP) accidents, recovery of injection follows the restoration of offsite power. In other types of accidents, an injection system is operating when core degradation commences, but no injection is taking place because the RCS pressure is too high. If a break in the RCS pressure boundary allows the RCS pressure to decrease to the point where the operating system can inject, there is some chance of arresting the core degradation process. The probability of arresting core degradation depends on the time the injection starts relative to the state of the core. The RCS failure that allows injection to commence may be an initiating break or a temperature-induced failure that occurs after the onset of core damage such as a break in the hot leg or surge line, the failure of an RCP seal, or the sticking-open of a power-operated relief valve (PORV).

For the internally initiated PDS groups, core damage arrest is possible for all groups except the interfacing systems LOCA, Event V. Offsite power may be recovered for the two SBO groups. Some PDSs in the transients, LOCAs, ATWS, and SGTR groups have LPIS, or LPIS and HPIS operating. The initiating break in the interfacing LOCA fails the LPIS by diverting the flow out the break. Figure S.3 contains no plot for the bypass accidents. Core damage arrest is not possible for Event V and some of the SGTRs. Furthermore, the fission products escape to the environment whether or not the vessel and containment fail. Thus, vessel failure is not of particular interest for the bypass accidents. Figure S.3 indicates that core damage arrest before VB is especially likely for the Transients PDS group. The dominant PDS in this group has both LPIS and HPIS operating at the onset of core damage. The probability of core damage arrest for this group reflects the probability that one of the five means of depressurizing the RCS reduces the RCS to a sufficiently low pressure to allow injection.

Core damage arrest does not necessarily mean that there will be no radionuclide releases during the accident. For accidents in which the containment is not bypassed, both hydrogen and radionuclides are released to the containment during the core damage process. If a large amount of hydrogen is generated during core damage and is subsequently ignited, it is possible that the resulting load will fail the containment.

If the containment fails, a pathway is established for the radionuclides to enter the outside environment. In contrast to the bypass accidents, this radionuclide release is generally small, however, because in the majority of the cases in which VB is averted these releases are scrubbed as they pass through the ice condenser.

<u>RCS Depressurization</u>. The reduction of the RCS pressure in the period between the onset of core damage and VB has two consequences that are important in determining offsite risk. First, pressure reduction may allow the LPIS to function and thus avoid vessel failure in accidents where the LPIS is operable but not injecting due to high RCS pressure. Second, lower RCS pressures at VB reduce the loads placed on the containment structure at that time and reduce the probability of containment failure at VB.

Four of the five means of depressurizing the RCS considered in the Sequoyah accident progression analysis are temperature-induced (T-I) and inadvertent. The five mechanisms are:

1. T-I hot leg or surge line failure;
2. PORVs or SRVs stuck open;
3. T-I RCP seal failure;
4. T-I SGTR; and
5. Deliberate opening of the PORVs by the operators.

T-I failures of the RCP seals and PORVs sticking open are also considered in the accident frequency analysis. Of these five mechanisms, only the first three are effective for most accidents. Distributions for the probability of hot leg failure, SGTR, and RCP seal failure were provided by expert panels. Acting together, the effective means of RCS depressurization in this analysis ensured that only about 10% or less of the accidents that were at the PORV setpoint pressure (about 2500 psi) at the onset of core damage remained at that pressure until the time of lower head failure.

<u>Early Containment Failure</u>. For those accidents in which the containment is not bypassed, the offsite risk depends strongly on the probability that the containment will fail early, i.e., anytime before VB, at VB, or within a few minutes after VB. There are four possibilities for early containment failure:

1. Pre-existing containment leak;
2. Isolation failure;
3. Containment failure before VB due to hydrogen combustion or detonation; and
4. Containment failure at VB due to the events at VB.

The probability of a pre-existing leak or isolation failure at Sequoyah is low, about 0.005. The design pressure of the Sequoyah containment is 11 psig and the assessed mean failure pressure is 65 psig. Because of its somewhat low failure pressure, the Sequoyah containment is susceptible not only to loads from hydrogen deflagrations and detonations but can also be threatened by slow pressurization events that are associated with the accumulation of steam and noncondensibles.

The production of hydrogen during the core damage process and later during VB, should it occur, is a key factor that affects the probability of containment failure. If the hydrogen ignition and air return fan systems are not operating, which is the case in an SBO, the hydrogen will accumulate in the ice condenser and upper plenum of the ice condenser. The lack of steam in these locations allows mixtures to form that have a high hydrogen concentration. Subsequent ignition of this hydrogen by either random sources, by the recovery of ac power, or by mechanisms occurring at VB can result in loads that can threaten the containment.

Hydrogen combustion events are the dominant events that cause early containment failure in the LOSP summary group. The containment is predicted to fail with a mean probability of 0.13 for this group when VB occurs, and with a mean probability of 0.04 when VB does not occur. The LOSP summary group is the only group in which early containment failure occurs without VB with significant probability. For the LOSP group, failures at VB are dominated by HPME/hydrogen events (system pressure greater than 200 psia) with an almost equal contribution from hydrogen burns alone (RCS pressure less than 200 psia). For the ATWS summary group, early containment failure with VB occurs with a mean probability of 0.05, with about equal contribution from hydrogen burns augmented with ex-vessel steam explosion (low system pressure at VB) and HPME/hydrogen events. For the transient summary group, early containment failure is predicted to occur very infrequently, the mean failure probability is about 0.02. For the LOCAs summary group, the containment is predicted to fail early with a mean probability of 0.05, and the failures are dominated by containment failure at VB involving HPME/hydrogen events.

Figure S.4 shows the probability distribution for early containment failure at Sequoyah. The probability distributions displayed in this figure are conditional on core damage. For the bins included in these distributions, VB occurs. For accidents other than Bypass, Figure S.4 shows that the mean probability of early containment failure is about 0.06 and the median is about one order of magnitude lower. If early containment failure without VB is included, the mean is about 0.07. The low failure probability is due to the effectiveness of the RCS depressurization mechanisms, as well as to mitigation of HPME events by deep flooding of the cavity (dispersal of debris from the cavity is inhibited).

S.6 Source Term Analysis

S.6.1 Description of the Source Term Analysis

The source term for a given bin consists of release fractions for the nine radionuclide classes for the early release and for the late release, and additional information about the timing of releases, the energy associated with the releases, and the height of the releases. It consists of information required for calculating consequences in the succeeding analysis. A source term is calculated for each APB for each observation in the sample. The nine radionuclide classes are: inert gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium.

The source term analysis is performed by a relatively small computer code: SEQSOR. The purpose of this code is not to calculate the behavior of the fission products from their chemical and physical properties and the flow and temperature conditions in the reactor and the containment. Instead, SEQSOR provides a means of incorporating into the analysis the results of the more detailed codes that do consider these quantities. This approach is needed because the detailed codes require too many computer resources to compute source terms for the numerous accident progression bins and the 200 observations that result from the sampling approach used in NUREG-1150.

SEQSOR is a fast-running, parametric computer code used to calculate the source terms for each APB for each observation for Sequoyah. As there are typically a few hundred bins for each observation, and 200 observations in the sample, the need for a source term calculation method that requires few computer resources for one evaluation is obvious. SEQSOR provides a framework for synthesizing the results of experiments and mechanistic codes, as interpreted by experts in the field. The reason for "filtering" the detailed code results through the experts is that no code available treats all the phenomena in a manner generally acceptable to those knowledgeable in the field. Thus, the experts extend the code results in areas where the codes are deficient and to judge the applicability of the model predictions. They also factor in the latest experimental results and modify the code results in areas where the codes are known or suspected of oversimplifying. Since the majority of the parameters used to compute the source term are derived from distributions determined by an expert panel, the dependence of SEQSOR on various detailed codes reflects the preferences of the experts on the panel.

It is not possible to perform a separate consequence calculation for each of the approximately 110,000 source terms computed for the Sequoyah integrated risk analysis. Therefore, the interface between the source term analysis and the consequence analysis is formed by grouping the source terms into a much smaller number of source term groups. These groups are defined so that the source terms within them have similar properties, and a single consequence calculation is performed for the mean source term for each group. This grouping of the source terms is performed with the PARTITION program, and the process is referred to as "partitioning."

The partitioning process involves the following steps: definition of an early health effect weight (EH) for each source term, definition of a chronic health effect weight (CH) for each source term, subdivision (partitioning) of the source terms on the basis of EH and CH, a further subdivision on the basis of the time the evacuation starts relative to the start of the release, and calculation of frequency-weighted mean source terms.

The result of the partitioning process is that the source term for each APB is assigned to a source term group. In the risk computations, each APB is represented by the mean source term for the group to which it is assigned, and the consequences calculated for that mean source term.

S.6.2 Results of the Source Term Analysis

When all the internally-initiated accidents at Sequoyah are considered together, the plots shown in Figure S.5 are obtained. These plots show four statistical measures of the 200 curves (one for each observation in the sample) that give the frequencies with which release fractions are exceeded. Figure S.5 summarizes the complementary cumulative distribution functions (CCDFs) for all of the radionuclide groups except for the noble gases. The mean frequency of exceeding a release fraction of 0.10 for iodine is 4×10^{-6}/yr; for cesium, it is 3×10^{-6}/yr; for tellurium, it is 2×10^{-6}/yr; and for strontium and barium, it is 3×10^{-7}/yr. The mean frequency of exceeding a release fraction of 0.01 for the lanthanum radionuclide class is 3×10^{-7}/yr.

S.7 Consequence Analysis

S.7.1 Description of the Consequence Analysis

Offsite consequences are calculated with MACCS for each of the source term groups defined in the partitioning process. MACCS tracks the dispersion of the radioactive material in the atmosphere from the plant and computes its deposition on the ground. MACCS then calculates the effects of this radioactivity on the population and the environment. Doses and the ensuing health effects from 60 radionuclides are computed for the following pathways: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, inhalation of resuspended ground contamination, ingestion of contaminated water and ingestion of contaminated food.

MACCS treats atmospheric dispersion by the use of multiple, straight-line Gaussian plumes. Each plume can have a different direction, duration, and initial radionuclide concentration. Cross-wind dispersion is treated by a multi-step function. Dry and wet deposition are treated as independent processes. The weather variability is treated by means of a stratified sampling process.

For early exposure, the following pathways are considered: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, and inhalation of resuspended ground contamination. For the long-term exposure, MACCS considers following four pathways: groundshine, inhalation of resuspended ground contamination, ingestion of contaminated water and ingestion of contaminated food. The direct exposure pathways, groundshine, and inhalation of resuspended ground contamination, produce doses in the population living in the area surrounding the plant. The indirect exposure pathways, ingestion of contaminated water and food produce doses in those who ingest food or water emanating from the area around the accident site. The contamination of water bodies is estimated for the washoff of land-deposited material as well as direct deposition. The food pathway model includes direct deposition onto the crop species and uptake from the soil.

Both short-term and long-term mitigative measures are modeled in MACCS. Short-term actions include evacuation, sheltering, and emergency relocation from the vicinity of the plant (i.e., relocation may not be restricted to the emergency planing zone). Long-term actions include relocation and restrictions on land use and crops. Relocation and land decontamination,

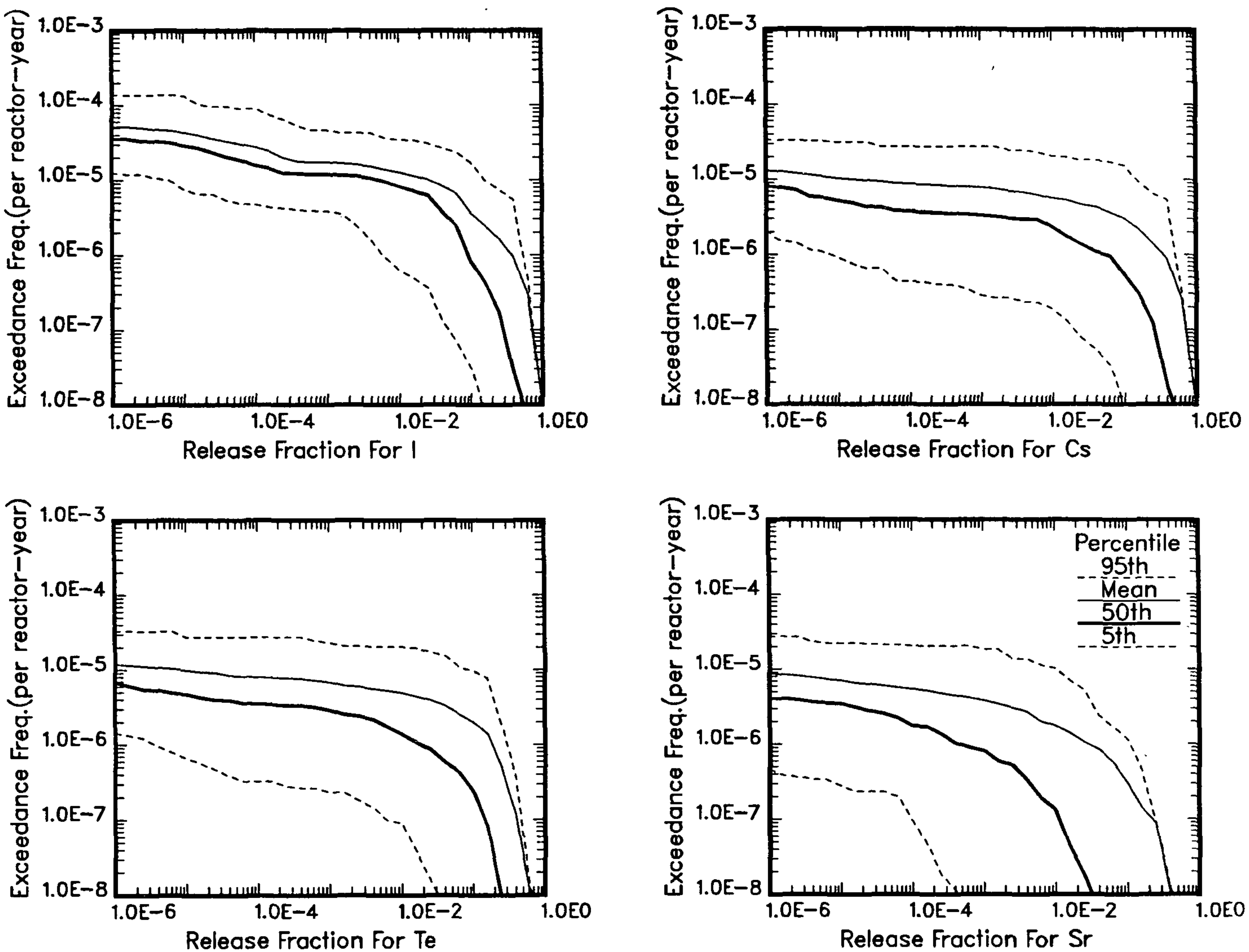


Figure S.5. Exceedance Freuqencies for Release Fractions for Sequoyah: All Internal Initiators

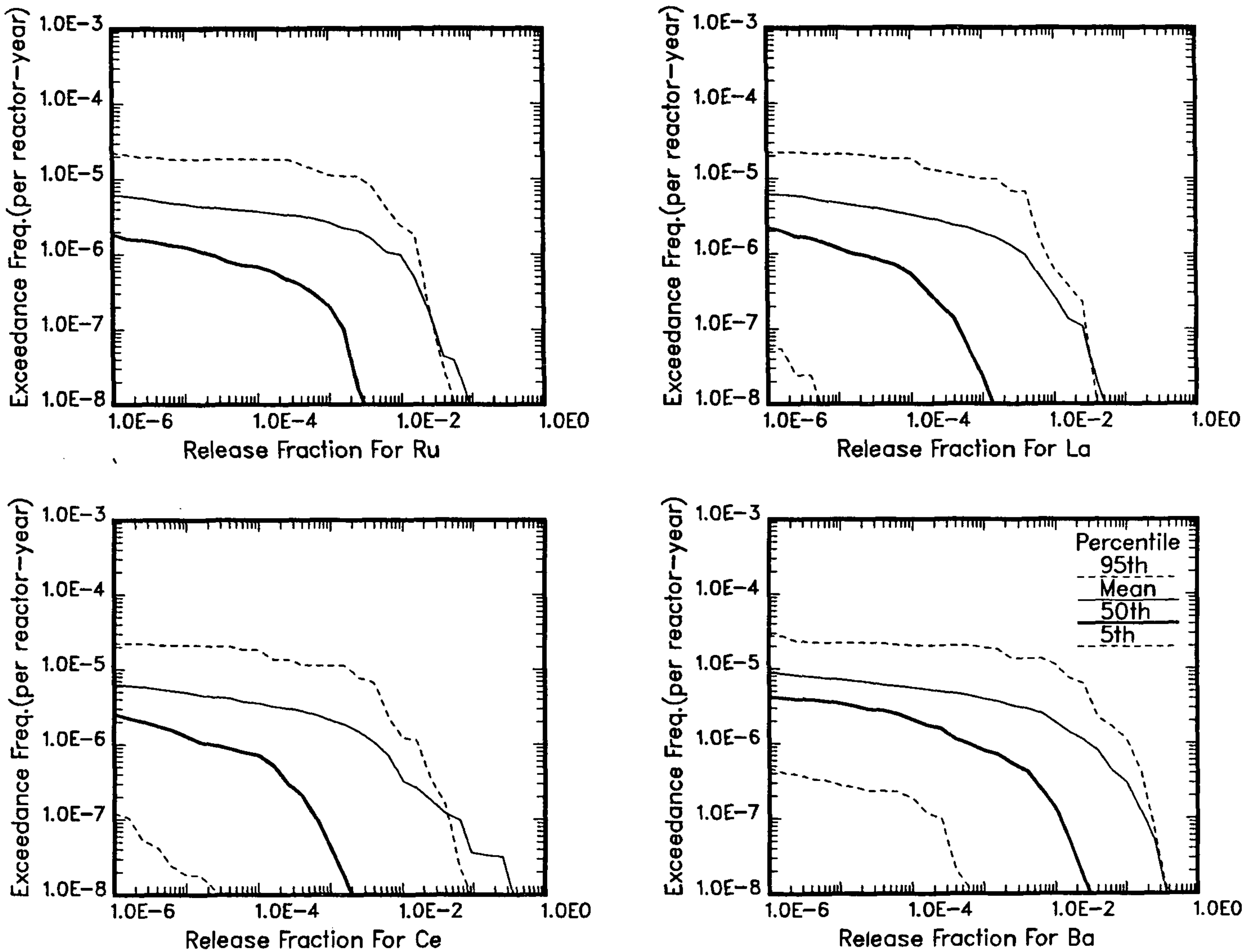


Figure S.5. (continued)

interdiction, and condemnation are based on projected long-term doses from groundshine and the inhalation of resuspended radioactivity. The disposal of agricultural products and the removal of farmland from crop production are based on contamination criteria.

The health effects models link the dose received by an organ to morbidity or mortality. The models used in MACCS calculate both short-term and long-term effects to a number of organs.

Although the variables thought to be the largest contributors to the uncertainty in risk are sampled from distributions in the accident frequency, accident progression, and source term analyses, there is no analogous treatment of uncertainties in the consequence analysis. Variability in the weather is fully accounted for, but the uncertainty in other parameters such as the dry deposition velocity or the evacuation rate is not considered.

The MACCS consequence model calculates a large number of different consequence measures. Results for the following six consequence measures are given in this report: early fatalities, total latent cancer fatalities, population dose within 50 miles, population dose for the entire region, early fatality risk within 1 mile, and latent cancer fatality risk within 10 miles. For NUREG-1150, 99.5% of the population evacuates and 0.5% of the population continues normal activity. For internal initiators at Sequoyah, the evacuation delay time between warning and the beginning of evacuation is 2.3 h.

S.7.2 Results of the Consequence Analysis

The results presented in this section are conditional on the occurrence of a source term group. That is, given that a release takes place, with release fractions and other characteristics as defined by one of the source term groups, then the tables and figures in this section give the consequences expected. This section contains no indication at all about the frequency with which these consequences may be expected. Implicit in the results given in this section are that 0.5% of the population does not evacuate and that there is a 2.3-h delay between the warning to evacuate and the actual start of the evacuation.

CCDFs display the results of the consequence calculation in a compact and complete form. The CCDFs in Figure S.6 for early fatalities and latent cancer fatalities display the relationship between consequence size and consequence frequency due to variability in the weather for each source term group which has a non-zero frequency. Conditional on the occurrence of a release, each of these CCDFs gives the probability that individual consequence values will be exceeded due to the uncertainty in the weather conditions that will exist at the time of an accident. Figure S.6 shows that there is considerable variability in the consequences that is solely due to the weather. There is, of course, considerable variability among the consequences that is due to the size and timing of the release as well.

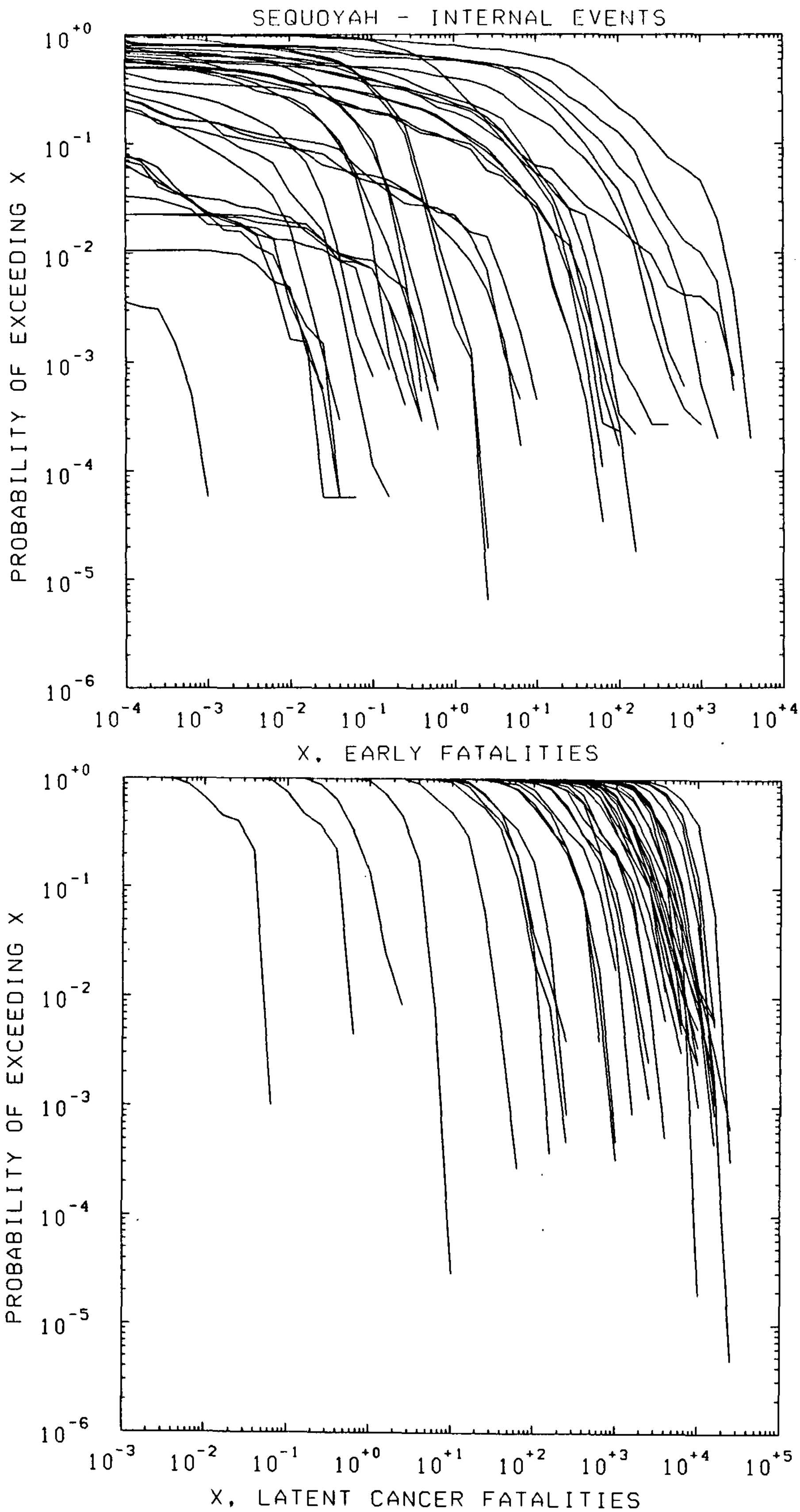


Figure S.6. Consequences Conditional on Source Terms.
Sequoyah: Internal Initiators

S.8 Integrated Risk Analysis

S.8.1 Determination of Risk

Risk is determined by bringing together the results of the four constituent analyses: the accident frequency analysis, the accident progression analysis, the source term analysis, and the consequence analysis. This process is described in general terms in Section S.2 of this summary, and in mathematical terms in Section 1.4 of this volume. Specifically, the accident frequency analysis produces a frequency for each PDS group for each observation, and the accident progression analysis results in a probability for each APB, conditional on the occurrence of the PDS group. The absolute frequency for each bin for each observation is obtained by summing the product of the PDS group frequency for that observation and the conditional probability for the APB for that observation over all the PDS groups.

For each APB for each observation, a source term is calculated; this source term is then assigned to a source term group in the partitioning process. The consequences are then computed for each source term group. The overall result of the source term calculation, the partitioning, and the consequence calculation is that a set of consequence values is identified with each APB for each observation. As the absolute frequency of each APB is known from the accident frequency and accident progression results, both frequency and consequences are known for each APB. The risk analysis assembles and analyzes all these separate estimates of offsite risk.

S.8.2 Results of the Risk Analysis

Measures of Risk. Figure S.7 shows the basic results of the integrated risk analysis for internal initiators at Sequoyah. This figure shows four statistical measures of the families of complementary CCDFs for early fatalities, latent cancer fatalities, individual risk of early fatality within one mile of the site boundary, and individual risk of latent cancer fatality within ten miles of the plant. The CCDFs display the relationship between the frequency of the consequence and the magnitude of the consequence. As there are 200 observations in the sample for Sequoyah, the actual risk results at the most basic level are 200 CCDFs for each consequence measure. Figure S.7 displays the 5th percentile, median, mean, and 95th percentile for these 200 curves, and shows the relationship between the magnitude of the consequence and the frequency at which the consequence is exceeded, as well as the variation in that relationship.

The 5th and 95th percentile curves provide an indication of the spread between observations, which is often large. This spread is due to uncertainty in the sampled variables, and not to differences in the weather at the time of the accident. As the magnitude of the consequence measure increases, the mean curve typically approaches or exceeds the 95th percentile curve. This results when the mean is dominated by a few observations, which often happens for large values of the consequences. Only a few observations have nonzero exceedance frequencies for these large consequences. Taken as a whole, the results in Figure S.7 indicate that large consequences are relatively unlikely to occur.

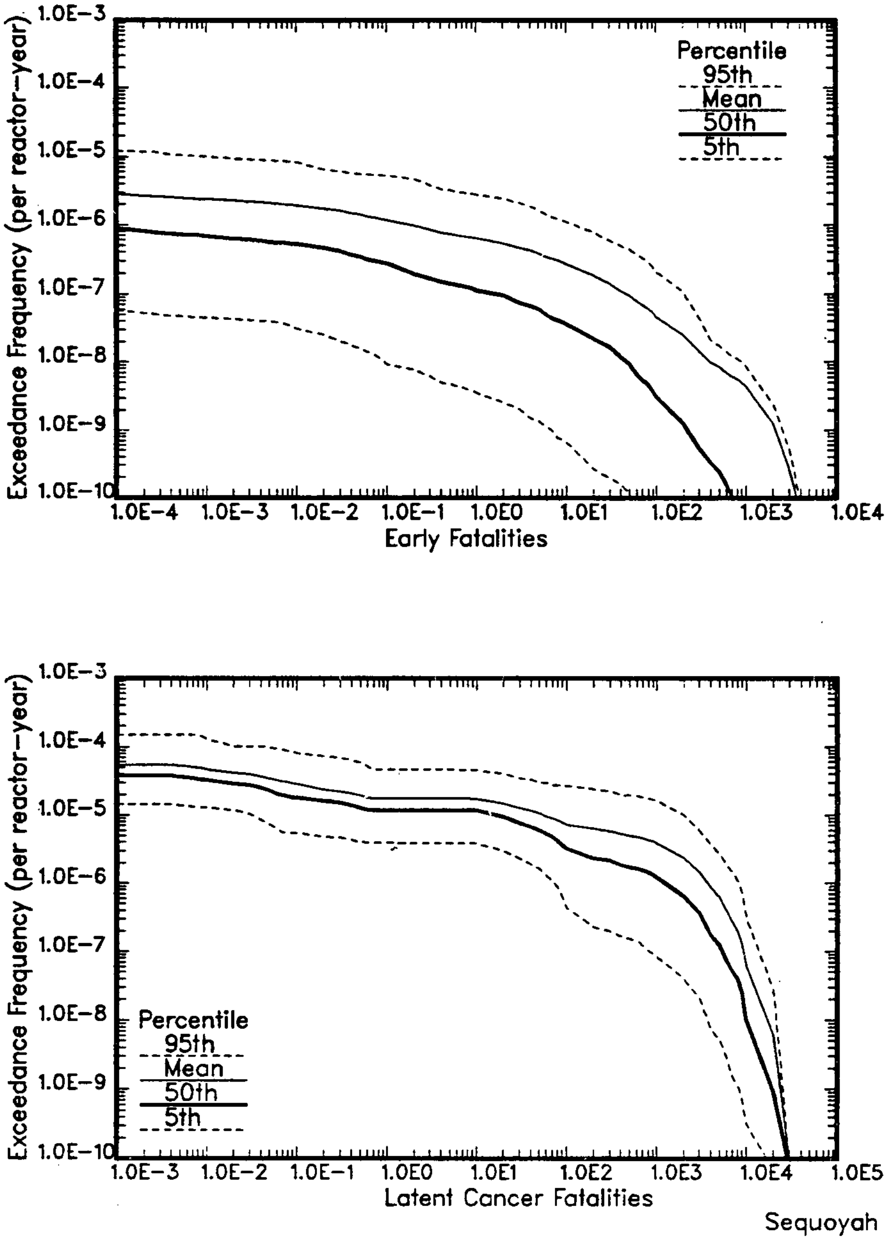
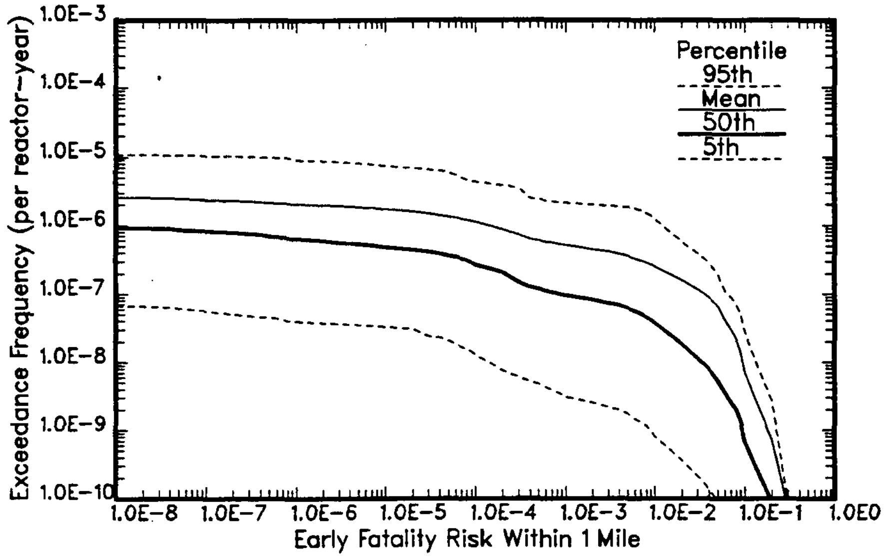


Figure S.7. Exceedance Frequencies for Risk.
Sequoyah: Internal Initiators



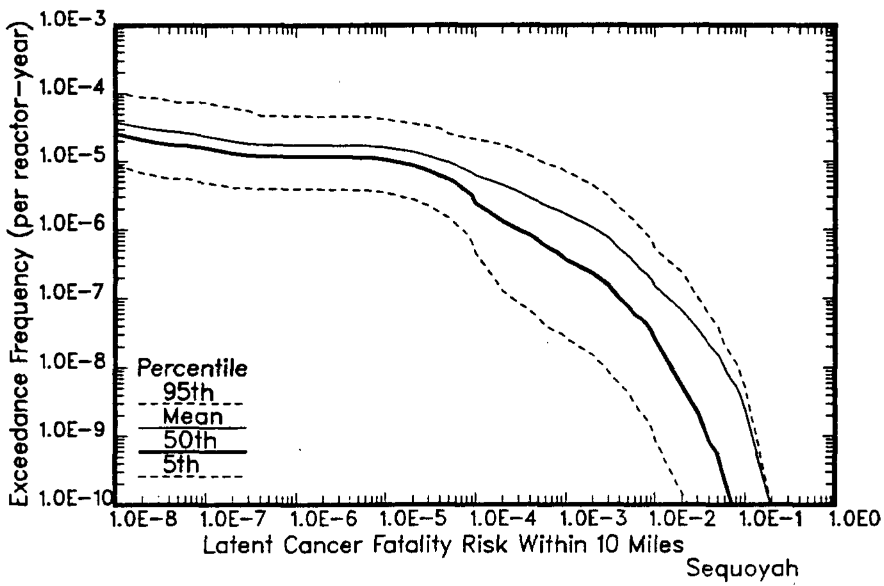
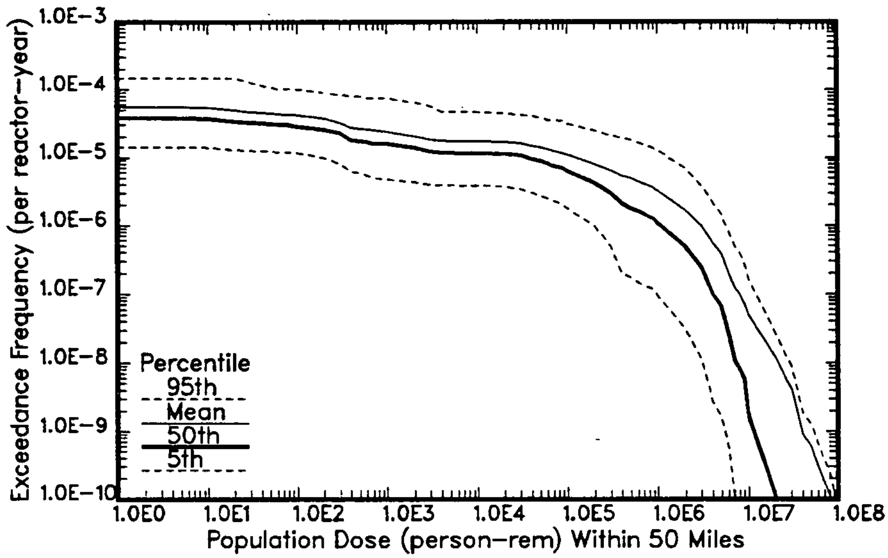


Figure S.7. (continued)



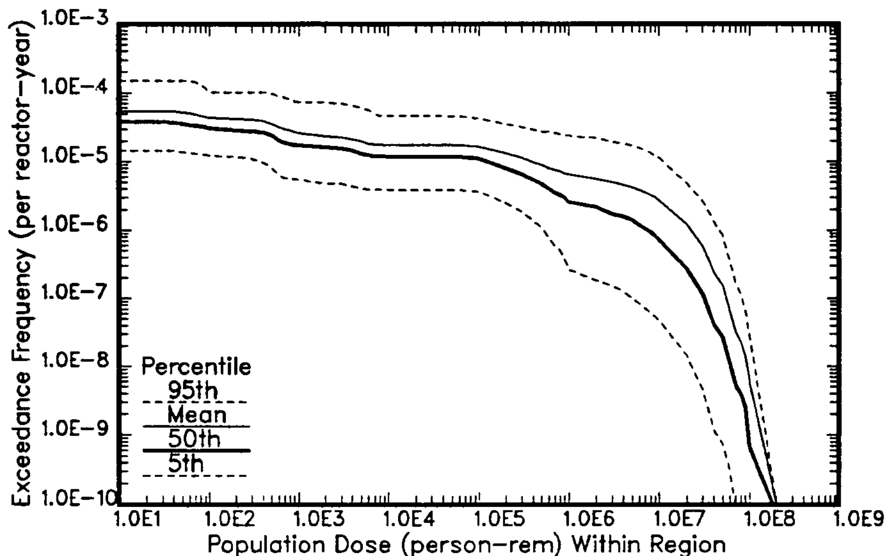


Figure S.7. (continued)

Although the CCDFs convey the most information about the offsite risk, summary measures are also useful. Such a summary value, denoting annual risk, may be determined for each observation in the sample by summing the product of the frequencies and consequences for all the points used to construct the CCDF. This has the effect of averaging over the different weather states as well as over the different types of accidents that can occur. Since the complete analysis consisted of a sample of 200 observations, there are 200 values of annual risk for each consequence measure. These 200 values may be ranked and plotted as histograms, which is done in Figure S.8. The same four statistical measures used above are shown on these plots as well. Note that considerable information has been lost in going from the CCDFs in Figure S.7 to the histograms of annual values in Figure S.8; the relationship between the size of the consequence and its frequency has been sacrificed to obtain a single value for risk for each observation.

The plots in Figure S.8 show the variation in the annual risk for internal initiators for four consequence measures. Where the mean is close to the 95th percentile, a relatively small number of observations dominate the mean value. This is more likely to occur for the early fatality consequence measures than for the latent cancer fatality or population dose consequence measures due to the threshold effect for early fatalities.

The safety goals are written in terms of mean individual fatality risks. The plots in Figure S.8 for individual early fatality risk and individual latent cancer fatality risk show that essentially the entire risk distribution for Sequoyah fall below the safety goals, and the means are well below the safety goals.

A single measure of risk for the entire sample may be obtained by taking the mean value of the distribution for annual risk. This measure of risk is commonly called mean risk, although it is actually the average of the annual risk, or the mean value of the mean risk. Mean risk values for internal initiators for four consequence measures are given in Figure S.8.

S.8.3 Important Contributors to Risk

There are two ways to calculate the contribution to mean risk. The fractional contribution to mean risk (FCMR) is found by dividing the average risk for the subset of interest for the sample by the average total risk for the sample. The mean fractional contribution to risk (MFCR) is found by determining the ratio of the risk for the subset of interest to the total risk for each observation, and then averaging over the sample.

Results of computing the contributions to the mean risk for internal initiators by the two methods are presented in Table S.3. Percentages are shown for early fatalities and latent cancer fatalities for the seven PDS groups.

Pie charts for contributions of the PDS groups to mean risk for internal initiators for these two risk measures for both methods are shown in Figure S.9. Figure S.10 displays similar pie charts for contributions of the summary APBs to mean risk. Not surprisingly, the two methods of calculating contribution to risk yield different values. Both methods of computing the contributions to risk are conceptually valid, so the conclusion is clear: contributors to mean risk can only be interpreted in a very broad sense. That is, it is valid to say that Event V is a major contributor to mean early fatality risk at Sequoyah; it is not valid to state that Event V group contributes 68% of the early fatality risk at Sequoyah.

Table S.3
Two Methods of Calculating Contribution
to Mean Risk

**Contributors (%) to Mean
Early Fatality Risk for Internal Initiators**

PDS Group	FCMR	MFCR
1 Fast SBO	6.9	6.7
2 Slow SBO	16.0	18.2
3 LOCAs	1.7	13.0
4 Event V	68.0	40.5
5 Transients	0.1	1.3
6 ATWS	1.9	6.8
7 SGTRs	5.3	13.5

**Contributors (%) to Mean Latent
Cancer Fatality Risk for Internal Initiators**

PDS Group	FCMR	MFCR
1 Fast SBO	12.5	8.4
2 Slow SBO	28.6	25.4
3 LOCAs	14.2	20.9
4 Event V	10.3	10.0
5 Transients	0.5	1.4
6 ATWS	3.8	5.7
7 SGTRs	30.1	28.1

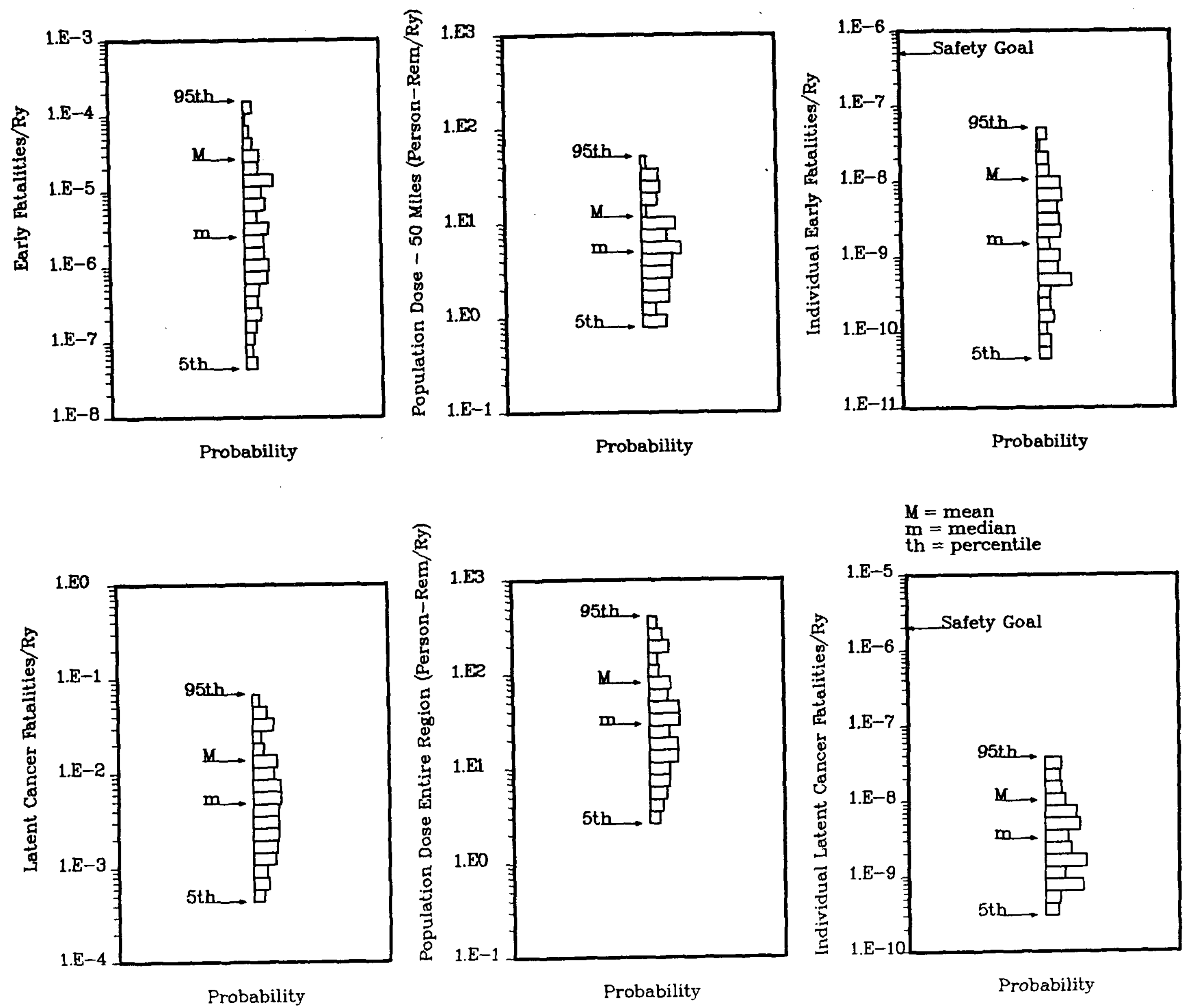


Figure S.8. Distributions of Annual Risk: Sequoyah: All Internal Initiators.

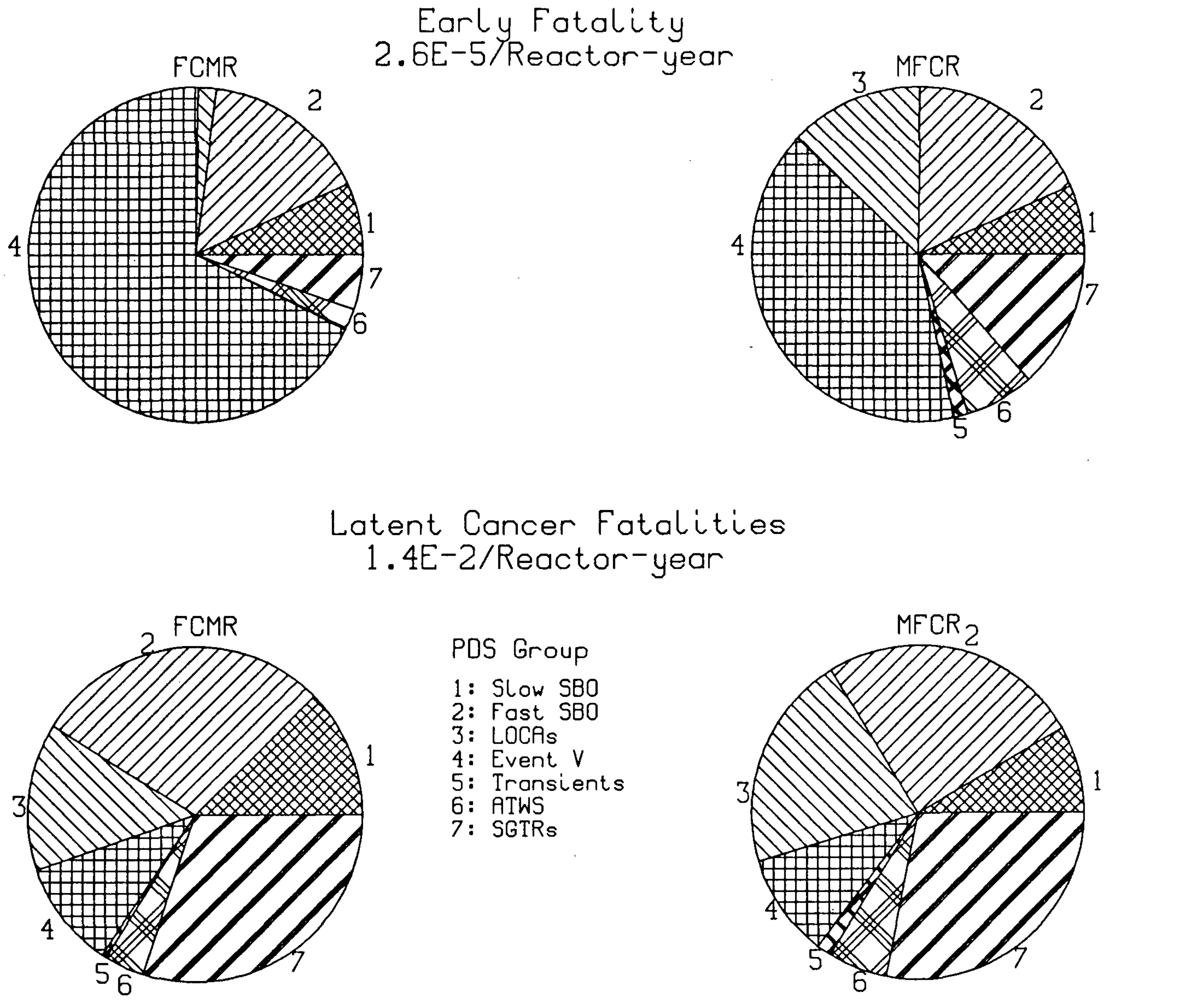


Figure S.9. Fractional PDS Contributions to Annual Risk; Sequoyah: Internal Initiators (MFCR = Mean Fractional Contribution to Risk; FCMR = Fractional Contribution to Mean Risk)

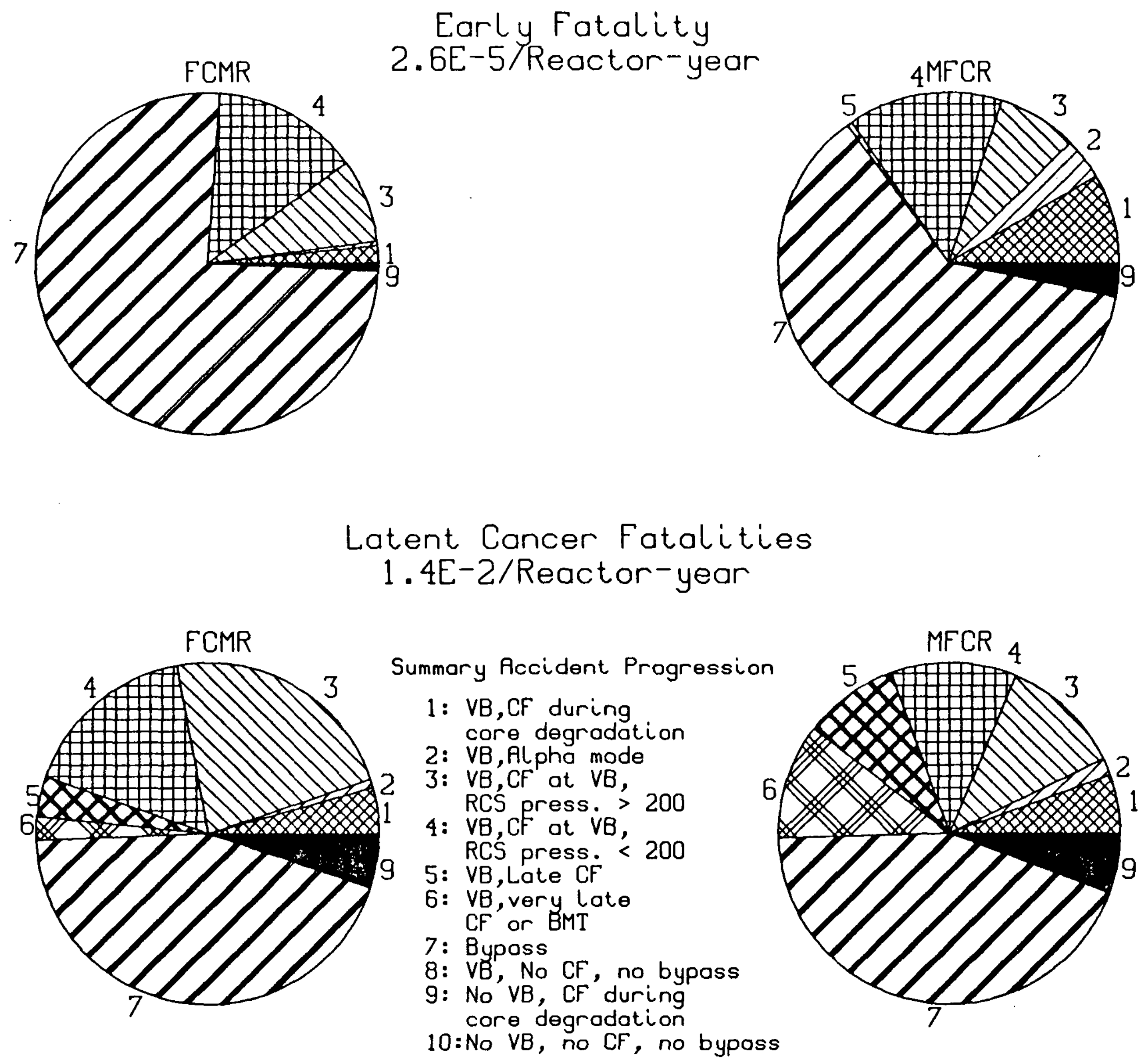


Figure S.10. Fractional APB Contributions to Annual Risk; Sequoyah: Internal Initiators (MFCR = Mean Fractional Contribution to Risk; FCMR = Fractional Contribution to Mean Risk)

Although the exact values are different for each method, the basic conclusions that can be drawn from these results are the same. For early fatalities, which depend on a large early release, the mean risk is dominated by Event V and to a lesser degree, station blackouts. Event V not only proceeds quickly to VB, but it creates a bypass of the containment as well. The blackout accidents are the most likely non-bypass accidents to progress to VB and involve early containment failures. Accidents in which the containment fails late are much less significant.

Latent cancer fatalities and population dose depend primarily on the total amount of radioactivity released. Thus, unlike early fatality risk, the timing of containment failure is not particularly important for this risk measure. However, if the containment fails late, there is more residence time in containment for the radionuclides to deposit by mitigative systems (sprays, ice condenser) and natural mechanisms before containment failure, than there is when early containment failure occurs. The mean latent cancer fatality risk and mean population dose are dominated by station blackouts, SGTRs, and LOCAs. For station blackouts and LOCAs, the early failures of containment dominate the contributions, with less contribution from the later failures. The SGTR accidents contribute more toward latent cancer fatalities than they do toward early fatalities because the dominant SGTR sequences with the higher releases are very lengthy accidents. Thus, evacuation occurs before the release has begun.

S.8.4 Important Contributors to the Uncertainty in Risk

The important contributors to the uncertainty in risk are determined by performing regression-based sensitivity analyses for the mean values for risk. The regression analyses for internally initiated events for early fatalities and individual risk of early fatality within 1 mile only account for about 50% of the observed variability. The independent variables that account for this variability are those that determine the frequency and the magnitude of an early release. The regression analysis for the other four consequence measures is somewhat less successful, as it is able to account for only 30% of the variability. The independent variables that account for this variability are predominantly those variables that determine the frequencies of the accident.

Because the regression results for all internal events do not account for much of the variability, the same type of stepwise regression analysis was performed for each PDS group for the consequences of early fatalities and latent cancer fatalities. The most robust results are exhibited for bypass accidents, PDS Groups 4 and 7, and to a lesser degree, for the anticipated transient without scram (ATWS) accidents, PDS Group 6. For PDS Group 4, Event V, more than 95% of the variability is explained for each consequence: at least 90% is accounted for by the initiating event frequency of check valve failure in one of the LPIS trains, the remainder involves the probability that the releases are scrubbed by fire sprays and the decontamination factor associated with the sprays. For PDS Group 7, SGTRs, about 80% is explained: the variables involved include the release fraction from the vessel to the environment, the initiating event frequency for SGTRs, and the fraction of the fission products released from the core to the vessel. The bypass accidents lend themselves best to analysis with a linear regression model, because the consequences are directly related to a product of several variables.

For the ATWS PDS group much of the risk is associated with the PDS that involves an SGTR. For this group, 65% of the variability is explained for early fatalities, and 86% for latent cancers. The variables involved include the same as mentioned for SGTR, as well as the probability of failure to effect manual scram due to operator error and the probability of failure of automatic insertion of control rods.

For the SBO, LOCA, and Transient PDS Groups, less than 60% of the variability is explained for both early fatalities and latent cancer fatalities. The models involved with these PDS Groups are more complex and nonlinear than for the bypass accidents, and different variables come into play for different degrees of consequences. Some of the variables that are involved with explaining the variability in the early and latent cancer fatality risks for these PDS Groups include: the containment failure pressure, the pressure rise in containment at VB, the fraction of core involved in HPME, and the decontamination factor of the ice condenser.

S.9 Insights and Conclusions

Core Damage Arrest. The inclusion of the possibility of arresting the core degradation process before vessel failure is an important feature of this analysis. For internal initiators, there is a good chance that non-bypass accidents will be arrested before vessel failure. This may be due to the recovery of offsite power or the reduction of RCS pressure to the point where an operable system can inject. The arrest of core damage before VB plays an important part in reducing the risk due to the most frequent types of internal accidents: LOCAs and SBOs.

Depressurization of the RCS. Depressurization of the RCS before the vessel fails is important in reducing the loads placed upon the containment at VB and in arresting core damage before VB. For accidents in which the RCS is at the PORV setpoint pressure during core degradation, the effective mechanisms for pressure reduction are T-I failure of the hot leg or surge line, T-I failure of the RCP seals, and the sticking open of the PORVs. All of these mechanisms are inadvertent and beyond the control of the operators. The apparent beneficial effects of reducing the pressure in the RCS when lower head failure is imminent indicate that further investigation of depressurization may be warranted. The dependency of the probability of containment failure on RCS pressure boundary failures that occur at unpredictable locations and at unpredictable times is somewhat unsettling. Studies of the effects of increasing PORV capacity, providing the means to open the PORVs in blackout situations, and changing the procedures to remove restrictive conditions on deliberate RCS pressure reduction might decrease the probability of early containment failure at PWRs. Depressurization may involve the loss of considerable inventory from the RCS. Any studies undertaken should consider possible drawbacks as well as benefits.

Containment Failure. If a core damage accident proceeds to the point where the lower head of the reactor vessel fails, the containment is not likely to fail at this time. This is partially due to the depressurization of the

RCS before vessel failure, partially due to deep-flooding of the reactor cavity which inhibits dispersal of core debris from the cavity in high pressure accidents, and partially due to the strength of the Sequoyah containment relative to the loads expected. Hydrogen burns before VB for the SBO accidents and hydrogen burn/DCH events are the factors that lead to early containment failures when they do occur. Early containment failures contribute significantly to the risks that depend on a large early release (early fatalities), and are major contributors to the risks that are functions of the total release (latent cancer fatalities and population dose). For SBOs, late failures occur from hydrogen burns upon power recovery during CCI. Very late failures that are many hours after VB depend upon the availability of CHR. If CHR is recovered within a day or so, BMT is the most probable failure mode. If CHR is not recovered, an overpressure failure within a day or two after the start of the accident is the likely mode.

Bypass Accidents. Bypass accidents are major contributors to the risks that depend on a large early release as well as those which are functions of the total release. Event V is the accident most likely to result in a large, early release for internal initiators. SGTRs are also important contributors to large releases, but most of the large releases due to SGTRs occur many hours after the start of the accident, and thus they contribute significantly to the risks that depend on the total release. The most important SGTRs are those in which the SRVs on the secondary system stick open. Although the bypass accidents are not the most frequent types of internal accidents, the somewhat low probability of containment failure, especially early containment failure, for the non-bypass accidents results in the large contributions of the bypass accidents to risk.

Fission Product Releases. There is considerable uncertainty in the release fractions for all types of accidents. There are several features of the Sequoyah plant that tend to mitigate the release. First, the in-vessel releases are generally directed to the ice condenser where they experience some decontamination. If the sprays are operating, the radionuclides will also contribute to the decontamination of the releases. The reactor cavity pool also offers a mechanism for reducing the release of radionuclides from CCI. The largest releases tend to occur when the containment is bypassed, or when early failure of containment involving catastrophic rupture occurs. Catastrophic rupture is assumed to cause bypass of the ice condenser and failure of the containment sprays.

Uncertainty in Risk. Considerable uncertainty is associated with the risk estimates produced in this analysis. The largest contributors to the uncertainty in early fatalities and latent cancer fatalities for the bypass sequences are the variability in the frequencies of the initiating events and the uncertainty in some of the parameters that determine the magnitude of the fission product release to the environment. For non-bypass accidents, the variability in frequencies of the initiating events and the uncertainty in the accident progression parameters and probabilities contribute to the uncertainty in latent cancers. The contribution to the uncertainty in early fatalities for non-bypass accidents arises from variability in all the constituent analyses that were incorporated into the uncertainty analysis: initiating events, accident progression, and fission product release.

Comparison with the Safety Goals. For both the individual risk of early fatality within one mile of the site boundary and the individual risk of latent cancer fatality within 10 miles, the mean annual risk and the 95th percentile for annual risk fall more than an order of magnitude below the safety goals. Indeed, even the maximum of the 200 values that make up the annual risk distributions fall well below the safety goals.

References

1. USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants", NUREG-1150, June, 1989.

2. R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency from Internal Events: Sequoyah, Unit 1," Sandia National Laboratories, NUREG/CR-4550, Vol. 5, SAND86-2084, 1989.

1. INTRODUCTION

The United States Nuclear Regulatory Commission (NRC) has recently completed a major study to provide a current characterization of severe accident risks from light water reactors (LWRs). The characterization was derived from the analysis of five plants. The report of that work, NUREG-1150[1] has recently been issued as a second draft for comment. NUREG-1150 is based on extensive investigations by NRC contractors. Several series of reports document these analyses as discussed in the Foreword.

These risk assessments can generally be characterized as consisting of four analysis steps, an integration step, and an uncertainty analysis step.

1. Accident frequency analysis: the determination of the likelihood and nature of accidents that result in the onset of core damage.

2. Accident progression analysis: an investigation of the core damage process, both within the reactor vessel before it fails and in the containment afterwards, and the resultant impact on the containment.

3. Source term analysis: an estimation of the radionuclide transport within the reactor coolant system (RCS) and the containment, and the magnitude of the subsequent releases to the environment.

4. Consequence analysis: the calculation of the offsite consequences, primarily in terms of health effects in the general population.

5. Risk integration: the combination of the outputs of the previous tasks into an overall expression of risk.

6. Uncertainty analysis: the propagation of uncertainties through the first three analyses above, and the determination of which of these uncertainties contribute the most to the uncertainty in risk.

This volume is one of seven that comprise NUREG/CR-4551. NUREG/CR-4551 presents the details of the last five of the six analyses listed above. The subject matter starts with the onset of core damage and concludes with an integrated estimate of overall risk and uncertainty in risk. This volume, Volume 5, describes the inputs used in these analyses and the results obtained for Sequoyah Power Station, Unit 1. The methods used in these analyses are described in detail in Volume 1 of this report and are only briefly discussed here.

1.1 Background and Objectives of NUREG-1150

Assessment of risk from the operation of nuclear power plants, involves determination of the likelihood of various accident sequences and their potential offsite consequences. In 1975, the NRC completed the first comprehensive study of the probabilities and consequences of core meltdown accidents--the "Reactor Safety Study" (RSS).[2] This report showed that the probabilities of such accidents were higher than previously believed, but that the consequences were significantly lower. The product of probability and consequence--a measure of the risk of core melt accidents--was estimated to be quite low when compared with natural events such as floods and earthquakes and with other societal risks such as automobile and airplane accidents. Since that time, many risk assessments of specific plants have been performed. In general, each of these has progressively reflected at least some of the advances that have been made in reactor safety and in the ability to predict the frequency of several accidents, the amount of radioactive material released as a result of such accidents, and the offsite consequences of such a release.

In order to investigate the significance of more recent developments in a comprehensive fashion, it was concluded that the current efforts of research programs being sponsored by the NRC should be coalesced to produce an updated representation of risk for operating nuclear power plants. "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants"[1] is the result of this program. The five nuclear power plants are Surry, Peach Bottom, Sequoyah, Grand Gulf, and Zion. The analyses of the first four plants were performed by Sandia National Laboratories (SNL). The analyis of Zion was performed by Idaho National Engineering Laboratory (INEL) and Brookhaven National Laboratory (BNL).

The overall objectives of the NUREG-1150 program are:

1. Provide a current assessment of the severe accident risks to the public from five nuclear power plants, which will:

    a. Provide a "snapshot" of the risks reflecting plant design and operational characteristics, related failure data, and severe accident phenomenological information extant in 1988;

    b. Update the estimates of the NRC's 1975 risk assessment, the "Reactor Safety Study";[2]

    c. Include quantitative estimates of risk uncertainty, in response to the principal criticism of the "Reactor Safety Study;" and

    d. Identify plant-specific risk vulnerabilities, in the context of the NRC's individual plant examination process.

2. Summarize the perspectives gained in performing these risk analyses, with respect to:

    a. Issues significant to severe accident frequencies, consequences, and risk;

    b. Uncertainties for which the risk is significant and which may merit further research; and

    c. Potential for risk reduction.

3. Provide a set of methods for the prioritization of potential safety issues and related research.

These objectives required special considerations in the selection and development of the analysis methods. This report describes those special considerations and the solutions implemented in the analyses supporting NUREG-1150.

1.2 Overview of Sequoyah Power Station, Unit 1

The subject of the analyses reported in this volume is the Sequoyah Power Station, Unit 1. It is operated by the Tennessee Valley Authority (TVA) and is located on the west shore of the Chickamauga Lake in southeastern Tennessee, about 10 miles northeast of Chattanooga, Tennessee. Two units are located on the site; Unit 2 is essentially identical to Unit 1.

The nuclear reactor of Sequoyah Unit 1 is a 1148 MWe pressurized water reactor (PWR) designed and built by Westinghouse. The reactor coolant system (RCS) has four U-tube steam generators (SGs) and four reactor coolant pumps (RCPs). The containment and the balance of the plant were designed and built by the utility, TVA. Unit 1 began commercial operation in 1981.

There are four diesel generators (DGs) at the Sequoyah site to supply emergency ac power if offsite power from the grid is lost. Two of these DGs are dedicated to Unit 1, and two are dedicated to Unit 2. Each unit has its own set of batteries to supply general emergency dc power. Each DG obtains starting power from a separate set of batteries.

The auxiliary feedwater system (AFWS) has three pumps: two are driven by electric motors; the third is driven by a steam turbine. The AFWS takes suction from the condensate storage tank (CST). There are two charging pumps and two safety injection pumps; together, the charging system and the safety injection system (SIS) perform the high pressure injection (HPI) functions. There are two low pressure injection (LPI) pumps. Both the high pressure injection system (HPIS) and the low pressure injection system (LPIS) can function in a recirculation mode as well as in an injection mode. In the injection mode they take suction from the refueling water storage tank (RWST); in the recirculation mode the LPI pumps take suction from the sump, and the HPIS uses the LPIS as a fluid source.

Sequoyah also has four cold leg accumulators to provide immediate, high-flow, low-pressure injection. RCS overpressure protection is provided by three-code safety relief valves (SRVs) and two power-operated relief valves (PORVs). The component cooling water (CCW) system that provides cooling for the reactor coolant pump (RCP) seals and other ECCS equipment has five pumps for the two units. Service water is provided to both units by eight self-cooled pumps.

The Sequoyah containment is a free-standing steel cylinder with a hemispherical dome. A concrete shield building surrounds the containment and provides radiation shielding, as well as protection from the elements and external missiles. Figure 1.1 shows a section through the Sequoyah containment. The volume is 1.2 million ft^3, and the design pressure is 10.8 psig.

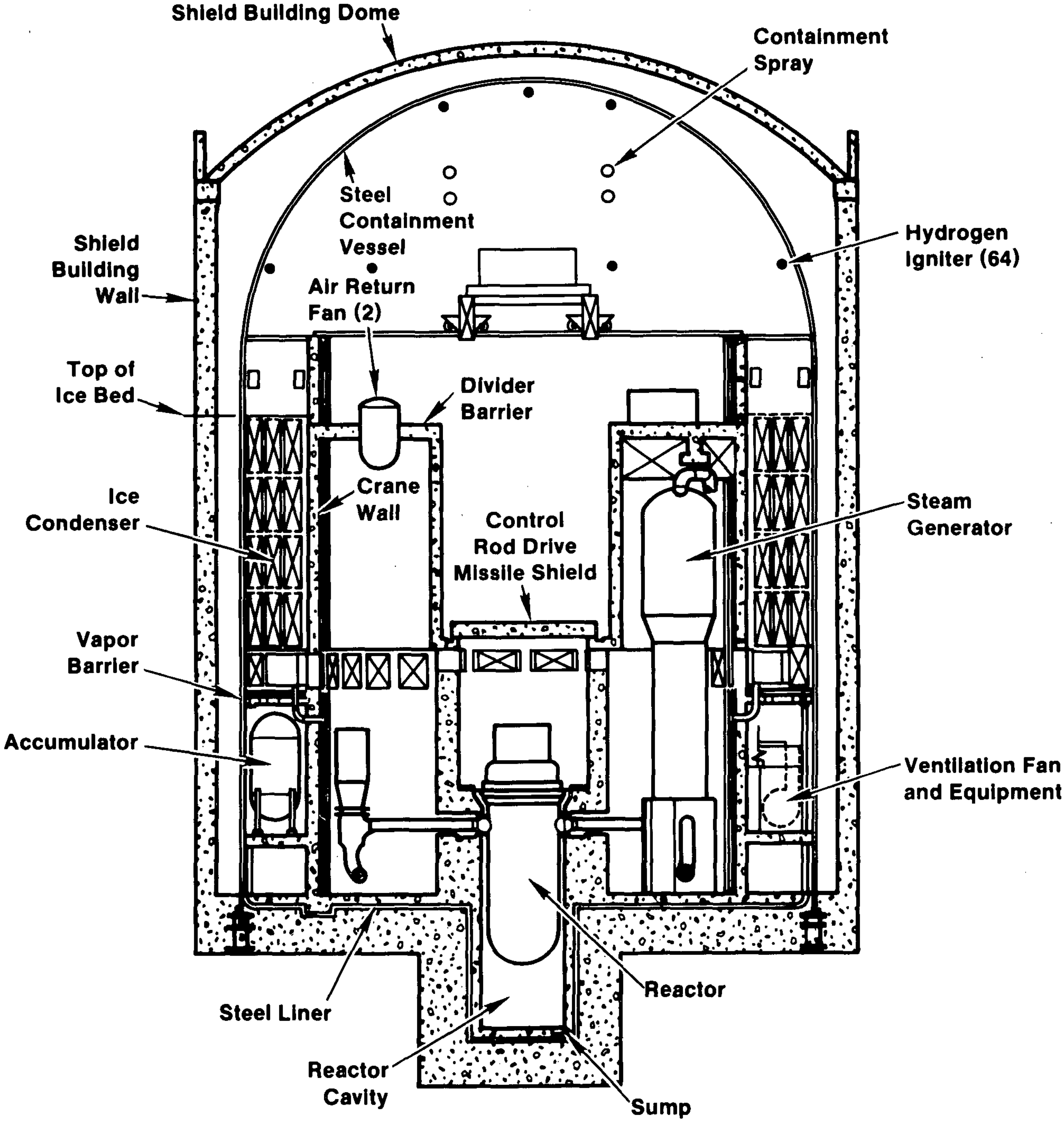


Figure 1.1. Section of the Sequoyah Containment

Pressure suppression during accident conditions is provided passively by the ice condenser system (ICS). Blowdown steam from the RCS is directed through the ice condenser (IC), thus reducing the containment pressure. Long-term emergency containment heat removal is by spray systems. The containment spray system (CSS) has two pumps which take suction from the RWST in injection and from the sump in recirculation.

There is no connection between the sump and the reactor cavity at a low elevation in the Sequoyah containment. Water from a pipe break in containment or from ice melt will flow to the sump. The reactor cavity will remain dry unless the water that has accumulated on the lower containment floor is enough for overflow into the cavity. This requires injection of the RWST contents into containment and melting of about one-quarter of the ice.

There is an air return fan (ARF) system at Sequoyah, in which two fans provide mixing of the containment atmosphere and ensure that gas displaced into the upper conatinment by the blowdown steam is returned rapidly to the lower containment. The hydrogen injection system (HIS) is provided to help preclude large hydrogen burns by burning relatively small quantities of hydrogen as it is produced.

More detail on the features of the plant that are important to the progression of the accident and the performance of the containment is contained in Section 2.1 of this volume.

1.3 Changes Since the Draft Report

The Sequoyah analyses for the February 1987 draft of NUREG-1150 were presented in Volume 2 of the original "Draft for Comment" versions of NUREG/CR-4551 and NUREG/CR-4700, published in April 1987. The analyses performed for NUREG-1150, Second Draft for Peer Review, June 1989, and reported in this volume, are new. While they build on the previous analyses and the basic approach is the same, very little from the first analyses is used directly in these analyses. This section presents the major differences between the two analyses. Essentially, the accident progression analysis and the source term analysis were redone to incorporate new information and to take advantage of expanded methods and analysis capabilities.

Quantification. A major change since the previous analyses is the expert elicitation process used to quantify variables and parameters thought to be large contributors to the uncertainty in risk. This process was used both for the accident progression analysis and the source term analysis. The sizes of the panels were expanded, with each panel containing experts from industry and academia in addition to experts from NRC contractors. The number of issues addressed was also increased to about 30. Separate panels of experts were convened for In-Vessel Processes, Containment Loads, Containment Structural Response, Molten Core-Containment Interactions (MCCI), and Source Term Issues.

To ensure that expert opinion was obtained in a manner consistent with the state of the art in this area, specialists in the process of obtaining expert judgments in an unbiased fashion were involved in designing the elicitation process, explaining it to the experts, and training them in the methods used. The experts were given several months between the meeting at which the problem was defined and the meeting at which their opinions were elicited so that they could review the literature, discuss the problem with colleagues, and perform independent analyses. The results of the elicitation of each expert were carefully recorded, and the reasoning of each expert and the process by which their individual conclusions were aggregated into the final distribution are thoroughly documented.

Accident Progression Analysis. Not only was a substantial fraction of the Accident Progression Event Tree (APET) for Sequoyah rewritten for this analysis, but the capabilities of EVNTRE, the code that evaluates the APET, were considerably expanded. The major improvements to EVNTRE were the ability to utilize user functions and the ability to treat continuous distributions. A user function is a FORTRAN subprogram which is linked with the EVNTRE code. When referenced in the APET, the user function is evaluated to perform calculations too complex to be handled directly in the APET. In the current Sequoyah APET, the user function is called to: compute the amount and distribution of hydrogen in containment during the various time periods; compute the concentration and the flammability of the atmosphere in the containment during the various time periods; calculate the pressure rise due to hydrogen burns and adjust the amounts of gases consumed in the burns accordingly; and determine whether the containment fails and the mode of failure. These problems were handled in a much simpler fashion in the previous analysis.

The event tree used for the analysis for the 1987 draft of NUREG-1150 could only treat discrete distributions. In the analysis reported here continuous distributions are used. Use of continuous distributions removes a significant constraint from the expert elicitations and eliminates any errors introduced by discrete levels in the previous analysis.

The event tree that forms the basis of this analysis was modified to address new issues and to incorporate new information. Thus, not only was the structure of the tree changed but new information was used to quantify the tree. A major modification was the way hydrogen combustion events were modeled and quantified. The amount of hydrogen in the containment is tracked throughout the accident. The probability of ignition, the probability of detonation, and the loads from a combustion event are all a function of the hydrogen concentration. In the current APET, loads are assigned to both deflagrations and detonations. These loads are then compared to the structural capacity of the containment to determine whether it fails or not and the mode of failure.

Another major modification to the APET was consideration of offsite electric power recovery during core degradation, i.e., between uncovering of the top of active fuel (TAF) and vessel breach (VB). This led to a significant portion of the station blackout (SBO) accidents terminating not with VB, but in an arrested core damage state similar to TMI-2. Additional means of depressurizing the RCS are now in the event tree. These additional mechanisms, along with the higher probabilities for some of them that resulted from the expert elicitations, mean that the likelihood is small that an accident that is at full system pressure at the onset of core damage will still be at that pressure when the vessel fails. Accidents in which core damage begins with LPIS, or both LPIS and HPIS operating are treated in the current APET whereas they were omitted in the previous version. If an event occurs to reduce the RCS pressure in these situations, core damage may be arrested before the vessel fails, leading, by another path, to an arrested core damage state similar to that of TMI-2.

Another change in the accident progression analysis is in the binning or grouping of the results of evaluating the APET. In the first analysis, all results were placed in one of about 20 previously defined bins. There were many pathways through the tree that did not fit well into these previously defined bins. For the current analysis, a flexible bin structure, defined by the characteristics important to the subsequent source term analysis was used. This eliminates a major problem in the original analysis process.

Source Term Analysis. While the basic parametric approach used in the original version of SEQSOR, the code used to compute source terms, has been retained in the present version of SEQSOR, the code has been completely rewritten with a different orientation. The previous version was designed primarily to produce results that could be compared directly with the results of the source term code package (STCP). Discrete values for the parameters that differed from those that produced results close to STCP results were then used in the sampling process, with the probabilities for each value or level determined by a small panel of experts. Thus, the first version of SEQSOR determined uncertainty in the amount of fission products released for the limited number of predefined bins from the STCP as a base.

The current version of SEQSOR is quite different. First, it is not tied to the STCP in any way. It was recognized before the new version was developed that most of the parameters would come from continuous distributions defined by an expert panel. Thus, the current version does not rely on results from the STCP or any other specific code. The experts used the results of one or more codes to derive their distributions, but SEQSOR itself merely combines the parameters defined by the expert panel. Furthermore, SEQSOR now treats any consistent accident progression state defined by 14 characteristics that constitute an accident progression bin (APB) for Sequoyah. It is not limited to a small number of pre-defined bins as it was in the original version.

Finally, a new method to group the source terms computed by SEQSOR has been devised. A source term is calculated for each accident progression bin (APB) for each observation in the sample. As a result, there are too many source terms to perform a consequence calculation for each and the source terms have to be grouped before the consequence calculations are performed. The "clustering" method used in the previous analysis was somewhat subjective and not as reproducible as desired. The new "partitioning" scheme developed for grouping the source terms in this analysis eliminates these problems.

Consequence Analysis. The consequence analysis for the current NUREG-1150 does not differ so markedly from that for the previous version of NUREG-

1150 as do the accident progression analysis and the source term analysis. Version 1.4 of MACCS was used for the original analysis, while Version 1.5 is used for this analysis. The major difference between the two versions is in the data used in the lung model. Version 1.4 used the lung data contained in the original version of "Health Effects Models for Nuclear Power Plant Accident Consequence Analysis",[3] whereas Version 1.5 of MACCS uses the lung data from Revision 1 (1989) of this report.[4] Other changes were made to the structure of the code in the transition from 1.4 to 1.5, but the effects of these changes on the consequence values calculated are small.

Another difference in the consequence calculation is that the NRC specified evacuation of 99.5% of the population in the evacuation area for this analysis, as compared with the previous analysis in which 95% of the population was evacuated.

Risk Analysis. The risk analysis combines the results of the accident frequency analysis, the accident progression analysis, the source term analysis, and the consequence analysis to obtain estimates of risk to the offsite population and the uncertainty in those estimates. This combination of the results of the constituent analyses was performed essentially the same way for both the previous and the current analyses. The only differences are in the number of variables sampled and the number of observations in the sample.

1.4 Structure of the Analysis

The NUREG-1150 analysis of the Sequoyah plant is a Level 3 probabilistic risk assessment composed of four constituent analyses:

1. Accident frequency analysis, which estimates the frequency of core damage for all significant initiating events;

2. Accident progression analysis, which determines the possible ways in which an accident could evolve given core damage;

3. Source term analysis, which estimates the source terms (i.e., environmental releases) for specific accident conditions; and

4. Consequence analysis, which estimates the health and economic impacts of the individual source terms.

Each of these analyses is a substantial undertaking. By carefully defining the interfaces between these individual analyses, the transfer of information is facilitated. At the completion of each constituent analysis, intermediate results are generated for presentation and interpretation. An overview of the assembly of these components into an integrated analysis is shown in Figure 1.2.

The NUREG-1150 plant studies are fully integrated probabilistic risk assessments in the sense that calculations leading to both risk and uncertainty in risk are carried through all four components of the individual plant studies. The frequency of the initiating event, the conditional

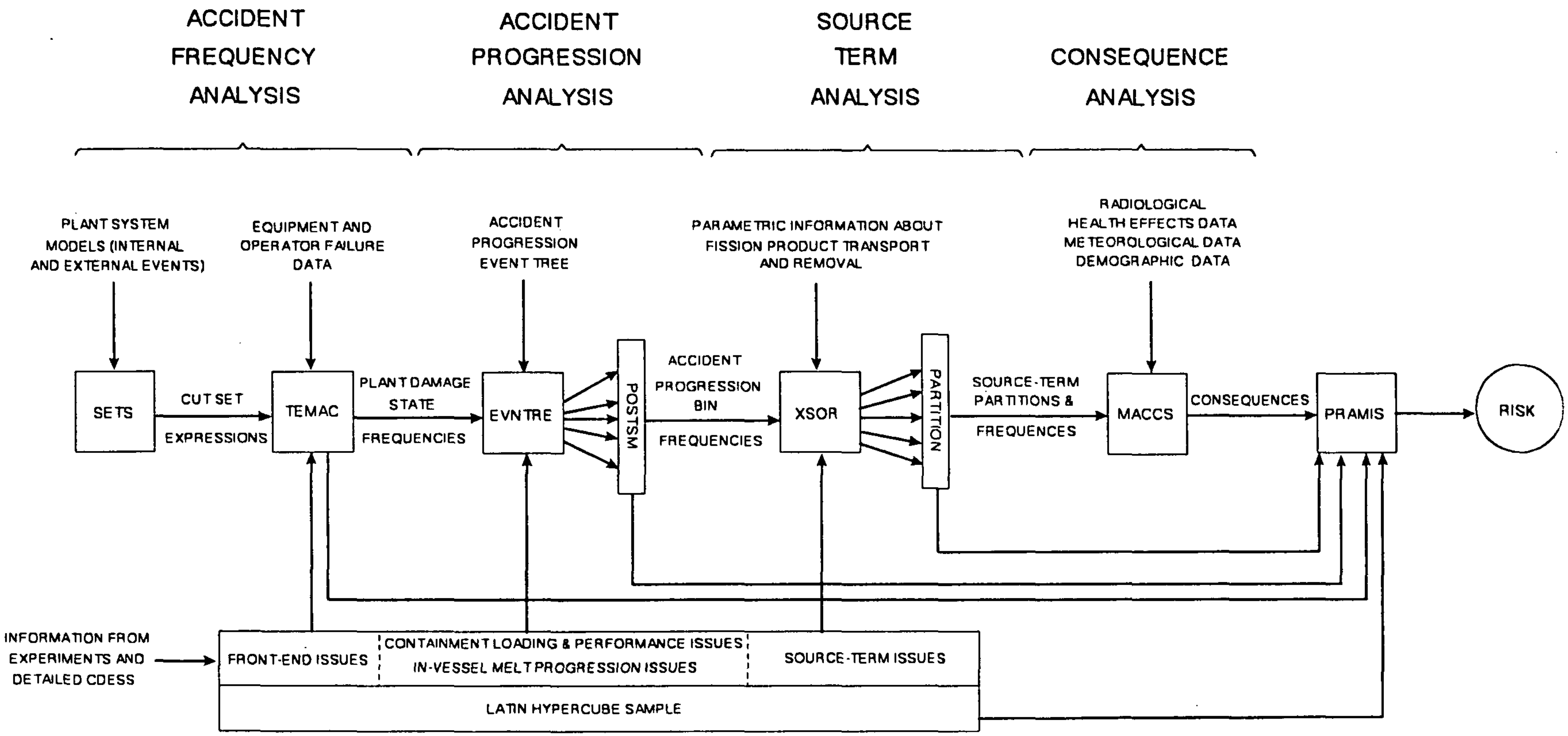


Figure 1.2. Overview of Integrated Plant Analysis in NUREG-1150

probability of the paths leading to the consequence, and the value of the consequence itself can then be combined to obtain a risk measure. Measures of uncertainty in risk are obtained by repeating the calculation just indicated many times with different values for important parameters. This provides a distribution of risk estimates that is a measure of the uncertainty in risk.

It is important to recognize that a probabilistic risk assessment is a procedure for assembling and organizing information from many sources; the models actually used in the computational framework of a probabilistic risk assessment serve to organize this information, and as a result, are rarely as detailed as most of the models that are actually used in the original generation of this information. To capture the uncertainties, the first three of the four constituent analyses use all available sources of information for each analysis component, including past observational data, experimental data, mechanistic modeling and, as appropriate or necessary, expert judgment. This requires the use of relatively quick running models to assemble and manipulate the data developed for each analysis.

To facilitate both the conceptual description and the computational implementation of the NUREG-1150 analyses, a matrix representation[5,6] is used to show how the overall integrated analysis fits together and how the progression of an accident can be traced from initiating event to offsite consequences.

Accident Frequency Analysis. The accident frequency analysis uses event tree and fault tree techniques to investigate the manner in which various initiating events can lead to core damage. In initial detailed analyses, the SETS program[7] combines experimental data, past observational data and modeling results into estimates of core damage frequency. The ultimate outcome of the initial accident frequency analysis for each plant is a group of minimal cut sets that lead to core damage. Detailed descriptions of the systems analyses for the individual plants are available elsewhere.[8,9,10,11,12] For the final integrated NUREG-1150 analysis for each plant, the group of risk-significant minimal cut sets is used as the systems model. In the integrated analysis, the TEMAC program[13,14] is used to evaluate the minimal cut sets. The minimal cut sets themselves are grouped into PDSs, where all minimal cut sets in a PDS provide a similar set of conditions for the subsequent accident progression analysis. Thus, the PDSs form the interface between the accident frequency analysis and the accident progression analysis.

With use of the transition matrix notation, the accident progression analysis may be represented by

$$\mathrm{fPDS} = \mathrm{fIE}\ P(\mathrm{IE} \rightarrow \mathrm{PDS}), \qquad \text{(Eq. 1.1)}$$

where **fPDS** is the vector of frequencies for the PDSs, **fIE** is the vector of frequencies for the initiating events, and **P(IE→PDS)** is the matrix of transition probabilities from initiating events to the PDSs. Specifically:

$\mathbf{fIE} = [fIE_1, \ldots, fIE_{nIE}]$,
fIE_i = frequency (yr^{-1}) for initiating event i,
nIE = number of initiating events,
$\mathbf{fPDS} = [fPDS_1, \ldots, fPDS_{nPDS}]$,
$fPDS_j$ = frequency (yr^{-1}) for PDS j,
nPDS = number of PDSs,

$$\mathbf{P(IE \rightarrow PDS)} = \begin{bmatrix} pPDS_{11} & \cdots & pPDS_{1,nPDS} \\ \vdots & & \vdots \\ pPDS_{nIE,1} & \cdots & pPDS_{nIE,nPDS} \end{bmatrix}$$

and

$pPDS_{ij}$ = probability that initiating event i will lead to PDS j.

The elements $pPDS_{ij}$ of **P(IE→PDS)** are conditional probabilities: given that initiating event i has occurred, $pPDS_{ij}$ is the probability that PDS j will also occur. The elements of **P(IE→PDS)** are determined by the analysis of the minimal cut sets with the TEMAC program. In turn, both the cut sets and the data used in their analysis come from earlier studies that draw on many sources of information. Thus, although the elements $pPDS_{ij}$ of **P(IE→PDS)** are represented as though they are single numbers, in practice these elements are functions of the many sources of information that went into the accident frequency analysis.

Accident Progression Analysis. The accident progression analysis uses event tree techniques to determine the possible ways in which an accident might evolve from each PDS. Specifically, a single event tree is developed for each plant and evaluated with the EVNTRE computer program.[15] The definition of each PDS provides enough information to define the initial conditions for the APET analysis. Due to the large number of questions in the Sequoyah APET and the fact that many of these questions have more than two outcomes, there are far too many paths through each tree to permit their individual consideration in subsequent source term and consequence analysis. Therefore, the paths through the trees are grouped into APBs, where each bin is a group of paths through the event tree that define a similar set of conditions for source term analysis. The properties of each APB define the initial conditions for the estimation of the source term.

Past observations, experimental data, mechanistic code calculations, and expert judgment were used in the development and parameterization of the model for accident progression that is embodied in the APET. The transition matrix representation for the accident progression analysis is

$$\mathbf{fAPB} = \mathbf{fPDS}\ \mathbf{P(PDS \rightarrow APB)} \qquad \text{(Eq. 1.2)}$$

where **fPDS** is the vector of frequencies for the PDSs defined in Eq. 1.1, **fAPB** is the vector of frequencies for the APBs, and **P(PDS→APB)** is the matrix of transition probabilities from PDSs to APBs. Specifically:

fAPB = $[fAPB_1, \ldots, fAPB_{nAPB}]$,

$fAPB_k$ = frequency (yr^{-1}) for accident progression bin k,

nAPB = number of APBs,

$$\mathbf{P(PDS \rightarrow APB)} = \begin{bmatrix} pAPB_{11} & \ldots & pAPB_{1,nAPB} \\ \vdots & & \vdots \\ pAPB_{nPDS,1} & \ldots & pAPB_{nPDS,nAPB} \end{bmatrix}$$

and

$pAPB_{jk}$ = probability that PDS j will lead to APB k.

The properties of **fPDS** are given in conjunction with Eq. 1.1. The elements $pAPB_{jk}$ of **P(PDS→APB)** are determined in the accident progression analysis by evaluating the APET with EVNTRE for each PDS group.

Source Term Analysis. The source terms are calculated for each APB with a non-zero conditional probability by a fast-running parametric computer code entitled SEQSOR. SEQSOR is not a detailed mechanistic model and is not designed to simulate the fission product transport, physics, and chemistry from first principles. Instead, SEQSOR integrates the results of many detailed codes and the conclusions of many experts. The experts, in turn, based many of their conclusions on the results of calculations with codes such as the source term code package,[16,17] MELCOR, and MAAP. Most of the parameters utilized calculating the fission product release fractions in SEQSOR are sampled from distributions provided by an expert panel. Because of the large number of APBs, use of fast-executing code like SEQSOR is absolutely necessary.

The number of APBs for which source terms are calculated is so large that it was not practical to perform a consequence calculation for every source term. That is, the consequence code, MACCS,[18,19,20] required so much computer time to calculate the consequences of a source term that the source terms had to be combined into source term groups. Each source term group is a collection of source terms that result in similar consequences. The frequency of the source term group is the sum of the frequencies of all the APBs which make up the group. The process of determining which APBs go to which source term group is denoted partitioning. It involves considering the potential of each source term group to cause early fatalities and latent cancer fatalities. Partitioning is a complex process; it is discussed in detail in Volume 1 of this report and in the User's Guide for the PARTITION Program.[21]

The transition matrix representation of the source term calculation and the grouping process is

$$\mathbf{fSTG} = \mathbf{fAPB}\ \mathbf{P(APB{\rightarrow}STG)} \qquad \text{(Eq. 1.3)}$$

where **fAPB** is the vector of frequencies for the APBs defined in Eq. 1.2, **fSTG** is the vector of frequencies for the source term groups, and **P(APB→STG)** is the matrix of transition probabilities from APBs to source term groups. Specifically,

$$\mathbf{fSTG} = [fSTG_1, \ldots, fSTG_{nSTG}],$$

$$fSTG_\ell = \text{frequency } (yr^{-1}) \text{ for source term group } \ell,$$

$$nSTG = \text{number of source term groups},$$

$$\mathbf{P(APB{\rightarrow}STG)} = \begin{bmatrix} pSTG_{11} & \cdots & pSTG_{1,nSTG} \\ \vdots & & \vdots \\ pSTG_{nAPB,1} & \cdots & pSTG_{nAPB,nSTG} \end{bmatrix}$$

and

$$pSTG_{k\ell} = \text{probability that APB k will be assigned to source term group } \ell.$$

$$= \begin{cases} 1 & \text{if APB k is assigned to source term group } \ell \\ 0 & \text{otherwise.} \end{cases}$$

The properties of **fAPB** are given in conjunction with Eq. 1.2. Note that the source terms themselves do not appear in Eq. 1.4. The source terms are used only to assign an APB to a source term group. The consequences for each APB are computed from the average source term for the group to which the APB has been assigned.

Consequence Analysis. The consequence analysis is performed for each source term group by the MACCS program. The results for each source term group include estimates for both mean consequences and distributions of consequences. When these consequence results are combined with the frequencies for the source term groups, overall measures of risk are obtained. The consequence analysis differs from the preceding three constituent analyses in that uncertainties are not explicitly treated in the consequence analysis. That is, important values and parameters are determined from distributions by a sampling process in the accident frequency analysis, the accident progression analysis, and the source term analysis. This is not the case for the consequences in the analyses performed for NUREG-1150.

In the transition matrix notation, the risk may be expressed by

$$\mathbf{rC} = \mathbf{fSTG}\ \mathbf{cSTG} \qquad \text{(Eq. 1.4)}$$

where **fSTG** is the vector of frequencies for the source term groups defined in Eq. 1.3, **rC** is the vector of risk measures, and **cSTG** is the matrix of mean consequence measures conditional on the occurrence of individual source term groups. Specifically,

$$\mathbf{rC} = [rC_1, \ldots, rC_{nC}],$$

rC_m = risk (consequence/yr) for consequence measure m,

nC = number of consequence measures,

$$\mathbf{cSTG} = \begin{bmatrix} cSTG_{11} & \ldots & cSTG_{1,nC} \\ \vdots & & \vdots \\ cSTG_{nSTG,1} & \ldots & cSTG_{nSTG,nC} \end{bmatrix}$$

and

$cSTG_{\ell m}$ = mean value (over weather) of consequence measure m conditional on the occurrence of source term group ℓ.

The properties of **fSTG** are given in conjunction with Eq. 1.3. The elements $cSTG_{\ell m}$ of **cSTG** are determined from consequence calculations with MACCS for individual source term groups.

<u>Computation of Risk</u>. Equations 1.1 through 1.4 can be combined to obtain the following expression for risk:

$$\mathbf{rC} = \mathbf{fIE}\ \mathbf{P(IE{\rightarrow}PDS)}\ \mathbf{P(PDS{\rightarrow}APB)}\ \mathbf{P(APB{\rightarrow}STG)}\ \mathbf{cSTG}. \qquad \text{(Eq. 1.5)}$$

This equation shows how each of the constituent analyses enters into the calculation of risk, starting from the frequencies of the initiating events and ending with the calculation of consequences. Evaluation of the expression in Eq. 1.5 is performed with the PRAMIS[22] and RISQUE codes.

The description of the complete risk calculation so far has focused on the computation of mean risk (consequences/year) because doing so makes the overall structure of the NUREG-1150 PRAs more easy to comprehend. The mean risk results are derived from the frequency of the initiating events, the conditional probabilities of the many ways that each accident may evolve and the probability of occurrence for each type of weather sequence at the time of an accident. The mean risk, then, is a summary risk measure.

More information is conveyed when distributions for consequence values are displayed. The form typically used for this is the complementary cumulative distribution function (CCDF). CCDFs are defined by pairs of values

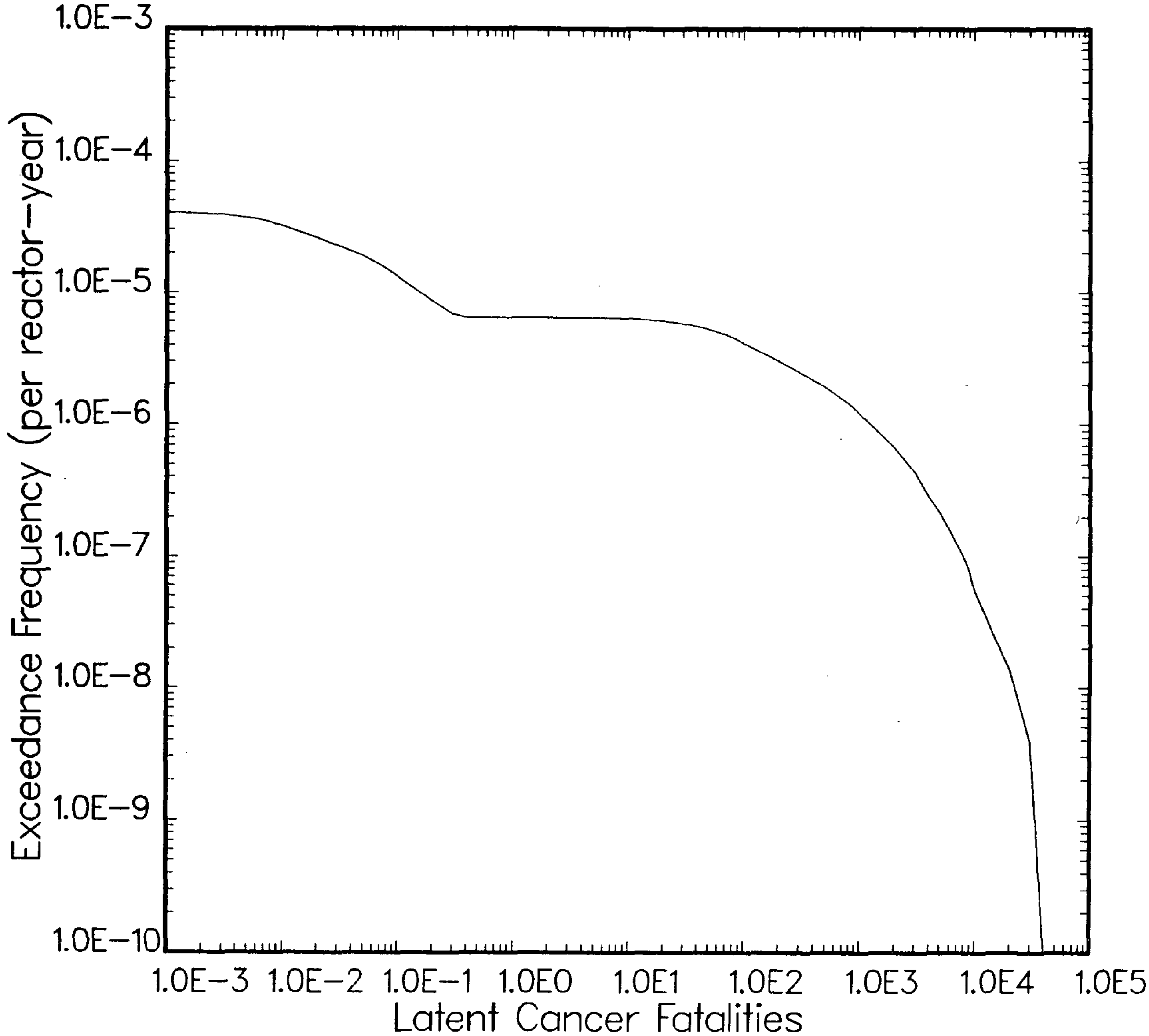


Figure 1.3. Example Risk CCDF

(c,f), where c is a consequence value and the f is the frequency with which c is exceeded. Figure 1.3 is an example of a CCDF. The construction of CCDFs is described in Volume 1 of this report. Each mean risk result is the outcome from reducing a curve of the form shown in Figure 1.3 to a single value. While the mean risk results are often useful for summaries or high-level comparisons, the CCDF is the more basic measure of risk because it displays the relationship between the size of the consequence and frequency exceedance. The nature of this relationship, i.e., that high consequence events are much less likely than low consequence events is lost when mean risk results alone are reported. This report uses both mean risk and CCDFs to report the risk results.

<u>Propagation of Uncertainty through the Analysis</u>. The integrated NUREG-1150 analyses use Monte Carlo procedures as a basis for both uncertainty and the sensitivity analysis. This approach utilizes a sequence:

$$X_1,\ X_2,\ \ldots,\ X_{nV} \qquad \text{(Eq. 1.6)}$$

of potentially important variables, where nV is the number of variables selected for consideration. Most of these variables were considered by a panel of experts representing the NRC and its contractors, the academic world, and the nuclear industry. For each variable treated in this manner, two to six experts considered all the information at their disposal and provided a distribution for the variable. Formal decision analysis techniques[23] (also in Volume 2 of this report) were used to obtain and record each expert's conclusions and to aggregate the assessments of the individual panel members into summary distribution for the variable. Thus, a sequence of distributions

$$D_1,\ D_2,\ \ldots,\ D_{nV}, \qquad \text{(Eq. 1.7)}$$

is obtained, where D_i is the distribution assigned to variable X_i.

From these distributions, a stratified Monte Carlo technique, Latin Hypercube Sampling,[24,25] is used to obtain the variable values that will actually be propagated through the integrated analysis. The result of generating a sample from the variables in Eq. 1.6 with the distributions in Eq. 1.7 is a sequence

$$S_i = [X_{i1},\ X_{i2},\ \ldots,\ X_{i,nV}],\ i = 1,\ 2,\ \ldots,\ nLHS, \qquad \text{(Eq. 1.8)}$$

of sample elements, where X_{ij} is the value for variable X_j in sample element i and nLHS is the number of elements in the sample. The expression in Eq. 1.5 is then determined for each element of the sample. This creates a sequence of results of the form

$$rC_i = fIE_i\ P_i(IE\rightarrow PDS)\ P_i(PDS\rightarrow APB)\ P_i(APB\rightarrow STG)\ cSTG, \qquad \text{(Eq. 1.9)}$$

where the subscript i is used to denote the evaluation of the expression in Eq. 1.5 with the i^{th} sample element in Eq. 1.8. The uncertainty and sensitivity analyses in NUREG-1150 are based on the calculations summarized in

Eq. 1.9. Since **P(IE→PDS)**, **P(PDS→APB)** and **P(APB→STG)** are based on results obtained with TEMAC, EVNTRE and SEQSOR, determination of the expression in Eq. 1.9 requires a separate evaluation of the cut sets, the APET, and the source term model for each element or observation in the sample. The matrix **cSTG** in Eq. 1.9 is not subscripted because the NUREG-1150 analyses do not include consequence modeling uncertainty other than the stochastic variability due to weather conditions.

1.5 Organization of this Report

This report is published in seven volumes as described briefly in the Foreword. Volume 1 of NUREG/CR-4551 describes the methods used in the accident progression analysis, the source term analysis, and the consequence analysis, in addition to presenting the methods used to assemble the results of these constituent analyses to determine risk and the uncertainty in risk. Volume 2 describes the results of convening expert panels to determine distributions for the variables thought to be the most important contributors to uncertainty in risk. Panels were formed to consider in-vessel processes, loads to the containment, containment structural response, molten CCIS, and source term issues. In addition to documenting the results of these panels for about 30 important parameters, Volume 2 includes supporting material used by these panels and presents the results of distributions that were determined by other means.

Volumes 3 through 6 present the results of the accident progression analysis, the source term analysis, and the consequence analysis, and the combined risk results for Surry, Peach Bottom, Sequoyah, and Grand Gulf, respectively. These analyses were performed by SNL. Volume 7 has analogous results for Zion. The Zion analyses were performed by BNL.

This volume gives risk and constituent analysis results for Unit 1 of the Sequoyah Nuclear Station, operated by the TVA. Part 1 of this volume presents the analysis and the results is some detail; Part 2 consists of appendices that contain further detail. Following a summary and an introduction, Chapter 2 consists of results of the accident progression analysis for internal initiating events. Chapter 3 deals with the results of the source term analysis, and Chapter 4 gives the result of the consequence analysis. Chapter 5 summarizes the risk results, including the contributors to uncertainty in risk, for Sequoyah, and Chapter 6 contains the insights and conclusions of the complete analysis.

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2. ANALYSIS OF THE ACCIDENT PROGRESSION

This chapter describes the analysis of the progression of the accident. The analysis begins at the time of the uncovering of the top of active fuel (UTAF) and continues until the release of the major portion of radioactive material is complete (a duration of about 24 h). As the last barrier to the release of the fission products to the environment, the response of the containment to the stresses placed upon it by the degradation of the core and failure of the reactor vessel is an important part of this analysis. The main tool for performing the accident progression analysis is a large and complex event tree. The methods used in the accident progression analysis are presented in Volume 1, Part 1. The accident progression analysis starts with information received from the accident frequency analysis: frequencies and definitions of the plant damage states (PDSs). The results of the accident progression analysis are passed to the source term analysis and the risk analysis.

Section 2.1 reviews the plant features that are important to the accident progression analysis and the containment response. Section 2.2 summarizes the results of the accident frequency analysis, defines the PDSs, and presents the PDS frequencies. Section 2.3 contains a brief description of the accident progression event tree (APET). A detailed description of the APET is contained in Appendix A. Section 2.4 describes the way in which the results of the evaluation of the APET are grouped together into bins. This grouping is necessary to reduce the information resulting from the APET evaluation to a manageable amount while still preserving the information required by the source term analysis. Section 2.5 presents the results of the accident progression analysis for internal initiators.

2.1 Sequoyah Features Important to Accident Progression

The entire Sequoyah plant was briefly described in Section 1.2 of this volume. This section provides more detail on the features that are important to the progression of a core degradation accident and the response of the containment to the stresses placed upon it. These features are:

- The containment structure;
- The ice condenser (IC);
- The containment spray system (CSS);
- The air return fan system (ARFS);
- The hydrogen ignition system (HIS);
- The compartmental structure of the containment; and
- The sump and cavity arrangement.

2.1.1 The Sequoyah Containment Structure

The Sequoyah containment is a free-standing steel cylinder with a dome-shaped roof and a bottom liner plate encased in concrete. The thickness of the cylindrical portion of the containment is 1-3/8 in. at the bottom and decreases to 1/2 in. at the spring line, where the cylinder transitions to the hemispherical dome. The dome is 7/16 in. thick at the spring line and decreases to 15/16 in. at the apex. The bottom liner plate is 1/4 in. thick, sits on a base of concrete about 8 ft thick, and upon which is cast a 2-ft-thick concrete slab, which serves as the containment floor. A concrete shield building with a wall thickness of 3 ft surrounds the steel containment providing radiation shielding, and protection of the containment from adverse atmospheric conditions and external missiles. Figure 1.1 shows a section through the Sequoyah containment.

The design pressure of the Sequoyah containment is 10.8 psig. Due to conservatisms in design and construction, most estimates of the failure pressure are well above the design pressure. The mean of the aggregate distribution for the failure pressure of the Sequoyah containment provided by the Structural Response Expert Panel was 65 psig. The concrete shield building is not a significant pressure barrier since its pressure capacity is substantially less than that of the shell.

2.1.2 The Ice Condenser

The free volume of the Sequoyah containment is 1.2 million ft^3, which is about half the volume of a typical large dry PWR containment. To compensate for this smaller volume in accommodating steam pressures generated during accident conditions, a compartment containing borated ice is located between the upper and lower portions of the containment. The ice condenser compartment is annular, subtending an angle of 300° at the containment center, and is located between the crane wall and the steel containment shell. As steam is blown down from the primary system during an accident, it is driven up through the ice where it is condensed, thereby limiting the pressure in containment. The condensed water then drains back into the lower compartment of the containment.

2.1.3 The Containment Spray System

At Sequoyah, long-term containment heat removal (CHR) is provided by the CSS. The spray system consists of two pump trains capable of drawing suction from the refueling water storage tank (RWST) and discharging through spray headers in the dome of the containment building. Water sprayed into containment passes through drains in the upper compartment floor to the containment sump. When the RWST reaches a low level, the pump suction is transferred by operator action to the sump. In this mode of operation, heat is removed from the containment atmosphere by a heat exchanger in each of the pump trains; the heat exchangers are in turn cooled by a service water system. It is worth noting that the failure to remove the upper compartment drain covers following refueling operations was assessed in RSSMAP[1] to be an important source of failure for both the spray and core cooling systems in the recirculation phase, since water from spray flow would be trapped in the upper compartment and would never reach the sump. Recent improvements in maintenance procedures have significantly reduced the likelihood that the drain covers could be left in place.

2.1.4 The ARFS

The ARFS consists of two recirculation fans, each supplied with its own separate duct system and dampers. The operation of the fans ensures that gas, displaced into the upper containment by the blowdown of steam from the primary system, is returned rapidly to the lower containment. The fans provide mixing of the containment atmosphere, thereby reducing the hydrogen concentration in stagnant areas of containment. The fans draw gases from the dome and dead-ended regions of containment and exhaust into the lower compartment. This maintains forced circulation from the lower compartment through the ice condenser to the dome. A signal for high containment pressure (3 psig) actuates the fans after a short delay time. The ARFS is ac-powered.

2.1.5 The Hydrogen Ignition System

Hydrogen combustion is a concern for an ice condenser containment because of the relatively small containment volume and low failure pressure. The hydrogen ignition system is provided to help preclude large hydrogen burns by burning relatively small quantities of hydrogen as it is generated. Hydrogen igniters are located in the upper plenum of the ice condenser, the dome, and the lower compartment. Unlike the spray and ARFS, which are both actuated automatically when containment pressure reaches 3 psig, the hydrogen igniters must be initiated by the operators. The igniters are dependent upon ac power for their operation.

2.1.6 The Compartmental Structure of the Containment

The Sequoyah containment is divided into three major compartments: the lower compartment, the ice condenser, and the upper compartment. This compartmental nature adds concern regarding high local hydrogen concentrations. Without operation of the ARFS, hydrogen can stagnate within the ice condenser at potentially detonable levels. If hydrogen were to collect in either the upper or lower compartment, the likelihood of a burn capable of leading to containment failure might be increased. This is particularly true for burns occurring in the upper containment, since doors at the entrance and exit of the ice condenser are designed to open only to flow from the lower to the upper compartment. Thus, the pressures from a hydrogen burn in the upper compartment would not be relieved by flow through the ice condenser.

2.1.7 Sump and Cavity Arrangement

The design of the reactor cavity is such that it is essentially a large room, with a keyway located some distance from the reactor vessel. For sequences in which the RWST contents are injected into containment and there is melting of more than one quarter of the ice, the reactor cavity would invariably be flooded at the time of vessel failure. Only for sequences involving failure of both emergency coolant injection and containment spray injection would it be likely that the cavity would be dry at VB. Whether the cavity is dry at VB has implications for the magnitude of the containment pressure rise at VB and whether CCI occurs. If the cavity is dry, the water in the sump is unavailable to mitigate the effects of VB or to cool the core after VB.

The design of the cavity and the adjacent in-core instrumentation room (ICIR) is such that a postulated containment failure mode becomes important for Sequoyah. The seal table forms part of the ceiling of the ICIR, and is located between the crane wall and the containment wall. If high pressure melt ejection (HPME) accompanies VB, it may fail the seal table and allow hot core debris to accumulate in the vicinity of the seal table. The hot debris could attack and fail the steel containment wall. A negligible failure mechanism at Sequoyah related to the cavity design is a direct impulse resulting from an ex-vessel steam explosion (EVSE) at VB. In plants which have a direct water pathway from the reactor cavity to the containment wall, it is possible that the impulse from an EVSE could be transmitted in water to the containment wall and fail it. There is no such pathway at Sequoyah.

2.2 Interface with the Core Damage Frequency Analysis

2.2.1 Definition of Plant Damage States

Information about the many different accidents that lead to core damage is passed from the core damage frequency analysis to the accident progression analysis by means of PDSs. Because most of the accident sequences identified in the core damage frequency analysis will have accident progressions similar to other sequences, these sequences have been grouped together into PDSs. All the sequences in one PDS should behave similarly in the period following the uncovering of the top of active fuel (TAF). For the PWRs, the PDS is denoted by a seven-letter indicator that defines seven characteristics that largely determine the initial and boundary conditions of the accident progression. More information about the accident sequences may be found in NUREG/CR-4550, Volume 5.[2] The methods used in the accident frequency analysis are presented in NUREG/CR-4550, Volume 1.[3]

Table 2.2-1 lists the seven characteristics used to define the PDSs for PWRs. Under each characteristic are given the possible values for that characteristic. For example, the first characteristic denotes the condition of the reactor cooling system (RCS) pressure boundary at the time core damage begins (assumed to be approximately when the TAF is uncovered). Table 2.2-1 shows that there are eight possibilities for this characteristic: T for transient or no break; A, S_1, S_2, and S_3 for the four sizes of break which do not bypass the containment; G and H for SGTRs, and V for the large bypass pipe failure.

The first characteristic in the PDS is not necessarily an indication of the initiating event. It is an indicator of the RCS integrity at the time the core uncovers. That is, if the initiating event is a transient, say loss of offsite power, but a reactor coolant pump (RCP) seal failure occurs before the onset of core degradation, then there is a small hole in the RCS pressure boundary at the time that core damage begins, which is the time the accident progression analysis begins. The PDS for this accident would begin with S_3 to reflect the fact that there is a small hole in the RCS when this analysis starts. It is the plant condition at the onset of core damage that is important for the accident progression analysis, not what the original initiator may have been.

The first character in the PDS indicates the condition of the RCS at the onset of core degradation. As a carry-over from the use of this character to indicate the original initiator, "T" is used to indicate no break (transient). An S_2 break is a break equivalent to a double-ended guillotine break of a pipe, between 0.5 and 2 in. in diameter; an S_3 break is a break of a pipe less than 0.5 in. in diameter. an A Break is a break of a pipe greater than 6 in. in diameter and an S_1 break is a break of a pipe between 2 and 6 in. in diameter. A and S_1 breaks are considered together in the accident progression analysis since both result in low pressure in the RCS. SGTRs are S_3 size. Almost all pump seal failures result in a leak area equivalent to an S_3 break. A stuck-open PORV is equivalent to an S_2 break. Event V is such a well known and unique type of accident that the subsequent six characteristics are usually not written out.

The second characteristic concerns the status of the ECCS. Recoverable means that the ECCS will operate if or when electric power is recovered. The value "L" for the second characteristic is used when the LPIS is available to inject when the core is uncovered but cannot because the RCS pressure is too high. "L" implies that HPIS is failed.

The letter "L" is chosen for the second characteristic, for example, for the S_2H_2 sequence. This is a small break with failure of HPI and it is placed in PDS S_2LYY-YYN. The LPI pumps are operable, so if the operators recognize the situation and depressurize to allow injection by the LPIS, there is no core damage. The only portion counted toward core damage is the small (about 2%) fraction where the operator does not recognize the situation and does not depressurize the primary system.

The use of the letter "B" for the second characteristic indicates that both the HPIS and the LPIS are operating but are unable to inject because the RCS pressure is too high. In sequence $T_2L_1P_1$, PDS TBYY-YNY, for example, the operators cannot open the PORVs and all auxiliary feedwater (AFW) is failed. Thus bleed and feed is not possible using the HPIS, nor can the operators depressurize the system to use the LPIS. As in S_2LYY-YYN, a temperature-induced failure of the RCS pressure boundary or the sticking open of the PORVs or the SRVs will allow injection when the RCS pressure falls to the appropriate level.

The third characteristic concerns the status of CHR. For Sequoyah, this characteristic refers to the active CHR systems only (sprays and associated systems), not the passive CHR through the functioning of the ice condenser. Recoverable means that the CHR systems will operate if, or when, electric power is recovered. The value "S" for the third characteristic is used when the sprays are available, but there is no heat removal from the spray heat exchangers. Even if there is no heat removal, it is important to know if the sprays are operating because they reduce the aerosol concentrations in the containment atmosphere.

The fourth characteristic concerns the status of ac power. Recoverable means that power can be restored within the timeframe of the accident, roughly 24 h. Electric power in the plant, in general, is always considered to be recoverable in those PDSs where it is not available.

The fifth characteristic concerns the status of the water in the RWST. It is important for the accident progression to know if the water from the RWST is inside the containment. If the water is injected into containment, it is available to fill the sumps and along, with water from ice melt, can overflow into the reactor cavity. The value "N" for this characteristic is used when some failure prevents the injection of the RWST contents, such as when the water from the RWST has been injected into the RCS but has ended up outside the containment. This occurs in event V when the water is injected into the RCS but flows out through the break into the auxiliary building, and thus is not available inside the containment.

The sixth characteristic concerns the heat removal from the steam generators (SGs). There are six possible values for this characteristic since the auxiliary feedwater system (AFWS) may operate for some time in a blackout accident, and the secondary system may or may not be depressurized by the operators. The following abbreviations are used in describing the sixth characteristic in Table 2.2-1:

E-AFWS = Electric-motor-driven auxiliary feedwater system; and
S-AFWS = Steam-turbine-driven auxiliary feedwater system.

The seventh characteristic concerns cooling for the RCP seals. Recoverable means that cooling will become available if or when electric power is recovered.

2.2.2 PDS Frequencies

Table 2.2-2 lists 26 PDSs for Sequoyah for internal initiated events as placed into seven PDS groups. These 26 PDSs are those with mean frequencies of 1E-7/R-yr or higher, and they account for over 99% of the total mean core damage frequency (TMCDF), 5.7E-5/R-yr.

Note that while Table 2.2-2 reports 26 PDSs, the accident frequencies actually used in the integrated risk analysis were those of the seven PDS groups. That is, the accident progression analysis was performed for each of the seven PDS groups individually. The 26 PDSs were used in determine the branching for some of the initialization questions in the APET, but the APET was not evaluated for each PDS separately.

The accident frequency analysis reports the PDS frequencies based on a sample size of 1000 (see Section 5 of NUREG/CR-4550, Vol. 5,[2] Part 1). When considered as a separate entity, a great many variables could be sampled in the accident frequency analysis, and a sample size of 1000 was used. A sample this large was not feasible for the integrated risk analysis. Based on the results from the 1000-observation sample, those variables which were not important to the uncertainty in the core damage frequency were eliminated from the sampling, and the cut sets were re-evaluated using 200 observations for the integrated risk analysis.

Table 2.2-1
PWR Plant Damage State Characteristics

1. Status of RCS at Onset of Core Damage
   T = no break (transient)
   A = large break in the RCS pressure boundary
   S_1 = medium break in the RCS pressure boundary
   S_2 = small break in the RCS pressure boundary
   S_3 = very small break in the RCS pressure boundary
   G = SGTR
   H = SGTR with loss of secondary system integrity
   V = large break in an interfacing system

2. Status of ECCS
   B = operated in injection and now operating in recirculation
   I = operated in injection only
   R = not operating, but recoverable
   N = not operating, not recoverable
   L = LPIS available in both injection and recirculation modes

3. Status of CHR
   Y = operating or operable if/when initiated
   R = not operating, but recoverable
   N = never operated, not recoverable
   S = sprays operable, but no CHR (no service water [SW] to heat exchangers [HXs])

4. Ac Power
   Y = available
   P = partially available
   R = not available, but recoverable
   N = not available, not recoverable

5. Contents of RWST
   Y = injected into containment
   R = not injected, but could be injected if power recovered
   N = not injected, cannot be injected in the future

6. Heat Removal from the Steam Generators (SGs)
   X = at least one AFWS operating, SGs not depressurized
   Y = at least one AFWS operating, SGs depressurized
   S = S-AFWS failed at beginning, E-AFWS recoverable
   C = S-AFWS operated until battery depletion, E-AFWS recoverable, SGs not depressurized
   D = S-AFWS operated until battery depletion, E-AFWS recoverable, SGs depressurized
   N = no AFWS operating, no AFWS recoverable

7. Cooling for RCP Seals
   Y = operating
   R = not operating, but recoverable
   N = not operating, not recoverable

Table 2.2-2
PDSs for Sequoyah

Group Number	Group Name	Mean CD Freq.(1) (1/R-yr)	Group % TMCD Freq.	Plant Damage States	Mean CD Freq.(1) (1/R-yr)	% TMCD Freq.
1	Slow Blackout	5.0E-6	9	TRRR-RDR	3.0E-7	< 1
				S_3RRR-RDR	4.2E-6	7
				S_3RRR-RCR	1.1E-7	< 1
				S_2RRR-RCR	3.7E-7	< 1
2	Fast Blackout	9.6E-6	17	TRRR-RSR	9.6E-6	17
3	LOCAs	3.6E-5	63	ALYY-YYY	1.3E-6	2
				ALYY-YYN	3.4E-7	< 1
				AINY-YYN	4.4E-7	< 1
				AIYY-YYN	5.6E-7	1
				S_1INY-YYN	1.4E-6	2
				S_1LYY-YYN	4.9E-6	9
				S_1IYY-YYN	9.0E-7	2
				S_2INY-YYN	8.9E-7	2
				S_2LYY-YYN	4.5E-6	8
				S_2IYY-YYN	8.5E-7	2
				S_3INY-YYN	2.9E-6	5
				S_3LYY-YYN	1.4E-5	24
				S_3IYY-YYN	3.0E-6	5
4	Event V	6.5E-7	1	V	6.5E-7	1
5	Transients	2.5E-6	4	TBYY-YNY	2.3E-6	4
				TINY-YNY	1.1E-7	< 1
6	ATWS	1.9E-6	3	TLYY-YXY	2.4E-7	< 1
				GLYY-YXY	3.0E-7	< 1
				S_3NYY-YXN	1.4E-6	2
7	SGTRs	1.7E-6	3	GLYY-YNY	4.1E-7	< 1
				HINY-NXY	1.3E-6	2
	Total	5.7E-5	Internal Initiators			

(1) Based on the sample of 1000 observations used in the accident frequency analysis.

As some variation from sample to sample is observed even when the sample size and the variables sampled remain the same, there are variations between the 1000-observation sample used for the stand-alone accident frequency analysis and the 200-observation sample used for the integrated risk analysis. These differences are summarized in Table 2.2-3.

For each PDS group, the first line of Table 2.2-3 contains the 5th percentile, median, mean, and 95th percentile core damage frequencies for the 1000-observation sample used in the stand-alone accident frequency analysis. These values are taken from Table 5-5 of NUREG/CR-4550, Volume 5,[2] Part 1. Samples containing 200 observations are used for the integrated risk analysis at Sequoyah. The 5th percentile, median, mean, and 95th percentile core damage frequencies for first sample are shown on the second line of Table 2.2-3 for each PDS group. The differences between distributions for core damage frequency for the two samples are within the statistical variation to be expected.

PDS Group 1 consists of four slow blackout PDSs. In these accidents, offsite power is lost and the diesel generators fail to start or run. The steam-turbine-driven (STD) AFWS operates until the batteries are depleted. Without power for instruments and controls, the STD-AFWS eventually fails. Battery depletion is estimated to take about 4 h. During this time, the RCP seals may fail or the PORVs may stick open. Thus, the four PDSs in this group have the RCS in different conditions when core damage begins.

In one of the PDSs in this group, the RCS is intact at the time of core uncovering. Another two of the PDSs have S_3-size breaks (failures of the RCP seals), and the final PDS in this group has an S_2-size break (stuck-open PORV). The differences between the two "S_3" PDSs is whether the secondary system is depressurized before the core uncovers and while the AFW is operating.

PDS Group 2 consists solely of the fast blackout PDS, TRRR-RSR. This group is similar to PDS Group 1, except that the STD-AFW fails at the beginning. The accident proceeds to the onset of core damage before the RCP seals are likely to fail or the PORVs are likely to stick open.

PDS Group 3 consists of 13 loss-of-coolant accident (LOCA) PDSs. Four of the PDSs have an A-size break and three of the PDSs have an S_1-size break. For this analysis, A-size and S_1-size breaks are indistinguishable and are grouped together in the "A" category. There are three PDSs with an S_2-size break and three PDSs with an S_3-size break. Five of the PDSs in this group have the low pressure injection system (LPIS) operating. In PDSs ALYY-YYY and ALYY-YYN, the accumulators have failed and the LPIS is operating successfully (all trains). For an A break, the success criteria require both accumulator injection and LPIS operation. Thus, even though the RCS pressure is low and the LPIS is injecting water successfully, core damage has been assumed. In PDS S_1LYY-YYN, the high pressure injection system (HPIS) has failed in recirculation and the LPIS is operating successfully (all trains). For an S_1 break, the success criteria require high pressure (HP) systems operating during the accident. In this PDS also, the RCS pressure is low and the LPIS is injecting water successfully, but core damage has been assumed since the success criteria have not been met. In PDS S_2LYY-YYN and S_3LYY-YYN, the break does not depressurize the RCS enough

Table 2.2-3
PDS Comparison
Sequoyah

PDS	LHS Sample Size(1)	Core Damage Frequency (1/R-yr) 5%	Median	Mean	95%	% Mean TCD Freq.(2)
1 Slow SBO	1000	1.0E-07	1.4E-06	5.0E-06	1.7E-05	9
	200	1.4E-07	1.6E-06	4.6E-06	1.6E-05	
2 Fast SBO	1000	4.2E-07	3.8E-06	9.6E-06	3.6E-05	17
	200	5.5E-07	3.8E-06	9.3E-06	3.5E-05	
3 LOCAs	1000	4.4E-06	1.8E-05	3.6E-05	1.2E-04	63
	200	6.6E-06	2.0E-05	3.5E-05	1.1E-04	
4 Event V	1000	1.5E-11	2.0E-08	6.5E-07	2.1E-06	1
	200	1.5E-11	2.0E-08	6.5E-07	3.4E-06	
5 Transient	1000	2.5E-07	1.1E-06	2.5E-06	7.2E-06	4
	200	2.2E-07	1.2E-06	2.3E-06	8.2E-06	
6 ATWS	1000	4.3E-08	5.3E-07	1.9E-06	7.5E-06	3
	200	4.2E-08	5.0E-07	2.1E-06	8.5E-06	
7 SGTR	1000	2.4E-08	4.1E-07	1.7E-06	7.1E-06	3
	200	2.2E-08	3.8E-07	1.7E-06	9.4E-06	
Total	1000	1.2E-05	3.6E-05	5.7E-05	1.7E-04	
	200	1.5E-05	3.9E-05	5.6E-05	1.6E-04	

(1) The accident frequency analysis used a LHS sample size of 1000. The accident progression analysis used a LHS sample size of 200.

(2) Percentages based on the LHS sample size of 1000.

to allow low pressure injection (LPI). Thus, the accident will progress to vessel failure at a pressure too high to allow LPI unless a large temperature-induced break occurs or the primary system is deliberately depressurized.

Group 4 consists solely of Event V. The V sequence results from a failure of any one of the four pairs of series check valves used to isolate the high pressure RCS from the low pressure injection system. The resultant flow into the low pressure system is assumed to result in rupture of the low pressure piping or components. The break is outside containment in the auxiliary building, so the break both fails the RCS pressure boundary and bypasses the containment.

Group 5 consists of two PDSs that have failure of both AFW and Bleed and Feed. This PDS group is denoted Transients. In PDS TBYY-YNY, both LPIS and HPIS are available, but the PORVs cannot be opened. The operators have failed to depressurize before the onset of core damage. In PDS TINY-NNY, all ECCS and AFW have failed.

As the operators have already failed to follow procedures and depressurize the system, no credit may be given for their depressurizing the RCS after the onset of core damage for PDS TBYY-YNY. Since there is RCP seal cooling and SGTRs are not very likely, the only effective means of depressurizing the RCS are the PORVs/safety relief valves (SRVs) sticking open or the failure of the hot leg/surge line. (Even though the PORVs cannot be opened from the control room, they may still open as part of their safety function. If they do not open at all, then the SRVs will open at a slightly higher pressure. The probability that the SRVs stick open is assumed to be the same as for PORVs sticking open.) If the RCS pressure decreases to the high or intermediate range, the HPIS will inject. If the RCS pressure decreases to the low range, then the LPIS will inject.

Group 6 contains the three ATWS PDSs, in which failure to scram the reactor has occurred. They differ in the status of the RCS at the time the core uncovers, the status of the ECCS, and whether cooling for the RCP seals is operating or failed. This group contains an accident which is initiated by an SGTR, GLYY-YXY, in which the secondary side SRV is not stuck-open. The LPIS is available in two of the PDSs, TLYY-YXY and GLYY-YXY, and will inject if the RCS reaches low pressure.

Group 7 consists of two PDSs that are initiated by SGTRs and which do not have scram failures. HINY-NXY is an SGTR with stuck-open SRVs in the secondary system. GLYY-YNY has no stuck-open SRVs on the secondary side, but the RCS PORVs are open since the operators are attempting to keep the core cooled by feed and bleed. HINY-NXY has no possibility of the water from the RWST being injected into the containment; the HPIS pumps the water through the broken tube and out of the containment through the main steam line. In GLYY-YNY, the sprays operate while there is still water in the RWST or in the sump, so if there is enough ice melt, the cavity might be full when the TAF uncovers, or shortly thereafter. For the GLYY-YNY PDS, LPIS is available, and will inject if the RCS reaches low pressure.

In grouping the PDSs into the seven groups shown in Table 2.2-2, no information is lost, nor are inappropriate assumptions made to facilitate this grouping. For example, all the breaks in PDS Group 2 are not treated as very small (S_3) LOCAs simply because the majority of the group frequency is in the very small LOCA PDSs. The appropriate division between large (A), small (S_2), and very small (S_3) LOCAs is made by using fractions for the branching ratios in Question 1 in the APET. By using fractional branch ratios in Question 1 and other places in the first 11 questions, placing the 26 PDSs into the seven PDS groups causes no loss of information.

For incorporation of the uncertainty and data distributions into each part of the analysis, values are sampled for given variables. The accident frequency analysis uses a larger sample size than was used for the accident progression, source term, and risk integration analyses. The sample size was reduced due to computer limitations in terms of central processing unit

(CPU), storage and memory. Table 2.2-3 illustrates the differences in the PDS frequencies for the two sample sizes.

2.2.3 High-Level Grouping of PDSs

To provide simpler, more easily understood summaries for NUREG-1150, the seven plant damage groups described above were further condensed into the following five groups:

1. Loss of Offsite Power (LOSP)
2. LOCAs
3. Transients
4. Bypass LOCAs
5. ATWS

These five groups are denoted summary PDS Groups. The mapping from the seven groups described in the previous section into the five summary groups used in the presentation of many of the results is given in Table 2.2-4. In combining two groups to form one summary group, frequency weighting by observation is employed. The percentages of the total mean core damage frequency given above provide only approximate weightings.

2.2.4 Variables Sampled in the Accident Frequency Analysis

In the stand-alone accident frequency analysis for internal events, a large number of variables were sampled. (A list of these variables may be found in NUREG/CR-4550, Vol. 5,[2] Part 1.) Only those variables found to be important to the uncertainty in the accident frequencies were selected for sampling in the integrated risk analysis. These variables are listed and defined in Table 2.2-5. For the regression analysis, identifiers of eight characters or less were required, and these are listed in the first column. The identifiers used in the fault trees are listed in the description in brackets. Generally, the eight-character identifiers have been selected to be as informative as possible to those not familiar with the conventions used in systems analysis. For example, while Event K is commonly used to indicate the failure of the reactor protection system (RPS) to insert enough control rods to make the reactor subcritical, the identifier AU-SCRAM was chosen since it was felt that "auto scram" conveys more meaning to most readers than "K".

The second column in Table 2.2-5 gives the range of the distribution for the variable and the third column indicates the type of distribution used and its mean value for the sample distribution used in the analysis. The entry "Experts" for the distribution indicates that the distribution came from the accident frequency analysis expert panel. The fourth and fifth columns in Table 2.2-5 show whether the variable is correlated with any other variable and the last column describes the variable. More complete descriptions and discussion of these variables may be found in the Sequoyah accident frequency analysis report (NUREG/CR-4550, Vol. 5).[2] This report also gives the source or the derivation of the distributions for all these variables.

Table 2.2-4
Relationship between PDS Groups and Summary Groups

Summary Group	% TMCDF	PDS Groups	% TMCDF
1. LOSP	26	1. Slow Blackout	9
		2. Fast Blackout	17
2. LOCAs	63	3. LOCAs	63
3. Bypass LOCAs	4	4. V	1
		7. SGTRs	3
4. Transients	4	5. Transients	4
5. ATWS	3	6. ATWS	3

Most of the variable distributions come from the generic accident sequence evaluation (ASEP) data base. Others were derived specifically for the Sequoyah equipment using plant data. The distribution for the frequency of the LOSP initiating event was derived by combining data from all nuclear power plant sites with the historical experience at Sequoyah, utilizing the methods of NUREG/CR-5032.[4] The distribution for the frequency of transient initiating events was derived from Sequoyah data as described in NUREG/CR-3862.[5] The distribution for the probability of failure to scram (AU-SCRAM, Event K) was derived from the information in NUREG-1000.[6] The human error probability distributions were derived using the human reliability analysis (HRA) methodology as described in NUREG/CR-4772.[7]

Failure of the RCP seals due to lack of cooling was sampled in the following manner in the accident frequency analysis: seven states were defined, and one of these states had a probability of 1.0 in each observation while the other six states had a probability of 0.0. (When all the probability is assigned to one branch in every observation, the sampling is denoted zero-one.) The seven RCP seal states are:

State	Total Leak Rate	Start Time	Probability	Fault Tree Identifier
1	240 gpm	90 min	0.050	RCP-LOCA-240GPM
2	240-1000 gpm	150 min	0.125	RCP-LOCA-620AVG
3	433 gpm	90 min	0.005	RCP-LOCA-433GPM
4	433-1000 gpm	150 min	0.005	RCP-LOCA-717AVG
5	1000 gpm	90 min	0.525	RCP-LOCA-1000GPM
6	1920 gpm	90 min	0.005	RCP-LOCA-1920GPM
7	Normal	N.A.	0.270	NO RCP SEAL LOCA

The probability for each state was determined by a special expert panel as described in NUREG/CR-4550, Volume 2.[8] The use of this information in the Sequoyah accident frequency analysis is described in more detail in NUREG/CR-4550, Volume 5.[2] The last state represents success, i.e., no failure of the RCP seals. Design leakage through the seals is about 3 gpm/pump during normal operation, but non-failure leakage could be as high as 21 gpm/pump when there is no flow of cooling water to the seals. Leakage following seal failure could be as high as 480 gpm/pump or 1920 gpm total. As there were 200 observations in the sample used to determine risk for Sequoyah, state 1 (a total leak of 240 gpm from the four pump seals starting at 90 minutes) had a probability of 1.0 for 10 observations and a probability of 0.0 for 190 observations. State 6 (1920 gpm starting at 90 minutes) had a probability of 1.0 for only one observation. A random number generator was used to determine which state had the unity probability for which observation.

Table 2.2-5
Variables Sampled in the Accident Frequency Analysis for Internal Initiators

Variable	Range	Distribution	Correlation	Correlation With	Description
AUTO-ACT	4.8E-5 0.020	Lognormal Mean=0.0016	None		Probability of failure of one train of an automatic actuation system (generic). [ACT-FA]
AOV-FTRN	1.0E-4 0.0063	Lognormal Mean=0.0010	None		Probability of failure to transfer (per demand) for air-operated valves (AOVs) (generic). [AOV-FT]
DG-FRUN1	9.9E-6 0.057	Lognormal Mean=0.0019	Rank 1	DG-FRUN6	Probability that the diesel generator fails to run for 1 h, given that it starts (generic). [OEP-DGN-FR-1H]
DG-FRUN6	6.0E-5 0.34	Lognormal Mean=0.011	Rank 1	DG-FRUN1	Probability that the diesel generator fails to run for 6 h, given that it starts (generic). [OEP-DGN-FR-6H]
DG-FSTRT	0.0030 0.19	Lognormal Mean=0.030	None		Probability that the diesel generator fails to start, given a demand to start (generic). [OEP-DGN-FS]
DG-UNAV	3.0E-5 0.17	Lognormal Mean=0.0061	None		Probability that the diesel generator is unavailable due to maintenance (generic). [OEP-DGN-MA]
AC-UNIT2	0.056 1.0	Max. Entropy Mean=0.28	None		Probability of failure to restore ac power via Unit 2 diesel generators (recovery action). [ACP-DGN-RC-U2]
AFW-STMB	2.0E-9 7.0E-4	Lognormal Mean=1.0E-5	None		Probability of common cause failure of all AFWS due to steam-binding. [STEAM-BINDING]

Table 2.2-5 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
MDP-FRN6	8.9E-7 0.0051	Lognormal Mean=1.7E-4	None		Probability of failure of a motor-driven pump to run for 6 h (generic). [MDP-FR-6H]
MDP-FSTR	1.5E-5 0.085	Lognormal Mean=0.003	None		Probability of failure (per demand) of a motor-driven pump to start (generic). [MDP-FS]
MDP-UNAV	9.9E-6 0.057	Lognormal Mean=0.0019	None		Probability of unavailability of a motor-driven pump due to test and maintenance (generic). [MDP-TM]
MOV-FOPN	1.5E-5 0.085	Lognormal Mean=0.0029	Rank 1	PORV-BLK MOV-FCLS	Probability of failure (per demand) to open a motor-operated valve (generic). [MOV-CC]
PORV-BLK	1.5E-5 0.085	Lognormal Mean=0.0029	Rank 1	MOV-FOPN MOV-FCLS	Probability of failure (per demand) to open the PORV motor-operated block valves (generic). [PPS-MOV-FT]
MOV-FCLS	1.5E-5 0.085	Lognormal Mean=0.0029	Rank 1	MOV-FOPN PORV-BLK	Probability of failure (per demand) to close a motor-operated valve (generic). [MOV-OO]
PORV-FOP	3.1E-5 0.18	Lognormal Mean=0.0061	None		Probability of failure (per demand) of the PORVs to open (generic). [PPS-SOV-FT]
TDP-FRN6	0.0030 0.30	Max. Entropy Mean=0.030	None		Probability of failure of the AFW turbine-driven pump to run for 6 h (generic). [AFW-TDP-FR-6H]

Table 2.2-5 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
TDP-FSTR	0.0030 0.30	Max. Entropy Mean=0.030	None		Probability of failure (per demand) of the AFW turbine-driven pump to start (generic). [AFW-TDP-FS]
TDP-UNAV	5.0E-5 0.28	Lognormal Mean=0.0096	None		Probability of unavailability of the AFW turbine-driven pump due to test and maintenance (generic). [AFW-TDP-TM]
HE-DPRSG	0.0029 0.29	Max. Entropy Mean=0.029	None		Probability of operator failure (per demand) to cooldown and depressurize during SGTR (human error). [RCS-XHE-DPRZ-TSG]
HE-FCV	1.0E-5 0.058	Lognormal Mean=0.0021	Rank 1	HE-SIM1 HE-SIM2	Probability of operator failure (per demand) to close an flow control valve (FCV) during switch to recirculation (human error). [HPR-XHE-FO-FCV]
HE-SIM1	1.4E-5 0.081	Lognormal Mean=0.0028	Rank 1	HE-FCV HE-SIM2	Probability of operator failure (per demand) to close SI miniflow to RWST for an S_2 sequence (human error). [HPR-XHE-FO-SIMIN]
HE-SIM2	1.2E-5 0.071	Lognormal Mean=0.0025	Rank 1	HE-FCV HE-SIM1	Probability of operator failure (per demand) to close SI miniflow to RWST for an S_3O_D sequence (human error). [HPR-XHE-FO-SIMN2]
HE-SGBL	1.7E-5 0.096	Lognormal Mean=0.0034	None		Probability of operator failure (per demand) to close SG blowdown line valve (human error). [MSS-XHE-FO-SGBL]

Table 2.2-5 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
HE-FDBLD	0.0022 0.22	Max. Entropy Mean=0.022	None		Probability of operator failure (per demand to initiate feed and bleed (human error). [HPI-XHE-FO-FDBLD]
HE-ISADV	0.010 1.0	Max. Entropy Mean=0.10	None		Probability of operator failure (per demand to isolate atmospheric dump valves (human error). [MSS-XHE-FO-ADV]
HE-XTIE	0.0064 0.64	Max. Entropy Mean=0.065	None		Probability of operator failure (per demand) to open AOV cross-tie from SG to AFW turbine driven pump (human error). [AFW-XHE-OPNVALVE]
IE-SGTR	5.0E-5 0.28	Lognormal Mean=0.0095	None		Initiating event: frequency (1/yr) of SGTRs (presuurized water reactor [PWR] data). [IE-TSG]
MFW-FRST	0.011 1.0	Max. Entropy Mean=0.11	None		Probability of failure to restore MFW after loss of AFW during SGTR (recovery action). [RA3]
IE-S3	0.0013 0.082	Lognormal Mean=0.013	None		Initiating event: frequency (1/yr) of a very small (dia. < 0.5 in.) break in the RCS (PWR data). [IE-S3]
SRV-DPRZ	7.0E-5 0.40	Lognormal Mean=0.014	None		Failure to depressurize the RCS to limit flow from open SG safety relief valve (SRV) during an SGTR (recovery action). [RA14]

Table 2.2-5 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
UNFV-MOD	1.8E-4 0.27	Lognormal Mean=0.014	None		Fraction of the time that the reactor operates with an unfavorable moderator temperature coefficient (PWR data). [Z]
ADV-DPRZ	7.0E-5 0.40	Lognormal Mean=0.013	None		Failure to depressurize the RCS to limit flow from open atmospheric dump valve during an SGTR (recovery action). [RA11]
MN-SCRAM	0.034 1.0	Max. Entropy Mean=0.34	None		Probability of failure to effect manual scram due to operator error and hardware faults. [R]
IE-BATT	2.5E-5 0.14	Lognormal Mean=0.0050	None		Initiating event: frequency (1/yr) of loss of dc vital battery (generic). [IE-TDC]
IE-A	5.1E-5 0.0032	Lognormal Mean=5.0E-4	None		Initiating event: frequency (1/yr) of a large (dia. > 6 in.) break in the RCS (PWR data). [IE-A]
AU-SCRAM	1.8E-6 7.6E-4	Lognormal Mean=5.9E-5	None		Probability of failure of the RPS to automatically insert sufficient control rods to terminate the reaction. [K]
IE-TTRIP	1.6 21.2	Lognormal Mean=6.3	None		Initiating event: frequency (1/yr) of turbine trip with main feedwater (MFW) and power control system (PCS) available. [IE-T3]
IE-T-HIP	1.2 16.2	Lognormal Mean=4.8	None		Initiating event: frequency (1/yr) of high power (>25%) transients that require reactor scram. [IE-TZ]

Table 2.2-5 (continued)

Variable	Range	Distribution	Correlation	Correlation With	Description
IE-T-ALL	1.3 17.8	Lognormal Mean=5.3	None		Initiating event: frequency (1/yr) of all transients that require reactor scram. [IE-T]
IE-LMFWS	0.18 2.4	Lognormal Mean=0.72	None		Initiating event: frequency (1/yr) of transients due to loss of the MFW system. [IE-T2]
BETA-2DG	0.0039 0.24	Lognormal Mean=0.038	None		Beta factor for common cause failure of the DGs (generic). [BETA-2DG]
BETA8AOV	0.0035 0.22	Lognormal Mean=0.034	None		Beta factor for common cause failure of eight AOVs (generic). [BETA-8AOV]
MS-LIAS	5.0E-7 0.0028	Lognormal Mean=9.5E-5	None		Probability of loss (per demand) of instrument air system (IAS) to main steam AOVs. [IAS-PTF-LF-AOV]
V-TRAIN	1.8E-13 1.5E-5	Experts Mean=5.4E-7	None		Initiating event: frequency (1/yr) of check valve failure in one of the LPIS trains. [IE-V-TRAIN]
IE-LOSP	4.0E-4 0.35	LOSP Data Mean=0.091	None		Initiating event: frequency (1/yr) of of LOSP. [IE-T1]
RCP-SL-F		Experts Mean=0.27	None		Probability of RCP seal LOCA before the onset of core damage. [See text]

2.3 Description of the APET

This section describes the APET that is used to perform the accident progression analysis for Sequoyah. The APET itself forms a high-level model of the accident progression. The APET is too large to be drawn out in a figure as smaller event trees usually are. Instead, the APET exists only as a computer input file. The APET is evaluated by the code EVNTRE, which is described elsewhere.[9]

The APET is not meant to be a substitute for detailed, mechanistic codes such as the STCP, CONTAIN, MELCOR, and MAAP. Rather, it is an integrating framework for synthesizing the results of these codes together with expert judgment on the strengths and weaknesses of the codes. The detailed, mechanistic codes require too much computer time to be run for all the possible accident progression paths. Therefore, the results from these codes are represented in the Sequoyah APET, which can be evaluated very quickly. In this way, the full diversity of possible accident progressions can be considered and the uncertainty in the many phenomena involved can be included.

The following section contains a brief overview of the Sequoyah APET. Details, including a complete listing of the APET and a discussion of each question, can be found in Appendix A of this volume. Section 2.3.2 is a summary of how the APET was quantified, that is, how the many numerical values for branching ratios and parameters were derived. Section 2.3.3 presents the variables that were sampled in the accident progression analysis for Sequoyah.

2.3.1 Overview of the APET

The APET for Sequoyah considers the progression of the accident from the time the TAF in the core is uncovered, which is assumed to be the onset of core damage, through the core-concrete interaction (CCI). Although the CCI may progress at increasingly slower rates for days, the end of this analysis for most accident progressions has been arbitrarily set at 24 h after the accident initiator. The exception to the 24 hour end limit is in the case of the initiation of CCI after very late overpressure failure, in which the end of the accident progression analysis is set at 40 h. The time limit is chosen such that the bulk of the release of fission products is complete.

Table 2.3.1 lists the 111 questions in the Sequoyah APET. The APET is divided into five time periods. To facilitate understanding of the APET and referencing between questions, each branch of every question is assigned a mnemonic abbreviation. The mnemonic branch abbreviations for most branches start with a character or characters which indicate the time period of the question. The time periods and their abbreviations are:

B - Initial	Questions 1 through 15 determine the conditions at the beginning of the accident.

E, E2 - Early	Questions 16 through 63 concern the progression of the accident from the uncovering of the TAF, through core degradation, and until the time before VB. Questions 17 through 21 concern events or actions which may depressurize the RCS before breach. The possibility that core degradation may be arrested and VB prevented is considered in Question 26. Questions 38 through 58 address the early threat of hydrogen to containment, and whether the containment fails before VB. Questions 59 through 61 address the effect that hydrogen events, containment failure, or the containment environment have on engineered safety features. Questions 62 and 63 establish conditions in containment immediately before VB.
I, I2 - Intermediate	Questions 64 through 85 address the time period in which VB occurs. Questions 64 through 82 address containment loading and ex-vessel phenomena, including the possibility of containment failure due to events associated with vessel failure. Questions 83 through 85 determine the effect that events associated with VB have on engineered safety features.
L, L2 - Late	Questions 86 through 109 determine the progression of the accident for the time period in which CCI occurs. Questions 86 through 103 address the accident during the initial period of CCI, up to a nominal period of 5 h after the start of CCI. Containment failure due to late hydrogen burns is addressed in this time regime. Questions 104 through 109 determine the progression the accident in the latter part of CCI. The status of systems in containment immediately after late hydrogen burns is considered. The possibility of containment failure due to late overpressure or basemat melt-through (BMT) is addressed.
L3 - Final	Questions 110 and 111 address the final stages of the accident. The impairment of sprays due to very late containment failure is considered in question 110. Question 111 concerns core-concrete attack after late overpressure of containment and subsequent late boiloff of cavity water.

The clock time for each period will vary depending upon the type of accident being modeled.

The Sequoyah APET does not contain any questions to resolve core-vulnerable sequences. A core-vulnerable PDS involves a LOCA with failure of CHR. The continual deposition of decay heat in the containment by operation of the ECCS in the recirculation mode is predicted to lead to eventual ice melt and containment failure after an extended period of time. Containment failure, in turn, may lead to ECCS failure. The Sequoyah PDSs with frequencies exceeding 1.0E-7/yr did not contain any accidents of this type.

In several places in the evaluation of the APET, a User Function is called. This is a FORTRAN function subprogram which is executed at that point in the evaluation of the APET. The user function allows computations to be carried out that are too complex to be treated directly in the event tree. The user function itself is listed in Appendix A.2. The calculations performed by the user function are described for each question in Appendix A.1, and are briefly mentioned below. The user function is called to:

- Compute the distribution of hydrogen and other gases in containment, and determine the flammability of the atmosphere;
- Calculate the burn completeness if ignition occurs;
- Compute the pressure rise and consumption of hydrogen and oxygen due to hydrogen burns;
- Determine whether the containment fails and its mode of failure;
- Compute the peak containment pressure at VB when the ice condenser is bypassed;
- Compute the amount of hydrogen released to the containment at VB;
- Calculate the amounts of hydrogen, carbon monoxide, and carbon dioxide generated during CCI.

2.3.2 Overview of the APET Quantification

This section summarizes the ways in which the questions in the Sequoyah APET were quantified and discusses these methods briefly. A detailed discussion of each question, which includes comments on quantification, may be found in Appendix A.1.1.

Table 2.3-1 lists the 111 questions in the Sequoyah APET. In addition to the number and name of the question, Table 2.3-1 indicates if the question was sampled, and the source of evaluation or quantification of the question. The item sampled may be either the branching ratios or the parameter defined at that question. For questions that are sampled, the entry ZO in the sampling column indicates that the question was sampled zero-one, and the entry SF means the question was sampled with split fractions. An entry of DS in the sampling column indicates that the branch probabilities are obtained from a distribution; sampling of the distribution is done in both the split fraction and zero-one manner.

The difference between split fraction and zero-one sampling may be illustrated by a simple example. Consider a question that has two branches, and a uniform distribution from 0.0 to 1.0 for the probability for the first branch. If the sampling is zero-one, in half the observations, the probability for the first branch will be 1.0, and in the other half of the observations it will be 0.0. If the sampling is split fraction, the probability for the first branch for each observation is a random fractional value between 0.0 and 1.0. The average over all the fractions in the sample is 0.50. The implications of ZO or SF sampling are discussed in the methodology volume (Volume 1).

If the sampling column is blank, the branching ratios for that question, and the parameter values defined in that question, if any, are fixed. The branching ratios of the PDS questions change to indicate which PDS is being considered. Some of the branching ratios depend on the relative frequency of the PDSs which make up the PDS group being considered. These branching ratios change for every sample observation, but may do so for some PDS groups and not for others. If the branching ratios change from observation to observation for any one of the seven PDS groups, SF is placed in the sampling column for the PDS questions.

Sometimes a question may have been quantified by more than one source; e.g., some of the cases in the question may have been quantified by an expert panel and some may have been quantified internally by the project staff. If this is the case, the entry in the quantification source column in Table 2.3-1 represents the major contributor to the quantification. At other times a question may have some cases in which the branching ratios or parameters are sampled and some cases in which they are not. For these questions the entry under the sampling column in Table 2.3-1 will address those cases that are sampled.

The abbreviations in the quantification source column of Table 2.3-1 are given below, with the number of questions which have that type of quantification.

Type of Quantity	Number of Questions	Comments
PDS	11	Determined by the PDS.
AcFrqAn	5	Determined by the Accident Frequency Analysis.
Other	4	See notes 1 through 4 in Table 2.3-1.
Internal	34	Quantified internally in this analysis.
Summary	16	The branch taken at this question follows directly from the branches taken at previous questions.
ROSP	3	The probability of the recovery of offsite power is determined by distributions derived from electric power recovery data for this plant.
UFUN-Str	4	Calculated in the User Function subroutine, using distributions from the Structural Expert Panel.
UFUN-Int	8	Calculated in the User Function subroutine, using models and distributions generated by the project staff.
UFUN-Lds	6	Calculated in the User Function subroutine, using models and distributions generated by the Containment Loads Expert Panel.
In-Vessel	5	Distributions from the In-Vessel Expert Panel.

Loads	15	Distributions from the Containment Loads Expert Panel.
Struct.	1	Distributions from the Structural Expert Panel.

Table 2.3-1
Questions in the Sequoyah APET

Question Number	Question	Sampling	Quant. source
1	Size and location of the RCS break when the core uncovers?	SF	PDS
2	Has the reaction been brought under control?		PDS
3	For SGTR, are the secondary SRVs stuck open?	SF	PDS
4	Status of ECCS?	SF	PDS
5	Is the RCS depressurized by the operators?		PDS
6	Status of sprays?	SF	PDS
7	Status of ac power?		PDS
8	Are the RWST contents injected into containment?		PDS
9	Heat removal from the steam generators?		PDS
10	Is the secondary depressurized before the core uncovers?	SF	PDS
11	Cooling for RCP seals?	SF	PDS
12	Initial containment leak or isolation failure?	SF	AcFrqAn
13	Do the operators turn on the hydrogen igniters?		AcFrqAn
14	Status of air return fans?		AcFrqAn
15	Event V - break location scrubbed by sprays?	SF	Note 1
16	RCS pressure at the start of core degradation (CD)?		Summary
17	Do the pressurizer PORVs stick open?	SF	Note 2
18	Temperature-induced RCP seal failure?	ZO	Note 3
19	Is the RCS depressurized by opening the PORVs?		Summary
20	Temperature-induced SGTR?	DS	In-Vessel
21	Temperature-induced hot leg or surge line break?	DS	In-Vessel
22	Is ac power recovered early?	SF	ROSP
23	After ac recovery, is core cooling re-established?		Internal
24	Rate of blowdown to containment?		Summary
25	Vessel pressure before VB?	ZO	Internal
26	Is core damage arrested? No VB?	SF	Internal
27	Early sprays?		Summary
28	Early air return fans?		Summary
29	Is the ice melted from the IC before VB?		Internal
30	Have bypass paths developed in the IC before VB?		Internal

Table 2.3-1 (continued)

Question Number	Question	Sampling	Quant. source
31	Are the ARFs effective before H_2 ignition?	SF	Internal
32	Is the bulk of blowdown flow diverted from the LC to the UC via the floor drains?	ZO	Loads
33	What is the steam concentration in the LC and O_2 distribution in containment during CD?		Internal
34	What is the steam concentration in the IC during core degradation?		Internal
35	What is the steam concentration in the UC during core degradation?		Internal
36	Early baseline pressure?		Internal
37	Time of accumulator discharge?		Summary
38	Amount of H_2 released in-vessel during CD?	P	In-Vessel
39	Amount of zirconium oxidized in-vessel during CD?		Summary
40	Fraction of in-vessel H_2 released from the RCS during CD?	P	Loads
41	To what degree is the H_2 mixed in the UC?	ZO	Loads
42	Distribution of H_2 in containment during CD?		UFUN-Lds
43	What is the H_2 concentration in the LC and burn completeness, if ignited?		UFUN-Lds
44	What is the H_2 concentration in the IC and burn completeness, if ignited?		UFUN-Lds
45	What is the H_2 concentration in the UP and burn completeness, if ignited?		UFUN-Lds
46	What is the H_2 concentration in the UC and burn completeness, if ignited?		UFUN-Lds
47	Are the hydrogen igniters operating during CD?		AcFrqAn
48	Does H_2 ignition occur in the LC during CD?	SF	Internal
49	Does H_2 ignition occur in the IC during CD?	SF	Loads
50	Does H_2 ignition occur in the UP during CD?	SF	Loads
51	Does H_2 ignition occur in the UC during CD?	SF	Loads
52	Is there DDT in the IC during CD?	SF	Loads
53	Is there DDT in the UP during CD?	SF	Loads
54	Pressure rise in containment due to early burn?		UFUN-Lds
55	Impulse from detonation in ice condenser?	P	Loads

Table 2.3-1 (continued)

Question Number	Question	Sampling	Quant. source
56	Impulse from detonation in upper plenum?	P	Loads
57	Containment failure criteria for pressure and impulse loadings?	P	Struct
58	Early containment failure and mode of failure?	ZO	UFUN-Str
59	Status of ice condenser before VB?		Internal
60	Are ARFs or ducting impaired due to early burns?		Internal
61	Are sprays impaired due to CF or environment?		Internal
62	What fraction of H_2 released in-vessel is in containment at VB?		Summary
63	Level of cavity flood at VB?	ZO	Internal
64	Does an alpha mode event fail both the vessel and containment?	SF	Note 4
65	Type of VB?	ZO	In-Vessel
66	Fraction of core released from vessel at VB?	P	In-Vessel
67	Level of core released from vessel at VB?		Summary
68	Fraction of core released at VB that is diverted to the in-core instrumentation room (ICIR)?	P	Internal
69	Level of core ejected to ICIR?		UFUN-Int
70	Does the vessel become a "rocket" and fail the containment or bypass the IC?		Internal
71	Ex-vessel steam explosion at VB?		Internal
72	Size of hole in vessel (after ablation)?	ZO	Internal
73	Maximum peak pressure rise at VB? (Low pressure and non-HPME cases)	P	Loads
74	Maximum peak pressure rise at VB? (Some of the intermediate pressure cases)	P	Loads
75	Maximum peak pressure rise at VB? (Intermediate, high, and system pressure cases)	P	Loads
76	Level of ice bypass at vessel breach?		Internal
77	Peak pressure rise at VB?		UFUN-Int
78	Containment failure by direct core contact with containment wall?	ZO	Internal
79	What fraction of potentially oxidizable metal in the ejected core is oxidized at VB?	P	Loads
80	Amount of H_2 released to containment at VB?		UFUN-Int

Table 2.3-1 (continued)

Question Number	Question	Sampling	Quant. source
81	Fraction of hydrogen in containment consumed at VB?	P	Loads
82	Containment failure at VB and mode of failure?	ZO	UFUN-Str
83	Status of IC immediately after VB?		Summary
84	Are ARFs or ducting impaired due to burns at VB?		Internal
85	Are sprays impaired due to CF or environment at VB?		Internal
86	Fraction of core not participating in HPME that is available for CCI?		Summary
87	Level of core not participating in HPME that is available for CCI?		Summary
88	Is the debris bed in a coolable configuration?		Internal
89	What is the nature of the prompt CCI?		Summary
90	Is ac power recovered late?	SF	ROSP
91	Late sprays?		Summary
92	Late air return fans?		Summary
93	Is the ice melted or bypassed at the start of prompt CCI?		Internal
94	Late baseline pressure?	P	Internal
95	Amount of H_2 (plus equivalent CO) and CO_2 generated during prompt CCI?		UFUN-Int
96	What amount of oxygen remains in containment late?		UFUN-Int
97	Amount of hydrogen in containment after CCI?		UFUN-Int
98	How much steam is in containment late?		Internal
99	What is the inert level in containment late, and is there sufficient H_2 or O_2 for burns?		UFUN-Int
100	Late hydrogen igniters?		AcFrqAn
101	Is there a late deflagration in containment?		Internal
102	Pressure rise due to late deflagration?		UFUN-Int
103	Late containment failure and mode of failure?		UFUN-Str
104	Are sprays impaired due to late CF or environment?		Internal
105	Is ac power recovered very late?	SF	ROSP
106	Very late sprays?		Summary
107	Basemat meltthrough?		Internal
108	What is the very late pressure in containment?	P	Internal
109	What is the mode of very late containment failure?		UFUN-Str
110	Sprays after very late containment failure?		Internal
111	Does CCI occur after late boiloff and very late CF?	ZO	Internal

Table 2.3-1 (continued)

Notes to Table 2.3-1

Note 1. Whether fire sprays would be available to scrub the releases from the break for Event V was determined by a special panel which considered only this problem for the draft version of this analysis. As there was no new information available, there was no reason to change the conclusions reached by this group. See the discussion of Question 15 in Appendix A.1.1.

Note 2. There is little or no data on the failure rate of PORVs when passing gases at temperatures considerably in excess of their design temperature. The quantification was arrived at by discussions between the accident frequency analyst and the plant analyst. See the discussion of Question 17 in Appendix A.1.1.

Note 3. In the accident frequency analysis, a special panel was convened to consider the issue of the failure of RCP seals. The quantification of this question is not as detailed as that done in the accident frequency analysis, but relies on the information produced by this panel. See the discussion of Question 18 in Appendix A.1.1.

Note 4. The Alpha mode of vessel and containment failure was considered by the Steam Explosion Review Group a few years ago. The distribution used in this analysis is based on information contained in the report of this group. See the discussion of Question 64 in Appendix A.1.1.

Key to Initialisms and Abbreviations in Table 2.3-1

AcFrqAn The quantification was performed by the Accident Frequency Analysis project staff.

DS The branch probabilities are obtained from a distribution; sampling of the distribution is done in both the split fraction and zero-one manner.

Internal The quantification for this question was performed at Sandia National Laboratories by the project team with the assistance of other members of the laboratory staff.

In-Vessel This question was quantified by sampling an aggregate distribution provided by the In-Vessel Expert Panel.

Table 2.3-1 (continued)

Key to Initialisms and Abbreviations in Table 2.3-1 (continued)

Loads	This question was quantified by sampling an aggregate distribution provided by the Containment Loads Expert Panel.
P	A parameter value introduced to the event tree in this question is obtained by sampling a distribution.
PDS	The quantification follows directly from the definition of the plant damage state.
ROSP	This question was quantified by sampling a distribution derived from the offsite power recovery data for the plant.
SF	Split fraction sampling - the branch probabilities are real numbers between zero and one.
Struct	This question was quantified by sampling an aggregate distribution provided by the Structural Expert Panel.
Summary	The quantification for this question follows directly from the branches taken at preceding questions, or the values of parameters defined in preceding questions.
UFUN-Int	This question is quantified by the execution of a module in the User Function subroutine, to apply models and distributions that were generated by the project staff.
UFUN-Str	This question is quantified by the execution of a module in the User Function subroutine, to apply models and distributions generated by the Structural Expert Panel.
UFUN-Lds	This question is quantified by the execution of a module in the User Function subroutine, to apply models and distributions generated by the Containment Loads Expert Panel.
ZO	Zero-one sampling - the branch probabilities are either 0.0 or 1.0.

2.3.3 Variables Sampled for the Accident Progression Analysis

There were 135 variables sampled for the accident progression analysis. That is, every time the APET was evaluated by EVNTRE, the original values of 135 variables were replaced with values selected for the particular observation under consideration. These values were selected by the LHS program from distributions that were defined before the APET was evaluated. Most of these distributions were determined by expert panels. Table 2.3-2 lists the variables in the APET that were sampled for the accident progression analysis. Some of them are branch fractions; the others are parameter values for use in calculations or comparisons performed while the APET is being evaluated.

In Table 2.3-2, the first column gives the variable abbreviation or identifier, and the question (and case if appropriate) in which the variable is used. The identifiers are limited to eight characters for the statistical package used to perform regression sensitivity studies. Where several variables are correlated, they are treated as one variable in the regression analysis, but are different variables as far as the accident progression analysis and sampling process are concerned. Some of these variables in Table 2.3-2 have a number in the last position to distinguish the actual variable number for the accident progression analysis. The number is dropped in the sensitivity analysis. For example, RCP-SL-P2 and RCP-SL-P3 are treated as one variable, RCP-SL-P, in the sensitivity analyses.

The second column gives the range of the distribution for the variable. An entry of "0.0/1.0" in this column indicates that the variable took on fractional values between 0.0 and 1.0. An entry of "Zero/One" in this column indicates that the variable was sampled Zero-One, i.e., it took on only the values 0.0 and 1.0. In each observation, one of these two values would be assigned.

The third column in Table 2.3-2 indicates the type of distribution used. The mean value of the distribution is given if appropriate. The entry "Experts" for the distribution indicates that the distribution came from an expert panel and the entry "Internal" distribution indicates that the distribution was determined internally by the project staff or others. (A listing of the input to the LHS program that contains many of the distributions in tabular form is given in Appendix E.) For zero-one variables, an indication of the probability of each state is given in this column.

The fourth and fifth columns in Table 2.3-2 show whether the variable is correlated with any other. "Rank 1" indicates a rank correlation of 1.0. An "n" is used to indicate any integer. In the entry for RCP-SL-P2, RCP-SL-Pn in the "Correlated with" column indicates that RCP-SL-P2 is correlated with RCP-SL-P3 and RCP-SL-P4.

Most of the variables listed in Table 2.3-2 need no further comment. The RCS pressure at VB variables, RCSPR-VB2 and RCSPR-VB3 (Question 25), are sampled Zero-One. The distribution column gives the fraction of the time each of the pressure ranges is chosen. RCP seal failure is considered both in the accident frequency analysis and in the accident progression analysis. The eight-character code is RCP-SL-F for RCP seal failures in the accident frequency analysis and RCP-SL-P for RCP seal failures in the accident progression analysis. These two variables should have been correlated with each other, but the ways in which seal failures were treated in the two constituent analyses were so different that this was not feasible.

Note that the temperature-induced (T-I) SGTR variable (Question 20), the T-I hot leg failure variables (Question 21), and the amount of in-vessel hydrogen variables (Question 38) are correlated with each other as the experts concluded that the oxidation of a large amount of zirconium before VB would result in high temperatures, which in turn, would make temperature-induced SGTRs, and hot leg or surge line failures more likely.

The degree of mixing in the upper containment when fans and igniters are not operating (Question 41) is sampled Zero-One. The entries under "Distribution" indicate the probability of each type of mixing. Mix2 indicates that the upper plenum and upper compartment are well-mixed and a clear path exists from the lower compartment to the upper plenum through the ice condenser. Mix3 indicates that the upper plenum and the upper compartment are well-mixed and a clear path does not exist. Unmix indicates that there is no mixing and a clear path does not exist. Mixing of the upper plenum and upper compartment atmosphere occurs when enough upper deck doors are open, and a clear path exists if enough intermediate deck doors are open.

The type of vessel failure (Question 65) is sampled Zero-One and the entries under "Distribution" indicate the probability of each type of vessel breach. HPME indicates ejection of the melt at high pressure through a hole that is small relative to the cross-section of the vessel. BtmHd indicates a gross failure of the entire bottom head of the vessel, and Pour indicates a slow release of the melt driven primarily by gravity.

The containment failure mode, as a function of failure pressure, was determined by the Structural Expert Panel. The containment failure mode variable, CF-MODE (Question 57), is only a random variable used to determine the failure mode. The method used to select the failure mode for each observation is explained in Volume 1, and the results of the expert panel on the failure pressure and failure mode for Sequoyah may be found in Volume 2.

The final variable in Table 2.3-2 (Questions 22, 90, and 105), POWERREC, is used to select the probability that offsite power will be recovered in a specified time interval given that it was not recovered in a previous time interval. Distributions were developed for 12 cases, each with different start and end times, corresponding to different classes of accidents. More detail on the methods for determining the probability of offsite power recovery can be found in Appendix A.3 and Appendix E. Additional information concerning the variable descriptions can be obtained from the detailed discussions of the indicated questions in Appendix A of this volume.

Table 2.3-2
Variables Sampled in the Accident Progression Analysis for Internal Initiators

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
CNT-ISOF Q12	2.5E-5 0.14	Lognormal Mean=0.005	None		Probability that the containment will not be isolated at the start of the accident.
V-SPRAYS Q15	0.60 1.0	Uniform Mean=0.80	None		Probability that the radioactive releases will be scrubbed by area fire sprays, given Event V.
PORV-STK Q17 C1	0.0 1.0	Uniform Mean=0.50	None		Probability that at least one pressurizer PORV or RCS SRV sticks open, given that the RCS is intact and the PORVs or SRVs are cycling.
RCP-SL-P2 Q18 C2	Zero One	Fail 0.71 NoFail 0.29	Rank 1	RCP-SL-Pn	Probability of a T-I failure of the RCP seals, given core damage, RCS at system setpoint pressure, and no cooling for the RCP seals.
RCP-SL-P3 Q18 C3	Zero One	Fail 0.65 NoFail 0.35	Rank 1	RCP-SL-Pn	Probability of a T-I failure of the RCP seals, given core damage, RCS at high pressure, and no cooling for the RCP seals.
RCP-SL-P4 Q18 C4	Zero One	Fail 0.60 NoFail 0.40	Rank 1	RCP-SL-Pn	Probability of a T-I failure of the RCP seals, given core damage, RCS at intermediate or low pressure, and no cooling for the RCP seals.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
TI-SGTR Q20 C1	0.0 0.12	Experts Mean=0.014	Rank 1	TI-HOTLGn H2-INVn	Probability of a T-I SGTR, given core damage, RCS at setpoint pressure, and no cooling for the SGs.
TI-HOTLG1 Q21 C1	0.0 1.0	Experts Mean=0.77	Rank 1	TI-SGTR TI-HOTLG2 H2-INVn	Probability of a T-I failure of the hot leg or surge line, given core damage, AFWS failure, and the RCS intact at system setpoint pressure.
TI-HOTLG2 Q21 C2	0.0 1.0	Experts Mean=0.035	Rank 1	TI-SGTR TI-HOTLG1 H2-INVn	Probability of a T-I failure of the hot leg or surge line, given core damage, AFWS failure, and an S_3 break in the RCS.
RCSPR-VB2 Q25 C2	Zero One	Low 0.20 Int 0.80	Rank 1	RCSPR-VB3	RCS pressure just before VB, given an initiating or induced S_2 break.
RCSPR-VB3 Q25 C3	Zero One	Low 0.335 Int 0.33 High 0.335	Rank 1	RCSPR-VB2	RCS pressure just before VB, given an initiating or induced S_3 break.
CDARREST2 Q26 C2	0.90 1.0	Uniform Mean=0.95	Rank 1	CDARRESTn	Probability that core damage can be arrested before VB, given that at UTAF, there was a large break and the LPIS was operating.
CDARREST3 Q26 C3,C5 Q26 C8,C9	0.80 1.0	Uniform Mean=0.90	Rank 1	CDARRESTn	Probability that core damage can be arrested before VB, given that at UTAF, the LPIS was operating or that power was recovered between 1 and 2.5 h, 4 and 10.5 h, or 7 and 12.5 h.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
CDARREST6 Q26 C6	0.56 1.0	Quadratic Mean=0.78	Rank 1	CDARRESTn	Probability that core damage can be arrested before VB, given that power was recovered between 1 and 4.5 h.
CDARREST7 Q26 C7	0.34 1.0	Quadratic Mean=0.67	Rank 1	CDARRESTn	Probability that core damage can be arrested before VB, given that power was recovered between 4 and 6 h.
IGN-RSBO Q31 C2	0.014 0.72	Internal Mean=0.17	None		Probability that hydrogen ignition occurs before the air return fans mix the containment atmosphere, given an SBO sequence in which ac power has been recovered.
FL-DRAIN Q32 C1	Zero One	Divert 0.25 NoDvrt 0.75	None		Probability that blowdown flow is diverted through the refueling canal floor drains, given an SBO sequence with blowdown rate typical of an S_3 break.
H2-INV1 Q38 C1	0.0 660.	Experts Mean=223.	Rank 1	TI-SGTR TI-HOTLGn H2-INVn	The amount of hydrogen, in kg-moles, that is generated in-vessel, given that the RCS is at setpoint pressure and the accumulators discharge before or after core melt.
H2-INV2 Q38 C2	0.0 660.	Experts Mean=255.	Rank 1	TI-SGTR TI-HOTLGn H2-INVn	The amount of hydrogen, in kg-moles, that is generated in-vessel, given that the RCS is at setpoint pressure and the accumulators discharge during core melt.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
H2-INV3 Q38 C3	0.0 400.	Experts Mean=164.	Rank 1	TI-SGTR TI-HOTLGn H2-INVn	The amount of hydrogen, in kg-moles, that is generated in-vessel, given that the RCS is at high pressure and the accumulators discharge before or after core melt.
H2-INV4 Q38 C4	0.0 430.	Experts Mean=192.	Rank 1	TI-SGTR TI-HOTLGn H2-INVn	The amount of hydrogen, in kg-moles, that is generated in-vessel, given that the RCS is at high pressure and the accumulators discharge during core melt.
H2-INV5 Q38 C5	0.0 600.	Experts Mean=244.	Rank 1	TI-SGTR TI-HOTLGn H2-INVn	The amount of hydrogen, in kg-moles, that is generated in-vessel, given that the RCS is at intermediate pressure and the accumulators discharge before or after core melt.
H2-INV6 Q38 C6	0.0 600.	Experts Mean=264.	Rank 1	TI-SGTR TI-HOTLGn H2-INVn	The amount of hydrogen, in kg-moles, that is generated in-vessel, given that the RCS is at intermediate pressure and the accumulators discharge during core melt.
H2-INV7 Q38 C7	0.0 600.	Experts Mean=228.	Rank 1	TI-SGTR TI-HOTLGn H2-INVn	The amount of hydrogen, in kg-moles, that is generated in-vessel, given that the RCS is at low pressure.
H2-EXV1 Q40 C1	0.25 0.85	Experts Mean=0.64	Rank 1	H2-EXVn	Fraction of in-vessel hydrogen that is released to containment, given that the blowdown to containment is typical of a transient sequence with a cycling PORV.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
H2-EXV2 Q40 C2	0.35 0.85	Experts Mean=0.66	Rank 1	H2-EXVn	Fraction of in-vessel hydrogen that is released to containment, given that the blowdown to containment is typical of an S_3 break in the RCS.
H2-EXV3 Q40 C3	0.55 0.85	Experts Mean=0.70	Rank 1	H2-EXVn	Fraction of in-vessel hydrogen that is released to containment, given that the blowdown to containment is typical of an S_2 break in the RCS.
H2-EXV4 Q40 C4	0.65 1.00	Experts Mean=0.85	Rank 1	H2-EXVn	Fraction of in-vessel hydrogen that is released to containment, given that the blowdown to containment is typical of a large break in the RCS.
H2-MIX Q41 C2	Zero One	Mix2 0.45 Mix3 0.45 Unmix 0.10	None		The degree of mixing of the atmosphere in the upper compartment, given that air return fans (ARFs) and H_2 ignition system (HIS) are not operating.
IGN-IC3 Q49 C3	0.0 0.9	Experts Mean=0.20	Rank 1	IGN-UPn IGN-UCn	Probability of H_2 ignition in the ice condenser, given that the ARFs and HIS are not operating, and the H_2 mole fraction is greater than 16%.
IGN-IC4 Q49 C4	0.0 0.9	Experts Mean=0.16	Rank 1	IGN-UPn IGN-UCn	Probability of H_2 ignition in the ice condenser, given that the ARFs and HIS are not operating, and the H_2 mole fraction is between 11 and 16%.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
IGN-IC5 Q49 C5	0.0 0.75	Experts Mean=0.12	Rank 1	IGN-UPn IGN-UCn	Probability of H_2 ignition in the ice condenser, given that the ARFs and HIS are not operating, and the H_2 mole fraction is between 5.5 and 11%.
IGN-UP6 Q50 C6	0.0 0.6	Experts Mean=0.35	Rank 1	IGN-ICn IGN-UCn	Probability of H_2 ignition in the upper plenum, given that the ARFs and HIS are not operating, and the H_2 mole fraction is greater than 16%.
IGN-UP7 Q50 C7	0.0 0.6	Experts Mean=0.26	Rank 1	IGN-ICn IGN-UCn	Probability of H_2 ignition in the upper plenum, given that the ARFs and HIS are not operating, and the H_2 mole fraction is between 11 and 16%.
IGN-UP8 Q50 C8	0.0 0.6	Experts Mean=0.18	Rank 1	IGN-ICn IGN-UCn	Probability of H_2 ignition in the upper plenum, given that the ARFs and HIS are not operating, and the H_2 mole fraction is between 5.5 and 11%.
IGN-UC6 Q51 C6	0.0 0.25	Experts Mean=0.097	Rank 1	IGN-ICn IGN-UPn	Probability of H_2 ignition in the upper compartment, given that the ARFs and HIS are not operating, and the H_2 mole fraction is greater than 16%.
IGN-UC7 Q51 C7	0.0 0.25	Experts Mean=0.092	Rank 1	IGN-ICn IGN-UPn	Probability of H_2 ignition in the upper compartment, given that the ARFs and HIS are not operating, and the H_2 mole fraction is between 11 and 16%.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
IGN-UC8 Q51 C8	0.0 0.25	Experts Mean=0.083	Rank 1	IGN-ICn IGN-UPn	Probability of H_2 ignition in the upper compartment, given that the ARFs and HIS are not operating, and the H_2 mole fraction is between 5.5 and 11%.
H2-DDT1 Q52 C1 Q53 C1	0.5 1.0	Experts Mean=0.72	Rank 1	H2-DDTn	Probability of deflagration to detonation transition given H_2 ignition in the ice condenser or upper plenum and H_2 mole fraction greater than 21%.
H2-DDT2 Q52 C2 Q53 C2	0.5 1.0	Experts Mean=0.62	Rank 1	H2-DDTn	Probability of deflagration to detonation transition given H_2 ignition in the ice condenser or upper plenum and H_2 mole fraction from 16 to 21%.
H2-DDT3 Q52 C3 Q53 C3	0.1 1.0	Experts Mean=0.45	Rank 1	H2-DDTn	Probability of deflagration to detonation transition given H_2 ignition in the ice condenser or upper plenum and H_2 mole fraction from 14 to 16%.
DET-IMP Q55 C1 Q56 C1	0.0 59.4	Experts Mean=10.4	None		Impulse, in kPa-s, delivered by H_2 detonation in the ice condenser or upper plenum, given DDT.
CF-PRES Q57	274. 929.	Experts Mean=551.	None		Containment failure pressure, in kPa.
CF-MODE Q57	0.0 1.0	Uniform Mean=0.5	None		Random number used to select the containment failure mode.
CF-IMPUP Q57	0.5 48.	Experts Mean=12.	Rank 1	CF-IMPIC	Impulsive failure criteria, in kPa-s, for the upper plenum.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
CF-IMPIC Q57	0.7 64.	Experts Mean=22.	Rank 1	CF-IMPUP	Impulsive failure criteria, in kPa-s, for the ice condenser.
R-CAVITY Q63 C2	Zero One	Wet 0.5 D-Flood 0.5	None		Probability that the reactor cavity is either wet or deeply flooded at vessel breach.
VB-ALPHA Q64 C1	0.0 1.0	Experts Mean=.0085	None		Probability that an alpha mode CF occurs, given that the RCS is at low pressure. (One-tenth this value is used for high pressure, Q64 C2.)
TYPE-VB3 Q65 C3	Zero One	Experts HPME 0.79 BtmHd 0.08 Pour 0.13	Rank 1	TYPE-VB4	Type of VB given that the RCS is at setpoint pressure.
TYPE-VB4 Q65 C4,C5	Zero One	Experts HPME 0.60 BtmHd 0.27 Pour 0.13	Rank 1	TYPE-VB3	Type of VB given that the RCS is at high pressure.
FR-HPME Q66	0.0 0.60	Experts Mean=0.30	None		Fraction of core which participates in HPME at VB.
FR-ICIR2 Q68 C2	0.0 0.5	Internal Mean=0.15	Rank 1	FR-ICIRn	Fraction of FR-HPME that is diverted to the ICIR, given core ejection from the cavity, RCS pressure 200 psia, and FR-HPME > 0.40.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
FR-ICIR3 Q68 C3	0.0 1.0	Internal Mean=0.33	Rank 1	FR-ICIRn	Fraction of FR-HPME that is diverted to the ICIR, given core ejection from the cavity, RCS pressure 200 to 600 psia, and FR-HPME > 0.40.
FR-ICIR4 Q68 C4	0.0 1.0	Internal Mean=0.32	Rank 1	FR-ICIRn	Fraction of FR-HPME that is diverted to the ICIR, given core ejection from the cavity, RCS pressure 200 to 600 psia, and 0.20 < FR-HPME < 0.40.
FR-ICIR5 Q68 C5	0.0 1.0	Internal Mean=0.31	Rank 1	FR-ICIRn	Fraction of FR-HPME that is diverted to the ICIR, given core ejection from the cavity, RCS pressure 200 to 600 psia, and FR-HPME < 0.20.
FR-ICIR6 Q68 C6	0.0 1.0	Internal Mean=0.42	Rank 1	FR-ICIRn	Fraction of FR-HPME that is diverted to the ICIR, given core ejection from the cavity, RCS pressure greater than 1000 psia, and FR-HPME > 0.40.
FR-ICIR7 Q68 C8	0.0 1.0	Internal Mean=0.42	Rank 1	FR-ICIRn	Fraction of FR-HPME that is diverted to the ICIR, given core ejection from the cavity, RCS pressure greater than 1000 psia, and 0.20 < FR-HPME < 0.40.
FR-ICIR8 Q68 C8	0.0 1.0	Internal Mean=0.42	Rank 1	FR-ICIRn	Fraction of FR-HPME that is diverted to the ICIR, given core ejection from the cavity, RCS pressure greater than 1000 psia, and FR-HPME < 0.20.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
V-HSIZE Q72 C1	Zero One	Large 0.1 Small 0.9	None		Size of the hole in the vessel after ablation, given HPME.
DP1-VB4 Q73 C4,C7	0.0 360.	Experts Mean=135.	Rank 1	DP1-VBn DPX1-VBn	Pressure rise at VB, in kPa, given that either the cavity is deeply flooded or there is no HPME, a wet cavity and significant H_2 burned before VB. The ice condenser (IC) is intact.
DPX1-VB4 Q73 C4,C7	0.0 400.	Experts Mean=148.	Rank 1	DP1-VBn DPX1-VBn	Pressure rise at VB, in kPa, given that either the cavity is deeply flooded or there is no HPME, a wet cavity and significant H_2 burned before VB. The IC is non-functional.
DP1-VB5 Q73 C5	0.0 1300.	Experts Mean=325.	Rank 1	DP1-VBn DPX1-VBn	Pressure rise at VB, in kPa, given no HPME, a wet cavity, little H_2 burned before VB and IC intact.
DPX1-VB5 Q73 C5	0.0 1500.	Experts Mean=358.	Rank 1	DP1-VBn DPX1-VBn	Pressure rise at VB, in kPa, given no HPME, a wet cavity, little H_2 burned before VB and IC non-functional.
DP1-VB6 Q73 C6	0.0 940.	Experts Mean=215.	Rank 1	DP1-VBn DPX1-VBn	Pressure rise at VB, in kPa, given no HPME, a dry cavity, little H_2 burned before VB and IC intact.
DPX1-VB6 Q73 C6	0.0 1300.	Experts Mean=292.	Rank 1	DP1-VBn DPX1-VBn	Pressure rise at VB, in kPa, given no HPME, a dry cavity, little H_2 burned before VB and IC non-functional.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
DP1-VB8 Q73 C8	0.0 130.	Experts Mean=56.	Rank 1	DP1-VBn DPX1-VBn	Pressure rise at VB, in kPa, given no HPME, a wet cavity, significant H_2 burned before VB and IC intact.
DPX1-VB8 Q73 C8	0.0 150.	Experts Mean=63.	Rank 1	DP1-VBn DPX1-VBn	Pressure rise at VB, in kPa, given no HPME, a wet cavity, significant H_2 burned before VB and IC non-functional.
DP2-VB2 Q74 C2	0.0 960.	Experts Mean=363.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, large hole in vessel, a wet cavity, little H_2 burned before VB, and IC intact.
DPX2-VB2 Q74 C2,C11 Q74 C14	0.0 1200.	Experts Mean=590.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, large hole in vessel, little H_2 burned before VB, and IC non-functional.
DP2-VB3 Q74 C3,C6 Q74 C9	0.0 650.	Experts Mean=253.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, a wet cavity, little H_2 burned before VB, and IC intact.
DPX2-VB3 Q74 C3,C6 Q74 C9,C12 Q74 C15,C18	0.0 940.	Experts Mean=413.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, little H_2 burned before VB, and IC non-functional.
DP2-VB4 Q74 C4,C7 Q74 C10	0.0 510.	Experts Mean=194.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, a wet cavity, little H_2 burned before VB, and IC intact.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
DPX2-VB4 Q74 C4,C7 Q74 C10,C13 Q74 C16,C19	0.0 550.	Experts Mean=238.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, little H_2 burned before VB, and IC non-functional.
DP2-VB5 Q74 C5	0.0 900.	Experts Mean=328.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, small hole in vessel, a wet cavity, high in-vessel zirconium oxidation, little H_2 burned before VB, and IC intact.
DPX2-VB5 Q74 C5,C17	0.0 1200.	Experts Mean=567.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, small hole in vessel, high in-vessel Zr oxidation, little H_2 burned before VB, and IC non-functional.
DP2-VB8 Q74 C8	0.0 880.	Experts Mean=311.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, small hole in vessel, a wet cavity, low in-vessel zirconium oxidation, little H_2 burned before VB, and IC intact.
DPX2-VB8 Q74 C8	0.0 1200.	Experts Mean=537.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, small hole in vessel, a wet cavity, low in-vessel Zr oxidation, little H_2 burned before VB, and IC non-functional.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
DP2-VB11 Q74 C11	0.0 1000.	Experts Mean=428.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, large hole in vessel, a dry cavity, high in-vessel zirconium oxidation, little H_2 burned before VB, and IC intact.
DP2-VB12 Q74 C12	0.0 720.	Experts Mean=323.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, large hole in vessel, a dry cavity, high in-vessel zirconium oxidation, little H_2 burned before VB, and IC intact.
DP2-VB13 Q74 C13	0.0 420.	Experts Mean=190.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, large hole in vessel, a dry cavity, high in-vessel Zr oxidation, little H_2 burned before VB, and IC intact.
DP2-VB14 Q74 C14	0.0 990.	Experts Mean=419.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, large hole in vessel, a dry cavity, low in-vessel Zr oxidation, little H_2 burned before VB, and IC intact.
DP2-VB15 Q74 C15	0.0 690.	Experts Mean=305.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, large hole in vessel, a dry cavity, low in-vessel Zr oxidation, little H_2 burned before VB, and IC intact.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
DP2-VB16 Q74 C16	0.0 410.	Experts Mean=181.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, large hole in vessel, a dry cavity, low in-vessel zirconium oxidation, little H_2 burned before VB, and IC intact.
DP2-VB17 Q74 C17	0.0 790.	Experts Mean=342.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, small hole in vessel, a dry cavity, little H_2 burned before VB, and IC intact.
DP2-VB18 Q74 C18	0.0 560.	Experts Mean=252.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, large hole in vessel, a dry cavity, little H_2 burned before VB, and IC intact.
DP2-VB19 Q74 C19	0.0 340.	Experts Mean=154.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, large hole in vessel, a dry cavity, little H_2 burned before VB, and IC intact.
DP3-VB2 Q75 C2	0.0 840.	Experts Mean=308.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, a wet cavity, significant H_2 burned before VB, and IC intact.
DPX3-VB2 Q75 C2,C5 Q75 C8	0.0 1200.	Experts Mean=498.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, significant H_2 burned before VB, and IC non-functional.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
DP3-VB3 Q75 C3	0.0 620.	Experts Mean=231.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, a wet cavity, significant H_2 burned before VB, and IC intact.
DPX3-VB3 Q75 C3,C6 Q75 C9	0.0 940.	Experts Mean=366.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, significant H_2 burned before VB, and IC non-functional.
DP3-VB4 Q75 C4	0.0 490.	Experts Mean=183.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, a wet cavity, significant H_2 burned before VB, and IC intact.
DPX3-VB4 Q75 C4,C7 Q75 C10	0.0 550.	Experts Mean=215.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, significant H_2 burned before VB, and IC non-functional.
DP3-VB5 Q75 C5	0.0 960.	Experts Mean=335.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, large hole in vessel, a dry cavity, significant H_2 burned before VB and IC intact.
DP3-VB6 Q75 C6	0.0 640.	Experts Mean=290.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, large hole in vessel, a dry cavity, significant H_2 burned before VB and IC intact.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
DP3-VB7 Q75 C7	0.0 390.	Experts Mean=173.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, large hole in vessel, a dry cavity, significant H_2 burned before VB and IC intact.
DP3-VB8 Q75 C8	0.0 780.	Experts Mean=311.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, high FR-HPME, small hole in vessel, a dry cavity, significant H_2 burned before VB and IC intact.
DP3-VB9 Q75 C9	0.0 520.	Experts Mean=232.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, medium FR-HPME, small hole in vessel, a dry cavity, significant H_2 burned before VB and IC intact.
DP3-VB10 Q75 C10	0.0 330.	Experts Mean=144.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at int. pressure, low FR-HPME, small hole in vessel, a dry cavity, significant H_2 burned before VB and IC intact.
DP3-VB11 Q75 C11	0.0 1100.	Experts Mean=372.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, high FR-HPME, a wet cavity, and IC intact.
DPX3-VB11 Q75 C11,C14 Q75 C17	0.0 1300.	Experts Mean=641.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, high FR-HPME, and IC non-functional.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
DP3-VB12 Q75 C12	0.0 740.	Experts Mean=290.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, medium FR-HPME, a wet cavity, and IC intact.
DPX3-VB12 Q75 C12,C15 Q75 C18	0.0 940.	Experts Mean=464.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, medium FR-HPME, and IC non-functional.
DP3-VB13 Q75 C13	0.0 550.	Experts Mean=212.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, low FR-HPME, a wet cavity, and IC intact.
DPX3-VB13 Q75 C13,C16 Q75 C19	0.0 550.	Experts Mean=264.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, low FR-HPME, and IC non-functional.
DP3-VB14 Q75 C14	0.0 1100.	Experts Mean=459.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, high FR-HPME, large hole in vessel, a dry cavity, and IC intact.
DP3-VB15 Q75 C15	0.0 740.	Experts Mean=337.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, medium FR-HPME, large hole in vessel, a dry cavity, and IC intact.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
DP3-VB16 Q75 C16	0.0 430.	Experts Mean=197.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, low FR-HPME, large hole in vessel, a dry cavity, and IC intact.
DP3-VB17 Q75 C17	0.0 890.	Experts Mean=364.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, high FR-HPME, small hole in vessel, a dry cavity, and IC intact.
DP3-VB18 Q75 C18	0.0 590.	Experts Mean=264.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, medium FR-HPME, small hole in vessel, a dry cavity, and IC intact.
DP3-VB19 Q75 C19	0.0 360.	Experts Mean=160.	Rank 1	DP2-VBn DPX2-VBn DP3-VBn DPX3-VBn	Pressure rise at VB, in kPa, given HPME at high or setpoint pressure, low FR-HPME, small hole in vessel, a dry cavity, and IC intact.
CF-DCON2 Q78 C2	Zero One	Fail 0.01 NoFail 0.99	Rank 1	CF-DCONn	Probability of containment failure by direct contact of liner with core debris, given that less that 10 metric tons of core debris enters the ICIR.
CF-DCON3 Q78 C3	Zero One	Fail 0.31 NoFail 0.69	Rank 1	CF-DCONn	Probability of containment failure by direct contact of liner with core debris, given that 10 to 30 metric tons of core debris enters the ICIR.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
CF-DCON4 Q78 C4	Zero One	Fail 0.53 NoFail 0.47	Rank 1	CF-DCONn	Probability of containment failure by direct contact of liner with core debris, given that 30 to 50 metric tons of core debris enters the ICIR.
CF-DCON5 Q78 C5	Zero One	Fail 0.60 NoFail 0.40	Rank 1	CF-DCONn	Probability of containment failure by direct contact of liner with core debris, given that more than 50 metric tons of core debris enters the ICIR.
FR-MOXVB1 Q79 C1	0.0 0.20	Max. Entropy Mean=0.075	Rank 1	FR-MOXVB2	Fraction of potentially oxidizable metal in ejected core is oxidized at VB, given that HPME does not occur.
FR-MOXVB2 Q79 C2	0.5 1.0	Uniform Mean=0.75	Rank 1	FR-MOXVB1	Fraction of potentially oxidizable metal in ejected core is oxidized at VB, given that HPME occurs.
FR-H2CNS Q81	0.7 0.9	Max. Entropy Mean=0.775	None		Fraction of hydrogen in containment at VB consumed by burns.
L-PRESS4 Q94 C4	207. 276.	Uniform Mean=241.	Rank 1	L-PRESSn	Late pressure in containment, in kPa, given prompt CCI with low steam generation and no CHR.
L-PRESS5 Q94 C5	241. 310.	Uniform Mean=276.	Rank 1	L-PRESSn	Late pressure in containment, in kPa, given prompt CCI with high steam generation and no CHR.

Table 2.3-2 (continued)

Variable Question and Case	Range	Distribution	Correlation	Correlation With	Description
L-PRESS6 Q94 C6	172. 241.	Uniform Mean=207.	Rank 1	L-PRESSn	Late pressure in containment, in kPa, given that prompt CCI does not occur and there is no CHR.
VL-PRESS4 Q108 C4	138. 241.	Uniform Mean=190.	Rank 1	VL-PRESS5	Late pressure in containment, in kPa, given that prompt CCI occurs with containment heat removal; pressure due to non-condensible gases.
VL-PRESS5 Q108 C5	138. 345.	Uniform Mean=241.	Rank 1	VL-PRESS4	Late pressure in containment, in kPa, given that prompt CCI occurs and the steam concentration in containment is low.
VL-CCI Q111 C2	Zero One	CCI 0.75 NoCCI 0.25	None		Probability that core concrete attack ensues after late boiloff and very late containment failure.
POWERREC Q22 C3-C7 Q90 C3-C7 Q105 C3-C4			None		Variable used to select the probability that offsite power will be recovered in a specified time interval given that it was not recovered in a previous time interval.

2.4 Description of the Accident Progression Bins

As each path through the APET is evaluated, the result of that evaluation is stored by assigning it to an APB. This bin describes the evaluation in enough detail that a source term (release of radionuclides) can be calculated for it. The APBs are the means by which information is passed from the accident progression analysis to the source term analysis. A bin is defined by specifying the attribute or value for each of 14 characteristics or quantities which define certain features of the evaluation of the APET. Section 2.4.1 describes the 14 characteristics, and the values that each characteristic can assume. A more detailed description of the binner, discussing each case in turn, is contained in Appendix A.1.3. The binner itself, which is expressed as a computer input file, is listed in Appendix A.1.4. Section 2.4.2 contains a discussion of rebinning, a process that takes place between evaluating the APET (in which binning takes place) and the source term analysis. Section 2.4.3 describes a set of summary binning characteristics which is used in presenting the results of evaluating the APET.

2.4.1 Description of the Bin Characteristics

The binning scheme for Sequoyah uses 14 characteristics. That is, there are 14 types of information required to define a path through the APET. A bin is defined by specifying a letter for each of the 14 characteristics, where each letter for each characteristic has a meaning defined below. For a characteristic, the possible states are termed attributes. The Sequoyah binning characteristics are:

Characteristic	Mnemonic	Description
1	CF-Time	Time of containment failure
2	Sprays	Periods in which sprays operate
3	CCI	Occurrence of core-concrete interactions
4	RCS-Pres	RCS pressure before VB
5	VB-Mode	Mode of VB
6	SGTR	Steam generator tube rupture
7	Amt-CCI	Amount of core available for CCI
8	Zr-Ox	Fraction of zirconium oxidized in vessel
9	HPME	Fraction of the core in HPME
10	CF-Size	Size or type of containment failure
11	RCS-Hole	Number of large holes in the RCS after VB
12	E2-IC	Early ice condenser function
13	I2-IC	Late ice condenser function
14	ARFans	Status of air return fans

Most of this information, organized in this manner, is needed by SEQSOR to calculate the fission product source terms. Characteristic 5, mode of VB, is not used by SEQSOR, but has been retained because it provides interesting output information about the APET outcome, or the paths taken through the APET. SEQSOR obtains the information it needs concerning HPME from Characteristic 9, fraction of the core in HPME.

The remainder of this section contains a listing of each attribute for each characteristic, followed by a brief description of each characteristic, and finally an explanation of an example bin. The listing below provides the letter identifier for each attribute, as well as the mnemonic descriptor and definition for the attribute.

Characteristic 1 - Containment Failure Time

A	V-Dry	Event V, releases not scrubbed by fire sprays.
B	V-Wet	Event V, releases scrubbed by fire sprays.
C	CF-Early	Containment failure during core degradation.
D	CF-atVB	Containment failure at VB.
E	CF-Late	Late containment failure (during the initial part of CCI, nominally a few hours after VB).
F	CF-VLate	Very late containment failure (from 12 to 24 h after VB).
G	NoCF	No containment failure.

Characteristic 2 - Sprays

A	Sp-Early	The sprays operate only in the early period.
B	Sp-E+I	The sprays operate only in the early and intermediate periods.
C	Sp-E+I+L	The sprays operate only in the early, intermediate, and late periods.
D	SpAlways	The sprays always operate during the periods of interest for fission product removal.
E	Sp-Late	The sprays operate only in the late period.
F	Sp-L+VL	The sprays operate only in the late and very late periods.
G	Sp-VL	The sprays operate only in the very late period.
H	Sp-Never	The sprays never operate during the accident.
I	Sp-Final	The sprays operate only during the final period (not of interest for fission product removal).

Characteristic 3 - Core-Concrete Interactions

A	Prmt-Dry	CCI takes place promptly following VB. There is no overlying water to scrub the releases.
B	Prmt-Shl	CCI takes place promptly following VB. There is a shallow (about 5 ft) overlying water pool to scrub the releases.
C	No-CCI	CCI does not take place.
D	Prmt-Dp	CCI takes place promptly following VB. There is a deep (at least 10 ft) overlying water pool to scrub the releases.
E	SDly-Dry	CCI takes place after a short delay. The debris is initially coolable but limited cavity water is not replenished.
F	LDly-Dry	CCI takes place after a long delay. The debris is initially coolable but the large amount of cavity water is not replenished.

Characteristic 4 - RCS Pressure Before VB

A	SSPr	System setpoint pressure (2500 psia).
B	HiPr	High pressure (1000 to 2000 psia).
C	ImPr	Intermediate pressure (200 to 1000 psia).
D	LoPr	Low pressure (less than 200 psia).

Characteristic 5 - Mode of VB

A	VB-HPME	HPME occurs - direct containment heating (DCH) always occurs to some extent.
B	VB-Pour	The molten core pours out of the vessel, driven primarily by the effects of gravity.
C	VB-BtmHd	There is gross failure of a large portion of the bottom head of the vessel.
D	Alpha	An Alpha mode failure occurs which also results in CF.
E	Rocket	Upward acceleration of the vessel occurs which also results in containment failure (Rocket mode).
F	No-VB	No VB occurs.

Characteristic 6 - Steam Generator Tube Rupture

A	SGTR	An SGTR occurs. The SRVs on the secondary system are not stuck open.
B	SG-SRVO	An SGTR occurs. The SRVs on the secondary system are stuck open.
C	No-SGTR	An SGTR does not occur.

Characteristic 7 - Amount of Core not in HPME Available for CCI

A	Hi-CCI	A CCI occurs and involves a large amount of the core (70 to 100%).
B	Med-CCI	A CCI occurs and involves an intermediate amount of the core (30 to 70%).
C	Lo-CCI	A CCI occurs and involves a small amount of the core (0 to 30%).
D	No-CCI	No CCI occurs.

Characteristic 8 - Zr Oxidation

A	Lo-ZrOx	A small amount of the core zirconium was oxidized in the vessel before breach. This implies a range from 0 to 40% oxidized, with a nominal value of 25%.
B	Hi-ZrOx	A large amount of the core zirconium was oxidized in the vessel before breach. This implies that more than 40% was oxidized, with a nominal value of 65%.

Characteristic 9 - High Pressure Melt Ejection (HPME)

A	Hi-HPME	A high fraction (> 40%) of the core was ejected under pressure from the vessel at failure.
B	Md-HPME	A moderate fraction (20-40%) of the core was ejected under pressure from the vessel at failure.
C	Lo-HPME	A low fraction (< 20%) of the core was ejected under pressure from the vessel at failure.
D	No-HPME	There was no HPME at vessel failure.

Characteristic 10 - Containment Failure Size

A	Cat-Rpt	The containment failed by catastrophic rupture; resulting in a very large hole and gross structural failure.
B	Rupture	The containment failed by the development of a large hole or rupture; nominal hole size is 7 ft^2.
C	Leak	The containment failed by the development of a small hole or a leak; nominal hole size is 0.1 ft^2.
D	BMT	The containment failed by BMT.
E	Bypass	The containment was bypassed by Event V or an SGTR.
F	No-CF	The containment did not fail or was not bypassed.

Characteristic 11 - Holes in the RCS

A	1-Hole	There is a large hole in the RCS after VB, so there is no effective natural circulation through the RCS.
B	2-Holes	There are two large holes in the RCS after VB, so there is effective natural circulation through the RCS.

Characteristic 12 - Early Ice Condenser Function

A	E2-InByp	There is no bypass of the ice condenser during core degradation. The IC is intact and is credited with the full DF for the RCS releases.
B	E2-IpByp	There is partial bypass of the ice condenser during core degradation. The effective bypass level is nominally 10%, i.e., the ice condenser is credited with an effective DF that is 90% of the DF for E2-InByp.
C	E2-IByp	There is total bypass of the ice condenser or the ice is completely melted from the ice condenser during CD. If the ice is melted and the fans are operating, the ice condenser is credited with an effective DF that is 20% of the DF for E2-InByp.

Characteristic 13 - Late Ice Condenser Function

A	I2-InByp	There is no bypass of the ice condenser during the initial phase of CCI. The ice condenser is intact and is credited with the full DF for the CCI releases.
B	I2-IpByp	There is partial bypass of the ice condenser during the initial phase of CCI. The effective bypass level is nominally 10%, i.e., the ice condenser is credited with an effective DF that is 90% of the DF for I2-InByp.
C	I2-IByp	There is total bypass of the ice condenser, or the ice is completely melted from the ice condenser during the initial phase of CCI. If the ice is melted and the fans operating, the ice condenser is credited with an effective DF that is 20% of the DF for I2-InByp.

Characteristic 14 - Status of Air Return Fans

A	ARF-Erly	The air return fans operate only in the early period, i.e., before and during the RCS releases.
B	ARF-E+L	The air return fans operate in both the early and late periods, i.e., during RCS and CCI releases.
C	ARF-Late	The air return fans operate only in the late period, i.e., during the CCI releases.
D	No-ARF	The air return fans do not operate for the early or late periods.

Characteristic 1 primarily concerns the time of containment failure. There are seven attributes. Four of these attributes concern the time of containment failure, two concern Event V, and one is for no containment failure. SGTRs are considered separately in Characteristic 6 since an SGTR can occur in addition to one of the modes of containment failure. BMT and eventual overpressure failure due to the inability to restore CHR within the day following the accident are the failures that occur in the very late period.

Characteristic 2 concerns the periods in which the sprays operate. The sprays are important for reduction of aerosol concentrations in the containment atmosphere. The division of this characteristic into the nine attributes is a straightforward sorting out of the various combinations of time periods. The final time period is of little consequence for the fission product release, but it must be included because there are cases where the sprays operate only in this period, and, for each characteristic, the binner must have a location in which to place every outcome. As SEQSOR does not distinguish between 'sprays never operate', Attribute H, and 'sprays operate only in the final period,' Attribute I, these two are combined in the rebinner for SEQSOR.

Characteristic 3 concerns the CCI. There are six possibilities which cover the meaningful combinations of prompt CCI, delayed CCI, and no CCI, with the amount of water in the cavity. The amount of water in the cavity may be divided into three cases. If the cavity was dry at VB and the accumulators discharge before breach, the cavity is dry at the start of CCI. If the cavity was dry at VB and the accumulators discharge at breach, the cavity will be holding about 5 ft of water. If the RWST is injected into containment and there is about half of the ice melted before breach, then the cavity will be holding about 22 ft of water.

Characteristic 4 concerns the pressure in the reactor vessel before VB; there are four levels. The pressures shown in parentheses above are approximate pressures just before VB. The RCS pressure during most of the core degradation period may be less than the parenthetical values except for SSPr where the reclosing of the PORVs will keep the system pressure at the setpoint value.

Characteristic 5 concerns the mode of VB; there are six possibilities, including no VB. Direct heating of the containment always occurs to some extent if there is high pressure melt ejection (HPME), so there is no simple way to distinguish whether direct containment heating occurs.

Characteristic 6 concerns SGTR. There are only three possibilities: no SGTR, SGTR, and SGTR with the SRVs on the secondary system stuck open. SGTR is considered separately from the other containment failure modes since it can occur in addition to the other failure modes. That is, occurrence of an SGTR before VB does not preclude containment failure at VB or late containment failure. The SGTR creates a bypass of the containment which may have no removal mechanisms operating in the escape path, so it is important to treat it separately.

Characteristic 7 concerns the amount of core not participating in HPME that is available to participate in the CCI. The fractions 0.30 and 0.70 divide the range into three portions. The fourth attribute is no CCI. As SEQSOR subtracts out the fraction of the core involved in HPME, when HPME occurs, the fraction of the core available for CCI is always set to the first attribute, 'Hi-CCI.'

Characteristic 8 concerns the amount of the core zirconium oxidized in-vessel before VB. There are two possible values for this characteristic: low and high. The demarcation point between the two ranges is 40%.

Characteristic 9 concerns the amount of the core involved in HPME; there are four attributes. The possible range is divided into three portions by 20% and 40%. No occurrence of HPME is the fourth attribute.

Characteristic 10 concerns the size of the hole that results from containment failure or the type of containment failure. There are six attributes. The first three attributes concern failure of the containment wall above ground. BMT results in a release from the containment below ground. As SEQSOR does not address late containment failures involving BMT, they are assigned to late containment leaks in the rebinner. SEQSOR determines whether the containment was bypassed from Characteristic 1

(Event V) and Characteristic 6 (SGTR), so the bypass attribute is combined with the no containment failure attribute in the rebinner.

Characteristic 11 concerns the number of large holes in the RCS after breach. The experts on the source term panel who provided distributions for revolatilization from the RCS surfaces after VB gave different distributions depending on whether an effective natural circulation flow would be set up within the vessel. A significant flow could be expected only if there were two large, effective holes in the RCS; for example the hole in the bottom head resulting from vessel failure and a large temperature-induced hole in the hot leg. SGTR, failure of the RCP seals, and Event V would not count as large effective holes since effective natural circulation through the RCS would not result in these cases. S_3-size holes are not considered large enough to result in effective natural circulation after VB.

Characteristic 12 concerns the status of the ice condenser during the core degradation process. The ice condenser DF is important for the RCS releases. There are three attributes for this characteristic: no ice bypass, partial ice bypass, and total ice bypass. The ice may be partially bypassed due to hydrogen detonations or preferential melting and subsequent channelling. The ice condenser may be totally bypassed due to a rupture failure of containment in the lower compartment or due to breach of the boundary between the lower and upper compartments. For times of containment failure in which catastrophic rupture occurs, the ice condenser is assumed to be totally bypassed; however, Characteristic 12 does not reflect this method of bypass because SEQSOR already assumes ice bypass when catastrophic rupture occurs. Complete ice melt also constitutes total ice bypass.

Characteristic 13 concerns the status of the ice condenser during the initial phase of CCI. The ice condenser DF is important for the CCI releases. The attributes are identical to those for Characteristic 12: no ice bypass, partial ice bypass, and total ice bypass.

Characteristic 14 concerns the operation of the air return fans before VB and during the initial phase of CCI. This characteristic has four attributes and is used in conjunction with Characteristics 12 and 13 to establish the ice condenser DF. The Source Term Expert Panel members who evaluated the ice condenser DF, determined that the DF was sensitive to the number of passes through the ice condenser. If fans are operating, there is more than one pass through the ice beds, and if not operating, the aerosol-laden gases make only a single pass through the ice.

A typical bin might be FFADBCABDDBABC; which, using the information presented above, is:

F - CF-VLate	Very late containment failure
F - Sp-L+VL	Sprays only in the late and very late periods
A - Prmpt-Dry	Prompt CCI, dry cavity
D - LoPr	Low pressure in the RCS at VB
B - VB-Pour	Core material poured out of the vessel at breach
C - No-SGTR	No steam generator tube rupture
A - Lrg-CCI	A large fraction of the core was available for CCI

B - Hi-ZrOx	A high fraction of the Zr was oxidized in-vessel
D - No-HPME	No high pressure melt ejection
D - BMT	Basemat melt-through
B - 2-Holes	Two holes in the RCS
A - E2-InByP	No early bypass of the ice condenser
B - I2-IpByP	Partial bypass of the ice condenser during CCI
C - ARF-Late	The ARFs operate during CCI

2.4.2 Rebinning

The binning scheme used for the evaluation of the APET does not exactly match the input information required by SEQSOR. The additional information in the initial binning is kept because it provides a better record of the outcomes of the APET evaluation. Therefore, there is a step between the evaluation of the APET and the evaluation of SEQSOR known as "rebinning." In the rebinning, a few attributes in some characteristics are combined because there are no significant differences between them for calculating the fission product releases. Characteristic 5, Mode of VB, is not used by SEQSOR, but is not eliminated in the rebinning. The information SEQSOR requires about HPME is obtained from Characteristic 9.

In the rebinning for Sequoyah, there are no changes for Characteristics 1, 3, 4, 5, 6, 7, 8, 9, 11, 12, 13 and 14. That is, for these 12 characteristics, the information produced by the APET is exactly that used by SEQSOR. For Characteristic 2, the two final attributes (H - Sp-Never, and I - Sp-Final) are combined into Attribute H, Sp-NonOp, since the operation of sprays in the final period does not affect the amount of fission products released. For Characteristic 10, the third and fourth attributes (C - Leak, and D - BMT) are combined into Attribute C (Leak) since SEQSOR considers the radionuclides released from BMT to be the same as those released from a leak in this period. Also for Characteristic 10, the fifth and sixth attributes (E - Bypass and F - No-CF) are combined into a new Attribute D (No-CF) since the containment pressure boundary is not failed by a bypass and the releases from the bypass events (V and SGTR) are treated separately in SEQSOR. For the rebinned APET pathways, the following listing describes each attribute for each characteristic:

Characteristic 1 - Containment Failure Time (Rebinned)

A	V-Dry	Event V, releases not scrubbed by fire sprays.
B	V-Wet	Event V, releases scrubbed by fire sprays.
C	CF-Early	Containment failure during core degradation.
D	CF-atVB	Containment failure at VB.
E	CF-Late	Late containment failure (during the initial part of CCI, nominally a few hours after VB).
F	CF-VLate	Very late containment failure (from 12 to 24 h after VB).

G	NoCF	No containment failure.

Characteristic 2 - Sprays (Rebinned)

A	Sp-Early	The sprays operate only in the early period.
B	Sp-E+I	The sprays operate only in the early and intermediate periods.
C	Sp-E+I+L	The sprays operate only in the early, intermediate, and late periods.
D	SpAlways	The sprays always operate during the periods of interest for fission product removal.
E	Sp-Late	The sprays operate only in the late period.
F	Sp-L+VL	The sprays operate only in the late and very late periods.
G	Sp-VL	The sprays operate only in the very late period.
H	Sp-NonOp	The sprays never operate during the accident or operate only during the final period, which is not of interest for fission product removal.

Characteristic 3 - Core-Concrete Interactions (Rebinned)

A	Prmt-Dry	CCI takes place promptly following VB. There is no overlying water to scrub the releases.
B	Prmt-Shl	CCI takes place promptly following VB. There is a shallow (about 5 ft) overlying water pool to scrub the releases.
C	No-CCI	CCI does not take place.
D	Prmt-Dp	CCI takes place promptly following VB. There is a deep (at least 10 ft) overlying water pool to scrub the releases.
E	SDly-Dry	CCI takes place after a short delay. The debris is initially coolable but limited cavity water is not replenished.
F	LDly-Dry	CCI takes place after a long delay. The debris is initially coolable but the large amount of cavity water is not replenished.

Characteristic 4 - RCS Pressure Before VB (Rebinned)

A	SSPr	System setpoint pressure (2500 psia).
B	HiPr	High pressure (1000 to 2000 psia).
C	ImPr	Intermediate pressure (200 to 1000 psia).
D	LoPr	Low pressure (less than 200 psia).

Characteristic 5 - Mode of VB (Rebinned)

A	VB-HPME	HPME occurs - DCH always occurs to some extent.
B	VB-Pour	The molten core pours out of the vessel, driven primarily by the effects of gravity.
C	VB-BtmHd	There is gross failure of a large portion of the bottom head of the vessel.
D	Alpha	An Alpha mode failure occurs which also results in CF.
E	Rocket	Upward acceleration of the vessel occurs which also results in containment failure (Rocket mode).
F	No-VB	No VB occurs.

Characteristic 6 - Steam Generator Tube Rupture (Rebinned)

A	SGTR	An SGTR occurs. The SRVs on the secondary system are not stuck open.
B	SG-SRVO	An SGTR occurs. The SRVs on the secondary system are stuck open.
C	No-SGTR	An SGTR does not occur.

Characteristic 7 - Amount of Core not in HPME Available for CCI (Rebinned)

A	Hi-CCI	A CCI occurs and involves a large amount of the core (70 to 100%).
B	Med-CCI	A CCI occurs and involves an intermediate amount of the core (30 to 70%).
C	Lo-CCI	A CCI occurs and involves a small amount of the core (0 to 30%).
D	No-CCI	No CCI occurs.

Characteristic 8 - Zr Oxidation (Rebinned)

A	Lo-ZrOx	A small amount of the core zirconium was oxidized in the vessel before breach. This implies a range from 0 to 40% oxidized, with a nominal value of 25%.
B	Hi-ZrOx	A large amount of the core zirconium was oxidized in the vessel before breach. This implies that more than 40% was oxidized, with a nominal value of 65%.

Characteristic 9 - High Pressure Melt Ejection (HPME) (Rebinned)

A	Hi-HPME	A high fraction (> 40%) of the core was ejected under pressure from the vessel at failure.
B	Md-HPME	A moderate fraction (20 to 40%) of the core was ejected under pressure from the vessel at failure.
C	Lo-HPME	A low fraction (< 20%) of the core was ejected under pressure from the vessel at failure.
D	No-HPME	There was no HPME at vessel failure.

Characteristic 10 - Containment Failure Size (Rebinned)

A	Cat-Rpt	The containment failed by catastrophic rupture; resulting in a very large hole and gross structural failure.
B	Rupture	The containment failed by the development of a large hole or rupture; nominal hole size is 7 ft^2.
C	Leak	The containment failed by the development of a small hole or a leak (nominal size 0.1 ft^2), or BMT has occurred.
D	No-CF	The containment did not fail. It may have been bypassed.

Characteristic 11 - Holes in the RCS (Rebinned)

A	1-Hole	There is a large hole in the RCS after VB, so there is no effective natural circulation through the RCS.
B	2-Holes	There are two large holes in the RCS after VB, so there is effective natural circulation through the RCS.

Characteristic 12 - Early Ice Condenser Function (Rebinned)

A	E2-InByp	There is no bypass of the ice condenser (IC) during core degradation (CD). The ice condenser is intact and is credited with the full DF for the RCS releases.
B	E2-IpByp	There is partial bypass of the ice condenser during CD. The effective bypass level is nominally 10%, i.e., the ice condenser is credited with an effective DF that is 90% of the DF for E2-InByp.
C	E2-IByp	There is total bypass of the ice condenser or the ice is completely melted from the the ice condenser during CD. If the ice is melted and the fans are operating, the ice condenser is credited with an effective DF that is 20% of the DF for E2-InByp.

Characteristic 13 - Late Ice Condenser Function (Rebinned)

A	I2-InByp	There is no bypass of the ice condenser during the initial phase of CCI. The ice condenser is intact and is credited with the full DF for the CCI releases.
B	I2-IpByp	There is partial bypass of the ice condenser during the initial phase of CCI. The effective bypass level is nominally 10%, i.e., the ice condenser is credited with an effective DF that is 90% of the DF for I2-InByp.
C	I2-IByp	There is total bypass of the ice condenser, or the ice is completely melted from the ice condenser during the initial phase of CCI. If the ice is melted and the fans are operating, the ice condenser is credited with an effective DF that is 20% of the DF for I2-InByp.

Characteristic 14 - Status of Air Return Fans (Rebinned)

A	ARF-Erly	The air return fans (ARFs) operate only in the early period, i.e., before and during the RCS releases.
B	ARF-E+L	The ARFs operate in both the early and late periods, i.e., during RCS and CCI releases.
C	ARF-Late	The ARFs operate only in the late period, i.e., during the CCI releases.
D	No-ARF	The ARFs do not operate for the early or late periods.

In the rebinning process, bin FFADBCABDDBABC used as an example above, becomes FFADBCABDCBABC since rebinning affects the tenth characteristic:

F - CF-VLate	Very late containment failure
F - Sp-L+VL	Sprays only in the late and very late periods
A - Prmpt-Dry	Prompt CCI, dry cavity
D - LoPr	Low pressure in the RCS at VB
B - VB-Pour	Core material poured out of the vessel at breach
C - No-SGTR	No steam generator tube rupture
A - Lrg-CCI	A large fraction of the core was available for CCI
B - Hi-ZrOx	A high fraction of the zirconium was oxidized in vessel
D - No-HPME	No high pressure melt ejection
C - Leak	Leak (includes BMT)
B - 2-Holes	Two holes in the RCS
A - E2-InByP	No early bypass of the ice condenser
B - I2-IpByP	Partial bypass of the ice condenser during CCI
C - ARF-Late	The ARFs operate during CCI

2.4.3 Summary Bins for Presentation

For presentation purposes in NUREG-1150,[10] a set of "summary" bins has been adopted. Instead of the 14 characteristics and thousands of possible bins that describe the evaluation of the APET in detail, the summary bins place the outcomes of the evaluation of the APET into a few, very general groups. The ten summary bins for Sequoyah are:

VB, very early CF, during CD or isolation failures
VB, early CF (at VB), Alpha mode
VB, early CF (at VB), RCS pressure > 200 psia
VB, early CF (at VB), RCS pressure < 200 psia
VB, late CF
VB, BMT and very late CF
Bypass
VB, no CF
No VB, very early CF, during CD or isolation failures
No VB, no CF

This order is that used in displays. It has containment failure with VB first, then bypass, then vb with no containment failure, then no VB with early containment failure, and finally, no VB. Containment failure is divided into seven subsets, which are listed roughly in decreasing order of the severity of the resulting release.

In assigning bins to one of these summary bins, however, the summary bins must be considered in the following order:

Bypass

VB, early containment failure, Alpha mode

No VB, very early containment failure, during CD or isolation failures

No VB, no containment failure

VB, very early containment failure, during CD or isolation failures

VB, early containment failure, RCS pressure >200 psia

VB, early containment failure, RCS pressure <200 psia

VB, late containment failure

VB, BMT and very late containment failure

VB, no containment failure

That is, if bypass and early containment failure both occur, the resulting bin assignment is the Bypass bin since bypass occurs first in this list. The reason that the summary bins must have a definite assignment priority is that all possible outcomes do not fit neatly into the 10 summary bins. There are certain combinations of events that can be put in different places in the summary bins and there are other combinations of events that do not fit well in any of the summary bins. None of these combinations are very frequent occurrences, but they must be assigned to one of the 10 summary bins. The principle determining the summary bin is that the release path that results in the highest offsite risk should determine the summary bin. Thus the summary bins reflect the logic used by SEQSOR in calculating the source terms.

As an example, consider Event V followed by an Alpha mode failure of the vessel and containment. This results in bypass and early containment failure. Should this be assigned to the Alpha summary bin, or the Bypass summary bin? By the priority list above, it is placed in the Bypass summary bin. The reason is that almost all of the fission products released from the core before VB will have escaped to the auxiliary building through the bypass before VB. Thus this path determines most of the risk. Although SEQSOR treats the CCI release as if all of it escapes through the ruptured containment, the early release is more important for determining offsite risk.

The placement in summary bins of four other ambiguous combinations of events is discussed below.

Combination 1: Event V and Containment Failure During CD

The fission product release from Event V with a very early containment failure (as calculated by SEQSOR) is very similar to the release from Event V without a very early failure, and quite dissimilar to the releases from accidents with a very early failure but no bypass of the containment. Therefore, this combination is placed in the Bypass summary bin.

Combination 2: Event V and Containment Failure at VB

This combination is analogous to the situation in which Event V is followed by an Alpha mode failure of the containment just discussed, except that the containment fails at VB for other reasons. It is also placed in the Bypass summary bin.

Combination 3: SGTR and noVB

In this scenario, vessel failure is avoided but there may be considerable core damage, and the fission products from the degradation of the core have an escape path to the environment through the secondary system. It is not possible in this analysis to determine how much core damage occurs before the arrest of the degradation process. For this combination of events, SEQSOR calculates a SGTR release assuming that the degradation proceeds to the point of VB. If the core degradation is arrested very late, this is probably a reasonable assumption. Thus, the SGTR and noVB combination is placed in the Bypass summary bin. This combination is very infrequent; there are only two PDSs with an initiating SGTR that may have no VB. These are GLYY-YXY in the ATWS PDS Group, and GLYY-YNY in the SGTR PDS Group, each of which contribute less than 1% to the total mean core damage frequency. PDSs in which temperature-induced SGTRs occur may result in this combination of events, but temperature-induced SGTRs are very unlikely.

Combination 4: SGTR and Containment Failure at VB

SEQSOR was designed to treat SGTRs in addition to other failures of the containment, so this combination of events poses no special problem for the source term calculation. As the SGTR largely determines the early release, and the early release is more important than the late release, this combination is placed in the Bypass summary bin. An Alpha mode failure is also a containment failure at VB, so an SGTR followed by an Alpha event is also placed in the Bypass summary bin.

Thus, in assigning combinations of events in the APET to summary bins, bypass failures (V and SGTR) take precedence no matter what else happens or does not happen. Alpha mode failures take precedence over other failure modes at VB, and over very early failures. No VB is above containment failure before VB and late containment failure in the priority list; these failures are not possible without breach of the vessel, so that combination will not arise. The 10 summary bins may now be defined as follows:

Bypass	Includes Event V and SGTRs no matter what happens to the containment after the start of the accident; it also includes SGTRs which do not result in VB.
Alpha	Includes all accidents that have an alpha mode failure of the vessel and the containment except those that follow Event V or an SGTR. It includes Alpha mode failures that follow very early failures due to hydrogen events or isolation failures because the alpha mode failure is of rupture size.

No VB, V Early CF — Includes the accident progressions in which failure of the vessel is avoided and in which containment is failed during the core degradation process and no bypass of containment occurs. The bins placed in this summary bin have very early containment failure that involve early hydrogen burns or detonations or involve failure to isolate the containment at the start of the accident.

No VB, no CF — Includes the accident progressions that avoid vessel failures except those which fail very early or bypass the containment. The bins placed in this summary bin involve no failures of containment due to events at VB, late hydrogen burns, late overpressure or BMT.

VB, V Early CF — Includes all accidents in which the vessel is breached and there is either an isolation failure at the start of the accident, or the containment fails before VB due to a hydrogen event. Not included are accidents involving bypass events and very early containment failures.

CF at VB, RCS HiPr — Implies containment failure at VB with the RCS above 200 psia when the vessel fails. It does not include Alpha mode failures, containment failures before VB, or bins in which containment failure at VB follows Event V or an SGTR.

CF at VB, RCS LoPr — Implies containment failure at VB with the RCS below 200 psia when the vessel fails. It does not include Alpha mode failures, containment failures before VB, or bins in which containment failure at VB follows Event V or an SGTR.

Late CF — Includes accidents in which the containment was not failed or bypassed before the onset of CCI and in which the vessel failed. The failure mechanism is hydrogen combustion during CCI.

V Late CF — Includes accidents in which the containment was not failed or bypassed before the latter stages of CCI. The failure mechanisms are eventual overpressure within 24 h due to noncondensibles and/or steam, or BMT in several days.

VB No CF — Includes all the accidents not in one of the previous summary bins. The vessel's lower head is penetrated by the core, but the containment does not fail and is not bypassed.

2.5 Results of the Accident Progression Analysis

This section presents the results of evaluating the APET. As evaluating the APET produces a number of APBs, the discussion is primarily in terms of APBs. Some intermediate results are also presented. Sensitivity analyses are discussed as well.

Section 2.5.1 presents the results for the internal initiators. Section 2.5.2 discusses the sensitivity analyses run for the internal initiators. Externally initiated events (seismic and fire) were not considered for the Sequoyah analysis. The tables in this section present only a very small portion of the output obtained by evaluating the APETs. Complete listings giving average bin conditional probabilities for each PDS Group, and listings giving the bin probabilities for each PDS Group for each observation are available on computer media by request.

2.5.1 Results for Internal Initiators

2.5.1.1 Results for PDS Group 1 - Slow SBO. This PDS Group consists of accidents in which all ac power is lost in the plant, but the steam turbine-driven AFWS operates for several hours. The operation of this system keeps the core covered and cooled as long as there is no water loss from the RCS. Until the batteries deplete, dc power is available. When the batteries deplete, control of the steam turbine-driven AFWS is lost and it fails.

This PDS Group contains four PDSs: one has the RCS intact at UTAF, two have failure of the RCP seals before UTAF, and one has stuck-open PORVs before UTAF. In two of the four PDSs, the operators depressurized the secondary system before UTAF, and in two PDSs they did not. The PDSs in this group are listed in Table 2.2-2.

Table 2.5-1 lists the five most probable APBs for the PDS Group, the five most probable APBs that have VB, and the five most probable APBs that have VB and early containment failure. Most probable means most probable when the whole sample of 200 observations is considered; that is, the five most probable bins are the top five when ranked by mean probability condi-tional on the occurrence of the PDS Group. In Table 2.5-1, the "Order" column gives the order of the bin when ranked by conditional probability. The "Prob." column lists mean APB probabilities conditional on the occurrence of the PDS Group. That is, this table shows the results averaged over the 200 observations that form the sample. If Bin A occurred with a probability of 0.005 for each observation, its probability would be 0.005 in Table 2.5-1. If Bin B occurred with a probability of 1.00 for one observation and did not occur in the other 199 observations, its probability would also be 0.005. The column headed "Occ." gives the number of observations out of the 200 in the sample in which this APB occurred with a non-zero probability.

The remaining eleven columns in Table 2.5-1 explain 11 of the 14 characteristics in the APB indicator. The sixth characteristic, SGTR, has been omitted since few of the bins and none of the 100 most probable bins for this PDS Group had T-I SGTR. The eleventh characteristic, RCS-Hole, and the last characteristic, ARFans, have been omitted since they are of less interest than the others. The abbreviations for each APB characteristic are explained in Section 2.4 above.

The first part of Table 2.5-1 shows the first five bins when they are ranked in order by probability. Evaluation of the APET produced 8184 bins for the Slow SBO PDS Group. To capture 95% of the probability, 1895 bins are required. The five most probable bins capture 22% of the probability

Table 2.5-1
Results of the Accident Progression Analysis for Sequoyah
Internal Initiators - PDS Group 1: Slow SBO

Five Most Probable Bins*

Order	Bin	Prob.**	Occ.	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
1	GDCCFCDADFAAAB	0.050	38	No-CF	Always	No-CCI	ImPr	No-VB	No-CCI	Lo	No	No-CF	noByP
2	GDCDFCDADFAAAB	0.044	36	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Lo	No	No-CF	noByP
3	GDCBFCDADFAAAB	0.044	42	No-CF	Always	No-CCI	HiPr	No-VB	No-CCI	Lo	No	No-CF	noByP
4	GDCDFCDADFBAAB	0.044	114	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Lo	No	No-CF	noByP
5	GDCBFCDBDFAAAB	0.042	31	No-CF	Always	No-CCI	HiPr	No-VB	No-CCI	Hi	No	No-CF	noByP

Five Most Probable Bins that have VB*

Order	Bin	Prob.**	Occ.	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
18	EEADBCAADABAAC	0.006	59	Late	Late	PrmDry	LoPr	Pour	Large	Lo	No	CatRu	noByP
20	GGADBCABDFBAAD	0.006	104	No-CF	VLate	PrmDry	LoPr	Pour	Large	Hi	No	No-CF	noByP
22	GGADBCAADFBAAD	0.005	87	No-CF	VLate	PrmDry	LoPr	Pour	Large	Lo	No	No-CF	noByP
23	GFADBCABDFBAAC	0.005	96	No-CF	L+VL	PrmDry	LoPr	Pour	Large	Hi	No	No-CF	noByP
24	EEADBCAADAAAAC	0.005	15	Late	Late	PrmDry	LoPr	Pour	Large	Lo	No	CatRup	noByP

Five Most Probable Bins that have VB and Early CF*

Order	Bin	Prob.**	Occ.	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
32	DHADBCAADAAAAD	0.004	7	CFatVB	Never	PrmDry	LoPr	Pour	Large	Lo	No	CatRup	noByP
45	DHADBCAADAAAAC	0.003	7	CFatVB	Never	PrmDry	LoPr	Pour	Large	Lo	No	CatRup	noByP
60	DHADBCAADABAAD	0.002	15	CFatVB	Never	PrmDry	LoPr	Pour	Large	Lo	No	CatRup	noByP
65	DHADBCAADABAAC	0.002	15	CFatVB	L+VL	PrmDry	LoPr	Pour	Large	Lo	No	CatRup	noByP
67	DHADBCABDABAAD	0.002	18	CFatVB	L+VL	PrmDry	LoPr	Pour	Large	Hi	No	CatRup	noByP

* A listing of all bins, and a listing by observation are available on computer media.
** Mean probability conditional on the occurrence of the PDS.

and have no VB and no containment failure. Two of the five most probable bins with VB result in late containment failure (due to hydrogen burns), and all have the RCS at low pressure (less than 200 psia) at VB.

The last part of Table 2.5-1 shows the five most probable APBs with VB and early containment failure. (Early containment failure means containment failure before or at VB). The five bins with containment failure at VB have the RCS at low pressure at VB, and have catastrophic rupture of the containment due to hydrogen burns at VB. As mentioned in Section 2.4.1, for times of containment failure in which catastrophic rupture occurs, the ice condenser is assumed to be bypassed; however, Characteristics 12 and 13 do not reflect this method of bypass because SEQSOR already assumes ice bypass when catastrophic rupture occurs.

In this PDS Group, the probability of recovering offsite electrical power early in the accident is about 0.69. The probability of subsequent arrest of the core degradation process and the prevention of VB is about 0.58. More detail on the arrest of core damage may be found in Appendix A.3.3.

Of the fraction of this PDS Group which resulted in VB, most had the RCS at low pressure at VB. The fractions of this PDS Group which are in the four pressure ranges at UTAF and just before VB (if it occurs) are:

	At UTAF	Just before VB
SSPr (2500 psia)	0.17	0.005
HiPr (600-2000 psia)	0.20	0.23
ImPr (200-600 psia)	0.63	0.26
LoPr (< 200 psia)	0.00	0.50

The relative frequencies of the "T", "S_3", and "S_2" PDSs, in conjunction with whether the secondary system has been depressurized while the AFWS is operating, result in about 17% the PDS Group being at the PORV setpoint pressure when the core uncovers (Question 16). Just before VB, the situation is quite different (Question 25). Five mechanisms for depressurizing the RCS are considered in the APET. Three of these are quite effective: RCP seal failures, PORVs sticking open, and temperature-induced hot leg (or surge line) failures. The result is that the probability of the accident continuing with the RCS pressure boundary intact from UTAF to VB is about 0.03. The determination of RCS pressure at VB is discussed further in Section 2.5.2.1.

The mean probability of containment failure during core degradation due to hydrogen burns or detonations for this PDS Group is 0.06; 0.01 of these failures also involve VB. The mean probability of containment failure at VB is 0.10. Note that the 0.90 probabi- lity of no containment failure at VB includes the times when the containment failed during core degradation and also when VB was arrested. The mean probability of late containment failure due to hydrogen burns is 0.10. The mean probability of very late containment failure due to overpressure by steam and/or noncondensibles is 0.004. The mean probability of BMT is 0.05.

2.5.1.2 <u>Results for PDS Group 2 - Fast SBO</u>. This PDS Group consists of accidents in which all ac power is lost in the plant and the steam turbine-driven AFWS fails at, or shortly after, the start of the accident. The Fast SBO PDS Group consists of only one PDS, TRRR-RSR. Table 2.5-2 lists the five most probable APBs for the Fast SBO PDS Group, the five most probable APBs that have VB, and the five most probable APBs that have VB and early containment failure (CF).

The first part of Table 2.5-2 shows the first five bins when they are ranked in order by probability. Evaluation of the APET produced 7883 bins for the Fast SBO PDS Group, of which 1768 are required to capture 95% of the probability. The five most probable bins capture 14% of the probability. Four have no containment failure, and three of them have no VB as well. Two of the five most probable bins that have VB have no containment failure; one has containment failure due to hydrogen burns at VB, and the other two have failures due to late hydrogen burns. The last part of Table 2.5-2 shows the five most probable APBs with both VB and early containment failure. (Early containment failure means containment failure before or at VB.) Four of these have containment failure due to hydrogen burns at VB and the other one has containment failure due to HPME and DCH at VB.

In this PDS Group, the probability of recovering offsite electrical power early in the accident is about 0.41. The probability of subsequent arrest of the core degradation process and the prevention of VB is about 0.35. More detail on the arrest of core damage may be found in Appendix A.3.3.

Of the fraction of this PDS Group that resulted in VB, most had the RCS at low pressure at VB. The fractions of this PDS Group which are in the four pressure ranges at UTAF and just before VB (if it occurs) are:

	At UTAF	Just before VB
SSPr (2500 psia)	1.00	0.03
HiPr (600-2000 psia)	0.00	0.11
ImPr (200-600 psia)	0.00	0.25
LoPr (< 200 psia)	0.00	0.61

As the only PDS in this Group has the RCS intact at UTAF, the RCS is at the PORV setpoint pressure at that time (Question 16). Just before VB (Question 25), the probability of being at SSPr is only about 0.03. As discussed with regard to PDS Group 1, three of the five depressurization mechanisms considered in the APET are quite effective: RCP seal failures, PORVs sticking open, and temperature-induced hot leg (or surge line) failures. The result is that the probability of the accident continuing with the RCS pressure boundary intact from UTAF to VB is fairly small. The determination of RCS pressure at VB is discussed in Sections 2.5.2.1 and 2.5.2.2.

The mean probability of containment failure during core degradation due to hydrogen events is 0.05; 0.02 of these failures also involve VB. The mean probability of containment failure at VB is 0.13. Note that the 0.87 probability of no containment failure at VB includes the times when the containment failed during core degradation and also when VB was arrested. The mean probability of late containment failure due to hydrogen burns is 0.18. The mean probability of very late containment failure due to overpressure by steam and/or noncondensibles is 0.002. The mean probability of BMT is 0.08.

Table 2.5-2
Results of the Accident Progression Analysis for Sequoyah
Internal Initiators - PDS Group 2: Fast SBO

Five Most Probable Bins*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
1	GDCDFCDBDFBAAB	0.050	122	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Hi	No	No-CF	noByP
2	GDCDFCDADFBAAB	0.034	114	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Lo	No	No-CF	noByP
3	GFADBCABDFBAAC	0.028	96	No-CF	L+VL	PrmDry	ImPr	Pour	Large	Hi	No	No-CF	noByP
4	GDCCFCDADFAAAB	0.016	38	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Lo	No	No-CF	noByP
5	EEADBCAADABAAC	0.015	59	Late	Late	PrmDry	ImPr	Pour	Large	Lo	No	CatRup	noByP

Five Most Probable Bins that have VB*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
3	GFADBCABDFBAAC	0.028	96	No-CF	L+VL	PrmDry	ImPr	Pour	Large	Hi	No	No-CF	noByP
5	EEADBCAADABAAC	0.015	59	Late	Late	PrmDry	ImPr	Pour	Large	Lo	No	CatRup	noByP
8	EEADBCABDABAAC	0.014	50	Late	Late	PrmDry	LoPr	Pour	Large	Hi	No	CatRup	noByP
12	GFADBCAADFBAAC	0.010	71	No-CF	L+VL	PrmDry	LoPr	Pour	Large	Lo	No	No-CF	noByP
13	DHADBCABDABAAC	0.010	18	CFatVB	Never	PrmDry	LoPr	Pour	Large	Hi	No	CatRup	noByP

Five Most Probable Bins that have VB and Early CF*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
13	DHADBCABDABAAC	0.010	18	CFatVB	Never	PrmDry	LoPr	Pour	Large	Hi	No	CatRup	noByP
26	DHADBCAADABAAC	0.006	15	CFatVB	Never	PrmDry	LoPr	Pour	Large	Lo	No	CatRup	noByP
31	DHADBCABDABAAD	0.006	18	CFatVB	Never	PrmDry	LoPr	Pour	Large	Hi	No	CatRup	noByP
46	DHADBCAADABAAD	0.004	15	CFatVB	Never	PrmDry	LoPr	Pour	Large	Lo	No	CatRup	noByP
53	DFABACABBCAACC	0.003	5	CFatVB	L+VL	PrmDry	HiPr	HPME	Large	Hi	Med	Leak	noByP

* A listing of all bins, and a listing by observation are available on computer media.
** Mean probability conditional on the occurrence of the PDS.

2.5.1.3 Results for PDS Group 3 - LOCAs. This PDS Group consists of accidents initiated by a break in the RCS pressure boundary. Four of the PDSs have A-size breaks, and three have S_1-breaks (treated as A breaks in this analysis). There are three PDSs with S_2-breaks and three PDSs with S_3-breaks in this group. These PDSs result in core damage because of failure of one or more of the ECCS that are required to respond. Five of the 13 PDSs in this group have the LPIS operating but not injecting at UTAF. The PDSs in this group are listed in Table 2.2-2.

Table 2.5-3 lists the five most probable APBs for this PDS Group, the five most probable APBs that have VB, and the five most probable APBs that have VB and early containment failure (CF). Evaluation of the APET produced 6728 bins for the LOCA PDS Group. To capture 95% of the probability, 1101 bins are required. The five most probable bins capture 25% of the probability. The five most probable bins all have no VB and no containment failure as well. One of the five most probable bins that have VB has no containment failure; the other four have very late failure due to steam overpressure. The last part of Table 2.5-3 shows the five most probable APBs with both VB and early containment failure. All of these bins have failure due to HPME and DCH, and occur infrequently; all five appear in either one or two sample observations.

In the LOCA PDS Group, the probability of arresting the core degradation process and avoiding VB is about 0.37. For three of the PDSs, the LPIS is operating at UTAF and the break (A or S_1) is large enough by itself to depressurize the RCS to the point where the LPIS may inject. These are core damage situations because the success criteria require the accumulators (A break) or HPIS (S_1 break) to function in addition to LPIS, and these systems failed. For two other PDSs, the LPIS is operating at UTAF, but the initiating break (S_2 or S_3) is not large enough to depressurize the RCS so the LPIS can inject. The RCS is partially depressurized at UTAF due to secondary side depressurization. During core degradation, repressurization or further depressurization may occur. If the RCS is sufficiently depressurized, then LPIS operation is likely to prevent VB by halting core degradation.

The fractions of the LOCA PDS Group which are in the four pressure ranges at UTAF and just before VB (if it occurs) are:

	At UTAF	Just before VB
SSPr (2500 psia)	0.00	0.00
HiPr (600-2000 psia)	0.00	0.17
ImPr (200-600 psia)	0.69	0.20
LoPr (< 200 psia)	0.31	0.63

As with all accidents in which ac power is initially available, the hydrogen threat is negligible due to the low probability of operator failure to initiate igniters and the low probability that the air return fans fail. The mean probability of containment failure during core degradation, due mainly to isolation failures, is low, only 0.004. The mean probability of containment failure at VB is 0.05. Note that the 0.95 probability of no containment failure at VB includes the times when the containment failed during core degradation and also when VB was arrested.

Table 2.5-3
Results of the Accident Progression Analysis for Sequoyah
Internal Initiators - PDS Group 3: LOCAs

Five Most Probable Bins*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
1	GDCDFCDADFBAAB	0.091	114	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Lo	No	No-CF	noByP
2	GDCDFCDBDFBAAB	0.074	122	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Hi	No	No-CF	noByP
3	GDCDFCDADFBAAA	0.030	114	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Lo	No	No-CF	noByP
4	GDCDFCDBDFAAAB	0.028	42	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Hi	No	No-CF	noByP
5	GDCDFCDADFAAAB	0.026	36	No-CF	Always	No-CCI	ImPr	No-VB	No-CCI	Lo	No	No-CF	noByP

Five Most Probable Bins that have VB*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
7	FHDDBCAADBBAAB	0.015	59	VLate	Never	PrmDp	LoPr	Pour	Large	Lo	No	Rupt	noByP
8	FHDDBCABDBBAAB	0.012	50	VLate	Never	PrmDp	LoPr	Pour	Large	Hi	No	Rupt	noByP
9	FHDDBCAADABAAB	0.011	41	VLate	Never	PrmDp	LoPr	Pour	Large	Lo	No	CatRup	noByP
10	FHDDBCABDABAAB	0.010	38	VLate	Never	PrmDp	LoPr	Pour	Large	Hi	No	CatRup	noByP
14	GDDDBCAADFBAAB	0.009	111	No-CF	Always	PrmDp	LoPr	Pour	Large	Lo	No	No-CF	noByP

Five Most Probable Bins that have VB and Early CF*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
132	DACBACDBBAAAAB	0.001	1	CFatVB	Early	No-CCI	HiPr	HPME	No-CCI	Hi	Med	CatRup	noByP
156	DACCACDAAAAAAB	0.001	2	CFatVB	Early	No-CCI	ImPr	HPME	No-CCI	Lo	Hi	CatRup	noByP
164	DACBACDBBAAAAA	0.001	1	CFatVB	Early	No-CCI	HiPr	HPME	No-CCI	Hi	Med	CatRup	noByP
165	DACBACDABAAAAB	0.001	1	CFatVB	Early	No-CCI	HiPr	HPME	No-CCI	Lo	Med	CatRup	noByP
193	DACCACDAAAAAAA	0.001	2	CFatVB	Early	No-CCI	ImPr	HPME	No-CCI	Lo	Hi	CatRup	noByP

* A listing of all bins, and a listing by observation are available on computer media.
** Mean probability conditional on the occurrence of the PDS.

The mean probability of late containment failure due to hydrogen burns is 0.001. Because the sprays are failed in many LOCA sequences, and the ice is melted at late time, the mean probability of very late containment failure due to overpressure by steam and/or noncondensibles is quite high, 0.22. The mean probability of BMT is 0.04.

2.5.1.4 Results for PDS Group 4 - Event V. This PDS Group consists of accidents in which the check valves between the RCS and the LPIS fail, and then the LPIS, subjected to pressures much higher than those for which it was designed, also fails. This produces a path from the RCS to the auxiliary building, bypassing the containment, and is known as Event V. It is expected, because of the location of the break in the LPIS, that there is a considerable probability (0.80) that that the fire sprays in the auxiliary building would scrub the releases.

Table 2.5-4 lists the 10 most probable APBs for the V PDS Group. Evaluation of the APET produced 105 bins, of which 15 are required to capture 95% of the probability. The 10 most probable bins capture 84% of the probability, and for eight of them, the releases are scrubbed.

There is no possibility of avoiding VB or CCI in this PDS Group. Due to the size of the containment bypass, containment failure is not of much interest; nonetheless, it will be reported here. The mean probability of containment failure during core degradation due to isolation failures is 0.004. The mean probability of containment failure at VB due to Alpha mode failure or hydrogen burn is 0.02. The mean probability of late containment failure due to hydrogen burns is 0.009. There are no very late containment failures due to overpressure by steam and/or noncondensibles. The mean probability of BMT is quite high, 0.39.

2.5.1.5 Results for PDS Group 5 - Transients. This PDS Group consists of accidents in which the RCS is intact but there is no way to remove heat from the core. The AFWS fails at the start of the accident; bleed and feed is ineffective because the HPIS fails or the PORVs cannot be opened. The Transient PDS Group consists of two PDSs, TBYY-YNY and TINY-NNY. Table 2.5-5 lists the five most probable APBs for the PDS Group, the five most probable APBs that have VB, and the five most probable APBs that have VB and early containment failure. Evaluation of the APET produced 2619 bins for the Transient PDS Group, of which 160 are required to capture 95% of the probability.

The five most probable bins capture 49% of the probability. They all have no VB, and no containment failure as well. All of the five most probable bins that have VB have no containment failure. The last part of Table 2.5-5 shows the five most probable APBs with both VB and early containment failure. One of the five has containment failure due to hydrogen burn at VB, and the remaining four have containment failure due to HPME and DCH at VB; all five of the failures are catastrophic ruptures. The five bins that have VB and early containment failure occur in only one or two out of 200 observations.

Table 2.5-4
Results of the Accident Progression Analysis for Sequoyah
Internal Initiators - PDS Group 4: Event V

Ten Most Probable Bins*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
1	BHADBCAADEAAAB	0.148	111	V-Wet	Never	PrmDry	LoPr	Pour	Large	Lo	No	Bypass	noByP
2	BHADBCAADEAAAA	0.115	111	V-Wet	Never	PrmDry	LoPr	Pour	Large	Lo	No	Bypass	noByP
3	BHADBCABDEAAAB	0.113	88	V-Wet	Never	PrmDry	LoPr	Pour	Large	Hi	No	Bypass	noByP
4	BHADBCAADDAAAB	0.098	111	V-Wet	Never	PrmDry	LoPr	Pour	Large	Lo	No	BMT	noByP
5	BHADBCABDEAAAA	0.088	88	V-Wet	Never	PrmDry	LoPr	Pour	Large	Hi	No	Bypass	noByP
6	BHADBCAADDAAAA	0.077	111	V-Wet	Never	PrmDry	LoPr	Pour	Large	Lo	No	BMT	noByP
7	BHADBCABDDAAAA	0.075	88	V-Wet	Never	PrmDry	LoPr	Pour	Large	Hi	No	BMT	noByP
8	BHADBCABDDAAAA	0.059	88	V-Wet	Never	PrmDry	LoPr	Pour	Large	Hi	No	BMT	noByP
9	AHADBCAADEAAAB	0.036	111	V-Dry	Never	PrmDry	LoPr	Pour	Large	Lo	No	Bypass	noByP
10	AHADBCABDEAAAB	0.029	88	V-Dry	Never	PrmDry	LoPr	Pour	Large	Hi	No	Bypass	noByP

* A listing of all bins, and a listing by observation are available on computer media.
** Mean probability conditional on the occurrence of the PDS.

Table 2.5-5
Results of the Accident Progression Analysis for Sequoyah
Internal Initiators - PDS Group 5: Transients

Five Most Probable Bins*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
1	GDCDFCDBDFBAAB	0.206	122	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Hi	No	No-CF	noByP
2	GDCDFCDADFBAAB	0.101	114	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Lo	No	No-CF	noByP
3	GDCDFCDBDFBCCB	0.074	122	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Hi	No	No-CF	ByP
4	GDCDFCDBDFBAAA	0.069	122	No-CF	Always	No-CCI	LoPr	No-VB	No-CCI	Hi	No	No-CF	noByP
5	GDCCFCDBDFBAAB	0.037	25	No-CF	Always	No-CCI	ImPr	No-VB	No-CCI	Hi	Lo	No-CF	noByP

Five Most Probable Bins that have VB*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
12	GDDAACAADFAAAB	0.015	79	No-CF	Always	PrmDp	SSPr	HPME	Large	Lo	No	No-CF	noByP
17	GDDDBCABDFBAAB	0.010	116	No-CF	Always	PrmDp	LoPr	Pour	Large	Hi	No	No-CF	noByP
20	GDCAACDACFAAAB	0.008	13	No-CF	Always	No-CCI	SSPr	HPME	No-CCI	Lo	Low	No-CF	noByP
23	GDCAACDACFAAAA	0.006	13	No-CF	Always	No-CCI	SSPr	HPME	No-CCI	Lo	Low	No-CF	noByP
25	GDCDBCDBDFBAAB	0.006	116	No-CF	Always	No-CCI	LoPr	Pour	No-CCI	Hi	No	No-CF	noByP

Five Most Probable Bins that have VB and Early CF*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
79	DACABCDADAAAAB	0.001	2	CFatVB	Early	No-CCI	SSPr	Pour	No-CCI	Lo	No	CatRup	noByP
89	DACAACDABAAAAB	0.001	1	CFatVB	Early	No-CCI	SSPr	HPME	No-CCI	Lo	Med	CatRup	noByP
91	DACABCDADAAAAA	0.001	2	CFatVB	Early	No-CCI	SSPr	HPME	No-CCI	Lo	No	CatRup	noByP
95	DACAACDAAAAAAB	0.001	2	CFatVB	Early	No-CCI	SSPr	HPME	No-CCI	Lo	Hi	CatRup	noByP
104	DACAACDABAAAAA	0.001	1	CFatVB	Early	No-CCI	SSPr	HPME	No-CCI	Lo	Med	CatRup	noByP

* A listing of all bins, and a listing by observation are available on computer media.

** Mean probability conditional on the occurrence of the PDS.

In this PDS Group, the probability of a temperature-induced failure of the RCS pressure boundary is quite high, almost 0.90. As a result, the probability of arresting the core degradation process and avoiding VB is also high, about 0.80. More detail on the arrest of core damage may be found in Appendix A.3.3.

The fractions of this PDS Group which are in the four pressure ranges at UTAF and just before VB (if it occurs) are:

	At UTAF	Just Before VB
SSPr (2500 psia)	1.00	0.11
HiPr (600-2000 psia)	0.00	0.001
ImPr (200-600 psia)	0.00	0.11
LoPr (< 200 psia)	0.00	0.78

As both PDSs in this group have the RCS intact at UTAF, the RCS is at the PORV setpoint pressure at that time (Question 16). Just before VB (Question 25), the probability of being at SSPr is only about 0.11. This probability is higher than PDS Group 2 (Fast SBO) because RCP seal cooling is available, thus rendering the failure of the pumps seals ineffective as a means of depressurization. The PORVs still function in their safety mode, so they may stick open even when hardware failures prevent their being opened from the control room. The two effective depressurization mechanisms for this PDS Group are the PORVs sticking open and the temperature-induced hot leg (or surge line) failures. Deliberate opening of the PORVs by the operators is ineffective because they cannot open the PORVs or have already failed to do so. Temperature-induced SGTRs are very unlikely according to the expert panel. The determination of RCS pressure at VB is discussed further in Sections 2.5.2.1 and 2.5.2.2.

As with all accidents in which ac power is initially available, the hydrogen threat is reduced due to the low probability of operator failure to initiate igniters and the low probability that the air return fans fail. The mean probability of containment failure during core degradation due mainly to isolation failures is low, only 0.005, and in these cases, VB does not occur. The mean probability of containment failure at VB is 0.02. Note that the 0.98 probability of no containment failure at VB includes the 0.77 of the group that had core damage and no VB. There are no late failures due to hydrogen burns. The mean probability of very late containment failure due to overpressure by steam and/or noncondensibles is 0.02. The mean probability of BMT is 0.02.

2.5.1.6 Results for PDS Group 6 - ATWS. This PDS Group consists of accidents in which neither control rod insertion nor boron injection bring the reaction under control shortly after the start of the accident. The core continues to generate large amounts of heat and steam until the water level drops far enough below TAF that the loss of the neutron moderating effect of the liquid water is lost for a substantial portion of the core. The ATWS PDS Group consists of three PDSs, one with the RCS intact at UTAF, one with an S_3 break, and one with an SGTR. In all three situations, the PORVs will be open at UTAF due to the rate of steam generation in the core. The LPIS is operating but not injecting in the RCS intact and SGTR PDSs.

Table 2.5-6 lists the 10 most probable APBs for the PDS Group, and the five most probable APBs that have VB and early containment failure or bypass. Evaluation of the APET produced 6627 bins for the ATWS PDS Group, of which 985 are required to capture 95% of the probability. Table 2.5-6 differs from the preceding tables in that the sprays characteristic has been omitted and the SGTR characteristic included. The PDSs in this group all have sprays initially, and the sprays usually do not fail throughout the accident.

The 10 most probable bins capture 34% of the probability; nine of them have no containment failure, and five of them have no VB as well. The APB in which containment failure occurs, is a very late failure due to BMT. The last part of Table 2.5-3 shows the five most probable APBs with VB and early containment failure or bypass. These APBs all have SGTR and no VB. Based on the MCDFs, a fraction of 0.13 of this PDS Group has an SGTR initiator, and thus, have containment bypass at the start of the accident. The most probable bin with containment failure at VB is 61st in order with a probability of 0.0025; the containment failure is due to a hydrogen burn at VB.

In this PDS Group, the mean probability of arresting the core degradation process and avoiding VB is about 0.17 when there is no bypass of containment due to SGTR, and about 0.10 when there is an SGTR. The arrest of core degradation is a result of the operation of the LPIS following a temperature-induced break in the RCS. The water from the RWST injected by the LPIS contains enough boron to shut down the reaction should the core be in a configuration where continued reaction is possible. More detail on the arrest of core damage may be found in Appendix A.3.3.

The fractions of this PDS Group which are in the four pressure ranges at UTAF and just before VB (if it occurs) are:

	At UTAF	Just Before VB
SSPr (2500 psia)	1.00	0.003
HiPr (600-2000 psia)	0.00	0.08
ImPr (200-600 psia)	0.00	0.22
LoPr (< 200 psia)	0.00	0.70

The RCS is at the PORV setpoint pressure at UTAF (Question 16) because the reaction has not been shut down and the steaming rate is high. Just before VB (Question 25), the probability of being at SSPr is only about 0.003. This probability is lower than in PDS Groups 1, 2, and 5 because the operators are allowed to deliberately open the PORVs in this PDS. In the human reliability analysis, it was judged that the operators would be too busy trying to bring the reaction under control before UTAF to consider opening the PORVs, and the PORVs would be kept open by the escaping steam in any event. Thus the effective depressurization mechanisms for this PDS Group are: the PORVs sticking open, temperature-induced hot leg (or surge line) failures, and deliberate opening of the PORVs by the operators. Pump seal cooling is available in the one PDS where it would be effective (the "T" PDS where the RCS is intact), so failure of the pumps seals is ineffective as a means of depressurization for the ATWS PDS Group. Temperature-induced SGTRs are very unlikely according to the expert panel.

Table 2.5-6
Results of the Accident Progression Analysis for Sequoyah
Internal Initiators - *PDS Group 6:* ATWS

Ten Most Probable Bins*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
1	GDCDFCDBDFBAAB	0.060	122	No-CF	No-CCI	LoPr	No-VB	No	No-CCI	Hi	No	No-CF	noByP
2	GDDDBCABDFBAAB	0.059	116	No-CF	PrmDp	LoPr	Pour	No	Large	Hi	No	No-CF	noByP
3	GDCDFCDADFBAAB	0.039	114	No-CF	No-CCI	LoPr	No-VB	No	No-CCI	Lo	No	No-CF	noByP
4	GDCDFADBDEBAAB	0.036	121	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Hi	No	Bypass	noByP
5	GDCDBCDBDFBAAB	0.034	116	No-CF	No-CCI	LoPr	Pour	No	No-CCI	Hi	No	No-CF	noByP
6	GDDDBCAADFBAAB	0.031	111	No-CF	PrmDp	LoPr	Pour	No	Large	Lo	No	No-CF	noByP
7	GDCDFADADEBAAB	0.022	102	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Lo	No	Bypass	noByP
8	GDDDBCABDFBAAA	0.020	116	No-CF	PrmDp	LoPr	Pour	No	Large	Hi	No	No-CF	noByP
9	GDCDFADADEBAAB	0.020	102	No-CF	No-CCI	LoPr	No-VB	No	No-CCI	Hi	No	No-CF	noByP
10	FDDDBCABDDBAAB	0.020	116	VLate	PrmDp	LoPr	Pour	No	Large	Hi	No	BMT	noByP

Five Most Probable Bins that have Bypass or VB and Early CF*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
4	GDCDFADBDEBAAB	0.036	121	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Hi	No	Bypass	noByP
7	GDCDFADADEBAAB	0.022	102	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Lo	No	Bypass	noByP
15	GDCDFADBDEBAAA	0.012	120	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Hi	No	Bypass	noByP
21	GDCDFADADEBAAA	0.007	102	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Lo	No	Bypass	noByP
40	GACDFADBDEBAAB	0.004	114	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Hi	No	Bypass	noByP

* A listing of all bins, and a listing by observation are available on computer media.
** Mean probability conditional on the occurrence of the PDS.

As with all accidents in which ac power is initially available, the hydrogen threat is reduced due to the low probability of operator failure to initiate igniters and the low probability that the air return fans fail. The mean probability of containment failure during core degradation due mainly to isolation failures is low, only 0.004. The mean probability of containment failure at VB is 0.05. Note that the 0.95 probability of no containment failure at VB includes the 0.17 of the group that had core damage and no VB. The mean probability of late failures due to hydrogen burns is 0.001. The mean probability of very late containment failure due to overpressure by steam and/or noncondensibles is 0.07. The mean probability of BMT is 0.08.

2.5.1.7 <u>Results for PDS Group 7 - SGTRs</u>. This PDS Group consists of accidents in which the initiating event is the rupture of a steam generator tube. The reaction is shut down successfully. The SGTR PDS Group includes one PDS in which the RCS is depressurized using the three unaffected SGs according to procedures, and the SRVs on the main steam lines from the affected SG do not stick open. These accidents, denoted "G" SGTRs, are indicated by "SGTR" in Table 2.5-7. The most frequent PDS in the SGTR PDS Group are accidents in which the RCS is not depressurized according to procedures, and the SRVs on the main steam lines from the affected SG stick open. These accidents, denoted "H" SGTRs, are indicated by "SRVO" in Table 2.5-7. Like Table 2.5-6, Table 2.5-7 omits the sprays characteristic to show the SGTR characteristic. All the APBs for this PDS Group have sprays most of the time.

Evaluation of the APET produced 2632 bins for the SGTR PDS Group, of which 354 are required to capture 95% of the probability. Table 2.5-7 lists the fifteen most probable APBs for the PDS Group; they all have bypass of the containment. Eleven of the 15 most probable APBs are "H" SGTR accidents in which the secondary SRVs are stuck open. The 15 most probable bins capture 39% of the probability.

In this PDS Group, the probability of avoiding VB is about 0.19. There is no ECCS operable in the "H" PDS; the LPIS is operating in the "G" PDS and there is an effective depressurization mechanism. This mechanism is the deliberate opening of the PORVs. RCP seal cooling is available, so there are no seal failures. The RCS is not at the PORV setpoint pressure, so there is no possibility of the PORVs sticking open, T-I hot leg failures, or T-I SGTRs.

The fractions of this PDS Group which are in the four pressure ranges at UTAF and just before VB (if it occurs) are:

	At UTAF	Just before VB
SSPr (2500 psia)	0.00	0.00
HiPr (600-2000 psia)	1.00	0.23
ImPr (200-600 psia)	0.00	0.32
LoPr (< 200 psia)	0.00	0.45

As the two PDSs in this group have an S_3-size SGTR at UTAF, the RCS pressure is in the high range at UTAF (Question 15). The two PDSs in this group are HINY-NXY and GLYY-YNY. In HINY-NXY the operators failed to

Table 2.5-7
Results of the Accident Progression Analysis for Sequoyah
Internal Initiators - PDS Group 7: SGTRs

Fifteen Most Probable Bins*

Order	Bin	Prob.**	Occ	CF Time	Sprays	CCI	RCS Pr	VB-Mode	Amt-CCI	ZrOx	HPME	CF-Size	E2-IC
1	GDCDFADADEBAAB	0.069	102	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Lo	No	Bypass	noByP
2	GDCDFADBDEBAAB	0.043	121	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Hi	No	Bypass	noByP
3	GHADBBABDEAAAB	0.035	31	No-CF	PrmDry	LoPr	Pour	SRVO	Large	Hi	No	Bypass	noByP
4	GHADBBAADEAAAB	0.034	29	No-CF	PrmDry	LoPr	Pour	SRVO	Large	Lo	No	Bypass	noByP
5	GHADBBABDEAAAA	0.028	31	No-CF	PrmDry	LoPr	Pour	SRVO	Large	Hi	No	Bypass	noByP
6	GHADBBAADEAAAA	0.026	29	No-CF	PrmDry	LoPr	Pour	SRVO	Large	Lo	No	Bypass	noByP
7	FHADBBABDDAAAB	0.024	31	VLate	PrmDry	LoPr	Pour	SRVO	Large	Hi	No	BMT	noByP
8	GDCDFADADEBAAA	0.023	102	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Lo	No	Bypass	noByP
9	FHADBBAADDAAAB	0.022	29	VLate	PrmDry	LoPr	Pour	SRVO	Large	Lo	No	BMT	noByP
10	FHADBBABDDAAAA	0.018	31	VLate	PrmDry	LoPr	Pour	SRVO	Large	Hi	No	BMT	noByP
11	FHADBBAADDAAAA	0.018	29	VLate	PrmDry	LoPr	Pour	SRVO	Large	Lo	No	BMT	noByP
12	GDCDFADBDEBAAA	0.014	120	No-CF	No-CCI	LoPr	No-VB	SGTR	No-CCI	Hi	No	Bypass	noByP
13	GHBBBBAADEAAAB	0.012	12	No-CF	PrmShl	HiPr	Pour	SRVO	Large	Lo	No	Bypass	noByP
14	GHBCBBAADEAAAB	0.011	12	No-CF	PrmShl	ImPr	Pour	SRVO	Large	Lo	No	Bypass	noByP
15	GHABABAACEAAAB	0.011	11	No-CF	PrmDry	HiPr	HPME	SRVO	Large	Lo	No	Bypass	noByP

* A listing of all bins, and a listing by observation are available on computer media.
** Mean probability conditional on the occurrence of the PDS.

follow procedures and open the PORVs before UTAF, so no credit is given for their opening the PORVs after UTAF. In GLYY-YNY, the PORVs are open at UTAF as the operators are or were attempting to cool the core by bleed and feed. In GLYY-YNY, the resulting pressure reduction in the RCS may allow the operating LPIS to inject water and arrest core damage before VB. As discussed in Section 2.5.2.1, it was estimated that with an S_3-size break in the system, the low, intermediate, and high pressure ranges were equally likely at VB. The probabilities of these three pressure ranges given above vary somewhat from 0.33 due to the open PORVs just discussed.

For the SGTR PDS Group, containment failure at VB is not particularly significant for risk as the bulk of the fission products escapes through the containment bypass. The mean probability of containment failure during core degradation due to isolation failures is 0.004. The mean probability of containment failure at VB is 0.16. The mean probability of late failures due to hydrogen burns is 0.003. The mean probability of very late containment failure due to overpressure by steam and/or noncondensibles is 0.01. The mean probability of BMT is 0.22.

2.5.1.8 Core Damage Arrest and Avoidance of VB. It is possible to arrest the core damage process and avoid VB if ECCS injection is restored before the core degradation process has gone too far. Recovery of injection is due to one of two events. In the LOSP accidents, recovery of injection follows the restoration of offsite power. In other types of accidents, the ECCS is operating at UTAF but no injection is taking place because the RCS pressure is too high. Any break in the RCS pressure boundary that allows the RCS pressure to decrease to the point where the ECCS can inject is likely to arrest the core degradation process. The break may be an initiating break or a temperature-induced break or other failure that occurs after UTAF.

PDSs ALYY-YYY and ALYY-YYN have the LPIS operating at UTAF. These are core damage situations because the success criteria require the accumulators to operate in addition to the LPIS, and the accumulators fail. PDS S_1LYY-YYN also has the LPIS operating at UTAF; it is a core damage situation because the success criteria require the HPIS to operate in addition to the LPIS, and the HPIS fails in recircirculation. For both of these PDSs, the initiating break depressurizes the RCS sufficiently for the LPIS to inject. In PDSs S_2LYY-YYN, S_3LYY-YYN, the LPIS is also operating but the system pressure is too high at UTAF to allow injection. During subsequent core degradation, the system pressure may sufficiently decrease such that injection will commence. In PDSs TLYY-YXY and TBYY-YNY, the RCS is intact at UTAF. For these situations, injection will commence only if one of the five depressurization means considered in this analysis operates, and if the RCS is depressurized to a low enough level. The five means of depressurizing the RCS after UTAF are:

1. PORVs or SRVs stick open;
2. T-I RCP seal failure;
3. Deliberate opening of the PORVs by the operators;
4. T-I SGTR; and
5. T-I hot leg or surge line failure.

Figure 2.5-1 shows the probability of halting the degradation of the core before the lower head of the vessel fails and thereby achieving a safe stable state with the vessel intact. For the LOSP summary PDS Group, the distribution in Figure 2.5-1 reflects the distribution for offsite ac power recovery in the APET early period. To avoid a gap in the times for which power recovery is considered, the start of the APET early period is the end of the period for which recovery of offsite power was considered in the accident frequency analysis. This time is nominally the onset of core damage, but for some PDSs this time precedes the current estimates of the onset of core damage (UTAF) by a significant amount. The end of the APET "early" period is the expected time of VB. The estimated core damage states at different times in this period were used to determine the probability of core damage arrest for each PDS involved as explained in Appendix A.1.1 (see the discussions of Questions 22 and 26) and in Appendix A.3.3.

For the ATWSs, Transients, and LOCAs, the distributions for core damage arrest show the combined effects of RCS depressurization mechanisms that allow ECCS injection in those PDSs that have ECCS operating at UTAF. The probability of core damage arrest is very high for Transients since one PDS in the group has LPIS operating and the other has both LPIS and HPIS operating. As the probability of occurrence of one or more of the depressurization mechanisms is high, so the probability of core damage arrest is high.

2.5.1.9 <u>Early Containment Failure</u>. For those accidents in which the containment is not bypassed, the offsite risk depends strongly on the probability that the containment will fail before or at VB. There are four possibilities:

1. Pre-existing containment leak or isolation failure,
2. Containment failure before VB due to hydrogen deflagration,
3. Containment failure before VB due to hydrogen detonation; and
4. Containment failure at VB due to the events at VB.

The probability of a pre-existing leak at Sequoyah is low. The main threat is due to isolation failures which are caused by air lock failures, purge valve failures or other similar, undetected failures of the containment boundary.

Hydrogen combustion before VB is a concern for the Sequoyah containment because of the relatively small containment volume and low failure pressure. The hydrogen ignition system, operating in conjunction with the air return fan system helps preclude large hydrogen burns by burning relatively small quantities of hydrogen as it is generated. Without operation of the fans and igniters (typical for an SBO), hydrogen can stagnate in the ice condenser and upper plenum of the ice condenser at potentially detonable levels. Sufficient accumulation of hydrogen in the dome for this scenario can pose a threat to containment by hydrogen deflagration. Thus, failures of containment during core degradation due to hydrogen events are contributors to early containment failure.

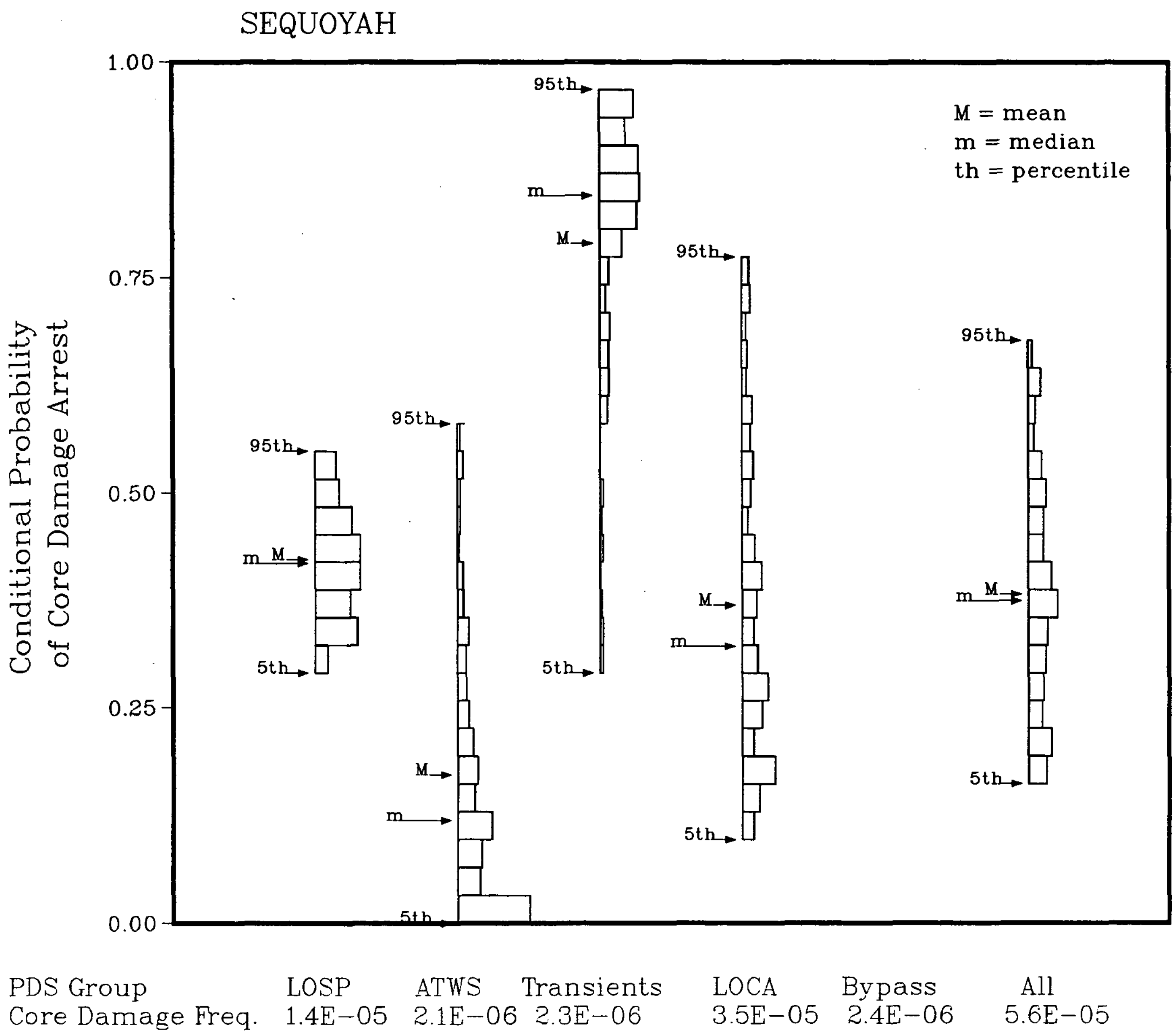

Figure 2.5-1. Probability of Core Damage Arrest.

The largest contribution to early containment failure (for non-bypass accidents) at Sequoyah comes from containment failures at VB. These failures are due to hydrogen burns at VB, with possible augmentation from ex-vessel steam explosions, HPME involving DCH and/or hydrogen burns, direct contact of the molten core debris on the containment wall, or in-vessel steam explosions (Alpha mode).

Figure 2.5-2 shows the probability distribution for early containment failure at Sequoyah (containment failure means containment failure before or at VB). The probability is conditional on core damage. All the no VB probability associated with no VB, including the small fraction which has containment failure during core degradation due to hydrogen events or isolation failures is not included in this figure. The conditional probability of early containment failure is particularly low for the Transient PDS Group because the probability of core damage arrest is quite high. There is no histogram for the Bypass summary PDS Group. When the containment function is bypassed by Event V or an SGTR, early containment failure ceases to be very important in determining the release of fission products and the offsite risk. Thus, the conditional probability of early containment failure was deliberately not plotted for the Bypass Group. For accidents other than Bypass, the mean conditional probability of early containment failure is on the order of 0.06.

2.5.1.10 Summary. Figure 2.5-3 shows the mean distribution among the summary accident progression bins for the summary PDS Groups. Only mean values are shown, so Figure 2.5-3 gives no indication of the range of values encountered. The distribution for core damage arrest is shown in Figure 2.5-1, and the distribution for early (at or before VB) failure of the containment is shown in Figure 2.5-2. Figure 2.5-3 gives a good idea of the relative likelihood of the possible results of the accident progression analysis. Except for the Bypass initiators, either no failure of the vessel (safe stable state) or no containment failure are by far the most likely outcomes. A late failure is more likely than failure at or before VB. The late failure may be due to hydrogen ignition some hours after VB, long-term overpressure by steam and/or noncondensibles, or BMT. Early containment failure is fairly unlikely, as was indicated by Figure 2.5-2.

Figure 2.5-3 shows only the mean frequencies for the summary PDS Groups and mean conditional probabilities for the summary APBs, where the mean is taken over all 200 observations in the sample. The core damage frequency of each PDS Group is different for each observation. Figure 2.5-4 displays the range of core damage frequencies for the 200 observations for the seven PDS Groups. The frequency range from the 5th percentile to the 95th percentile is about two or three orders of magnitude for all of the PDS Groups except Event V. The large range for Event V reflects the large uncertainty in the initiating event frequency for the interfacing system LOCA.

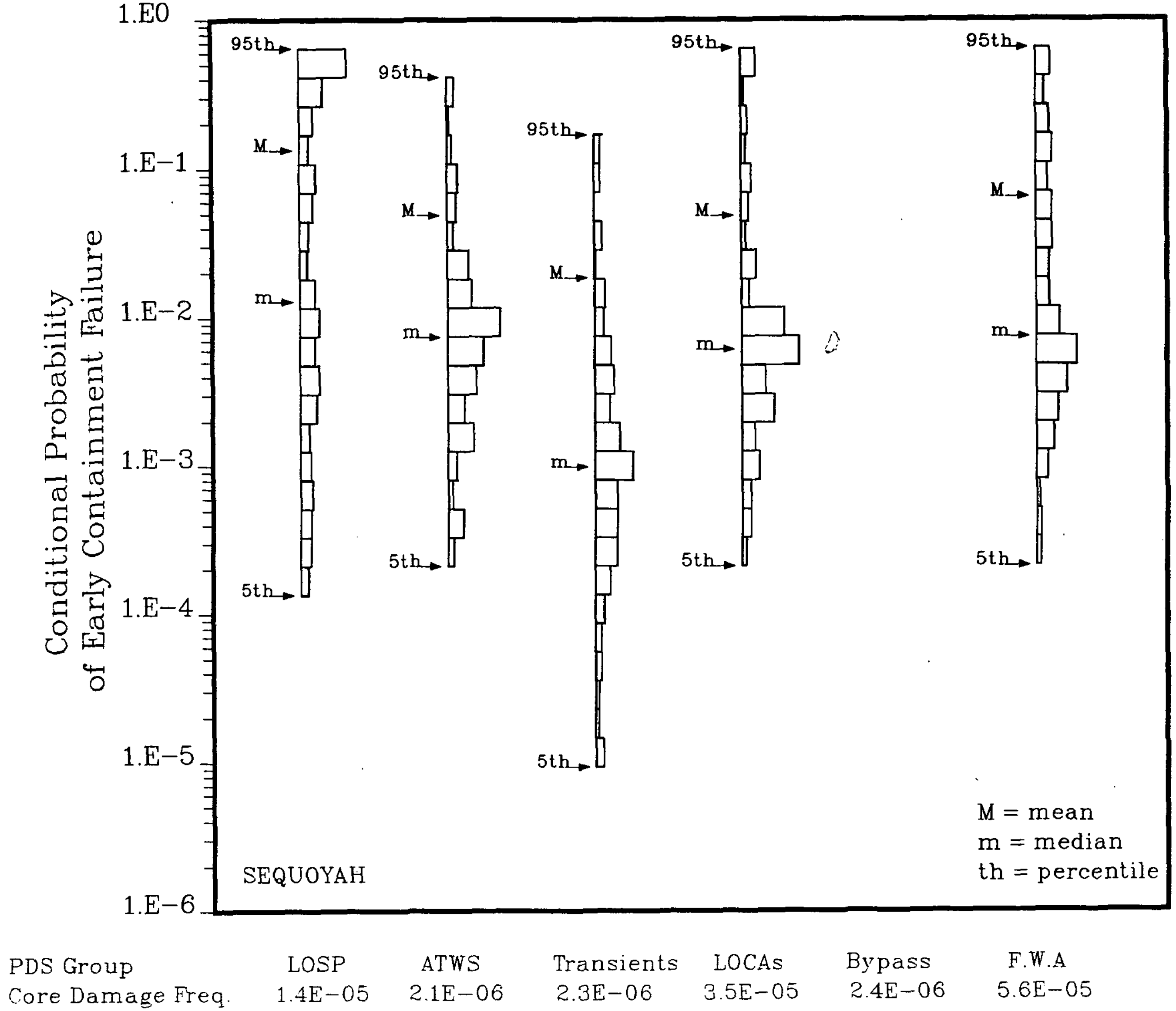

Figure 2.5-2. Probability of Early Containment Failure.

ACCIDENT PROGRESSION BIN	PLANT DAMAGE STATE (Mean Core Damage Frequency)					
	LOSP (1.38E−05)	ATWS (2.07E−06)	Transients (2.32E−06)	LOCAs (3.52E−05)	Bypass (2.39E−06)	Frequency Weighted Average (5.58E−05)
VB, early CF (during CD)	0.014	0.003		0.002		0.005
VB, alpha, early CF (at VB)	0.002	0.003		0.002		0.002
VB > 200 psi, early CF (at VB)	0.064	0.023	0.014	0.031		0.035
VB < 200 psi, early CF (at VB)	0.054	0.020	0.004	0.014		0.023
VB, late CF	0.153	0.001		0.001		0.038
VB, BMT, very late CF	0.065	0.151	0.039	0.260		0.171
Bypass	0.001	0.134	0.006		0.996	0.056
VB, No CF	0.200	0.471	0.137	0.301		0.269
No VB, early CF (during CD)	0.038	0.001	0.005	0.002		0.011
No VB	0.384	0.171	0.785	0.367		0.371

BMT = Basemat Meltthrough
CF = Containment Failure
VB = Vessel Breach
CD = Core Degradation

Sequoyah

Figure 2.5-3. Mean Probability of Summary APBs for Summary PDSs.

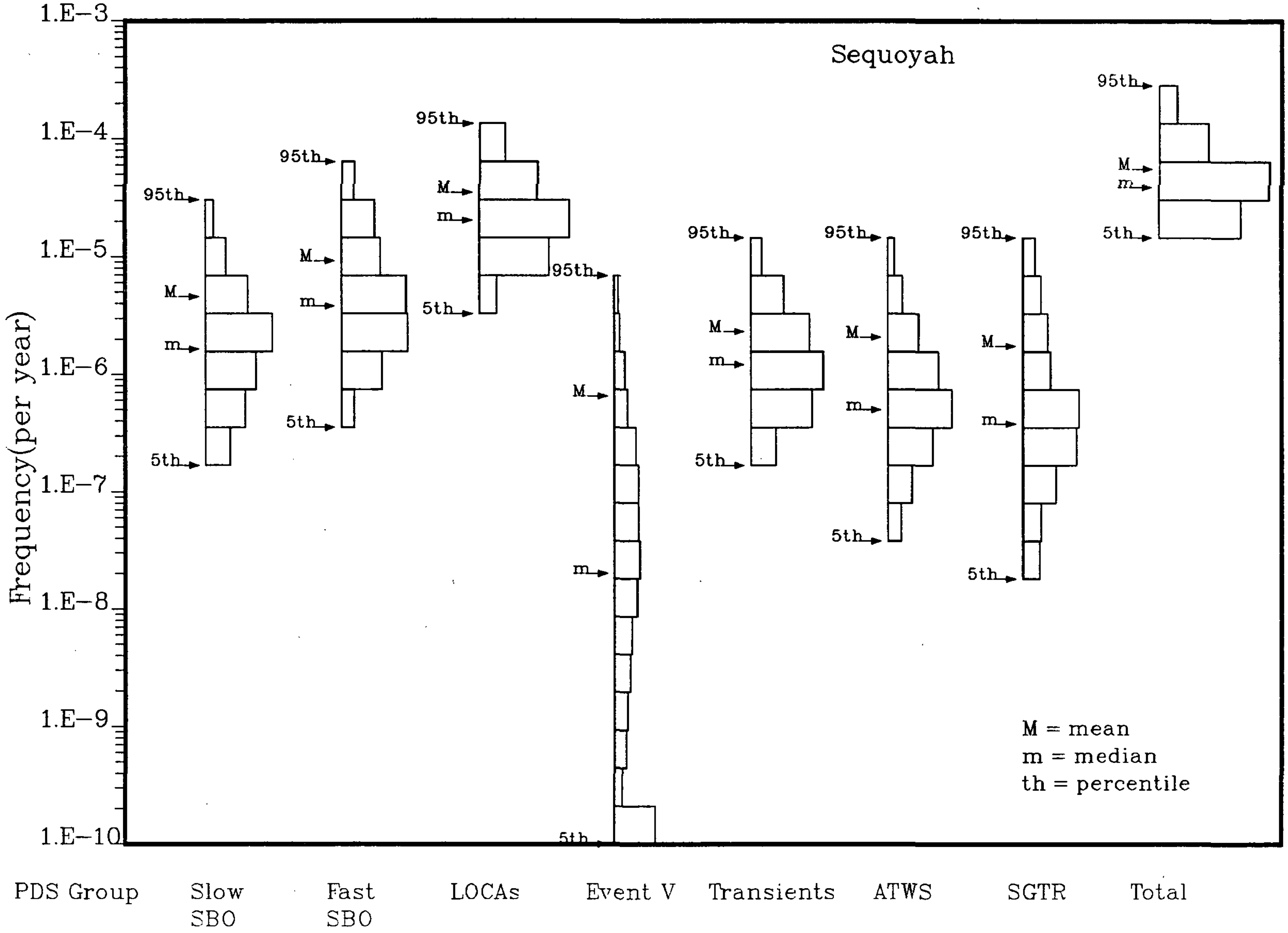

Figure 2.5-4. Core Damage Frequency Distributions for PDS Groups.

The mean conditional probability of each summary APB may be computed for each PDS Group for each observation. When combined with the PDS Group frequency, a frequency for each summary APB for each observation is obtained. The distribution of these values is displayed in Figure 2.5-5 The 95th percentiles of the distributions for VB coincident with early containment failure (the first three distributions) all fall below 1.0E-4/year. The means are much greater than the medians for these distributions, indicating that the means are largely determined by a small number of observations with high probability of VB followed by early containment failure. The bypass summary APB includes both Event V and the SGTRs. The long low frequency 'tail' of the distribution for Event V in Figure 2.5-4 is lost when the interfacing system LOCA and SGTR frequencies are summed for presentation in Figure 2.5-5.

The releases from accidents that result in VB and early containment failure are roughly comparable to releases from the most severe bypass accidents, and the releases from both of these types of accidents are much larger than non-bypass accidents in which the containment does not fail at all or fails some hours after VB. Therefore, since Figure 2.5-5 shows that bypass accidents have a comparable frequency distribution with accidents with VB and early containment failure, it may be inferred that the risk to the offsite population from internally initiated accidents at Sequoyah is likely to be dominated by bypass accidents and accidents in which VB and early containment failure occur.

2.5.2 Sensitivity Analysis for Internal Initiators

This section reports the results of a sensitivity analysis that was performed for the internally initiated accidents at Sequoyah. The sensitivity study was performed to determine the importance and the effects of the temperature-induced (T-I) hot leg (and surge line) breaks and the T-I SGTRs. These failures occur after the core melt has begun and when the hydrogen and superheated steam leaving the core have heated the hot leg, surge line, and steam generator inlet plenum to temperatures on the order of 1000 K. Aggregate cumulative failure probabilities for these phenomena were provided by the Expert Panel on In-Vessel Issues. Their conclusions were that these failures would occur only if the RCS was at the PORV setpoint pressure (about 2500 psia). The hot leg failures were judged to be relatively likely (mean failure probability about 0.70), while the SGTRs were estimated to be quite unlikely (mean failure probability about 0.015). In the sensitivity analysis, these two T-I failures were eliminated completely. Note that the distributions used for the other three depressurization mechanisms were not altered in this sensitivity analysis. The deliberate opening of the PORVs is not a particularly effective means of depressurizing the RCS, but the sticking open of the PORVs and the failure of the RCP seals are effective.

Of the seven internally initiated PDS groups at Sequoyah, three (LOCAs, Event V, and SGTRs) are completely unaffected by the elimination of the T-I hot leg failures and T-I SGTRs because the conditions for these events (RCS at PORV setpoint pressure) are not met. The other four PDS groups were evaluated in this sensitivity analysis, and the results for PDS Group 1, Slow SBO, will be discussed in some detail.

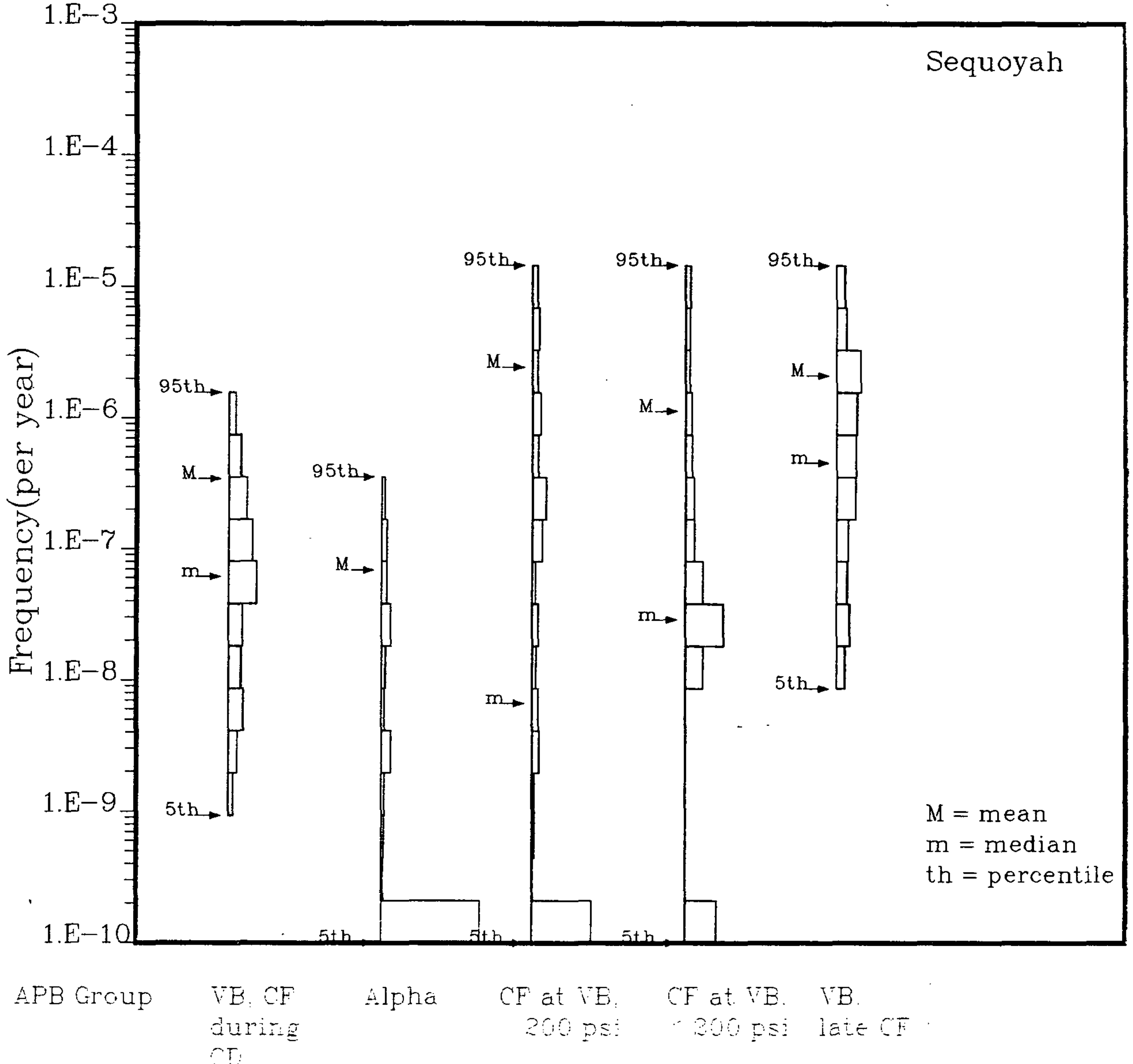

Figure 2.5-5. Distribution of Freuqencies for Summary APBs.

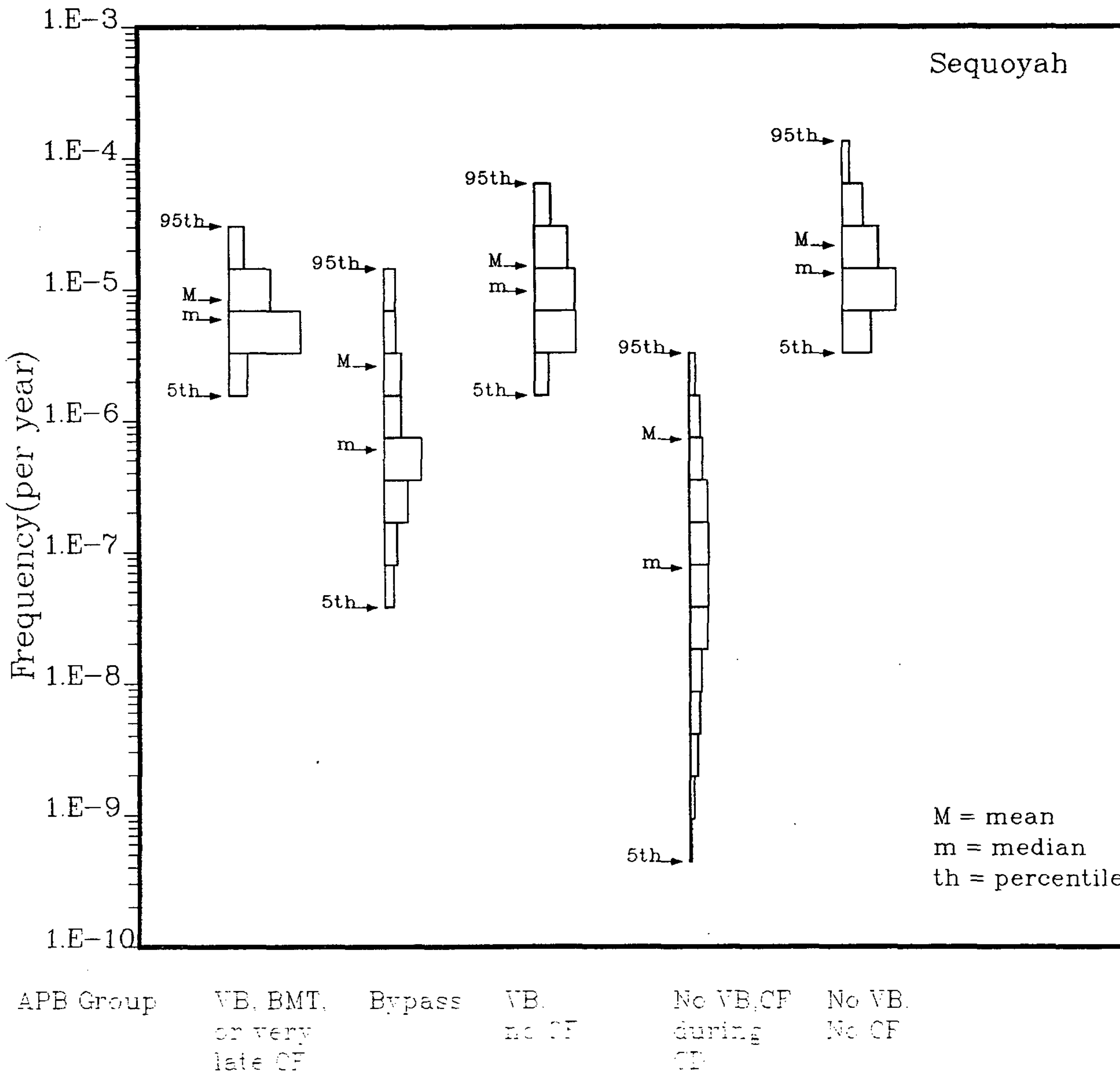

Figure 2.5-5 (continued).

In the Sequoyah APET, the occurrence of T-I SGTRs is addressed in Question 20, and the occurrence of T-I hot leg failures is addressed in Question 21. Thus, the base case (T-I failures as specified by the expert panel) and the sensitivity case (no T-I failures) are identical up through Question 19.

For slow blackouts, the mean RCS condition at the uncovering of the top of active fuel (UTAF) is:

No Break	0.171
S_3 Break	0.189
S_2 Break	0.640

This is the condition of the RCS at the start of the accident progression analysis as determined by averaging the 200 observations in the sample. Question 16 determined the RCS pressure at UTAF. As the RCS pressure depends upon the state of the AFWS as well as the condition of the RCS, the mean division among the pressure levels for the Slow SBO PDS Group at Question 16 does not exactly match the division among RCS states:

SSPr	0.171
HiPr	0.201
ImPr	0.628
LoPr	0.000

where:

SSPr = 2500 psia (PORV setpoint),
HiPr = roughly 1000 to 1400 psia, but perhaps as high as 2000 psia,
ImPr = 200 to 600 psia, and
LoPr = less than 200 psia.

The high pressure range includes all pressures from 600 psia to over 2000 psia, but the detailed mechanistic codes suggest that, during most of the core degradation process, the RCS pressure will be in the 1000 to 1400 psia range.

Question 17 is whether the PORVs stick open. The probability that the PORVs will stick open is 0.50 if they are cycling, that is, if there is no break in the RCS and the system is at the PORV setpoint pressure (SSPr). Thus, half of the no break states become effective S_2 states at this point. Question 18 is whether the RCP seals fail. The mean fraction of RCP seal failures is 0.615, but most of these failures occur for states in which there is already an S_3 or S_2 break, and so have no effect. As there is no electric power, the operators are prevented from opening the PORVs in Question 19.

Question 20 concerns the T-I SGTR. No SGTRs were computed in the sensitivity case vs. 0.0002 in the base case. Question 21 concerns the T-I hot leg (or surge line) failure. No failures were computed in the sensitivity case vs. 0.045 in the base case.

The pressure in the RCS just before VB is determined at Question 25. For this question, the mean division among the pressure levels is not noticeably different for the two analyses:

RCS Pressure at Vb (Q25)

Pressure Range	Sensitivity (No T-I Breaks)	Base (T-I Breaks)
SSPr	0.023	0.005
HiPr	0.24	0.23
ImPr	0.27	0.26
LoPr	0.47	0.50

These tables give results up to only two significant figures, so roundoff may cause the column sums to differ slightly from exactly 1.00. Since the PORVs stick open half the time for the "T" PDSs, and the RCP seals fail about 60% to 70% of the time when there is no pump seal cooling, there are two effective means of depressurizing the RCS in the sensitivity case. This PDS Group has no pump seal cooling. The stuck-open PORVs question alone has converted half the No Break PDSs in the Slow SBO Group to effective S_2 breaks. The base case has T-I hot leg breaks as well, and there is a small difference. As expected, the T-I hot leg failures and SGTRs affect only the SSPr and LoPr pressure ranges since hot leg failures occur only when the RCS pressure is at the PORV setpoint value.

The fractions of the Slow SBO Group that went to each case in Question 25 may also be of interest:

Break Size	Sensitivity (No T-I Breaks)	Base (T-I Breaks)
Case 1: A-size Breaks	0.000	0.045
Case 2: S_2-size Breaks	0.283	0.283
Case 3: S_3-size Breaks	0.693	0.667
Case 4: No Breaks	0.023	0.005

The effect of eliminating the T-I SGTRs is negligible, even in Question 25, but the effect of eliminating the T-I hot leg failures is to transfer about 2% of the Slow SBO Group from LoPr to SSPr. The reason the fraction is not greater is that only 17% of the group is in the "No Break" category to begin with, and the stuck-open PORVs eliminate half of this category before the hot leg failure question is asked. The RCP seal failures eliminate the remaining portion of the sequences initially at system setpoint pressure.

Containment failure during core degradation is due to hydrogen combustion or detonation events, and occurs with non-negligible probability only for the blackout sequences. For times when the system is at higher pressures, there is more hydrogen retained in the RCS, and thus the probability for threatening burns or detonations is lower. Elimination of T-I hot leg or SGTR failures might be favorable to reducing these early containment failures. The containment failures during core degradation are determined in Question 58. For this PDS Group, there is a slight increase in the fraction of times containment failure occurs during core degradation, when the T-I failures are eliminated. The mean branch probabilities for catastrophic rupture, rupture, leak and no containment failure are:

Early containment failure (during CD) and mode of failure (Q58)

CF Mode	Sensitivity (No T-I Breaks)	Base (T-I Breaks)
Cat. Rupture	0.019	0.018
Rupture	0.025	0.028
Leak	0.008	0.009
NoCF	0.948	0.945

The type of vessel failure is determined in Question 65 of the Sequoyah APET. The realized branching (mean values) is:

Type of Vb (Q65)

Type of VB	Sensitivity (No T-I Breaks)	Base (T-I Breaks)
PrEj	0.134	0.125
Pour	0.253	0.265
BtmHd	0.036	0.033
NoVB or α	0.577	0.577

The differences are not larger because the mean probability is 0.576 that offsite electric power and coolant injection is recovered before a large portion of the core is molten, and vessel failure is thus averted. It may be noted that the fraction for pressurized ejection is about the same. Alpha mode failures account for only about 0.1% of the vessel failures.

If eliminating the T-I SGTRs and hot leg failures is to increase risk significantly, it must do so by increasing the fraction of containment failures at VB. This is determined in Questions 78 and 82. Question 78 indicates the probability that the containment fails by direct contact of the core debris with the containment wall. In Question 82, containment failure by overpressure is determined and the rupture failures by alpha mode, upward acceleration of the vessel, and EVSE are summarized. Because the direct contact mode of failure may occur after overpressure failure, there is some overlap between the failure probabilities. The actual probability of failure at (or soon after) VB is determined in the binner. If both overpressure and direct contact failure occur, only overpressure is reported here. The mean branch probabilities for the Slow SBO Group are:

Containment Failure at VB (Q78,Q82)

CF Mode	Sensitivity (No T-I Breaks)	Base (T-I Breaks)
Cat. Rupture	0.039	0.038
Rupture	0.023	0.023
Leak	0.010	0.010
Dir. Contact	0.021	0.018
NoCF	0.907	0.911

There are slightly more containment failures when the T-I RCS breaks are set to zero. The increase is due mainly to the direct contact failure mode. The decrease in failures at VB for the sensitivity study are almost compensated by the increase in failures during core degradation.

The late failures of the containment due to hydrogen burns, long-term overpressurization (OP), and BMT are addressed in Questions 103, 107, and 109:

Late Containment Failures (Q103,Q107,Q109):

Failure	Sensitivity (No T-I Breaks)	Base (T-I Breaks)
Late CF by H_2 burn	0.095	0.096
Very late CF by OP	0.004	0.004
Very late CF by BMT	0.041	0.041

The differences are not significant. Given the results of Question 43, this is to be expected.

Tables 2.5-8 through 2.5-11 summarize the results of the sensitivity analysis for the four internally initiated PDS groups for which the elimination of the T-I breaks have any effect. The Slow SBO Group has already been discussed. The tables show the mean branch probabilities. The Fast SBO Group results are similar to those for the Slow SBO Group, although more pronounced for the Fast SBO, because there is a greater increase in the probability that the vessel fails at higher pressures when there are no T-I failures. The difference in containment failure at VB, the most important question for offsite risk, is quite significant; for Slow SBO the probability of containment failure at VB for the sensitivity study is about 1.04 times the base case, and for Fast SBO the probability is about 1.14 times the base case. For the Transient Group, Table 2.5-10, the major difference is in the probability of core damage arrest and no vessel failure; for the sensitivity study, the probability that no vb occurs is about half of what it was for the base case. The hot leg failure plays a very important role in depressurizing the RCS so that LPIS injection results. Further, RCP seal cooling is operating in this PDS Group, so the RCP seal failure mechanism is not effective. For the Transients, the probability of containment failure at VB for the sensitivity study is about 3.7 times the base case. While the relative increase in the probability of containment failure at VB is large, the low probability of occurrence of this PDS Group renders the impact of the increased failures to be insignificant. For the anticipated transient without scram (ATWS) PDS Group reported in Table 2.5-11, the differences between the base and the sensitivity cases are not significant.

Table 2.5-8
Comparison of APET Results With and Without
T-I Hot Leg Breaks and SGTRs
PDS Group 1: Slow SBO

Fraction With RCS Pressure in Four Ranges:

	At UTAF	At VB Base Case	At VB No T-I Breaks
SSPr	0.171	0.005	0.023
HiPr	0.201	0.231	0.237
ImPr	0.628	0.263	0.272
LoPr	0.000	0.502	0.467

	Base Case	Sensitivity Case
Fraction With CF during CD Total	0.055	0.052
Catastrophic Rupture	0.018	0.019
Rupture	0.028	0.025
Leak	0.009	0.008
Fraction With No VB	0.576	0.576
Fraction With CF at VB Total	0.089	0.093
Catastrophic Rupture	0.038	0.039
Rupture	0.023	0.023
Leak	0.010	0.010
Direct Contact	0.018	0.021
Fraction With VB, but No CF at VB	0.335	0.331
Fraction With CF by Late Burn	0.096	0.095
Fraction With CF by Very Late OP	0.004	0.004
Fraction With CF by Very Late BMT	0.041	0.041

Table 2.5-9
Comparison of APET Results With and Without
T-I Hot Leg Breaks and SGTRs
PDS Group 2: Fast SBO

Fraction With RCS Pressure in Four Ranges:

	At UTAF	At VB Base Case	At VB No T-I Breaks
SSPr	1.000	0.034	0.143
HiPr	0.000	0.108	0.108
ImPr	0.000	0.247	0.246
LoPr	0.000	0.611	0.502

	Base Case	Sensitivity Case
Fraction With CF during CD Total	0.047	0.043
Catastrophic Rupture	0.015	0.013
Rupture	0.026	0.024
Leak	0.006	0.006
Fraction With No VB	0.350	0.350
Fraction With CF at VB Total	0.134	0.156
Catastrophic Rupture	0.063	0.062
Rupture	0.024	0.030
Leak	0.016	0.016
Direct Contact	0.031	0.048
Fraction With VB, but No CF at VB	0.516	0.494
Fraction With CF by Late Burn	0.176	0.156
Fraction With CF by Very Late OP	0.002	0.002
Fraction With CF by Very Late BMT	0.075	0.072

Table 2.5-10
Comparison of APET Results With and Without
T-I Hot Leg Breaks and SGTRs
PDS Group 5: Transients

Fraction With RCS Pressure in Four Ranges:

	At UTAF	At VB Base Case	At VB No T-I Breaks
SSPr	1.000	0.112	0.500
HiPr	0.000	0.001	0.000
ImPr	0.000	0.108	0.105
LoPr	0.000	0.779	0.395

	Base Case	Sensitivity Case
Fraction With CF during CD Total	0.002	0.001
Catastrophic Rupture	0.0010	0.0004
Rupture	0.0007	0.0004
Leak	0.0002	0.0002
Fraction With No VB	0.798	0.450
Fraction With CF at VB Total	0.021	0.078
Cat. Rupture	0.009	0.041
Rupture	0.004	0.012
Leak	0.005	0.020
Direct Contact	0.003	0.005
Fraction With VB, but No CF at VB	0.181	0.472
Fraction With CF by Late Burn	0.000	0.001
Fraction With CF by Very Late OP	0.016	0.039
Fraction With CF by Very Late BMT	0.023	0.056

Table 2.5-11
Comparison of APET Results With and Without
T-I Hot Leg Breaks and SGTRs
PDS Group 6: ATWS

Fraction With RCS Pressure in Four Ranges:

	At UTAF	At VB Base Case	At VB No T-I Breaks
SSPr	1.000	0.003	0.012
HiPr	0.000	0.078	0.107
ImPr	0.000	0.218	0.237
LoPr	0.000	0.701	0.644

	Base Case	Sensitivity Case
Fraction With CF during CD Total	0.001	0.001
Catastrophic Rupture	0.0006	0.0005
Rupture	0.0006	0.0004
Leak	0.0002	0.0002
Fraction With No VB	0.275	0.263
Fraction With CF at VB Total	0.046	0.047
Catastrophic Rupture	0.013	0.013
Rupture	0.015	0.015
Leak	0.017	0.018
Direct Contact	0.001	0.001
Fraction With VB, but No CF at VB	0.679	0.690
Fraction With CF by Late Burn	0.001	0.001
Fraction With CF by Very Late OP	0.071	0.069
Fraction With CF by Very Late BMT	0.080	0.077

2.6 Insights from the Accident Progression Analysis

For internal initiators, there is a good chance that non-bypass accidents will be arrested before vessel failure. The arrest of core damage is due to the recovery of offsite power or the reduction of RCS pressure to the point where a system operating at the onset of core damage can inject successfully. Even if core damage proceeds to failure of the lower head, the containment is not likely to fail.

The occurrence of containment failure during the time of core degradation is not likely because for many sequences, ac power, and hence, the hydrogen ignition system and air return fans are operating. For SBOs, the probability of early containment failure is somewhat likely because hydrogen can accumulate in the ice condenser where there is no steam-inerting of the atmosphere. The probability that ignition of hydrogen occurs in areas of locally high concentration is low, however, because of lack of an ignition source in the timeframe considered. When power is recovered during core degradation for an SBO, it is more likely that an ignition source is present, although more often than not, the air return fans are effective in mixing the containment atmosphere before ignition occurs. This is mainly because it is assumed that mixing occurs after the bulk of the hydrogen is released. Overall, for SBOs, the mean conditional probability (the probability is conditional on occurrence of core damage for the SBO accidents) that the containment fails during core degradation is on the order of 0.05.

The occurrence of containment failure at vessel failure is more likely than failure during core degradation, although the likelihood is still quite low. The mechanisms causing failure of the containment at VB depend on the RCS pressure at the time the vessel fails. If the RCS is at low pressure (less than 200 psia), the pressure increase in containment is due primarily to hydrogen combustion and can be augmented by ex-vessel steam explosions, if there is water in the reactor cavity. If the RCS is at high pressure (greater than 200 psia), the pressure increase is due to hydrogen combustion and HPME acting together. The expulsion of molten core debris at high pressure from the reactor vessel results in a substantial portion of the core debris being injected into the containment atmosphere in the form of fine particles. This causes rapid transfer of sensible heat to the containment atmosphere and the rapid generation of additional hydrogen from the oxidation of the metal in the particles by the accompanying steam. Subsequent combustion of the hydrogen generated in the direct heating event as well as of pre-existing hydrogen in containment augments the direct heating pressure increase.

For the SBOs, the conditional probability of containment failure at VB is about 0.12; roughly half the failures occur by HPME/hydrogen events (high RCS pressure) and half by combustion of pre-existing hydrogen and hydrogen created at VB (low RCS pressure). For the ATWSs, containment failure at VB occurs with a conditional probability of about 0.05, with about equal contribution from HPME/hydrogen events and hydrogen burns coupled with ex-vessel steam explosions. For the Transients, containment failure at VB is predicted to occur very infrequently, the mean conditional probability is about 0.02. For the LOCAs, the containment is predicted to fail at VB with a conditional probability of roughly 0.05, mostly due to HPME/hydrogen events, while hydrogen burns coupled with ex-vessel steam explosions also contribute. All of the accidents have a very low conditional probability (on the order of 0.002) of containment failure at VB due to alpha mode failure, where an in-vessel steam explosion fails both the vessel and the containment.

The relatively low probability of containment failure at VB is due, in large part, to the depressurization of the RCS before VB. Depressurization of the RCS before the vessel fails is quite effective in reducing the loads placed upon the containment at VB. The effective mechanisms are temperature-induced failure of the hot leg or surge line, temperature-induced failure of the RCP seals, and the sticking open of the PORVs. All of these mechanisms are inadvertent and beyond the control of the operators. The apparent beneficial effects of depressurizing the RCS when lower head failure is imminent indicate that further investigation of depressurization may be warranted. The dependency of containment integrity on failures that occur at unpredictable locations and at unpredictable times is somewhat unsettling. Analysis of the effects of increasing PORV capacity, providing the means to open the PORVs in blackout situations, and changing the procedures to remove the restricting conditions on deliberate depressurization might prove rewarding in decreasing the probability of early containment failure at PWRs with ice condenser containments.

Another factor limiting the probability that the containment will fail at VB is that there is a high likelihood that the reactor cavity will contain large amounts of water at VB (the bottom of the vessel is submerged in nominally 8 ft of water). The presence of a large amount of water inhibits the dispersal of debris from the cavity, thus lowering the threat from direct containment heating at VB. The presence of water also contributes to the probability that core debris released from the vessel will be cooled. If CCI does initiate, the release will be scrubbed by the overlaying pool of water. On the other hand, water in the cavity can increase the possibility of ex-vessel steam explosions which can also threaten the integrity of the containment. Containment failure by ex-vessel steam explosion was investigated in this study and was found to be a minor threat. An ex-vessel steam explosion can also contribute to the radionuclide release at vessel breach.

Late failures of containment due to deflagration of combustible gases (hydrogen and carbon monoxide) occur with non-negligible probability only for the SBOs in which the mean conditional probability of occurrence is 0.15. When considering all PDSs, the mean conditional probability is a few percent. The mean conditional probability of very late failures due to BMT is low for the non-bypass accidents, the mean probabilities are less than 0.10. For SGTR initiators, the mean conditional probability that basemat melt-through occurs is 0.22, and for Event V it is 0.39. The high occurrence of basemat melt-through for bypass accidents is because there is virtually no cavity water in these sequences to prevent core-concrete interaction. Long-term overpressure of containment occurs most frequently for the LOCA accidents, with a mean conditional probability of occurrence of 0.22. This is because long-term containment heat removal through the containment sprays failed early in the accident. For the other plant damage states, the occurrence of long-term overpressure is unlikely.

Although their core damage frequency is relatively low, the bypass accidents are important for internal initiators. This is due to the low probability of early containment failure for the more frequent accidents, LOCAs and SBO. Given a core damage event, the occurrence of bypass is about as likely to defeat the containment function as a LOCA or SBO with early containment failure. For Event V, the importance of bypass is even greater, because the release occurs earlier than for an SGTR. Even though a bypass of the containment is created for the V-sequence, there is a mean probability of 0.80 that the break in the interfacing low pressure system will be located such that when the releases commence, they are scrubbed by the area fire sprays.

2.7 References

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3. RADIOLOGICAL SOURCE TERM ANALYSIS

The source term is the information passed to the next analysis so that the offsite consequences can be calculated for each group of accident progression bins (APBs). The source term for a given bin consists of the release fractions for the nine radionuclide groups for the early release and for the late release, additional information about the timing of the releases, the energy associated with the releases, and the height of the releases.

The source terms for Sequoyah are generated by the computer model, SEQSOR. The aim of this model is not to calculate in a mechanistic fashion the behavior of the fission products by application of first principles of chemistry, thermodynamics, and heat and mass transfer. Instead, it represents the results and interim results of the more detailed computer codes that do consider these principles. Although SEQSOR is a simple parametric model coded in FORTRAN, it will be referred to in this analysis as the SEQSOR code.

A more complete discussion of the source term analysis, and of SEQSOR in particular, may be found in NUREG/CR-5360.* The methods on which SEQSOR is based are presented in NUREG/CR-4551, Volume 1, and the source term issues considered by the expert panels are described more fully in NUREG/CR-4551, Volume 2, Part 4.

Section 3.1 summarizes the features of the Sequoyah plant that are important to the magnitude of the radionuclide release. Section 3.2 presents a brief overview of the SEQSOR code, and Section 3.3 presents the results of the source term analysis. Section 3.4 discusses the partitioning of the thousands of source terms into groups for the consequence analysis. Section 3.5 concludes this section with a summary of the insights gained from the source term analysis.

3.1 Sequoyah Features Important to the Source Term Analysis

The reactor system of Sequoyah Unit 1 consists of a four-loop pressurized water reactor (PWR). The reactor system is situated within a free-standing steel shell containment that forms a pressure boundary with the external environment. Figure 1.1 shows a section through the Sequoyah containment. More detail on the Sequoyah plant is contained in Sections 1.2 and 2.1 and is not repeated here.

The design pressure of the Sequoyah containment is 10.8 psig, although the mean value of the failure pressure distribution provided by the structural experts is six times the design pressure. The failure pressure, when compared with loads during the accident progression, leads to relatively low probabilities of containment failure (CCF). This is evidenced by the results of the accident progression analysis. If the containment fails, the timing, location, and mode of failure are important to the magnitude and character of the source term.

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*H.-N. Jow, W. B. Murfin, and J. D. Johnson, "XSOR Codes Users Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, (unpublished).

Emergency containment heat removal (CHR) at Sequoyah is by the ice condenser (IC) and the containment spray system (CSS) as described in Sections 2.1.2 and 2.1.3. Both the IC and the sprays are quite effective in removing fission products from the containment atmosphere. As long as the ice is not melted or bypassed, there are no accident situations at Sequoyah in which fission products will not be removed from the atmosphere as they pass through the IC. If the air return fans (ARFs) are operating, the decontamination of the IC is even more effective, especially for the first few passes through the ice. If electric power is available and the sprays have not failed due to hardware faults, they become a backup as well as a long-term means for decontamination of the containment atmosphere. Decontamination by the sprays or IC before and immediately following vessel breach (VB) is important in reducing the release if the containment fails early.

The Sequoyah reactor cavity is located such that for sequences with injection of the contents of the refueling water storage tank (RWST) into containment as well as melting of more than one quarter of the ice, the cavity will invariably be flooded at the time of vessel failure, as described in Section 2.1.7. If the reactor cavity is dry, core-concrete interaction (CCI) will occur upon VB, and the fission products released during CCI are unmitigated within the cavity. If the cavity is flooded, CCI is not as likely as when the cavity is dry, and furthermore, if CCI occurs, the releases are subject to scrubbing from the overlying water.

Two accident scenarios have been identified at Sequoyah that bypass the containment: Event V and steam generator tube ruptures (SGTRs). In Event V, the check valves that separate the low pressure injection system (LPIS) from the reactor coolant system (RCS) fail. The LPIS piping is not designed for full RCS pressure, and it fails outside the containment. This provides a direct pathway from the vessel to the auxiliary building. It is possible that the failure in the LPIS piping is at a location where there will be some scrubbing of the fission products released from the vessel by area fire sprays. If the break is not at such a location, there may be few effective removal mechanisms between the core and the environment, and releases could be quite high.

The magnitude of the source term from an SGTR accident depends on the integrity of the secondary system and the containment. If the integrity of both is maintained, the releases may be quite small. If the safety relief valves (SRVs) on the secondary system stick open, then a direct path from the vessel to the environment is created and the releases may be very high. If the SRVs on the secondary system do not stick open, then the releases depend on the time at which the containment fails (if at all) as in non-bypass accidents.

In summary, the Sequoyah containment is relatively robust, which reduces the likelihood of early containment failure. When functional, the IC and sprays are effective in decontamination of the atmosphere. While ice still remains in the condenser, the IC is a passive mitigation system not requiring power to be effective. Operation of the ARFs enhances the decontamination effects of the IC. If a water pool covers the core debris in the cavity after breach, releases from CCI can be mitigated by scrubbing. In Event V and SGTRs in which the secondary systems SRVs are stuck open, the release path bypasses the containment.

3.2 Description of the SEQSOR Code

This section describes how the source term is computed for each APB. The source term is more than the fission product release fractions for each radionuclide class; it also contains information about the timing of the release, the height of the release, and the energy associated with the release. The next subsection presents a brief overview of the parametric model used to calculate the source terms. Section 3.2.2 discusses the model in some detail; a complete discussion of SEQSOR may be found in Reference 1. Section 3.2.3 presents the variables sampled in the source term portion of this analysis.

3.2.1 Overview of the Parametric Model

SEQSOR is a fast-running, parametric computer code used to calculate the source terms for each APB for each observation for Sequoyah. As there are typically a few thousand bins for each observation and 200 observations in the sample, the need for a source calculation method that requires a minimum of computer time for one evaluation is obvious. SEQSOR does not mechanistically calculate the behavior of the fission products by application of first principles of chemistry, thermodynamics, and heat and mass transfer. SEQSOR does provide a framework for integrating the results and interim results of the more detailed codes that do consider these quantities. Since many of the variables SEQSOR uses to calculate the release fractions were determined by a panel of experts, the results of the detailed codes enter SEQSOR after "filtering" by the experts.

The 60 radionuclides (also referred to as isotopes, or fission products) considered in the consequence calculation are not dealt with individually in the source term calculation. Some different elements behave similarly enough both chemically and physically in the release path that they can be considered together. The 60 isotopes are placed in nine radionuclide classes as shown in Table 3.2-1. It is these nine classes that are treated individually in the source term analysis.

Table 3.2-1
Isotopes in Each Radionuclide Release Class

Release Class	Isotopes Included
1. Inert Gases	Kr-85, Kr-85M, Kr-87, Kr-88, Xe-133, Xe-135
2. Iodine	I-131, I-132, I-133, I-134, I-135
3. Cesium	Rb-86, Cs-134, Cs-136, Cs-137
4. Tellurium	Sb-127, Sb-129, Te-127, Te-127M, Te-129, Te-129M, Te-131M, Te-132
5. Strontium	Sr-89, Sr-90, Sr-91, Sr-92
6. Ruthenium	Co-58, Co-60, Mo-99, Tc-99M, Ru-103, Ru-105, Ru-106, Rh-105

Table 3.2-1 (continued)

Release Class	Isotopes Included
7. Lanthanum	Y-90, Y-91, Y-92, Y-93, Zr-95, Zr-97, Nb-95, La-140, La-141, La-142, Pr-143, Nd-147, Am-241, Cm-242, Cm-244
8. Cerium	Ce-141, Ce-143, Ce-144, Np-239, Pu-238, Pu-239, Pu-240, Pu-241
9. Barium	Ba-139, Ba-140

3.2.2 Description of SEQSOR

Since the largest consequences generally result from accidents in which the containment fails before VB or about the time of VB, the nomenclature and structure of SEQSOR reflect failure at VB. An early release occurs before, at, or a few tens of minutes after VB, and a late release occurs several hours after VB. In general, the early release is due to fission products that escape from the fuel while the core is still in the RCS, that is, before VB, and is often referred to as the RCS release. The late release is largely due to fission products that escape from the fuel during the CCI and is referred to as the CCI release. The late release includes not only fission products released from the core during CCI, but also material released from the fuel before VB that deposits in the RCS or the containment and is revolatilized after VB.

For situations in which the containment fails many hours after VB, the "early" release equation is still used, but the release is better termed the RCS release. After both releases are calculated in SEQSOR, they are combined into the late release, and the early release is set to zero. For radionuclide class i, the early (or RCS) release is calculated from the following equation:

$$\mathrm{ST(i) = [FCOR(i) * FVES(i) * FCONV(i)/DFE(i)] + DST[FDCH(i)]}. \quad \text{(Eq. 3.1)}$$

And the late or CCI release is calculated from

$$\mathrm{STL(i) = [(1\text{-}FCOR(i)) * FPART(i) * FCCI(i) * FCONC(i)/DFL(i)] + DLATE[FLATE(i)] + LATEI}. \quad \text{(Eq. 3.2)}$$

Both equations are valid for most APBs, but are not complete; the additional terms are either small or apply only to certain types of accidents not shown in this summary for reasons of expediency. For example, some of the omitted terms concern releases from Event V and SGTR accidents. The term LATEI applies only for the iodine radionuclide class. The complete equations used are presented in NUREG/CR-5360.*

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*H.-N. Jow, W. B. Murfin, and J. D. Johnson, "XSOR Codes Users Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, (unpublished).

The FORTRAN listing of SEQSOR is in Appendix B. The meaning of the terms in the equations above is as follows:

ST	=	fraction of the radionuclide in the core at start of accident released to environment as part of RCS release;
FCOR	=	fraction of the radionuclide in the core released to the vessel before VB;
FVES	=	fraction of the radionuclide released to the vessel that is subsequently released to the containment;
FCONV	=	fraction of the radionuclide in the containment from the RCS release that is released from the containment in the absence of any mitigating effects;
DFE	=	decontamination factor for the RCS releases (sprays, etc.);
DST	=	fraction of core radionuclide released to the environment due to DCH at VB;
FDCH	=	fraction of radionuclide in the portion of the core involved in DCH that is released to the containment at VB;
STL	=	fraction of the radionuclide in the core at the start of the accident released to environment as part of the CCI release;
FPART	=	fraction of the core participating in the CCI;
FCCI	=	fraction of the radionuclide in the core material at the start of CCI subsequently released to the containment;
FCONC	=	fraction of the radionuclide in the containment from the CCI release released from the containment in the absence of any mitigating effects;
DFL	=	decontamination factor for the late releases (sprays, etc.);
DLATE	=	fraction of core radionuclide released to the environment due to revolatilization from the RCS late in the accident;
FLATE	=	fraction of core radionuclide remaining in the RCS that is revolatilized late in the accident; and
LATEI	=	fraction of core iodine in the containment that assumes a volatile form and is released late in the accident.

Only the functional dependence of DLATE on FLATE and of DST on FDCH is indicated above, but DLATE and DST also depend on other variables such as FCOR. DST and DLATE are expressed as fractions of the initial core inventory like ST and STL. Complete expressions for DST and DLATE and an expanded discussion of them may be found in the XSOR dcoument.*

Figure 3.2-1 depicts the parametric equations schematically as a flow diagram. Coming in from the left is all the radioactivity in any radionuclide class. The black arrows represent releases to the environment, and the white arrows represent material retained in the RCS or in the containment. The first division of the radioactive material is indicated by FCOR. The top branch (FCOR) represents the fraction released from the core before VB, and the lower branch (1-FCOR) represents the amount still in the RCS at VB. The FCOR branch is then split into what leaves the RCS before or at VB (FVES) and what is retained in the RCS past VB (1-FVES). Of the material retained in the RCS at VB, a fraction FLATE is revolatilized later. Of the revolatilized fraction, a portion is removed by engineered removal mechanisms such as sprays (variable 1/DFL), and another portion is removed by natural mechanisms such as deposition (variable FCONRL). Part of the revolatilized fraction not removed escapes to the environment (DLATE in the equation) as indicated by the top black arrow in Figure 3.2-1. FCONRL is the containment release fraction for the late revolatilization release and is set equal to the FCONC value for tellurium.

When evaluated as part of the integrated risk analysis, SEQSOR is run in the "Sampling mode." That is, most of the variables in the release fraction equations are determined by sampling from distributions for that variable, and the value for each variable varies from observation to observation. Most of these distributions were provided by an expert panel.

Equations 3.1 and 3.2 contain 11 variables. Distributions for seven of these variables were provided by the source term expert panel: FCOR, FVES, FCONV, DST, FCCI, FCONC, and FLATE. Two other variables were also partially quantified by the expert panel; for DFE and DFL, distributions for the IC decontamination factor (DF) were provided. The distributions for the other DFs considered for DFE and DFL (such as the DFs for sprays or pool scrubbing) and the distribution for FPART and LATEI were determined either by the expert panel for the previous draft of this report or internally.

For each variable in Equations 3.1 and 3.2, a distribution is usually provided for the nine radionuclide release classes defined in Table 3.2-1, although release classes are sometimes grouped together. For example, for FCOR, the experts provided separate distributions for all nine classes; whereas for other variables, they stated that classes 5 through 9 should be considered together as an aerosol class. The distributions for the nine radionuclide classes are assumed to be completely correlated. That is, a single Latin Hypercube Sample (LHS) variable applies to each variable in the release fraction equation, and it applies to the distributions for all nine radionuclide classes. For example, if the random variable provided by the LHS for FCOR is 0.777, the 77.7th percentile value is chosen from the iodine distribution, the cesium distribution, the tellurium distribution etc., for FCOR.

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*H.-N. Jow, W. B. Murfin, and J. D. Johnson, "XSOR Codes Users Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, (unpublished).

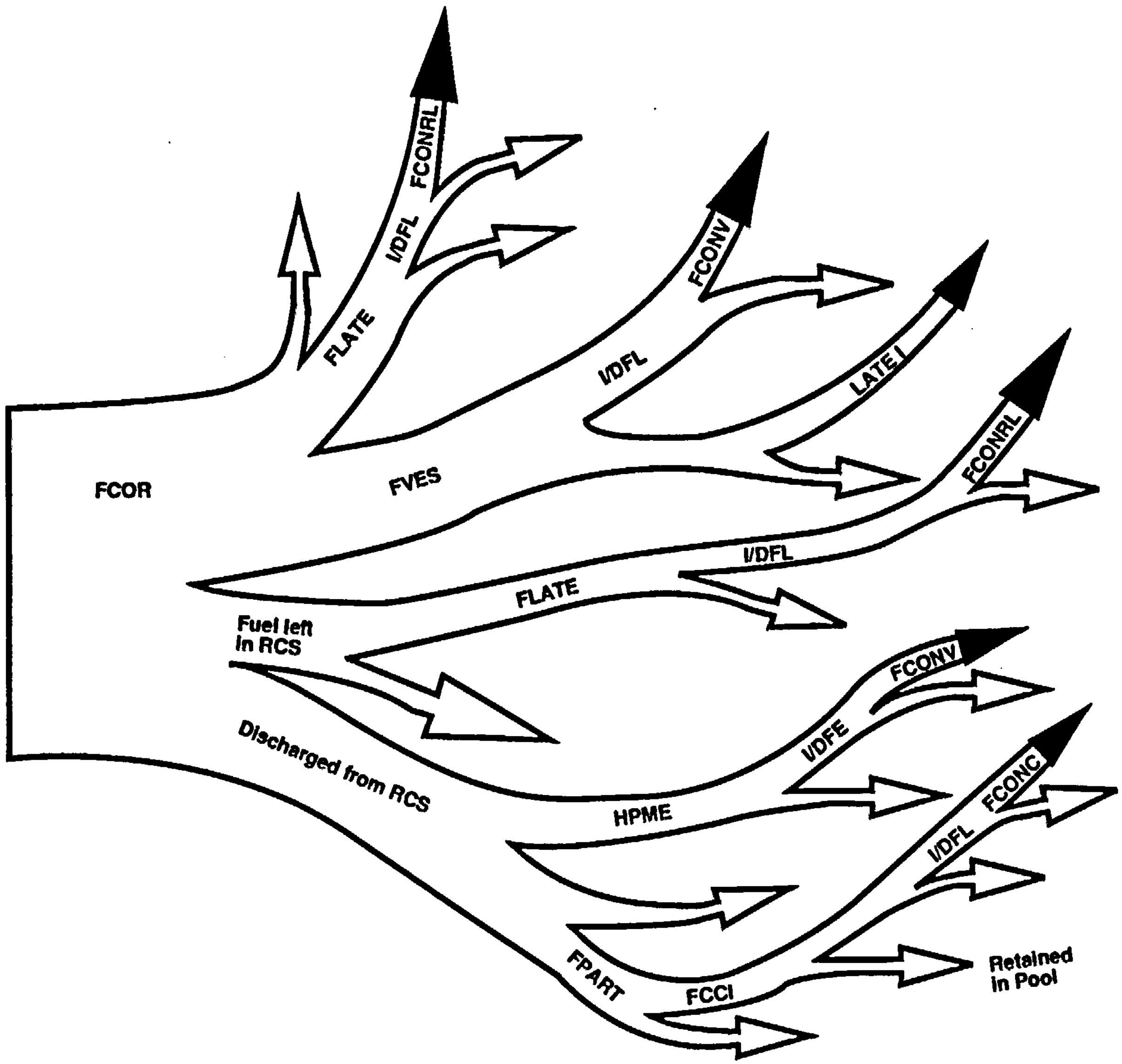

Figure 3.2-1. Blood Flow Diagram for SEQSOR

Many of the variables in Equations 3.1 and 3.2 are determined directly by sampling from distributions provided by a panel of experts (see NUREG/CR-4551, Volume 2, Part 4). Other variables are derived from such values, and still others were determined internally (see NUREG/CR-4551, Volume 2, Part 6 and the XSOR document*). A brief discussion of each variable in Equations 3.1 and 3.2 follows.

FCOR is the fraction of the fission products released from the core to the vessel before vessel failure. The value used in each sample observation is obtained directly from the experts' aggregate distribution. There are separate distributions for each fission product group for two cases: high and low in-vessel zirconium oxidation.

FVES is the fraction of the fission products released to the vessel that is subsequently released to the containment before or at vessel failure. As for FCOR, the value used in each sample observation is obtained directly from the experts' aggregate distribution, and there are separate distributions for each fission product group. There are four cases: RCS at system setpoint pressure, RCS at high or intermediate pressure, RCS at low pressure, and Event V.

FCONV is the fraction of the fission products in the containment from the RCS release that is released from the containment in the absence of mitigating factors such as sprays. The expert panel provided distributions for FCONV for five cases, each of which applies to all species except the noble gases. The five cases are containment leak at or before VB and the containment sprays not operating, containment leak at or before VB and the containment sprays operating, containment rupture in the upper compartment (UC) at or before VB, containment rupture in the lower compartment (LC) at or before VB, and late containment rupture. The case differentiation on spray operation is to account for differences in containment atmosphere temperature and humidity. Distributions for other levels and times of containment failure (except for very late failures) are derived in SEQSOR from these five distributions. A sixth distribution applies to Event V and was quantified internally. If the containment failure happens a day or more after the start of the accident, none of these distributions is used for FCONV. These very late failures occur due to long-term overpressurization or basemat melt-through (BMT). For very late failures, the long time period allows the engineered and natural removal processes to reduce the concentration of the fission products in the containment atmosphere, so the fraction of the fission products released before or at VB remaining airborne at the time of containment failure is very small. This fraction was estimated internally to be 1.0E-6, and FCONV is set to that value for containment failure at very late times.

DFE is the DF for early releases. At Sequoyah, the containment sprays and the IC are the mechanisms that contribute to DFE for non-bypass accidents. The variable for the early IC DF is DFICV and the variable for the early spray DF is DFSPV. DFE is the product of DFICV and DFSPV for non-bypass

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*H.-N. Jow, W. B. Murfin, and J. D. Johnson, "XSOR Codes Users Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, (unpublished).

accidents. For Event V, when the releases are scrubbed by fire sprays, the variable for the scrubbing DF is VDF. DFE is set equal to VDF when used for Event V. The distribution for VDF was determined internally.

DFICV is the DF for the IC for early releases. The source term expert panel determined the DFICV distributions for four cases: fans operating and no prior containment failure; fans operating and the containment is failed; fans not operating; and failure of the vessel involved a DCH event. Fans are considered because the DF for multiple passes through the IC is higher than for a single pass. The DCH event is considered separately because conditions are very different from normal blowdown. A bypass fraction is applied to DFICV, and can be one of three levels: no bypass, partial bypass, or the ice is completely bypassed or melted. DFICV is then described by:

$$\text{DFICV} = 1./\{(1. - \text{FBYPV})/\text{DFICV} + \text{FBYPV}\}, \qquad \text{(Eq. 3.3)}$$

where FBYPV is the effective bypass fraction for the RCS releases. For completely melted ice FBYPV = 1.0, except when fans are operating, in which case, FBYPV = 0.8. For partial bypass, FBYPV = 0.1, for catastrophic rupture, FBYPV = 1.0, and for no bypass, FBYPV = 0.0. More detail about DFICV is provided in the XSOR document.*

DFSPV is the DF for the sprays for early releases. The distributions for DFSPV were determined internally. There are two spray distributions which apply to the fission products released from the RCS before or at VB: the first applies when the containment fails before or at VB and the RCS is at high pressure at VB; and the second applies when the containment fails after VB or when the containment fails at VB but the RCS is at low pressure. Each distribution applies to all species except the noble gases. For failures of the containment in the very late time period, the value from the distribution is multiplied by 10 to account for the long time period which the sprays have to wash particulate material out of the containment atmosphere.

DST is the fission product release (in fraction of the original core inventory) from the fine core debris particles that are rapidly spread throughout the containment in a DCH event at VB. The experts provided distributions for the fractions of the fission products that are released from the portion of the core involved in DCH for VB at high pressure (1000 to 2500 psia) and for VB at intermediate pressure (200 to 1000 psia). There are separate distributions for each fission product group (inert gases, iodine, cesium, etc.). These distributions are used only if the containment fails at (or within a few minutes of) vessel failure. For containment failures that occur hours after VB, it was internally estimated that the amount of fission products from DCH remaining in the atmosphere many hours after VB would be negligible.

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*H.-N. Jow, W. B. Murfin, and J. D. Johnson, "XSOR Codes Users Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, (unpublished).

FPART is the fraction of the core leaving the vessel and not participating in high pressure melt ejection (HPME) that participates in CCI. The value of this variable is determined in the accident progression event tree (APET). There are four ranges of values for FPART: none, small (nominally 15%), moderate (nominally 50%), and large (nominally 100 percent). Five percent of the core is estimated to remain in the vessel indefinitely and is not available to participate in CCI under any circumstances; SEQSOR subtracts this 5% from FPART. The amount of the core participating in HPME is not included in FPART; that is, FPART always assumes the large range when HPME occurs.

FCCI is the fraction of the fission products present in the core material at the start of CCI that is released to the containment during CCI. The experts provided distributions for four cases that depended upon the fraction of the zirconium oxidized in the vessel and the presence or absence of water over the core debris during CCI. There are separate distributions for each fission product group.

FCONC is the fraction of the fission products released to the containment from the CCI that is released from the containment. The expert panel provided distributions for FCONC for five cases. There are separate distributions for each fission product group (inert gases, iodine, cesium, etc.). The five cases are the same as for FCONV, and there is also an additional sixth case for Event V. None of these cases is used for containment failure in the very late period (after 24 h). Since containment failure occurs many hours after most of the fission products have been released from CCI, only a very small fraction of these fission products will still be in the containment atmosphere at the time of containment failure. This fraction was estimated internally to be on the order of 1.0E-4. The exact value is determined by using the FCONC distribution for case 3, containment rupture in the UC at or before VB. The ratio of the LHS value from the distribution to the median value of the distribution is multiplied by 1.0E-4 to obtain the value of FCONC used for very late period containment failure. This value is used whether the release is due to BMT or aboveground failure by long-term overpressurization.

DFL is the DF for late releases. At Sequoyah, DFL can be due to the IC, the containment sprays, or a pool of water over the core debris during CCI. The variable for the late IC DF is DFICC, the variable for the late spray DF is DFSPC, and the variable for the pool scrubbing DF is VPS. For non-bypass accidents, DFL is the product of DFICC and the larger value of DFSPC and VPS. As with DFE, DFL is set equal to VDF when used for Event V.

DFICC is the DF for the IC for late releases. The source term expert panel determined the DFICC distributions for three cases that are identical to the first three cases for DFICV. The bypass fraction applied to DFICC is similar to that applied for DFICV, although the bypass is determined at a later time in the APET.

DFSPC is the DF for the sprays for late releases. There is a single distribution used for DFSPC, which was determined internally. The distribution applies to all species except the noble gases. As for DFSPV, if the containment fails in the very late period, the value from the late containment failure spray distribution is multiplied by 10 to account for the very long time the sprays have to wash particulate material out of the containment atmosphere.

VPS is the pool scrubbing DF and is obtained from one of two internally determined distributions. One distribution applies to a full cavity and the other to a partially full cavity (accumulator water only).

FLATE accounts for the release of radionuclides from the RCS late in the accident. Like DST, it is a fraction of the original core inventory. Fission products deposited in the RCS before VB may revert to a volatile form after the vessel fails and make their way to the environment. This term considers only revolatilization from the RCS. Revolatilization from the containment is considered to be significant only for iodine, and is included in the LATEI variable. The expert panel provided distributions for the fraction of the radionuclides remaining in the RCS that are revolatilized. The amount remaining in the RCS is a function of FCOR, FVES, and other terms and is calculated in SEQSOR. The experts concluded that whether there was effective natural circulation through the vessel was important in determining the amount of revolatilization. Thus, there are two cases: one large hole in the RCS, and two large large holes in the RCS. The experts provided separate distributions only for iodine, cesium, and tellurium. Revolatilization is not possible for the inert gases as they would not deposit, and the experts concluded that it is negligible for radionuclide classes 5 through 9. FLATE is computed in the following manner: the value from the experts' distributions is applied to the fraction of the radionuclide remaining in the RCS to obtain the fraction of the core inventory released to the containment by this mechanism. This is multiplied by the FCONC value for tellurium to determine the fraction that escapes to the environment. The tellurium value for FCONC is considered to be appropriate for revolatilized material because it, like tellurium, is slowly released over a long time period.

LATEI accounts for iodine in the containment that may assume a volatile form, such as methyl iodide, and be released late in the accident. The primary source of this iodine is the water in the reactor cavity and the containment sumps (separate at Sequoyah). This term is added to the late release only for radionuclide class 2, iodine. The experts provided a distribution for the fraction of iodine in the containment that is converted to volatile forms. The method of calculating the amount of iodine remaining in the containment depends upon FCOR, FVES, FCCI, and other variables and is explained in the XSOR document.*

FISG and FOSG are the release fractions used for the RCS release for SGTR accidents. FISG is the fraction released from the core that enters the steam generator (SG), and FOSG is the fraction entering the SG that is released from the SG to the environment. For SGTR accidents, Equation 3.1 for the early or RCS release becomes:

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*H.-N. Jow, W. B. Murfin, and J. D. Johnson, "XSOR Codes Users Manual," NUREG/CR-5360, SAND89-0943, Sandia National Laboratories, (unpublished).

$$ST(i) = [FCOR(i) * \{FISG(i) * FOSG(i) + [1.0 - FISG(i)] * FVES(i) * FCONV(i) / DFE(i)\}] + DST(i). \quad \text{(Eq. 3.4)}$$

As the material passing from the SG to the atmosphere bypasses the containment, the variables FCONV and DFE are not applied to this release path. FISG and FOSG each have two cases: SGTRs in which the secondary SRVs reclose and SGTRs in which the secondary SRVs stick open.

No differentiation is made between BMT and above-ground leaks in the very late period. Even though the release point for BMT is underground, no allowance is made for attenuation or decontamination of the late fission product release. The BMT release is often dominated by the iodine release due to the LATEI term. The very slow passage of the gases through wet soil with a low driving pressure would undoubtedly result in some reduction in this release. This reduction could be quite large. Although giving no credit for removal in the wet soil is conservative, it is unimportant for the sample as a whole. The total releases from all the BMT failures of the containment are small compared to the releases from accidents and pathways in which the containment fails at or before VB, or when the containment is bypassed.

3.2.3 Variables Sampled for the Source Term Analysis

The 13 variables sampled for the source term analysis are listed in Table 3.2-2. That is, when SEQSOR was evaluated for all the bins generated by the APET evaluation for a given sample observation, all the sampled variables in SEQSOR had values chosen specifically for that observation. These values were selected by the LHS program from distributions previously defined. Most of these distributions were determined by the source term expert panel.

The sampling process works somewhat differently for the source term analysis than it does for the accident progression analysis. For the source term analysis, the LHS provided only a random number between 0.0 and 1.0 for each variable to be sampled. The actual distributions are in a data file (listed in Appendix B) read by SEQSOR before execution. The variables provided by the LHS are used to define quantiles in the variable distributions; the values associated with these quantiles are used as variable values in SEQSOR.

As an example of the sampling process, assume that the LHS value is 0.05 for FCOR for Sample Observation 1. The data tables in Appendix B.2 show that for low zirconium oxidation in-vessel, the 0.05 quantile values for FCOR are 0.18 for inert gases, 0.084 for iodine, 0.067 for cesium, etc. There is no correlation between any of the source term variables, but complete correlation within a variable. FCOR is not correlated with FVES, FCONV, or any other variable, but the values for the different cases and for the different radionuclide classes are completely correlated. That is, if the 0.05 quantile value is chosen for iodine for low zirconium oxidation, the 0.05 quantile value is also chosen for all the other radionuclide classes and for all values for high zirconium oxidation.

As all the source term variables are uniformly distributed from 0.0 to 1.0, and are uncorrelated, there are no columns for this information in Table 3.2-2. There is a separate distribution for each radionuclide class for each variable in this table unless otherwise noted in the variable description. The different cases for each variable are noted in the description. Not all the cases considered by SEQSOR are listed in Table 3.2-2; variable values for other cases are determined internally in SEQSOR, often from the values for the cases listed. For example, there is no distribution for FCONV for late leak. The value of FCONV for late leak is derived from the distribution for another case. (See the listing of subroutine FCONVC in Appendix B.)

Table 3.2-2
Variables Sampled in the Source Term Analysis

Variable	Description
FCOR	Fraction of each fission product group released from the core to the vessel before or at VB.
FVES	Fraction of each fission product group released from the vessel to the containment before or at VB.
VDF	DF for Event V when the releases are scrubbed by fire sprays.
FCONV	Fraction of each fission product group in the containment from the RCS release that is released from the containment in the absence of mitigating factors such as sprays.
FCCI	Fraction of each fission product group in the the core material at the start of CCIs that is released to the containment.
FCONC	Fraction of each fission product group in the containment from the CCI release that is released from the containment in the absence of mitigating factors such as sprays.
DFSP	DF for sprays; DFSPV for early releases, DFSPC for late releases.
LATEI	Fraction of the iodine deposited in the containment that is revolatilized and released to the environment late in the accident.
FLATE	Fraction of the deposited amount of each fission product group in the RCS that is revolatilized after VB and released to the containment.
DST	Fraction of each fission product group in the the core material that becomes aerosol particles in a DCH event at VB that is released to the containment.
DFIC	DF for the IC; DFICV for the early releases, DFICC for the late releases.

Table 3.2-2 (continued)

Variable	Description
FISG FOSG	Fraction of each fission product group released from the reactor vessel to the SG, and from the SG to the environment, in an SGTR accident.
VPS	DF for a pool of water overlying the core debris during CCI.

The variable identifiers given in Table 3.2-2 are used in several ways in the source term analysis. Consider FCOR, the first variable in Table 3.2-2. FCOR in the equation for fission product release is the actual fraction of each fission product group released from the core to the vessel before or at VB for the sample observation in question. But, FCOR is also used to refer to the experts' aggregate distributions from which the nine values (one for each radionuclide class or fission product group) for FCOR are chosen. Further, in the sampling process, FCOR is used to refer to the random number from the LHS used to select the values from these distributions. That is, as used in sampling, FCOR defines a quantile in these distributions. The release fractions associated with this quantile are used in SEQSOR as the FCOR values. Thus, in Table 3.2-2, the end use of each variable is given although the actual sampled variable is a random number between 0.0 and 1.0 used to select an actual value.

The 13 variables in Table 3.2-2 have been described more fully in the preceding section. The distributions for FCOR, FVES, FCONV, FCCI, FCONC, FLATE, DST, and DFIC were provided by the source term expert panel. These distributions, the reasoning that led each expert to his conclusions, and the aggregation of the individual distributions are fully described in NUREG/CR-4551, Volume 2, Part 4. VDF, DFSP, LATEI, FISG, FOSG, and VSP are discussed briefly below; the distributions for these source term variables and more discussion of them can be found in Appendix B.

The SGTR accidents with the secondary SRVs stuck open were not known to be significant to risk at Sequoyah when the source term expert panel met for the last time. Therefore, a special ad hoc panel was convened to consider the variables FISG and FOSG. These variables are discussed briefly below; more detail can be found in NUREG/CR-4551, Volume 2, Part 6. The LATEI variable was considered by the expert panel for the boiling water reactors (BWRs), but the BWR distributions were not used directly for the PWRs as discussed in more detail in Appendix B of this report.

VDF is the DF used for Event V when the releases are scrubbed by fire sprays. These accidents are referred to as V-Wet accidents. For these types of accidents, SEQSOR sets DFE to the value of VDF. The distribution for VDF was determined by the project staff. The range for VDF is from 1.6 to 5100; the median value is 6.2. VDF represents only scrubbing by passage of the aerosols through the water sprays. Any additional removal in the auxiliary building is accounted for by FCONV. The distribution for VDF is given in Appendix B.

DFSP refers to both the spray DF for the RCS (vessel) release, DFSPV, and the CCI spray DF, DFSPC. There is only one value for each of these DFs; that is, each DF applies to all radionuclide groups except the inert gases. The same random value between 0.0 and 1.0 from the LHS program is used to select both the RCS and CCI spray DF values. That is, the spray DF distributions are completely correlated. The spray DF distributions were determined by the project staff. For the RCS release with containment failure at VB, there are two distributions for the spray DF. One applies if the RCS was at high pressure before VB. In this case, most of the RCS release will escape from the vessel just at VB, and the sprays will be very ineffective. The range of the spray DF distribution is from 1.0 (no effect) to 2.8 and the median value is 1.6. For the RCS release with containment failure at VB with the RCS at low pressure before VB, much of the RCS release will have escaped from the vessel before VB, and the sprays will be very effective for that portion of the RCS release. The range of this spray DF distribution is from 2.3 to 2800; the median value is 40. The distribution for the CCI spray DF distribution ranges from 6.7 to 3200; the median value is 28. The complete distributions are contained in Appendix B.

LATEI refers to the evolution of iodine in volatile form from water in the containment late in the accident. Because of its volatile form (typically organic), this volatile iodine is released to the environment because it is unaffected by all the removal mechanisms (pool scrubbing, sprays, deposition, etc.). The release fraction determined by LATEI applies to all the iodine released from the fuel and retained in the containment in aqueous solution, which is expected to be the bulk of the iodine released from the vessel and remaining in the containment. In Sequoyah, this iodine would be expected to be contained in the water in the sump. The sump water does not play the same role in heat removal that the suppression pool does in the BWR, so the results of the expert panel (which apply to BWRs only) were not used directly. Instead, the distribution obtained specifically for PWRs in the first draft of this report was used. This is discussed further in Appendix B. The distribution used for LATEI ranges from 0.0 to 0.10; the median value is 0.05.

For the SGTRs where the secondary system SRVs reclose, the distributions for FISG and FOSG were determined by the project staff. For the SGTRs where the secondary system SRVs stick open, the distributions for FISG and FOSG were determined by an ad hoc expert panel. The panel provided distributions for the product FISG * FOSG for iodine, cesium, tellurium, and aerosols. There is no retention in the SGs for the noble gases. Complete distributions for FISG and FOSG are listed in Appendix B.

SPV is the DF for the late pool scrubbing of the CCI release. This DF is applied when the core debris is not coolable and CCI takes place under water. There are two distributions: one applies for a shallow pool (approximately 5 ft deep) that results if only the accumulator water enters the cavity, and the other distribution applies when the cavity is full (at least 10 ft deep). For both the shallow and deep pool distributions, one distribution applies to the iodine, cesium, barium, ruthenium, lanthanum, and cerium radionuclide classes, and another applies to the tellurium and strontium radionuclide classes. The distributions were determined by the NUREG-1150 project staff and are listed in Appendix B.

3.3 Results of the Source Term Analysis

This section presents the results of computing the source terms for the APBs produced by evaluating the APET. The APET's evaluation produced a large number of APBs, so, as in Section 2.5, only more likely and more important APBs are discussed here. However, source terms were computed for all the APBs for each of the 200 observations in the sample. The source term is composed of release fractions for the nine radionuclide groups for an early and a late release as well as release timing, release height, and release energy. As discussed previously, the source terms are computed by a fast-running parametric computer code, SEQSOR.

Section 3.3.1 presents the results for the internal initiators. The tables in this section are only a very small portion of the output obtained by computing source terms for each APB. More detailed results are contained in Appendix B, and complete listings are available on computer media by request.

3.3.1 Results for Internal Initiators

As in Section 2.5.1, the results of the source term analysis for internal initiators are presented for each PDS group.

3.3.1.1 Results for PDS Group 1: Slow SBO. As discussed in Section 2.5.1.1, this plant damage state (PDS) group consists of accidents in which all ac power is lost in the plant, but the steam-turbine-driven (STD) auxiliary feedwater system (AFWS) operates for several hours. When the batteries deplete, control of the STD AFWS is lost and it fails. This PDS group contains four PDSs: one has the RCS intact at uncovering of top of active fuel (UTAF), two have failure of the RCP seals before UTAF, and one has stuck-open PORVs before UTAF. In two of the four PDSs, the operators depressurized the secondary system before UTAF, and in two PDSs they did not. The PDSs in this group are listed in Table 2.2-2.

For this PDS group, VB is not inevitable because electric power may be recovered before the vessel fails. Releases are calculated by SEQSOR in this case, as fission products may escape to the containment through the PORVs or a temperature-induced (T-I) break before the arrest of core damage. In a small fraction of the times that core damage is arrested, the containment fails during core degradation (CD) due to hydrogen events. If so, an appropriate source term is provided by SEQSOR.

Table 2.5-1 lists the five most probable APBs for PDS Group 1, the five most probable APBs that have VB, and the five most probable APBs that have VB and early containment failure. Table 3.3-1 lists the mean source terms for these same APBs. The source term consists of the release fractions, the release height and energy, and the times associated with the release. The release fractions give the early (RCS) and late (CCI) releases as fractions of the core inventory at the start of the accident. Table 3.3-1 shows the time (in seconds) when the warning is given to evacuate the surrounding area, when the release starts, and the duration of the release. The elevation of the release is given in meters, and the energy in watts.

Although the same bins are shown in both Tables 2.5-1 and 3.3-1 and the structures of both tables are roughly analogous, there are some important differences. First, Table 3.3-1 has two designators for each APB. The first designator is the APB definition initially produced in the analysis of the APET; the second designator is the rebinned definition input to SEQSOR. Consider the first APB in Table 3.3-1: GDCFCDADFAAAB. Following evaluation of the APET, it was rebinned to GDCCFCDADDAAAB, with the tenth characteristic changing from F to D (see Section 2.4.2). Another important feature of Table 3.3-1 is that the characteristics of the early release segment are provided on the first line for each bin, and the characteristics of the late release segment are provided on the second line.

The other difference between the nature of Tables 2.5-1 and 3.3-1 lies in the nature of the information presented. In Table 2.5-1, the bin itself was well defined; that is, the characteristics of the bin did not vary from observation to observation. The only item that varied from observation to observation was the probability of the occurrence of the bin itself. Thus, Table 2.5-1 lists a conditional probability averaged over the 200 observations in the sample. In Table 3.3-1, the bin is still well defined, but because the variables used in calculating the fission product release vary from observation to observation, the source term for a specific bin varies with the observation. Thus, the entries in all columns in Table 3.3-1 except the Order and Bin columns represent averages over the 200 observations in the sample.

For example, consider the first APB in Table 3.3-1: GDCCFCDADDAAAB. Of the 200 observations in the sample, 38 had non-zero conditional probabilities for this bin. Because source terms are not computed for zero-probability bins, 38 source terms are associated with APB GDCCFCDADDAAAB. These 38 source terms were summed and then divided by 38 to produce the mean source terms given in the first two lines of Table 3.3-1.

The five most probable APBs and three of the five most probable APBs with VB for PDS Group 1 did not have containment failure. As a result, the releases associated with these APBs are very small. The first and fifth bins listed for the most probable APBs with VB have late failures. These releases are relatively large when compared with the releases for no failures. When there is no containment failure or late containment failure, SEQSOR describes releases with a single release segment rather than the two release segments used when there is containment failure. The five most probable APBs with VB and early containment failure have low conditional probabilities (see Table 2.5-1) but larger releases than the APBs without containment failure or with late containment failure. The mean source terms in Table 3.3-1 can be used to compare the releases for specific APBs. However, as these mean source terms are typically not calculated over the same sample elements, fine distinctions between source terms associated with different APBs may be lost in the averaging.

Table 3.3-1 presents mean source terms but does not contain any frequency information. In contrast, Figure 3.3-1 presents information on both source term size and frequency. Figure 3.3-1 summarizes the release fraction (CCDFs) for the iodine, cesium, strontium, and lanthanum radionuclide classes. It indicates the frequency with which different values of the release fraction are exceeded, and displays the uncertainty in that frequency. The curves in Figure 3.3-1 are derived in the following manner: for each observation, evaluation of the APET produced a conditional probability for each APB. Multipling by the frequency of the PDS group for that observation gives a frequency for the APB. Calculation of the source term for the APB gives a total release fraction for each APB. When all the APBs are considered, a curve of exceedance frequency versus release fraction can be plotted for each observation. Figure 3.3-1 summarizes these curves for the 200 observations in the sample.

Instead of placing all 200 curves on one figure, only four statistical measures are shown. These measures are generated by analyzing the curves in the vertical direction. For each release fraction on the abscissa, there are 200 values of the exceedance frequency (one for each sample element). From these 200 values, it is possible to calculate mean, median (50th quantile), 95th quantile, and 5th quantile values. When this is done for each value of the release fraction, the curves in Figure 3.3-1 are obtained. Thus, Figure 3.3-1 provides information on the relationship between the size of the release fractions associated with PDS Group 1 and the frequency at which these release fractions are exceeded, as well as the variation in that relationship between the observations in the sample.

As an illustration of the information in Figure 3.3-1, the mean frequency (yr^{-1}) at which a release fraction of 10^{-6} is exceeded due to PDS Group 1 is 4×10^{-6}, 1×10^{-6}, 1×10^{-6}, and 8×10^{-7} for the iodine, cesium, strontium, and lanthanum release classes, respectively. For a release fraction of 0.1, the corresponding mean exceedance frequencies are 4×10^{-7}, 4×10^{-7}, 2×10^{-8}, and $<10^{-10}$, respectively. The three quantiles (i.e., the median, 95th, and 5th) indicated the often large spread between observations. Typically, the mean curves drop very rapidly and move above the 95th quantile curve. This happens when the mean curve is dominated by a few large observations. This often occurs for large release fractions because only a few of the sample observations have nonzero exceedance frequencies for these large release fractions. Taken as a whole, the results in Figure 3.3-1 indicate that large source terms (e.g., release fractions ≥ 0.1) occur infrequently with PDS Group 1.

Table 3.3-1
Mean Source Terms for Sequoyah
Internal Initiators (PDS Group 1: Slow SBO)

Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (W)	Release Start (s)	Release Duration (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
Five Most Probable Bins*															
1	GDCCFCDADFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCCFCDADDAAAB			0.00E+00	4.70E+04	8.60E+04	3.90E-03	2.80E-05	1.90E-10	6.30E-11	4.00E-12	1.10E-12	2.00E-13	6.90E-13	5.30E-12
2	GDCDFCDADFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDADDAAAB			0.00E+00	4.70E+04	8.60E+04	4.00E-03	4.60E-05	7.50E-10	4.80E-10	2.00E-10	2.90E-11	1.10E-11	5.00E-11	2.10E-10
3	GDCBFCDADFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCBFCDADDAAAB			0.00E+00	4.70E+04	8.60E+04	4.00E-03	1.90E-05	1.00E-10	4.20E-11	2.70E-12	1.20E-12	1.50E-13	4.50E-13	4.20E-12
4	GDCDFCDADFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDADDBAAB			0.00E+00	4.70E+04	8.60E+04	3.90E-03	4.10E-05	4.20E-10	2.30E-10	7.00E-11	1.10E-11	4.00E-12	1.80E-11	7.60E-11
5	GDCBFCDBDFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCBFCDBDDAAAB			0.00E+00	4.70E+04	8.60E+04	4.30E-03	1.90E-05	1.20E-10	4.50E-11	5.00E-12	1.90E-12	2.40E-13	7.90E-13	6.20E-12
Five Most Probable Bins with VB*															
18	EEADBCAADABAAC	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	EEADBCAADABAAC			3.50E+06	4.70E+04	2.00E+02	1.00E+00	5.00E-02	1.40E-02	5.80E-03	3.00E-04	7.90E-06	2.70E-05	4.30E-05	2.50E-04
20	GGADBCABDFBAAD	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GGADBCABDDBAAD			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.50E-04	1.60E-09	1.00E-09	2.90E-10	2.00E-11	2.50E-11	2.70E-11	2.60E-10
22	GGADBCAADFBAAD	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GGADBCAADDBAAD			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.70E-04	1.50E-09	9.70E-10	3.30E-10	2.20E-11	2.50E-11	6.10E-11	3.10E-10
23	GFADBCABDFBAAC	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GFADBCABDDBAAC			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.50E-04	1.70E-09	1.00E-09	2.90E-10	2.00E-11	2.50E-11	2.60E-11	2.60E-10
24	EEADBCAADAAAAC	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	EEADBCAADAAAAC			3.50E+06	4.70E+04	2.00E+02	1.00E+00	5.50E-02	5.10E-03	3.20E-03	4.70E-04	2.60E-05	2.30E-05	8.00E-05	4.20E-04
Five Most Probable Bins with VB and Early CF*															
32	DHADBCAADAAAAD	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	1.90E-01	2.10E-01	1.60E-02	5.20E-04	2.70E-04	2.50E-05	6.70E-05	8.90E-04
	DHADBCAADAAAAD			1.60E+06	2.90E+04	2.20E+04	1.20E-02	4.80E-02	4.10E-03	1.10E-01	2.80E-02	2.80E-04	3.10E-03	3.40E-03	2.30E-02
45	DHADBCAADAAAAC	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.30E-01	2.50E-01	1.60E-02	5.20E-04	2.70E-04	2.50E-05	6.70E-05	8.90E-04
	DHADBCAADAAAAC			1.60E+06	2.90E+04	2.20E+04	1.20E-02	4.50E-02	4.10E-03	1.30E-01	2.80E-02	2.70E-04	3.00E-03	3.30E-03	2.40E-02
60	DHADBCAADABAAD	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.30E-01	2.40E-01	3.20E-02	1.40E-03	8.40E-04	8.70E-05	2.50E-04	2.50E-03
	DHADBCAADABAAD			1.60E+06	2.90E+04	2.20E+04	1.20E-02	4.70E-02	1.80E-02	1.50E-01	6.60E-02	1.20E-03	7.10E-03	8.90E-03	5.50E-02
65	DHADBCAADABAAC	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.70E-01	2.80E-01	3.20E-02	1.40E-03	8.40E-04	8.70E-05	2.50E-04	2.50E-03
	DHADBCAADABAAC			1.60E+06	2.90E+04	2.20E+04	1.20E-02	4.50E-02	1.80E-02	1.80E-01	7.80E-02	1.40E-03	8.60E-03	1.10E-02	6.50E-02
67	DHADBCABDABAAD	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.20E-01	2.30E-01	1.60E-02	5.60E-04	3.50E-04	2.60E-05	6.90E-05	9.00E-04
	DHADBCABDABAAD			1.60E+06	2.90E+04	2.20E+04	1.10E-02	8.10E-02	5.70E-02	1.30E-01	3.70E-02	5.30E-04	3.20E-03	2.30E-03	3.00E-02

* A listing of source terms for all bins is available on computer media

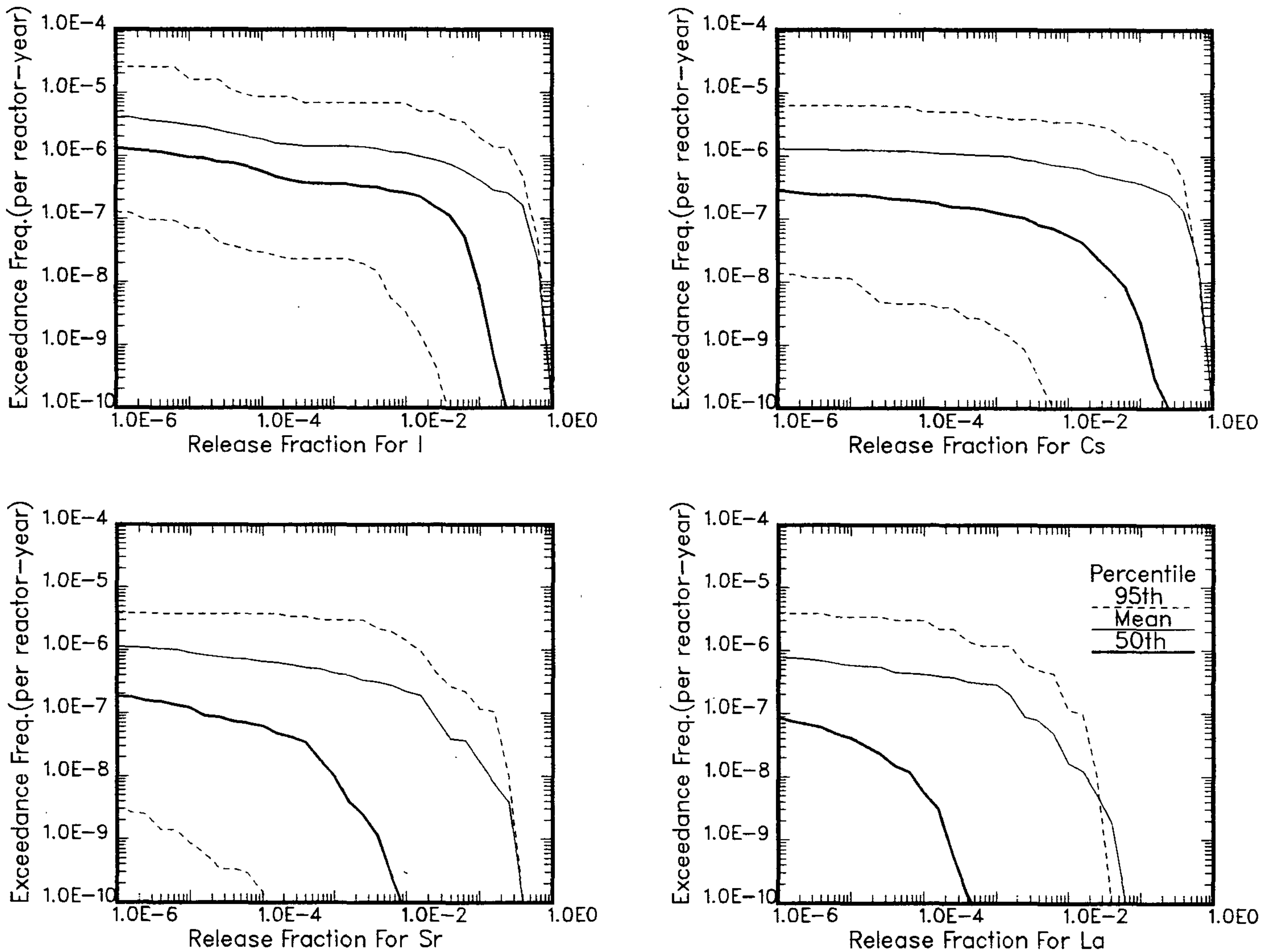


Figure 3.3-1. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 1: Slow SBO)

3.3.1.2 Results for PDS Group 2: Fast SBO. This PDS group consists of accidents in which all ac power is lost in the plant and the STD AFWS fails at, or shortly after, the start of the accident. As discussed in Section 2.5.1.2, the fast station blackout (SBO) PDS group consists of only one PDS, TRRR-RSR. As in the slow SBO PDS group, if offsite electrical power is recovered for a fast SBO accident before the vessel fails, it may be possible to arrest the CD process and avoid VB. Table 2.5-2 lists the five most probable APBs for the fast SBO PDS group, the five most probable APBs that have VB, and the five most probable APBs that have VB and early containment failure. Table 3.3-2 lists the mean source terms for these same APBs. The source term consists of the release fractions, the release height and energy, and the times associated with the release.

For the fast SBO PDS group, the four most probable bins have very low source terms because there is no containment failure. Three of these four bins have no VB as well. Of the five most probable bins that have VB, the first and fourth listed have no containment failure, the second and third have late containment failure, and the fifth has containment failure at VB. As discussed previously, for no containment failure or late containment failure, the early release is zero, and the late release contains the entire amount estimated to pass to the atmosphere.

The five most probable fast SBO APBs with VB and early containment failure have lower conditional probabilities (see Table 2.5-2) but larger releases than the APBs without containment failure. The release fractions for the fast PDS group are slightly higher than for the slow PDS group, in part because the PDS frequencies are higher and also because there are slightly more early failures for the fast SBOs. Some of these APBs give rise to source terms in which the release fractions exceed 0.10, but Figure 3.3-2 shows that the mean frequencies at which release fractions of 0.10 are exceeded are quite low: 1×10^{-6} for iodine, 9×10^{-7} for cesium, 1×10^{-7} for strontium, and less than 10^{-10} for lanthanum.

Table 3.3-2
Mean Source Terms for Sequoyah
Internal Initiators (PDS Group 2: Fast SBO)

Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (W)	Release Start (s)	Release Duration (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
Five Most Probable Bins*															
1	GDCDFCDBDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDBDDBAAB			0.00E+00	4.70E+04	8.60E+04	4.20E-03	4.70E-05	4.80E-10	2.30E-10	4.20E-11	1.10E-11	2.00E-12	7.60E-12	4.70E-11
2	GDCDFCDADFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDADDBAAB			0.00E+00	4.70E+04	8.60E+04	3.90E-03	4.10E-05	4.20E-10	2.30E-10	7.00E-11	1.10E-11	4.00E-12	1.80E-11	7.60E-11
3	GFADBCABDFBAAC	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GFADBCABDDBAAC			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.50E-04	1.70E-09	1.00E-09	2.90E-10	2.00E-11	2.50E-11	2.60E-11	2.60E-10
4	GDCCFCDADFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCCFCDADDAAAB			0.00E+00	4.70E+04	8.60E+04	3.90E-03	2.80E-05	1.90E-10	6.30E-11	4.00E-12	1.10E-12	2.00E-13	6.90E-13	5.30E-12
5	EEADBCAADABAAC	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	EEADBCAADABAAC			3.50E+06	4.70E+04	2.00E+02	1.00E+00	5.00E-02	1.40E-02	5.80E-03	3.00E-04	7.90E-06	2.70E-05	4.30E-05	2.50E-04
Five Most Probable Bins with VB*															
3	GFADBCABDFBAAC	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GFADBCABDDBAAC			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.50E-04	1.70E-09	1.00E-09	2.90E-10	2.00E-11	2.50E-11	2.60E-11	2.60E-10
5	EEADBCAADABAAC	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	EEADBCAADABAAC			3.50E+06	4.70E+04	2.00E+02	1.00E+00	5.00E-02	1.40E-02	5.80E-03	3.00E-04	7.90E-06	2.70E-05	4.30E-05	2.50E-04
8	EEADBCABDABAAC	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	EEADBCABDABAAC			3.50E+06	4.70E+04	2.00E+02	1.00E+00	4.80E-02	1.70E-02	5.80E-03	1.70E-04	7.50E-06	2.20E-05	2.10E-05	1.40E-04
12	GFADBCAADFBAAC	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GFADBCAADDBAAC			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.80E-04	1.40E-09	9.70E-10	3.40E-10	2.40E-11	2.60E-11	6.40E-11	3.20E-10
13	DHADBCABDABAAC	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.50E-01	2.60E-01	1.60E-02	5.60E-04	3.50E-04	2.60E-05	6.90E-05	9.00E-04
	DHADBCABDABAAC			1.60E+06	2.90E+04	2.20E+04	1.10E-02	7.90E-02	5.70E-02	1.50E-01	4.30E-02	6.00E-04	3.70E-03	2.60E-03	3.60E-02
Five Most Probable Bins with VB and Early CF*															
13	DHADBCABDABAAC	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.50E-01	2.60E-01	1.60E-02	5.60E-04	3.50E-04	2.60E-05	6.90E-05	9.00E-04
	DHADBCABDABAAC			1.60E+06	2.90E+04	2.20E+04	1.10E-02	7.90E-02	5.70E-02	1.50E-01	4.30E-02	6.00E-04	3.70E-03	2.60E-03	3.60E-02
26	DHADBCAADABAAC	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.70E-01	2.80E-01	3.20E-02	1.40E-03	8.40E-04	8.70E-05	2.50E-04	2.50E-03
	DHADBCAADABAAC			1.60E+06	2.90E+04	2.20E+04	1.20E-02	4.50E-02	1.80E-02	1.80E-01	7.80E-02	1.40E-03	8.60E-03	1.10E-02	6.50E-02
31	DHADBCABDABAAD	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.20E-01	2.30E-01	1.60E-02	5.60E-04	3.50E-04	2.60E-05	6.90E-05	9.00E-04
	DHADBCABDABAAD			1.60E+06	2.90E+04	2.20E+04	1.10E-02	8.10E-02	5.70E-02	1.30E-01	3.70E-02	5.30E-04	3.20E-03	2.30E-03	3.00E-02
46	DHADBCAADABAAD	2.20E+04	1.00E+01	2.80E+07	2.80E+04	2.00E+02	9.90E-01	2.30E-01	2.40E-01	3.20E-02	1.40E-03	8.40E-04	8.70E-05	2.50E-04	2.50E-03
	DHADBCAADABAAD			1.60E+06	2.90E+04	2.20E+04	1.20E-02	4.70E-02	1.80E-02	1.50E-01	6.60E-02	1.20E-03	7.10E-03	8.90E-03	5.50E-02
53	DFABACABBCAACC	2.20E+04	1.00E+01	5.20E+05	2.80E+04	1.00E+03	7.40E-01	2.50E-02	2.90E-02	6.40E-03	2.30E-04	8.00E-04	2.10E-04	2.10E-04	3.30E-04
	DFABACABBCAACC			1.60E+05	2.90E+04	2.20E+04	2.60E-01	4.80E-02	1.00E-02	2.70E-03	9.20E-05	2.70E-04	7.00E-05	7.00E-05	1.20E-04

* A listing of source terms for all bins is available on computer media

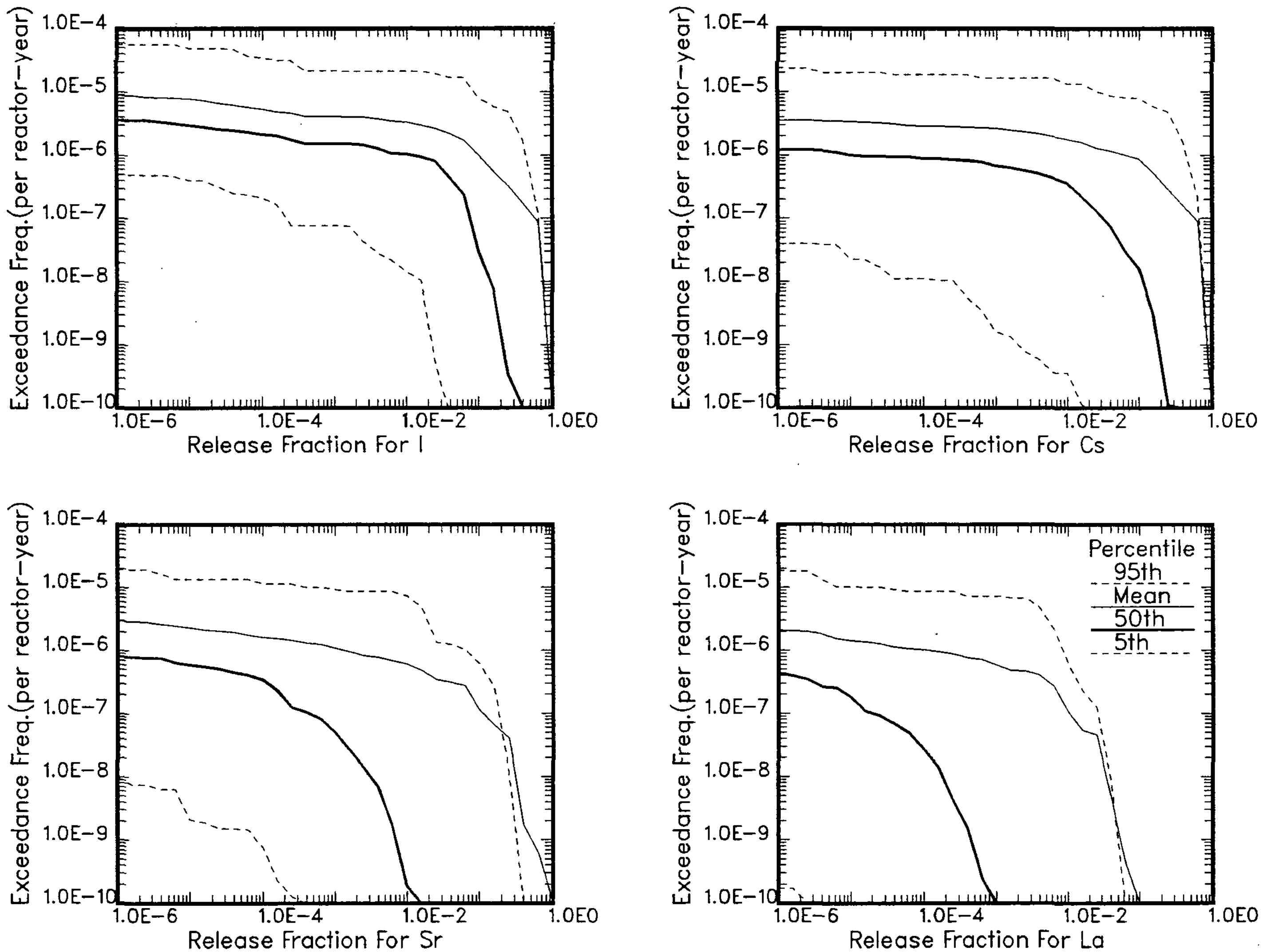


Figure 3.3-2. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 2: Fast SBO)

3.3.1.3 Results for PDS Group 3: LOCAs. This PDS group consists of accidents initiated by a break in the RCS pressure boundary, as discussed in Section 2.5.1.3. The breaks are of all (A, S_1, S_2, and S_3) sizes. These PDSs result in core damage because one or more emergency core cooling system (ECCS) required to respond does not operate. The PDSs in this group are listed in Table 2.2-2. Five of the 13 PDSs have the LPIS operating but not injecting at UTAF, so the arrest of core damage before vessel failure is possible as discussed in Section 2.5.1.3. Even though the containment does not fail in these core damage arrest cases, design basis leakage results in small but nonzero releases.

Table 2.5-3 lists the five most probable APBs for this PDS group, the five most probable APBs that have VB, and the five most probable APBs that have VB and early containment failure. Table 3.3-3 lists the mean source terms for these same APBs. The source term consists of the release fractions, the release height and energy, and the times of the release. The release fractions give the early (RCS) and late (CCI) releases as fractions of the core inventory at the start of the accident. However, when there is no containment failure, or late containment failure, SEQSOR sets the early release to zero and places the entire release into the late release portion.

The five most probable APBs for PDS Group 3 did not have containment failure or VB, and the releases for these APBs are extremely small. The four most probable APBs that have VB had long-term overpressure in the very late period. The releases for these APBs are larger than those with no containment failure, but are still quite small.

As with the APBs for PDS Groups 1 and 2 that have VB and containment failure at VB, some of these APBs give rise to source terms in which the mean release fractions for iodine and cesium exceed 0.10. Figure 3.3-3 summarizes the release fraction CCDFs and shows that the frequency at which iodine and cesium release fractions of 0.10 are exceeded are quite low, despite the high frequency of occurrence of this PDS group. Mitigation mechanisms for the releases (sprays, cavity water, etc.) are very likely for this PDS group. The frequency of occurrence of a large release is commensurate to that for PDS Group 1; for this PDS group, the mean exceedance frequencies for a release fraction of 0.1 are 4×10^{-7}, 3×10^{-7}, 5×10^{-9}, and $<10^{-10}$ for iodine, cesium, strontium, and lanthanum, respectively.

Table 3.3-3
Mean Source Terms for Sequoyah
Internal Initiators (PDS Group 3: Loss-of-Coolant Accidents)

Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (W)	Release Start (s)	Release Duration (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
Five Most Probable Bins*															
1	GDCDFCDADFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDADDBAAB			0.00E+00	4.70E+04	8.60E+04	3.90E-03	4.10E-05	4.20E-10	2.30E-10	7.00E-11	1.10E-11	4.00E-12	1.80E-11	7.60E-11
2	GDCDFCDBDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDBDDBAAB			0.00E+00	4.70E+04	8.60E+04	4.20E-03	4.70E-05	4.80E-10	2.30E-10	4.20E-11	1.10E-11	2.00E-12	7.60E-12	4.70E-11
3	GDCDFCDADFBAAA	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDADDBAAA			0.00E+00	4.70E+04	8.60E+04	3.90E-03	4.10E-05	4.20E-10	2.30E-10	7.00E-11	1.10E-11	4.00E-12	1.80E-11	7.60E-11
4	GDCDFCDBDFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDBDDAAAB			0.00E+00	4.70E+04	8.60E+04	4.10E-03	4.90E-05	5.60E-10	3.00E-10	8.50E-11	1.80E-11	3.70E-12	1.40E-11	9.30E-11
5	GDCDFCDADFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDADDAAAB			0.00E+00	4.70E+04	8.60E+04	4.00E-03	4.60E-05	7.50E-10	4.80E-10	2.00E-10	2.90E-11	1.10E-11	5.00E-11	2.10E-10
Five Most Probable Bins with VB*															
7	FHDDBCAADBBAAB	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	FHDDBCAADBBAAB			3.50E+06	1.30E+05	2.00E+02	1.00E+00	3.90E-02	5.20E-06	2.60E-06	1.10E-07	5.60E-09	1.40E-08	2.00E-08	9.50E-08
8	FHDDBCABDBBAAB	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	FHDDBCABDBBAAB			3.50E+06	1.30E+05	2.00E+02	1.00E+00	3.10E-02	7.80E-06	4.10E-06	1.80E-07	6.00E-09	2.70E-08	2.80E-08	1.50E-07
9	FHDDBCAADABAAB	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	FHDDBCAADABAAB			3.50E+06	1.30E+05	2.00E+02	1.00E+00	3.40E-02	4.80E-06	1.70E-06	1.80E-07	1.20E-08	2.00E-08	2.30E-08	1.70E-07
10	FHDDBCABDABAAB	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	FHDDBCABDABAAB			3.50E+06	1.30E+05	2.00E+02	1.00E+00	3.20E-02	6.80E-06	3.10E-06	2.00E-07	9.10E-09	2.40E-08	2.70E-08	1.70E-07
14	GDDDBCAADFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDDDBCAADDBAAB			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.80E-04	1.60E-09	7.90E-10	2.40E-10	2.60E-11	1.80E-11	4.80E-11	2.40E-10
Five Most Probable Bins with VB and Early CF*															
132	DACBACDBBAAAAB	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	1.00E+00	5.50E-02	5.00E-02	3.10E-02	6.30E-03	1.30E-02	3.00E-03	3.30E-03	8.90E-03
	DACBACDBBAAAAB			1.60E+06	1.00E+06	1.00E+06	0.00E+00	1.10E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
156	DACCACDAAAAAAB	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	9.70E-01	6.80E-02	9.80E-02	7.00E-02	4.40E-02	4.80E-02	1.20E-02	1.20E-02	5.50E-02
	DACCACDAAAAAAB			1.60E+06	1.00E+06	1.00E+06	2.90E-02	5.00E-03	8.90E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
164	DACBACDBBAAAAA	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	1.00E+00	5.50E-02	5.00E-02	3.10E-02	6.30E-03	1.30E-02	3.00E-03	3.30E-03	8.90E-03
	DACBACDBBAAAAA			1.60E+06	1.00E+06	1.00E+06	0.00E+00	1.10E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
165	DACBACDABAAAAB	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	7.60E-01	8.60E-02	9.40E-02	5.20E-04	1.20E-05	1.20E-05	1.20E-05	1.20E-05	1.20E-05
	DACBACDABAAAAB			1.60E+06	1.00E+06	1.00E+06	2.40E-01	1.90E-02	9.00E-03	6.50E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
193	DACCACDAAAAAAA	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	9.70E-01	6.80E-02	9.80E-02	7.00E-02	4.40E-02	4.80E-02	1.20E-02	1.20E-02	5.50E-02
	DACCACDAAAAAAA			1.60E+06	1.00E+06	1.00E+06	2.90E-02	5.00E-03	8.90E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

* A listing of source terms for all bins is available on computer media

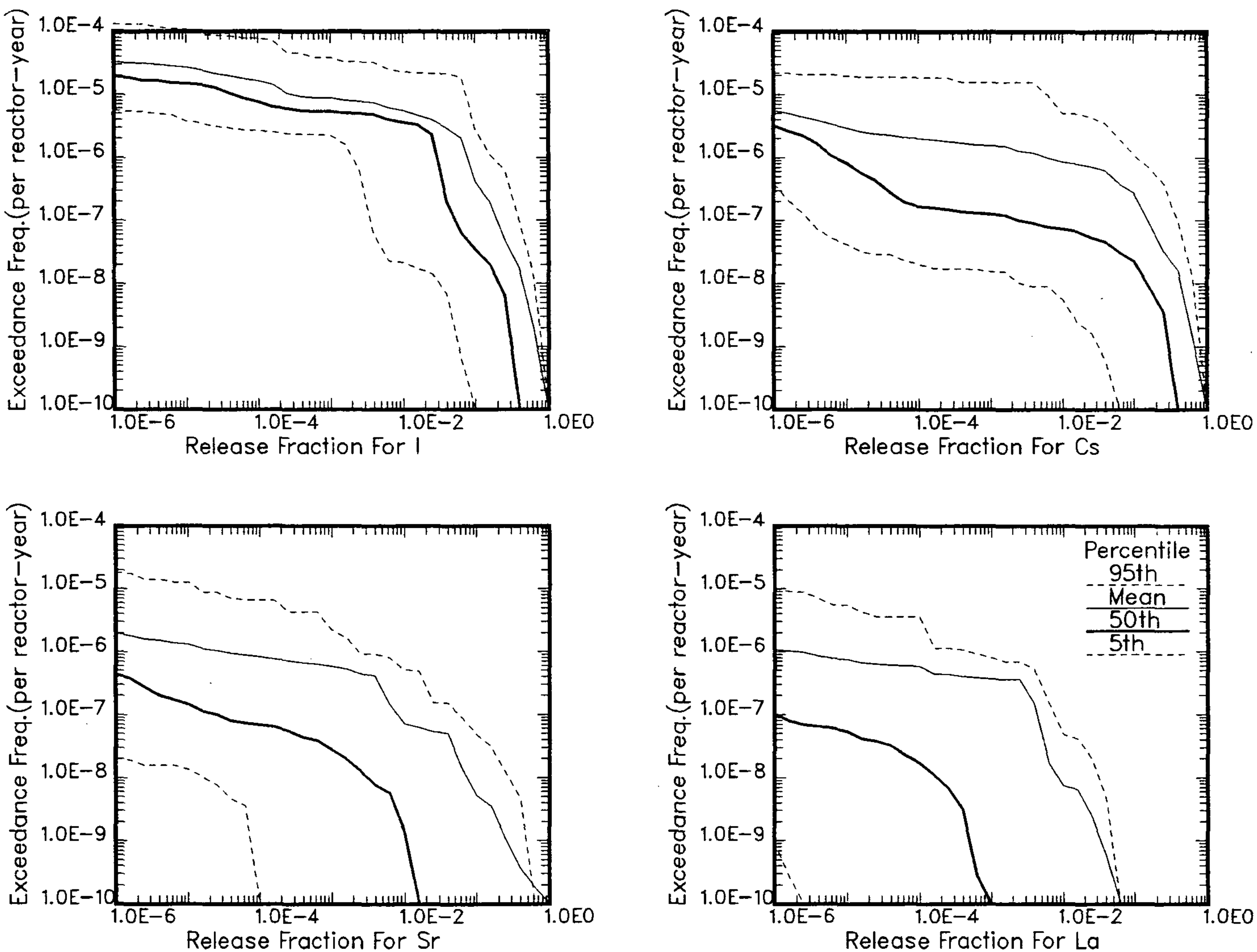


Figure 3.3-3. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 3: Loss-of-Coolant-Accidents)

3.3.1.4 Results for PDS Group 4: Event V. As discussed in Section 2.5.1.4, this PDS group consists of accidents in which the check valves between the RCS and the LPIS fail. Failure of the low pressure piping produces a direct path from the RCS to the auxiliary building, bypassing the containment, and failing the LPIS as well. It is expected that there is a considerable probability (0.80) that the area fire sprays in the auxiliary building will scrub the releases. These sprays can remove and retain a significant portion of the release. When the release is scrubbed, the accident is termed V-Wet, and when there is no pool, it is termed V-Dry. There is no possibility of avoiding VB or CCI in this PDS group. Due to the size of the containment bypass, containment failure is not of much interest.

Table 2.5-4 lists the 10 most probable APBs for the V PDS group, and Table 3.3-4 lists the mean source terms for these same APBs. The source term consists of the release fractions, the release height and energy, and the times associated with the release. The eight most probable bins are V-Wet and the next two are V-Dry. (The probability of the break location being under water is between 0.60 and 1.0.) As expected, the V-Wet release fractions are considerably lower than the V-Dry release fractions.

The release fraction CCDF summary curves in Figure 3.3-4 show that the frequency at which iodine and cesium release fractions of 0.10 are exceeded is below 10^{-6}/yr. For this PDS group, the mean exceedance frequencies for a release fraction of 0.1 are 4×10^{-7}, 3×10^{-7}, 1×10^{-7}, and $<10^{-10}$ for iodine, cesium, strontium, and lanthanum, respectively. Although the frequency of occurrence of this accident is low because it bypasses the containment, the releases are likely to be substantial when this accident occurs. This is indicated in Figure 3.3-4 by a pronounced drop (threshold effect) in the curves at values of high release fractions.

Table 3.3-4
Mean Source Terms for Sequoyah
Internal Initiators (PDS Group 4: Event V)

Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (W)	Release Start (s)	Release Duration (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
Ten Most Probable Bins*															
1	BHADBCAADEAAAB	1.30E+03	0.00E+00	1.90E+06	3.70E+03	1.80E+03	9.90E-01	7.00E-02	7.20E-02	9.00E-03	2.80E-03	5.80E-04	2.40E-04	1.30E-03	3.10E-03
	BHADBCAADDAAAB			1.70E+05	1.00E+04	2.20E+04	1.10E-02	4.40E-02	4.90E-03	3.60E-02	1.60E-02	7.20E-04	1.10E-03	2.60E-03	1.40E-02
2	BHADBCAADEAAAA	1.30E+03	0.00E+00	1.90E+06	3.70E+03	1.80E+03	9.90E-01	7.00E-02	7.20E-02	9.00E-03	2.80E-03	5.80E-04	2.40E-04	1.30E-03	3.10E-03
	BHADBCAADDAAAA			1.70E+05	1.00E+04	2.20E+04	1.10E-02	4.40E-02	4.90E-03	3.60E-02	1.60E-02	7.20E-04	1.10E-03	2.60E-03	1.40E-02
3	BHADBCABDEAAAB	1.30E+03	0.00E+00	1.90E+06	3.70E+03	1.80E+03	9.90E-01	6.10E-02	6.20E-02	1.30E-02	3.50E-03	8.30E-04	2.30E-04	1.00E-03	3.80E-03
	BHADBCABDDAAAB			1.70E+05	1.00E+04	2.20E+04	8.00E-03	4.60E-02	7.60E-03	2.60E-02	6.00E-03	1.30E-04	7.70E-04	5.50E-04	4.90E-03
4	BHADBCAADDAAAB	1.30E+03	0.00E+00	1.90E+06	3.70E+03	1.80E+03	9.90E-01	7.00E-02	7.20E-02	9.00E-03	2.80E-03	5.80E-04	2.40E-04	1.30E-03	3.10E-03
	BHADBCAADCAAAB			1.70E+05	1.00E+04	2.20E+04	1.10E-02	4.40E-02	4.90E-03	3.60E-02	1.60E-02	7.20E-04	1.10E-03	2.60E-03	1.40E-02
5	BHADBCABDEAAAA	1.30E+03	0.00E+00	1.90E+06	3.70E+03	1.80E+03	9.90E-01	6.10E-02	6.20E-02	1.30E-02	3.50E-03	8.30E-04	2.30E-04	1.00E-03	3.80E-03
	BHADBCABDDAAAA			1.70E+05	1.00E+04	2.20E+04	8.00E-03	4.60E-02	7.60E-03	2.60E-02	6.00E-03	1.30E-04	7.70E-04	5.50E-04	4.90E-03
6	BHADBCAADDAAAA	1.30E+03	0.00E+00	1.90E+06	3.70E+03	1.80E+03	9.90E-01	7.00E-02	7.20E-02	9.00E-03	2.80E-03	5.80E-04	2.40E-04	1.30E-03	3.10E-03
	BHADBCAADCAAAA			1.70E+05	1.00E+04	2.20E+04	1.10E-02	4.40E-02	4.90E-03	3.60E-02	1.60E-02	7.20E-04	1.10E-03	2.60E-03	1.40E-02
7	BHADBCABDDAAAA	1.30E+03	0.00E+00	1.90E+06	3.70E+03	1.80E+03	9.90E-01	6.10E-02	6.20E-02	1.30E-02	3.50E-03	8.30E-04	2.30E-04	1.00E-03	3.80E-03
	BHADBCABDCAAAA			1.70E+05	1.00E+04	2.20E+04	8.00E-03	4.60E-02	7.60E-03	2.60E-02	6.00E-03	1.30E-04	7.70E-04	5.50E-04	4.90E-03
8	BHADBCABDDAAAA	1.30E+03	0.00E+00	1.90E+06	3.70E+03	1.80E+03	9.90E-01	6.10E-02	6.20E-02	1.30E-02	3.50E-03	8.30E-04	2.30E-04	1.00E-03	3.80E-03
	BHADBCABDCAAAA			1.70E+05	1.00E+04	2.20E+04	8.00E-03	4.60E-02	7.60E-03	2.60E-02	6.00E-03	1.30E-04	7.70E-04	5.50E-04	4.90E-03
9	AHADBCAADEAAAB	1.30E+03	0.00E+00	3.70E+06	3.70E+03	1.80E+03	9.90E-01	3.70E-01	3.80E-01	6.10E-02	2.10E-02	3.90E-03	1.50E-03	8.20E-03	2.30E-02
	AHADBCAADDAAAB			1.70E+05	1.00E+04	2.20E+04	1.10E-02	2.70E-02	4.90E-03	1.70E-01	7.10E-02	2.40E-03	5.70E-03	1.10E-02	6.10E-02
10	AHADBCABDEAAAB	1.30E+03	0.00E+00	3.70E+06	3.70E+03	1.80E+03	9.90E-01	3.10E-01	3.10E-01	5.60E-02	1.30E-02	3.30E-03	7.90E-04	3.30E-03	1.50E-02
	AHADBCABDDAAAB			1.70E+05	1.00E+04	2.20E+04	8.00E-03	3.50E-02	7.60E-03	1.20E-01	4.10E-02	5.30E-04	3.90E-03	2.50E-03	3.30E-02

* A listing of source terms for all bins is available on computer media

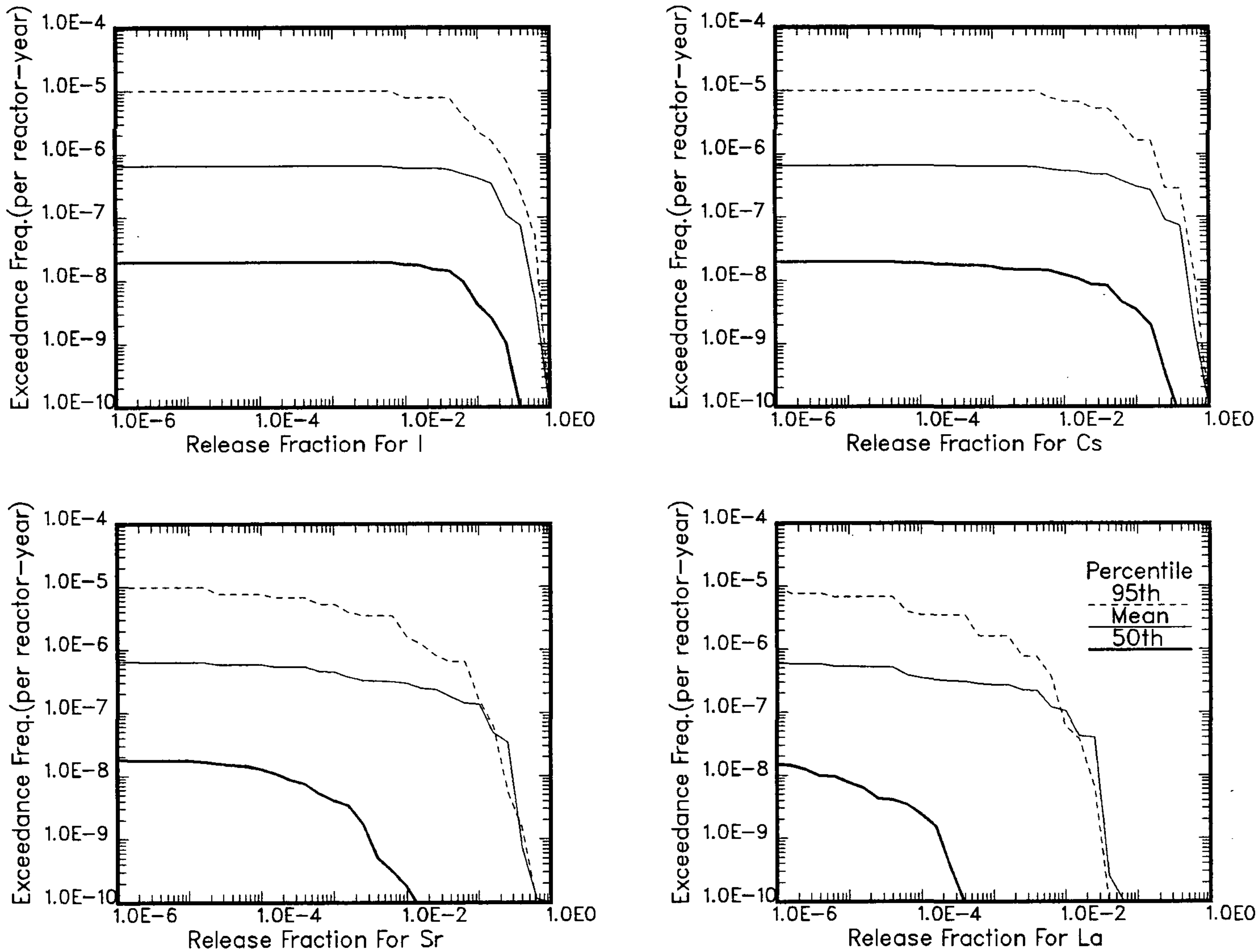


Figure 3.3-4. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 4: Event V)

3.3.1.5 Results for PDS Group 5: Transients. This PDS group consists of accidents in which the RCS is intact but there is no way to remove heat from the core (see Section 2.5.1.5). The AFWS fails at the start of the accident; bleed and feed is ineffective. In PDS TBYY-YNY, high pressure injection system (HPIS) and LPIS are available, but the operators cannot open the PORVs from the control room or have failed to do so before the onset of core damage. PDS TBYY-YNY is the dominant sequence for this PDS group. In the other PDS in this group, TINY-NNY, both HPIS and LPIS are failed. The probability of a T-I failure of the RCS pressure boundary is quite high, about 0.90. Since for the dominant PDS, the HPIS and LPIS are operating at the onset of core damage, the probability of arresting the CD process and avoiding VB is also high, about 0.80.

Table 2.5-5 lists the five most probable APBs for the PDS group, the five most probable APBs that have VB, and the five most probable APBs that have VB and early containment failure. Table 3.3-5 lists the mean source terms for these same 15 APBs. The five most probable bins and the five most probable bins that have VB all have no containment failure, and their release fractions are so low as to be negligible in an overall risk context.

The five most probable transient APBs with VB and early containment failure have lower conditional probabilities (see Table 2.5-5) but larger releases than the APBs without containment failure. Note that for these five APBs, CCI does not occur, and the late release fractions are essentially zero for the source term groups strontium, ruthenium, lanthanum, cerium, and barium. Figure 3.3-5 shows that the mean frequencies at which release fractions of 0.10 are exceeded is very low: 1×10^{-8} for iodine and cesium, 2×10^{-10} for strontium, and less than 10^{-10} for lanthanum.

Table 3.3-5
Mean Source Terms for Sequoyah
Internal Initiators (PDS Group 5: Transients)

Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (W)	Release Start (s)	Release Duration (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
Five Most Probable Bins*															
1	GDCDFCDBDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDBDDBAAB			0.00E+00	4.70E+04	8.60E+04	4.20E-03	4.70E-05	4.80E-10	2.30E-10	4.20E-11	1.10E-11	2.00E-12	7.60E-12	4.70E-11
2	GDCDFCDADFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDADDBAAB			0.00E+00	4.70E+04	8.60E+04	3.90E-03	4.10E-05	4.20E-10	2.30E-10	7.00E-11	1.10E-11	4.00E-12	1.80E-11	7.60E-11
3	GDCDFCDBDFBCCB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDBDDBCCB			0.00E+00	4.70E+04	8.60E+04	4.20E-03	4.70E-05	1.10E-09	5.30E-10	8.80E-11	2.40E-11	4.20E-12	1.60E-11	1.00E-10
4	GDCDFCDBDFBAAA	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDBDDBAAA			0.00E+00	4.70E+04	8.60E+04	4.20E-03	4.70E-05	4.80E-10	2.30E-10	4.20E-11	1.10E-11	2.00E-12	7.60E-12	4.70E-11
5	GDCCFCDBDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCCFCDBDDBAAB			0.00E+00	4.70E+04	8.60E+04	4.30E-03	3.80E-05	4.50E-10	2.00E-10	1.30E-11	6.30E-12	6.60E-13	2.00E-12	1.80E-11
Five Most Probable Bins with VB*															
12	GDDAACAADFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDDAACAADDAAAB			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.20E-04	9.80E-10	4.60E-10	1.60E-10	9.80E-12	1.50E-11	2.50E-11	1.50E-10
17	GDDDBCABDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDDDBCABDDBAAB			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.60E-04	1.70E-09	8.70E-10	1.80E-10	2.50E-11	1.30E-11	2.60E-11	1.80E-10
20	GDCAACDACFAAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCAACDACDAAAB			0.00E+00	4.70E+04	8.60E+04	5.00E-03	4.80E-05	4.60E-10	1.30E-10	3.10E-12	1.80E-12	1.80E-13	5.00E-13	5.60E-12
23	GDCAACDACFAAAA	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCAACDACDAAAA			0.00E+00	4.70E+04	8.60E+04	5.00E-03	4.80E-05	4.60E-10	1.30E-10	3.10E-12	1.80E-12	1.80E-13	5.00E-13	5.60E-12
25	GDCDBCDBDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDBCDBDDBAAB			0.00E+00	4.70E+04	8.60E+04	5.00E-03	9.50E-05	1.30E-09	6.20E-10	8.80E-11	2.30E-11	4.10E-12	1.60E-11	9.90E-11
Five Most Probable Bins with VB and Early CF*															
79	DACABCDADAAAAB	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	8.30E-01	8.40E-02	7.60E-02	3.80E-02	1.00E-03	8.40E-04	6.10E-05	1.40E-04	2.00E-03
	DACABCDADAAAAB			1.60E+06	1.00E+06	1.00E+06	1.70E-01	3.40E-02	1.20E-02	6.00E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
89	DACAACDABAAAAB	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	9.50E-01	2.30E-01	2.30E-01	7.60E-02	1.80E-03	3.20E-03	5.10E-04	5.40E-04	2.90E-03
	DACAACDABAAAAB			1.60E+06	1.00E+06	1.00E+06	5.10E-02	4.20E-02	1.90E-02	8.60E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
91	DACABCDADAAAAA	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	8.30E-01	8.40E-02	7.60E-02	3.80E-02	1.00E-03	8.40E-04	6.10E-05	1.40E-04	2.00E-03
	DACABCDADAAAAA			1.60E+06	1.00E+06	1.00E+06	1.70E-01	3.40E-02	1.20E-02	6.00E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
95	DACAACDAAAAAAB	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	9.80E-01	5.50E-02	8.40E-02	6.90E-02	4.80E-02	5.40E-02	1.60E-02	1.60E-02	6.00E-02
	DACAACDAAAAAAB			1.60E+06	1.00E+06	1.00E+06	2.20E-02	2.80E-02	1.50E-02	1.40E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
104	DACAACDABAAAAA	2.20E+04	1.00E+01	2.80E+06	2.80E+04	2.00E+02	9.50E-01	2.30E-01	2.30E-01	7.60E-02	1.80E-03	3.20E-03	5.10E-04	5.40E-04	2.90E-03
	DACAACDABAAAAA			1.60E+06	1.00E+06	1.00E+06	5.10E-02	4.20E-02	1.90E-02	8.60E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

* A listing of source terms for all bins is available on computer media

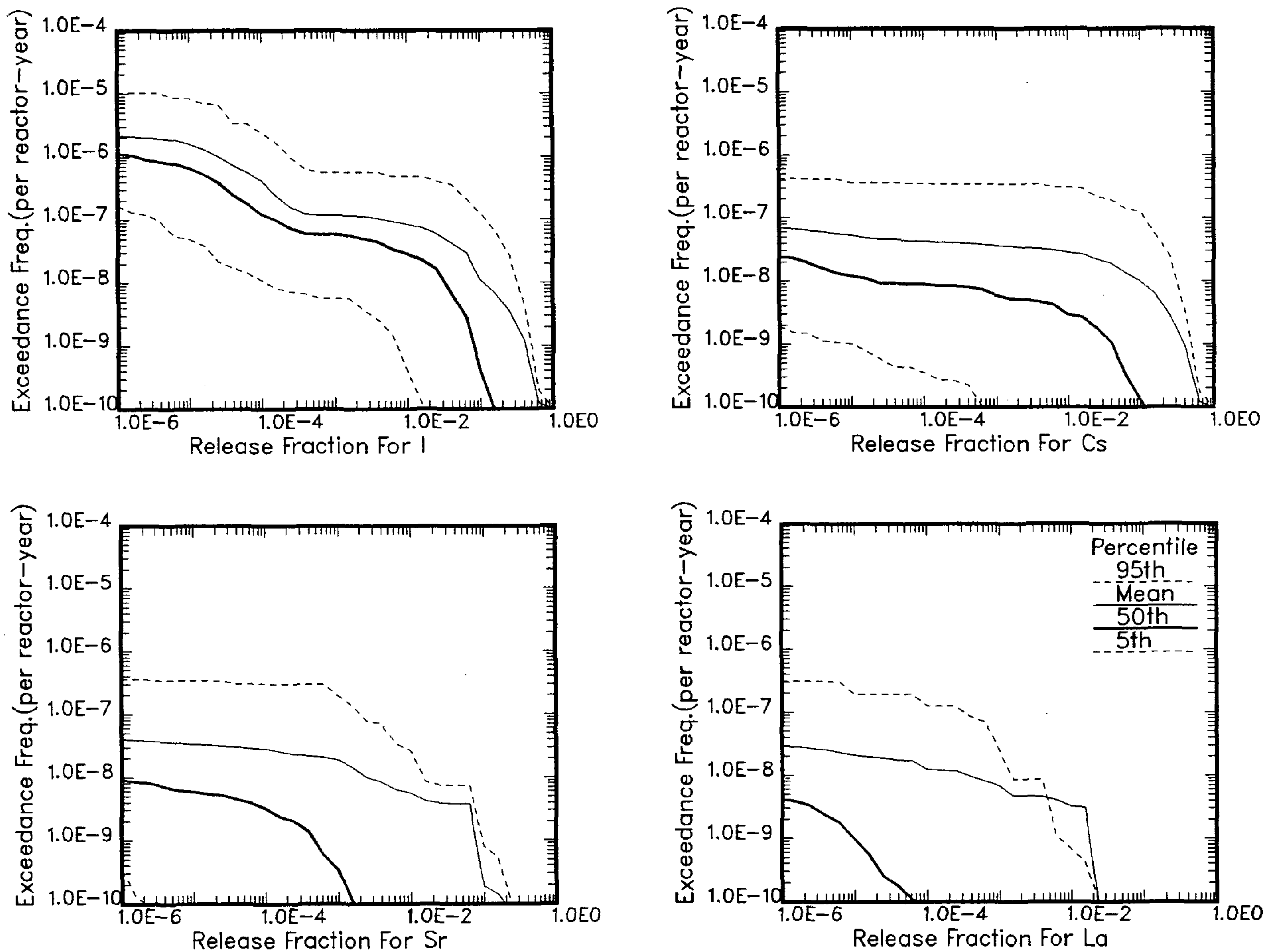


Figure 3.3-5. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 5: Transients)

3.3.1.6 Results for PDS Group 6: ATWS. This PDS group consists of accidents in which automatic control rod insertion fails to bring the nuclear reaction under control. The discussion in Section 2.5.1.6 points out that this PDS group consists of three PDSs, one with the RCS intact at UTAF, one with an S_3 break, and one with an SGTR. In all three situations, the PORVs will be open at UTAF due to the rate of steam generation in the core. The LPIS is operating but not injecting in the RCS-intact and SGTR PDSs. A T-I break in the RCS, however, will allow the LPIS to inject successfully. The water from the RWST injected by the LPIS contains enough boron to shut down the reaction should the core be in a configuration where continued reaction is possible.

Table 2.5-6 lists the 10 most probable APBs for the PDS group and the five most probable APBs that have VB and early containment failure or bypass, and Table 3.3-6 lists the source terms calculated for these same 15 APBs. Seven of the 10 most probable bins have neither failure nor bypass of the containment and thus have very low releases. The fourth and sixth most probable bins have bypass of the containment (SGTR) and therefore have substantial releases although they have no VB due to the operation of the LPIS throughout the accident. Even in the absence of VB, SEQSOR may calculate significant releases in these SGTR accidents since the CD may not be arrested until it is quite well advanced. By this time, a substantial portion of the fission products may have been released from the core. The tenth most probable APB has very late containment failure by BMT and the releases are larger than those without containment failure, but still quite small. The small source term is because failure occurs after many days, and the release point is below ground.

The five most probable APBs with VB and early containment failure or bypass all have SGTR and no containment failure. Whether the vessel fails or not does not have a large effect on the computed release fractions. Figure 3.3-6 shows that the mean frequencies at which release fractions of 0.10 are exceeded are fairly low for this PDS group in spite of the contribution from the SGTR initiators: 1×10^{-7} for iodine and cesium, 1×10^{-9} for strontium, and less than 10^{-10} for lanthanum.

Table 3.3-6
Mean Source Terms for Sequoyah
Internal Initiators (PDS Group 6: Anticipated Transient Without Scram)

							Release Fractions								
Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (W)	Release Start (s)	Release Duration (s)	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
Ten Most Probable Bins*															
1	GDCDFCDBDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDBDDBAAB			0.00E+00	4.70E+04	8.60E+04	4.20E-03	4.70E-05	4.80E-10	2.30E-10	4.20E-11	1.10E-11	2.00E-12	7.60E-12	4.70E-11
2	GDDDBCABDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDDDBCABDDBAAB			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.60E-04	1.70E-09	8.70E-10	1.80E-10	2.50E-11	1.30E-11	2.60E-11	1.80E-10
3	GDCDFCDADFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDFCDADDBAAB			0.00E+00	4.70E+04	8.60E+04	3.90E-03	4.10E-05	4.20E-10	2.30E-10	7.00E-11	1.10E-11	4.00E-12	1.80E-11	7.60E-11
4	GDCDFADBDEBAAB	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.70E-01	1.80E-01	1.60E-01	1.40E-01	1.60E-02	5.00E-03	1.10E-03	5.20E-03	1.90E-02
	GDCDFADBDDBAAB			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.80E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
5	GDCDBCDBDFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDCDBCDBDDBAAB			0.00E+00	4.70E+04	8.60E+04	5.00E-03	9.50E-05	1.30E-09	6.20E-10	8.80E-11	2.30E-11	4.10E-12	1.60E-11	9.90E-11
6	GDDDBCAADFBAAB	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDDDBCAADDBAAB			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.80E-04	1.60E-09	7.90E-10	2.40E-10	2.60E-11	1.80E-11	4.80E-11	2.40E-10
7	GDCDFADADEBAAB	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.60E-01	1.70E-01	1.50E-01	1.20E-01	2.60E-02	5.20E-03	2.10E-03	1.20E-02	2.80E-02
	GDCDFADADDBAAB			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.60E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
8	GDDDBCABDFBAAA	2.20E+04	0.00E+00	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	GDDDBCABDDBAAA			0.00E+00	4.70E+04	8.60E+04	5.00E-03	1.60E-04	1.70E-09	8.80E-10	1.90E-10	2.50E-11	1.40E-11	2.70E-11	1.90E-10
9	GDCDFADADEBAAB	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.60E-01	1.70E-01	1.50E-01	1.20E-01	2.60E-02	5.20E-03	2.10E-03	1.20E-02	2.80E-02
	GDCDFADADDBAAB			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.60E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
10	FDDDBCABDDBAAB	2.20E+04	1.00E+01	0.00E+00	4.70E+04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
	FDDDBCABDCBAAB			6.50E+03	1.30E+05	1.10E+04	1.00E+00	3.20E-02	7.10E-08	4.00E-08	9.80E-09	2.80E-10	1.00E-09	1.10E-09	8.20E-09
Five Most Probable Bins with Bypass or VB and Early CF*															
4	GDCDFADBDEBAAB	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.70E-01	1.80E-01	1.60E-01	1.40E-01	1.60E-02	5.00E-03	1.10E-03	5.20E-03	1.90E-02
	GDCDFADBDDBAAB			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.80E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
7	GDCDFADADEBAAB	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.60E-01	1.70E-01	1.50E-01	1.20E-01	2.60E-02	5.20E-03	2.10E-03	1.20E-02	2.80E-02
	GDCDFADADDBAAB			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.60E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
15	GDCDFADBDEBAAA	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.70E-01	1.80E-01	1.60E-01	1.40E-01	1.60E-02	5.00E-03	1.10E-03	5.20E-03	1.90E-02
	GDCDFADBDDBAAA			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.80E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
21	GDCDFADADEBAAA	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.60E-01	1.70E-01	1.50E-01	1.20E-01	2.60E-02	5.20E-03	2.10E-03	1.20E-02	2.80E-02
	GDCDFADADDBAAA			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.60E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
40	GACDFADBDEBAAB	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.60E-01	1.80E-01	1.60E-01	1.40E-01	1.50E-02	4.70E-03	9.90E-04	4.80E-03	1.80E-02
	GACDFADBDDBAAB			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.90E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

* A listing of source terms for all bins is available on computer media

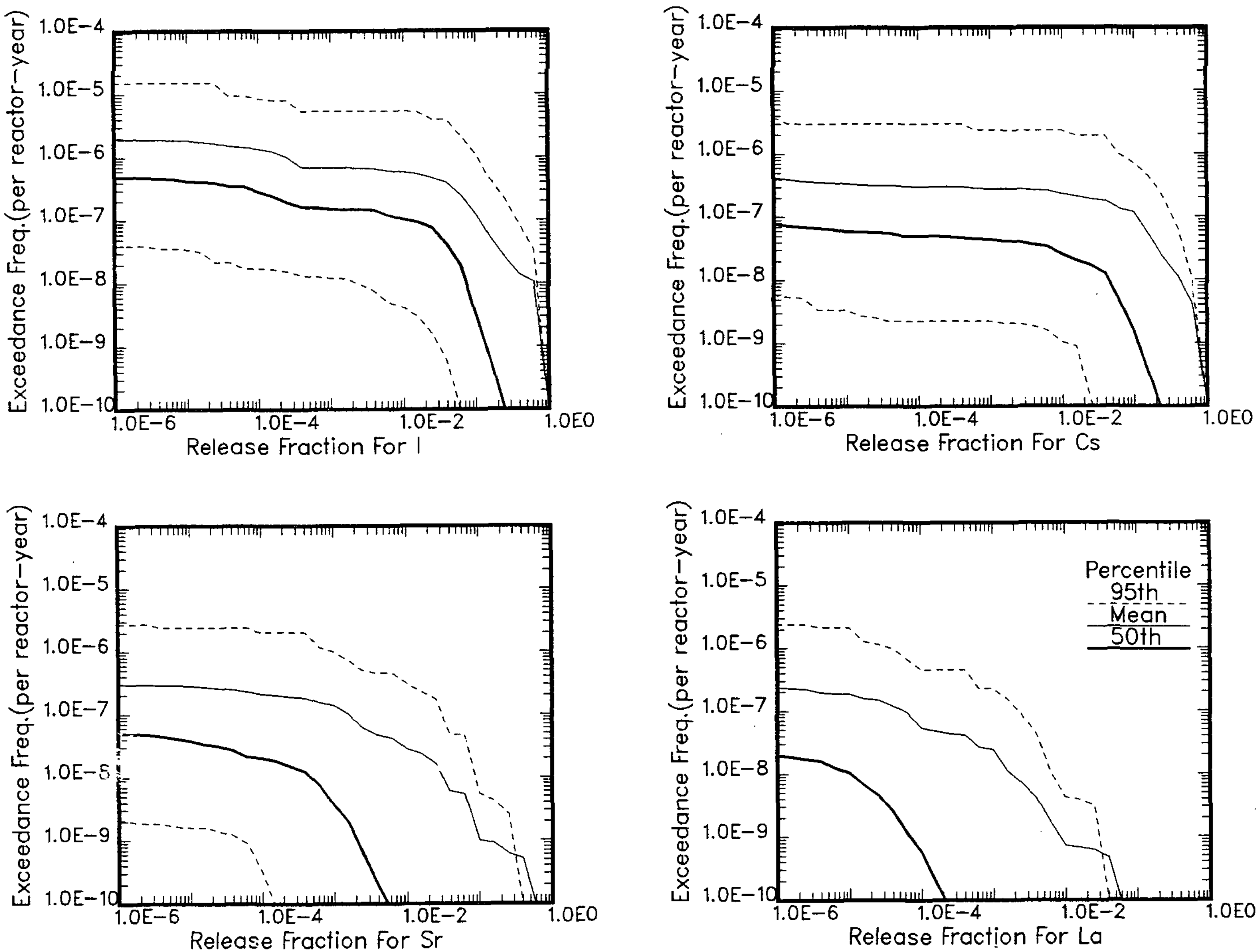


Figure 3.3-6. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 6: ATWS)

3.3.1.7 Results for PDS Group 7: SGTRs. As discussed in Section 2.5.1.7, this PDS group consists of accidents in which the initiating event is the rupture of an SG tube and the reaction is shut down successfully. In one of the PDSs in this group, the RCS is depressurized quickly using the three unaffected SGs according to procedures and the SRVs on the main steam lines from the affected SG do not stick open. These accidents, denoted "G" SGTRs, are indicated by "SGTR" in the SGTR column of Table 2.5-7. In the other PDS in the SGTR PDS group, the RCS is not depressurized in a timely fashion, and the SRVs on the main steam line from the affected SG stick open. These accidents, denoted "H" SGTRs, are indicated by "SRVO" in the SGTR column of Table 2.5-7. Since all the APBs for this PDS group have bypass of the containment, Table 2.5-7 lists the 15 most probable APBs. The "G" SGTR accidents occur less frequently than the "H" SGTR accidents; only four of the 15 most probable bins have the SRVs reclosing, and the other 11 bins have the secondary SRVs stuck open.

Table 3.3-7 lists the mean source terms for the same 15 APBs listed in Table 2.5-7. All the most probable APBs have fairly substantial release fractions. Note that the start of the release is about 14 h after the start of the accident for the "H" SGTRs. The evacuation warning time is estimated to be much earlier than this, so there is time for the evacuation to be completed. Thus, few early fatalities are to be expected even though the mean iodine release fractions are commonly higher than 0.10. The mean exceedance frequencies for release fractions of 0.10 are 1×10^{-6} for iodine and cesium, 3×10^{-8} for strontium, and less than 10^{-10} for lanthanum. As with PDS Group 4 (Event V), although the frequency of this accident is low because the containment is bypassed, the releases are likely to be substantial if the accident occurs. This is indicated in Figure 3.3-7 by the pronounced drop in the curves (threshold effect) at values of high release fractions, particularly for iodine and cesium.

Table 3.3-7
Mean Source Terms for Sequoyah
Internal Initiators (PDS Group 7: SGTRs)

Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (W)	Release Start (s)	Release Duration (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
Fifteen Most Probable Bins*															
1	GDCDFADADEBAAB	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.60E-01	1.70E-01	1.50E-01	1.20E-01	2.60E-02	5.20E-03	2.10E-03	1.20E-02	2.80E-02
	GDCDFADADDBAAB			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.60E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2	GDCDFADBDEBAAB	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.70E-01	1.80E-01	1.60E-01	1.40E-01	1.60E-02	5.00E-03	1.10E-03	5.20E-03	1.90E-02
	GDCDFADBDDBAAB			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.80E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
3	GHADBBABDEAAAB	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	7.40E-01	1.60E-01	1.40E-01	5.30E-02	4.20E-03	1.90E-03	2.00E-04	6.20E-04	5.70E-03
	GHADBBABDDAAAB			0.00E+00	5.20E+04	2.20E+04	3.70E-02	8.00E-03	6.80E-03	2.70E-03	2.10E-04	9.60E-05	1.00E-05	3.10E-05	2.90E-04
4	GHADBBAADEAAAB	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	7.60E-01	2.40E-01	2.10E-01	1.00E-01	4.70E-02	8.80E-03	3.90E-03	2.30E-02	5.00E-02
	GHADBBAADDAAAB			0.00E+00	5.20E+04	2.20E+04	3.80E-02	1.20E-02	1.10E-02	5.00E-03	2.40E-03	4.40E-04	1.90E-04	1.20E-03	2.50E-03
5	GHADBBABDEAAAA	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	7.40E-01	1.60E-01	1.40E-01	5.30E-02	4.20E-03	1.90E-03	2.00E-04	6.20E-04	5.70E-03
	GHADBBABDDAAAA			0.00E+00	5.20E+04	2.20E+04	3.70E-02	8.00E-03	6.80E-03	2.70E-03	2.10E-04	9.60E-05	1.00E-05	3.10E-05	2.90E-04
6	GHADBBAADEAAAA	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	7.60E-01	2.40E-01	2.10E-01	1.00E-01	4.70E-02	8.80E-03	3.90E-03	2.30E-02	5.00E-02
	GHADBBAADDAAAA			0.00E+00	5.20E+04	2.20E+04	3.80E-02	1.20E-02	1.10E-02	5.00E-03	2.40E-03	4.40E-04	1.90E-04	1.20E-03	2.50E-03
7	FHADBBABDDAAAB	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	9.40E-01	1.60E-01	1.40E-01	5.30E-02	4.20E-03	1.90E-03	2.00E-04	6.20E-04	5.70E-03
	FHADBBABDCAAAB			0.00E+00	5.20E+04	1.10E+04	5.80E-02	3.90E-02	6.80E-03	2.70E-03	2.20E-04	9.60E-05	1.10E-05	3.10E-05	2.90E-04
8	GDCDFADADEBAAA	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.60E-01	1.70E-01	1.50E-01	1.20E-01	2.60E-02	5.20E-03	2.10E-03	1.20E-02	2.80E-02
	GDCDFADADDBAAA			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.60E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
9	FHADBBAADDAAAB	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	9.40E-01	2.40E-01	2.10E-01	1.00E-01	4.70E-02	8.80E-03	3.90E-03	2.30E-02	5.00E-02
	FHADBBAADCAAAB			0.00E+00	5.20E+04	1.10E+04	5.70E-02	4.90E-02	1.10E-02	5.00E-03	2.40E-03	4.40E-04	1.90E-04	1.20E-03	2.50E-03
10	FHADBBABDDAAAA	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	9.40E-01	1.60E-01	1.40E-01	5.30E-02	4.20E-03	1.90E-03	2.00E-04	6.20E-04	5.70E-03
	FHADBBABDCAAAA			0.00E+00	5.20E+04	1.10E+04	5.80E-02	3.90E-02	6.80E-03	2.70E-03	2.20E-04	9.60E-05	1.10E-05	3.10E-05	2.90E-04
11	FHADBBAADDAAAA	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	9.40E-01	2.40E-01	2.10E-01	1.00E-01	4.70E-02	8.80E-03	3.90E-03	2.30E-02	5.00E-02
	FHADBBAADCAAAA			0.00E+00	5.20E+04	1.10E+04	5.70E-02	4.90E-02	1.10E-02	5.00E-03	2.40E-03	4.40E-04	1.90E-04	1.20E-03	2.50E-03
12	GDCDFADBDEBAAA	1.30E+04	1.00E+01	1.00E+06	2.00E+04	3.60E+03	3.70E-01	1.80E-01	1.60E-01	1.40E-01	1.60E-02	5.00E-03	1.10E-03	5.20E-03	1.90E-02
	GDCDFADBDDBAAA			0.00E+00	1.00E+06	1.00E+06	0.00E+00	2.80E-05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
13	GHBBBBAADEAAAB	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	8.80E-01	3.20E-01	2.90E-01	1.50E-01	5.40E-02	1.20E-02	5.20E-03	3.20E-02	5.90E-02
	GHBBBBAADDAAAB			0.00E+00	5.20E+04	2.20E+04	4.40E-02	1.60E-02	1.40E-02	7.60E-03	2.70E-03	5.90E-04	2.60E-04	1.60E-03	2.90E-03
14	GHBCBBAADEAAAB	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	7.00E-01	1.50E-01	1.30E-01	4.80E-02	7.50E-03	1.60E-03	3.90E-04	1.50E-03	9.10E-03
	GHBCBBAADDAAAB			0.00E+00	5.20E+04	2.20E+04	3.50E-02	7.40E-03	6.30E-03	2.40E-03	3.70E-04	8.20E-05	1.90E-05	7.50E-05	4.50E-04
15	GHABABAACEAAAB	3.60E+04	1.00E+01	1.00E+06	5.10E+04	1.00E+03	7.30E-01	1.40E-01	1.10E-01	3.30E-02	9.10E-04	4.60E-04	4.10E-05	9.90E-05	1.60E-03
	GHABABAACDAAAB			0.00E+00	5.20E+04	2.20E+04	3.70E-02	7.00E-03	5.60E-03	1.60E-03	4.50E-05	2.30E-05	2.00E-06	5.00E-06	8.10E-05

* A listing of source terms for all bins is available on computer media

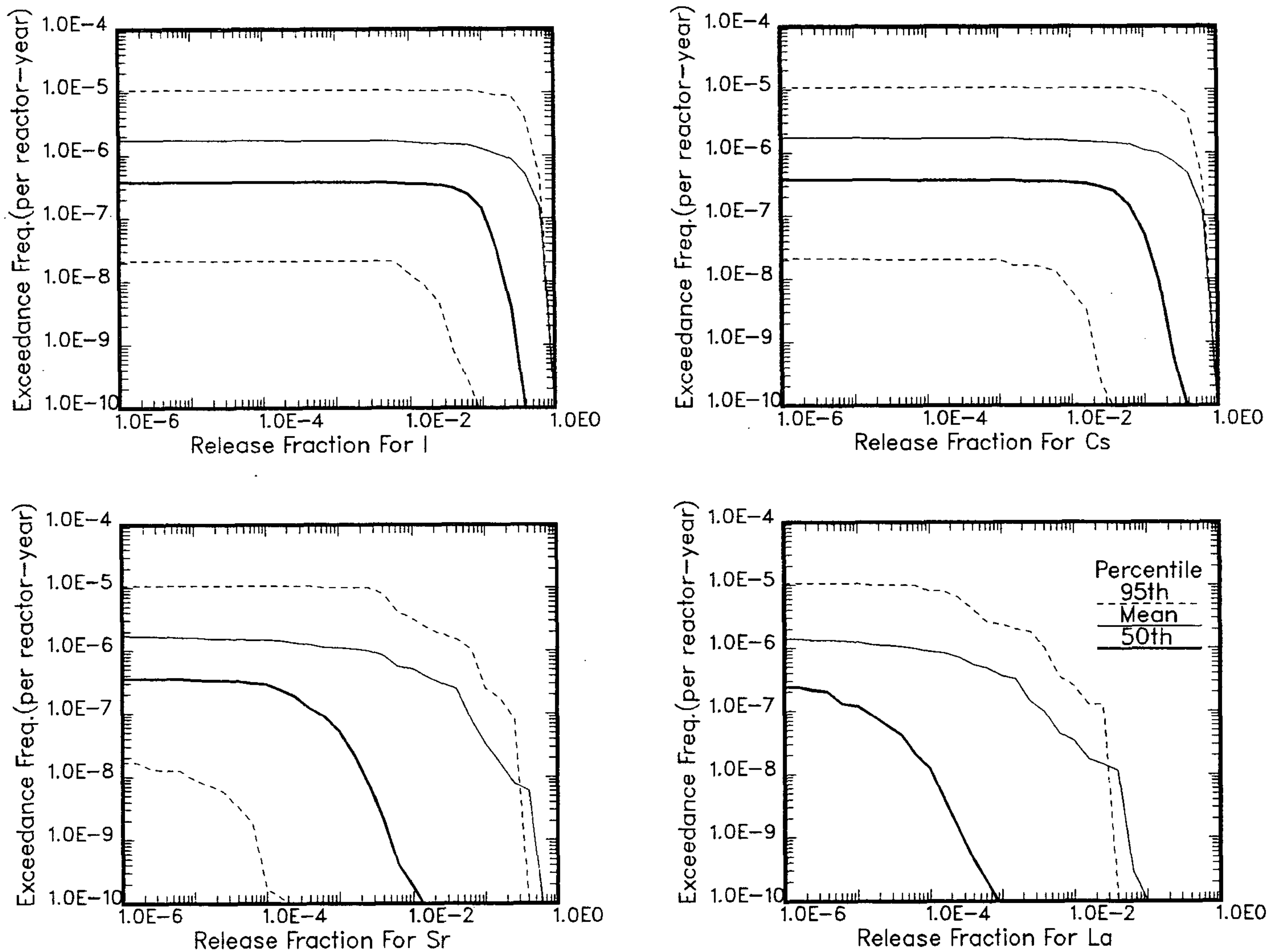


Figure 3.3-7. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (PDS Group 7: SGTRs)

3.3.1.8 Results for Generalized Accident Progression Bins. The preceding seven subsections presented the source term results by PDS group. It is also possible to group the source terms in other ways. These other groupings are called generalized APBs. In some cases, these generalized APBs break apart the results in a PDS group, and in others, they put results from several PDS groups together.

Figure 3.3-8 shows the variation of the exceedance frequency with release fraction for the iodine, cesium, strontium, and lanthanum radionuclide classes for all the APBs that had containment failure during CD. The containment failure is due to hydrogen burn or detonation, or isolation failure. None of the APBs included in Figure 3.3-8 involved a bypass event; that is, no SGTR or Event V APBs are included. This figure shows that the frequency of a sizeable release from containment failure during CD is quite low; however, the curves for iodine and cesium indicate that if the event occurs, the release fraction is likely to exceed 0.01. For strontium and lanthanum, it is more likely that the releases will be much lower.

Figures 3.3-9, 3.3-10, and 3.3-11 show the variation of the exceedance frequency with release fraction for all the APBs in which there was containment failure at VB and the containment was not bypassed. Figure 3.3-9 contains APBs with Alpha mode failure of the vessel and containment. Figure 3.3-10 contains APBs in which the containment failed at VB with the RCS at high (>200 psia) pressure at the time of VB and Figure 3.3-11 contains APBs in which the containment failed at VB with the RCS at low (<200 psia) pressure at the time of VB. These figures indicate that if containment failure occurs at VB, the release fractions for iodine and cesium are likely to exceed 0.01. Note that the qualitative features of the curves for the early containment failure in Figures 3.3-8 through 3.3-11 are similar. For example, with respect to the iodine and cesium mean curves, the curves are relatively flat until they begin to decrease slowly at release fractions between 10^{-3} and 10^{-2}. These are basically "threshold" release fractions that form a lower limit for the magnitude of the release. Variation between the curves is noted due to variation in functioning of mitigating features (sprays, ice, etc.) between and within the generalized bins.

Figure 3.3-12 considers all the APBs in which the containment failed some hours or days after the vessel failed, and the containment was not bypassed. Some of these failures are due to hydrogen burns a few hours after VB, some are by eventual overpressure due to lack of CHR, or they result from BMT. The figure shows that these types of containment failure are much more frequent than early containment failure but that the release fractions are likely to be much lower. The exceedance frequencies for late containment failure decrease more rapidly at lower release fractions than they do for early failures; that is, there is not a threshold effect at high release fractions. This also results in a greater spread in the magnitude of the source term for late containment failure than for early containment failure.

Figures 3.3-13 and 3.3-14 show the variation of the exceedance frequency with release fraction for Event V. All the source terms for the V-Dry APBs were analyzed to produce Figure 3.3-13, while all the source terms for the

V-Wet APBs were analyzed to produce Figure 3.3-14. As expected, the V-Dry release fractions are larger than the V-Wet release fractions due to the absence of the scrubbing by the fire sprays. The V-Dry releases are, however, about an order of magnitude less likely than the V-Wet releases. The "threshold" release fractions are higher for the V sequence releases (especially V-Dry) than for the early containment failures, and the range of release fractions is smaller for this accident.

Figures 3.3-15 and 3.3-16 consider all the APBs with SGTRs. Figure 3.3-15 shows the SGTRs in which the secondary SRVs reclose, termed "G" SGTRs, whereas Figure 3.3-16 shows the SGTRs in which the secondary SRVs stick open, termed "H" SGTRs. Almost all these SGTRs are initiating events; a very small portion of these APBs results from T-I SGTRs following the onset of core damage. The T-I SGTRs are all "G" SGTRs. As indicated by the discussion in subsection 2.5.1.6 and 3.3.1.6, the "H" SGTRs are both more likely and more harmful than the normal "G" SGTRs.

3.3.1.9 Summary. When all the types of internally initiated accidents at Sequoyah are considered together, the exceedance frequency plots shown in Figure 3.3-17 are obtained. The first sheet of Figure 3.3-17 shows the release fractions for iodine, cesium, tellurium, and strontium. The second sheet of Figure 3.3-17 shows the release fractions for ruthenium, lanthanum, cerium, and barium, which are often treated together as aerosol species. A plot is not shown for the noble gases because almost all of the noble gases (xenon and krypton) in the core are eventually released to the environment whether the containment fails or not. The mean frequency of exceeding a release fraction of 0.10 for iodine is 4×10^{-6}, 3×10^{-6} for cesium, 2×10^{-6} for tellurium, 3×10^{-7} for strontium, 4×10^{-9} for ruthenium, 1×10^{-10} for lanthanum, 4×10^{-8} for cerium, and 3×10^{-7} for barium.

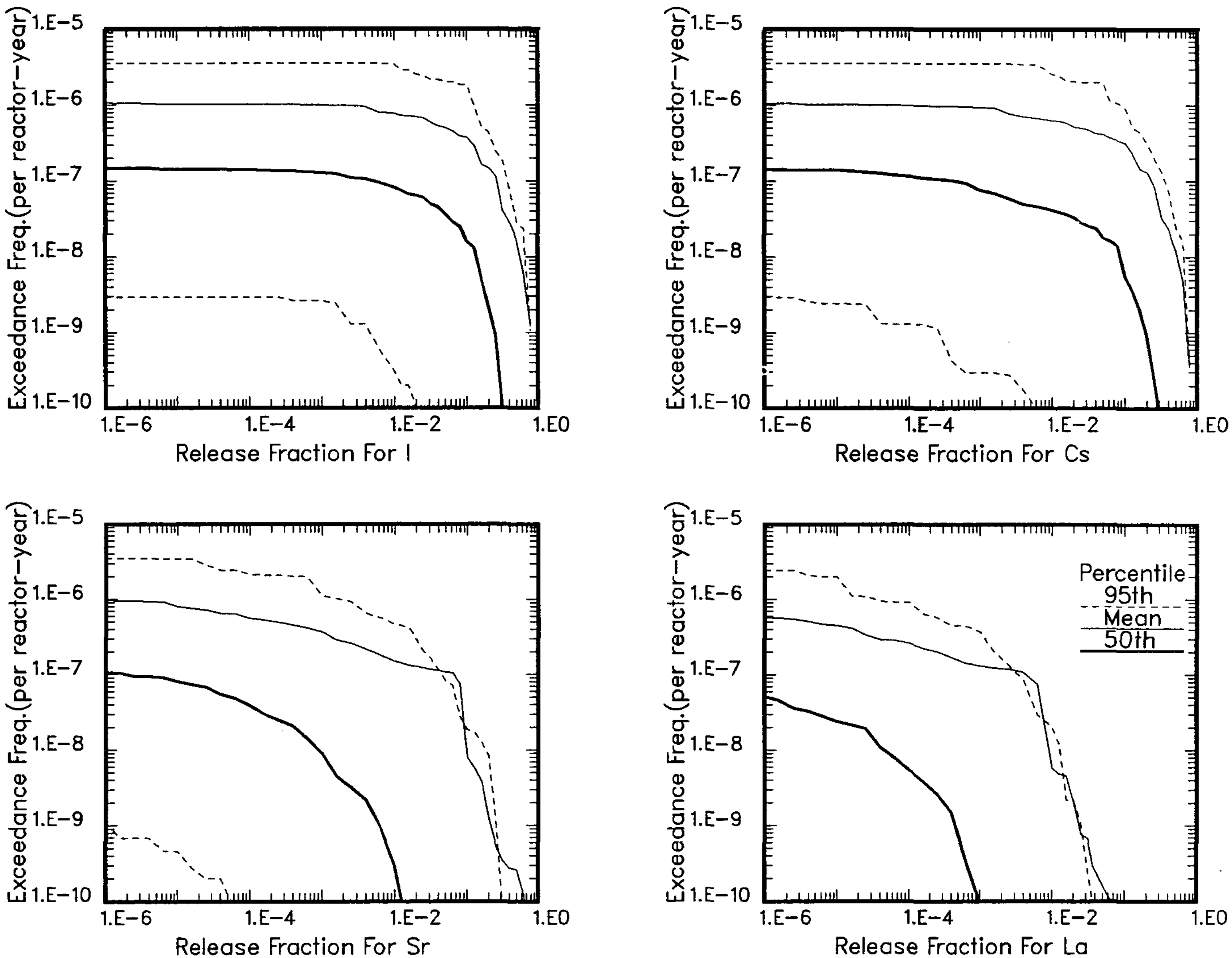


Figure 3.3-8. Exceedance Frequencies for Release Fractions for
Sequoyah Internal Initiators (CF During CD)

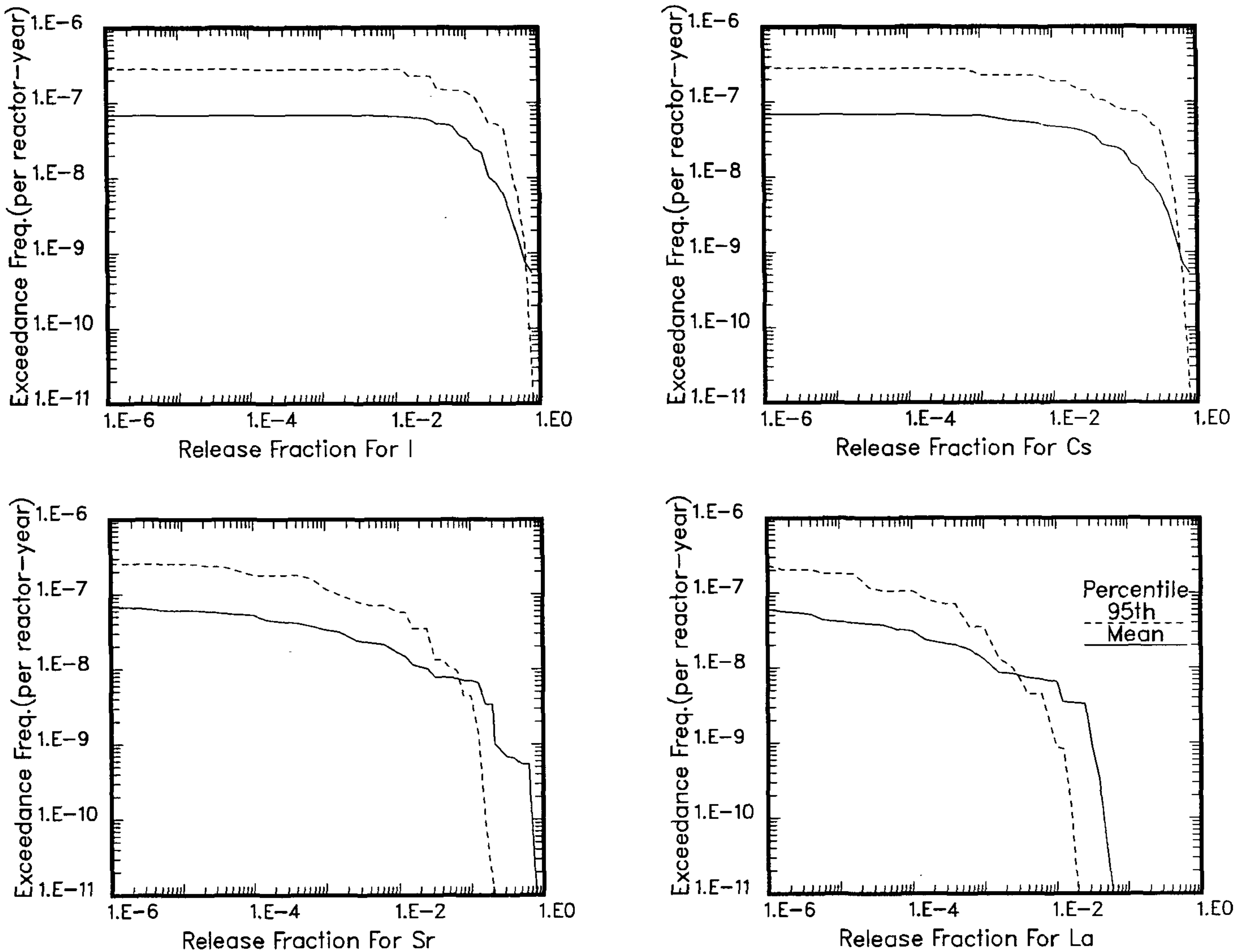


Figure 3.3-9. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (Alpha Mode)

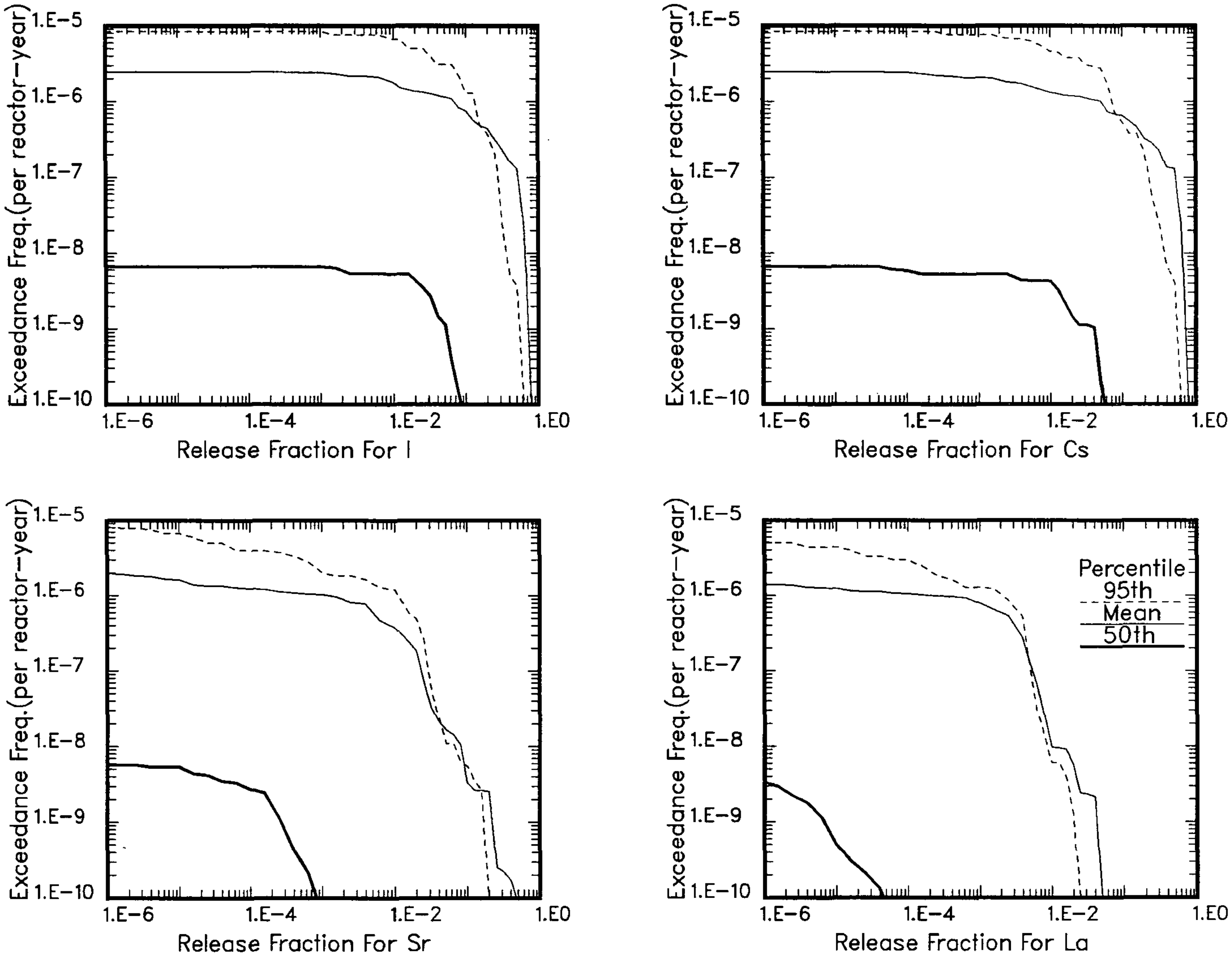


Figure 3.3-10. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (CF at VB with the RCS at High Pressure)

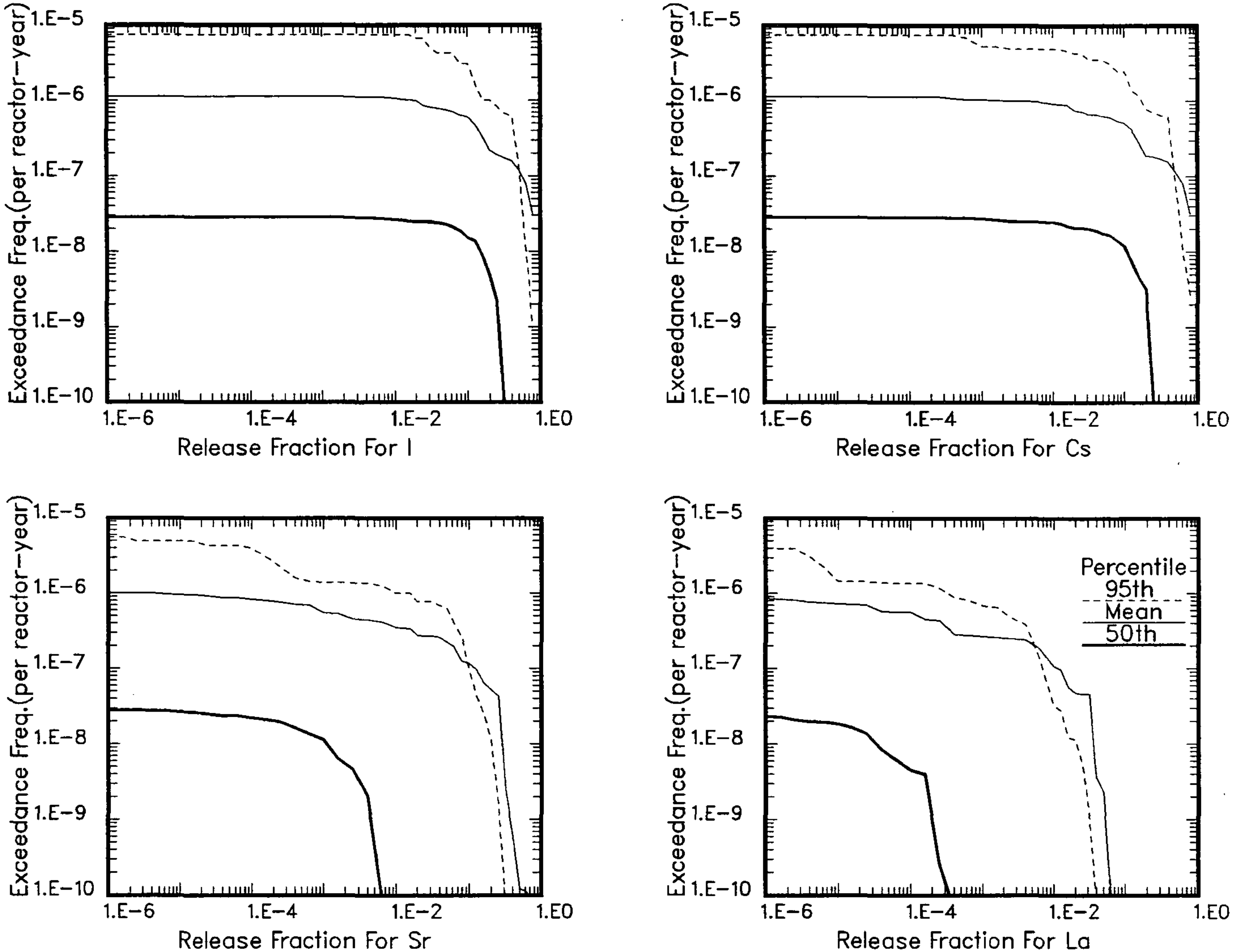


Figure 3.3-11. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (CF at VB with the RCS at Low Pressure)

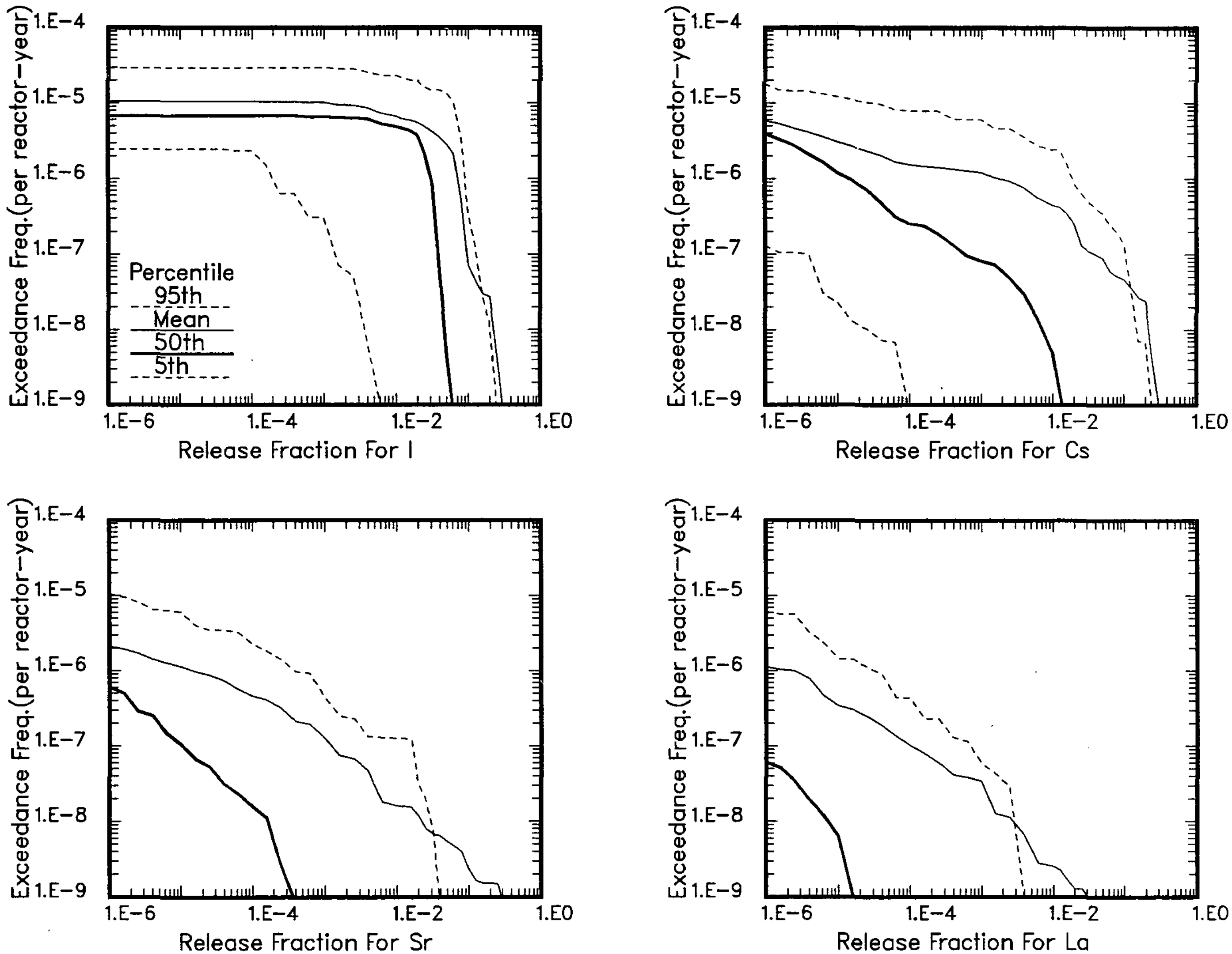


Figure 3.3-12. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (Late CF)

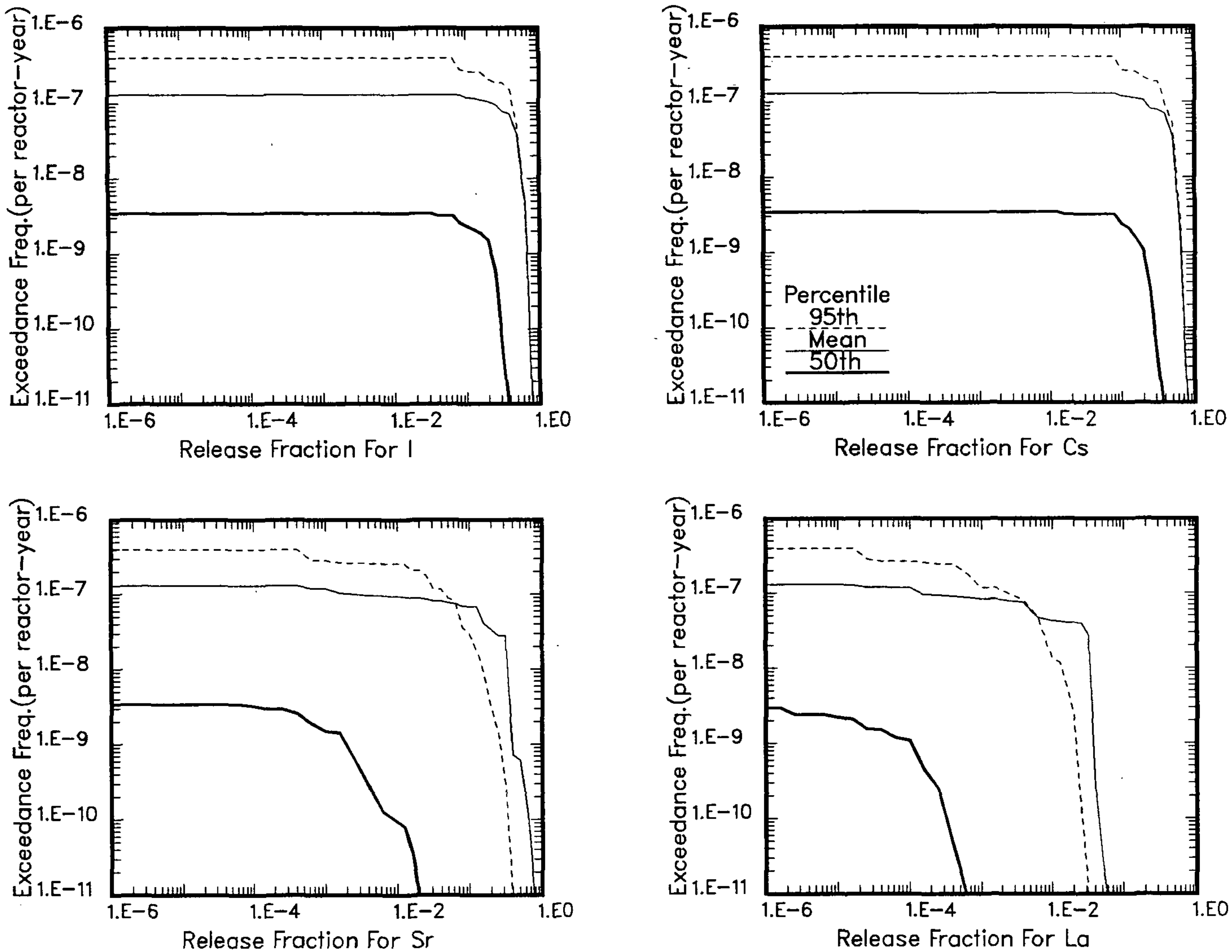


Figure 3.3-13. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (Event V, Dry)

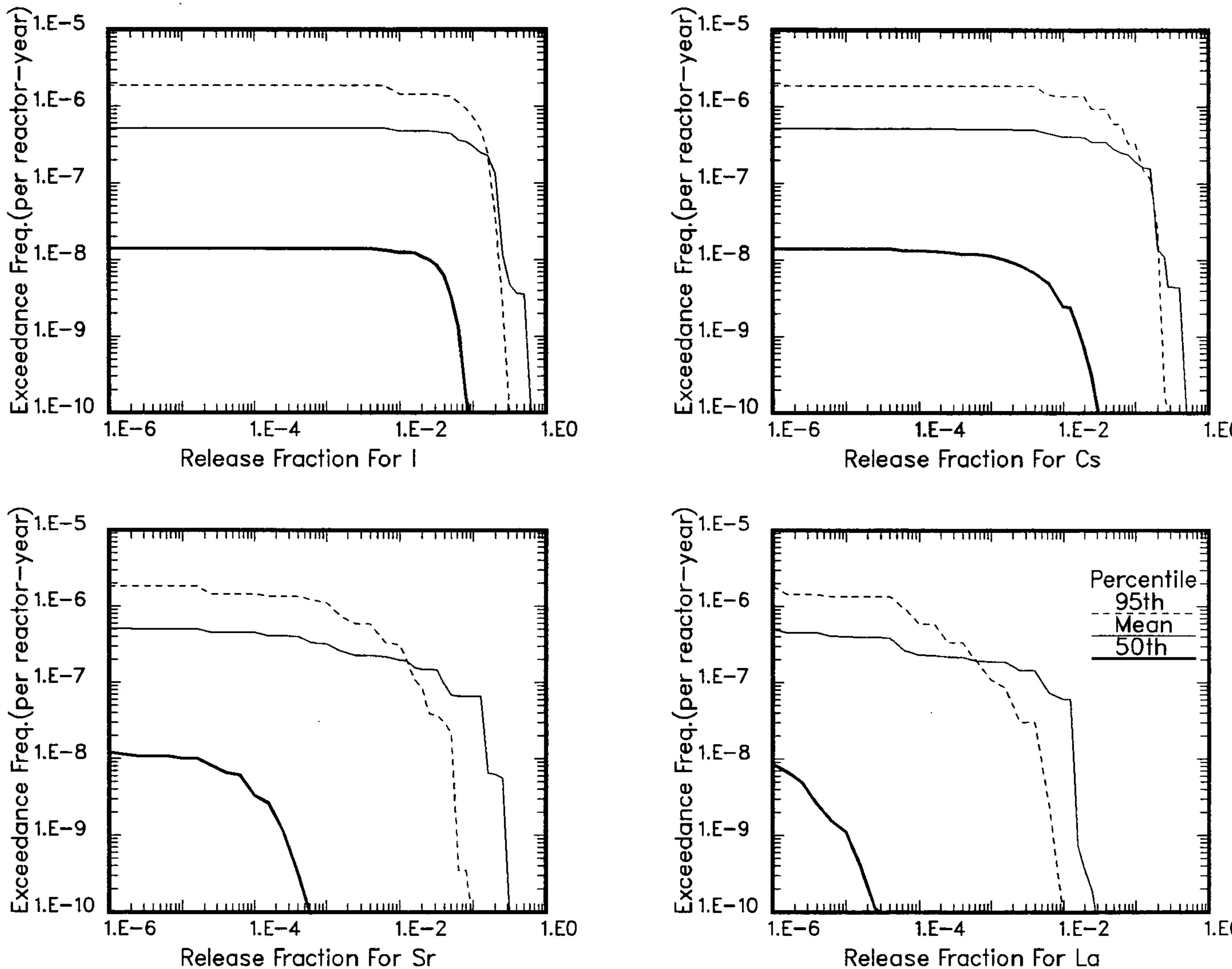


Figure 3.3-14. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators (Event V, Wet)

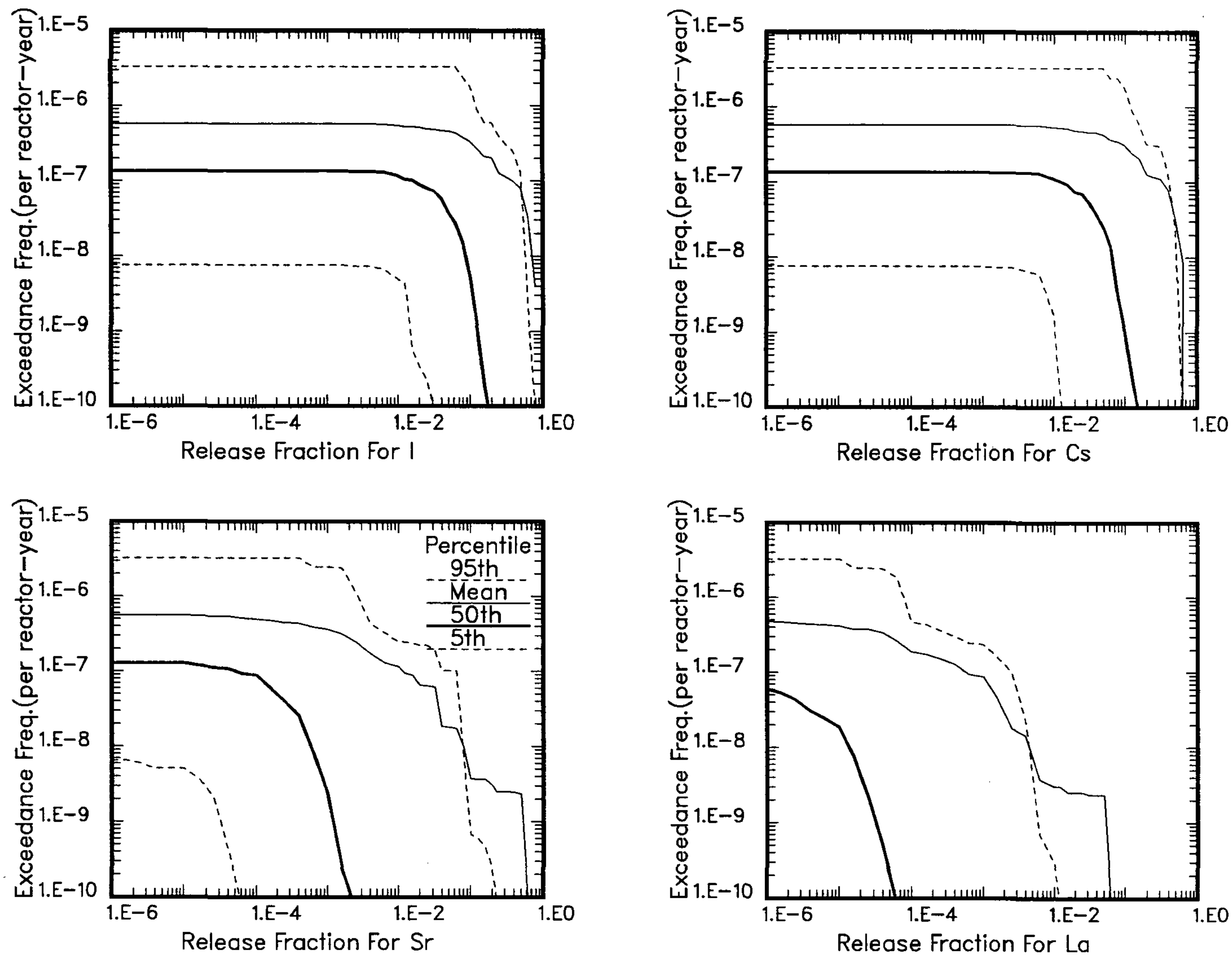


Figure 3.3-15. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators, "G" SGTRs (Secondary SRVs Reclosing)

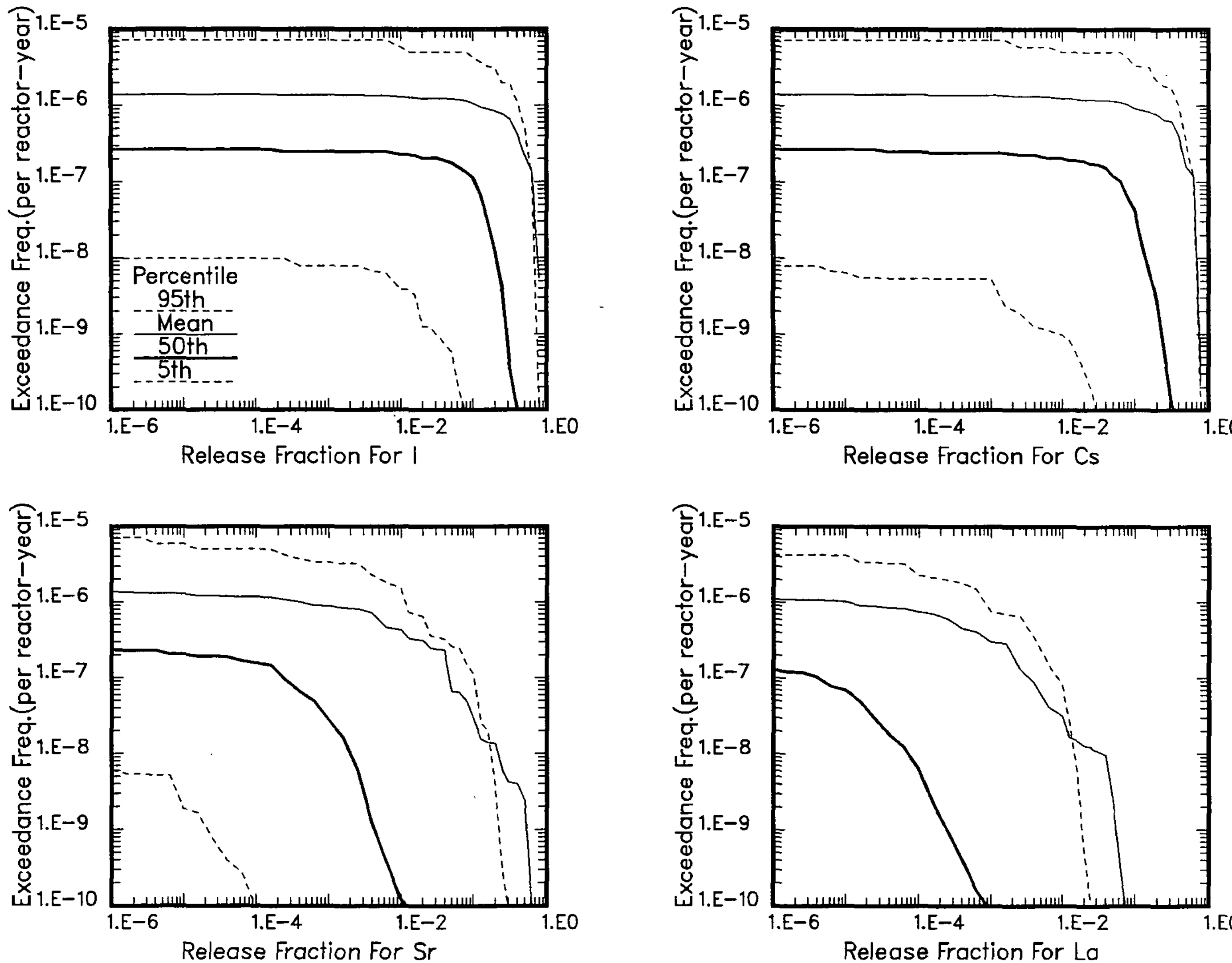


Figure 3.3-16. Exceedance Frequencies for Release Fractions for Sequoyah Internal Initiators, "H" SGTRs (Secondary SRVs Stuck Open)

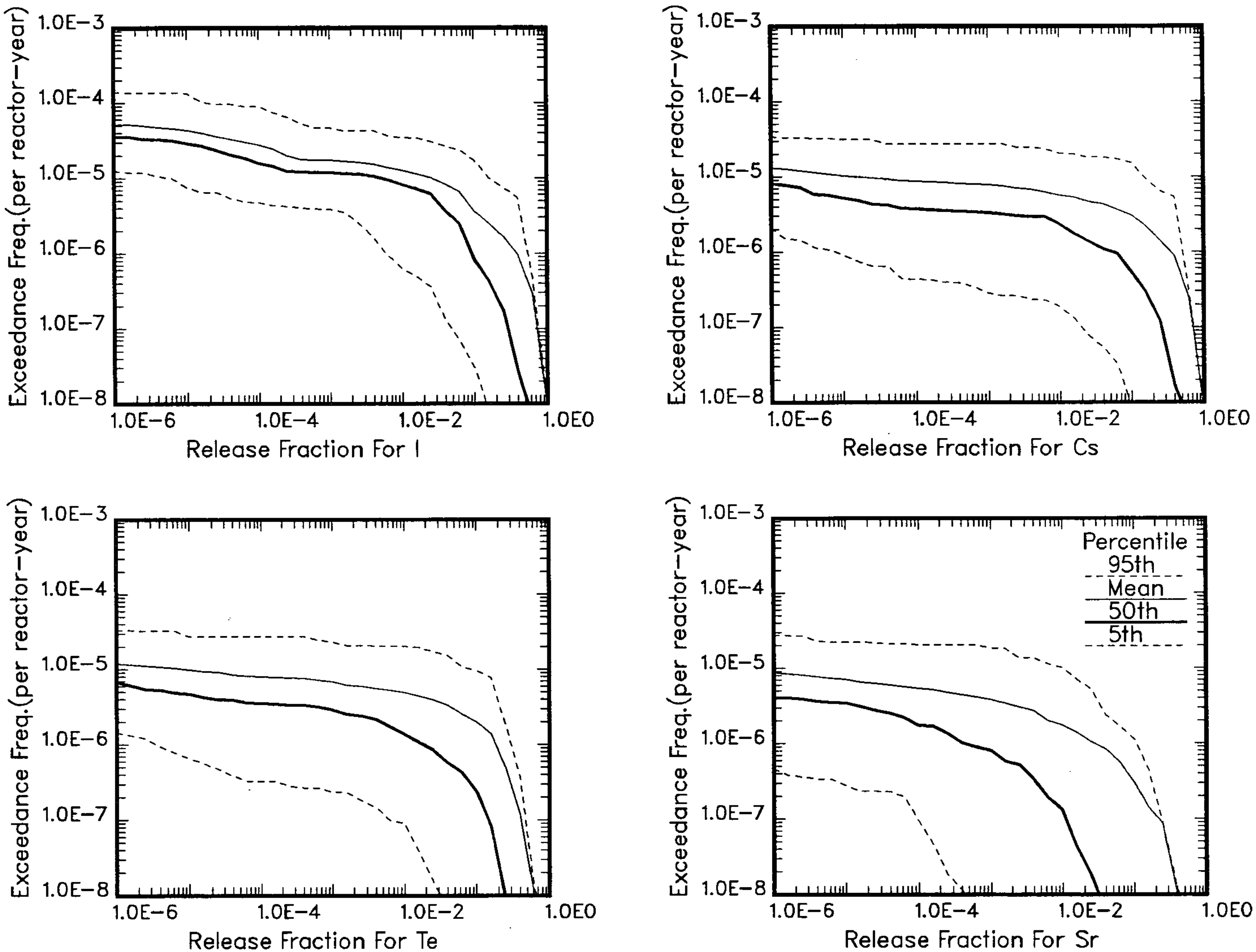


Figure 3.3-17. Exceedance Frequencies for Release Fractions for Sequoyah (All Internal Initiators)

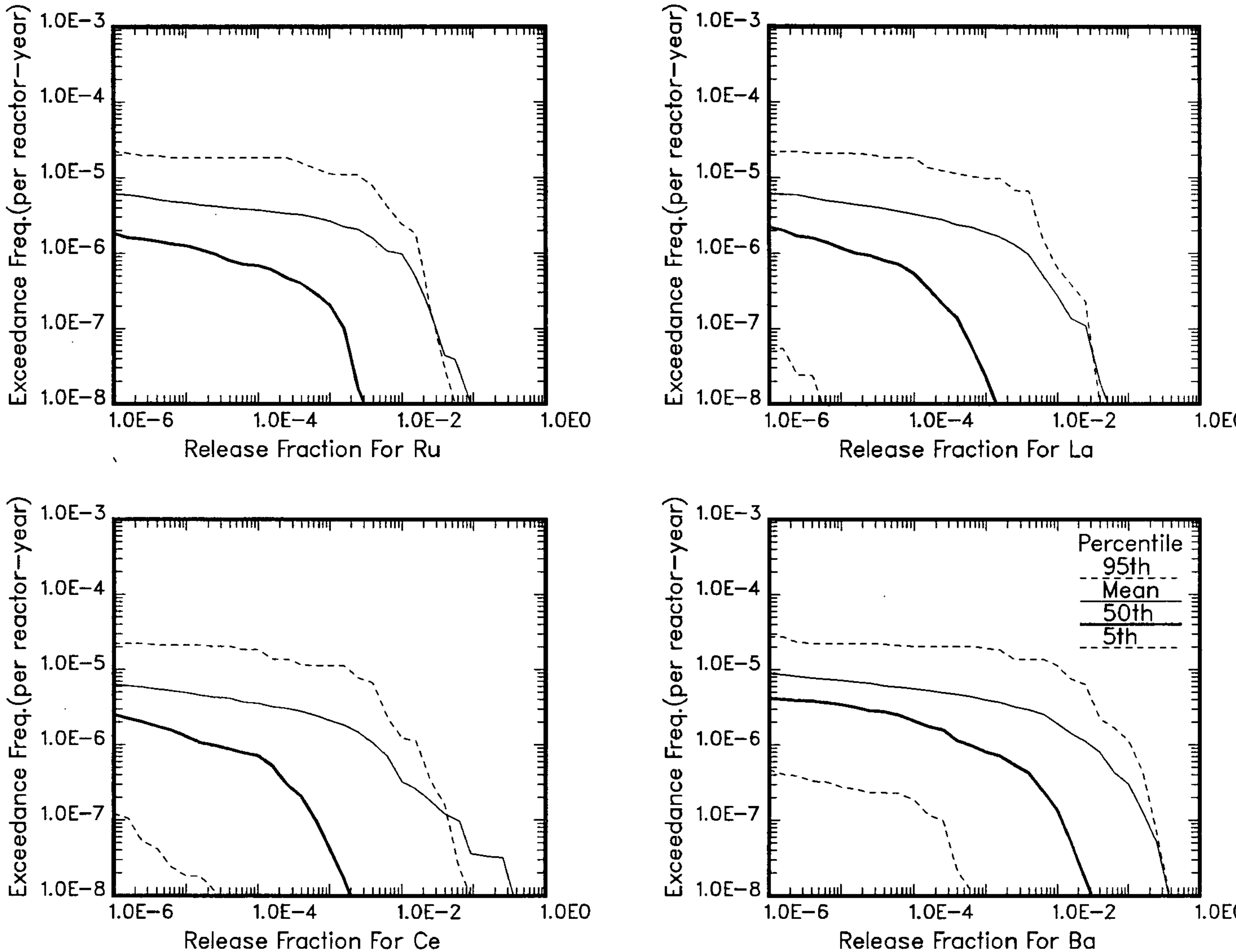


Figure 3.3-17. (continued).

3.4 Partitioning of the Source Terms for the Consequence Analysis

The following discusses the partitioning process in some detail as it presents the partitioning results for internal initiators.

3.4.1 Results for Internal Initiators

The accident progression analysis and the subsequent source term analysis resulted in the generation of 114,471 source terms for internal initiators. It is not computationally possible to perform a calculation with the MACCS consequence model[1] for each of these source terms. Therefore, the number of source term groups. These groups are defined so that the source terms within them have similar properties and a frequency-weighted mean source term is determined for each group. Then, a single MACCS calculation interface between the source term analysis and the consequence analysis is formed by grouping this large number of source terms into a much smaller is performed for each mean source term. This grouping of the source terms is performed with the PARTITION program,[2] and the process is referred to as "partitioning the source terms" or just "partitioning."

The partitioning process involves the following steps: definition of an early health effect weight (EH) for each source term, definition of a chronic health effect weight (CH) for each source term, subdivision (partitioning) of the source terms on the basis of EH and CH, a further subdivision on the basis of evacuation timing, and calculation of frequency-weighted mean source terms. The partitioning process is described in detail in NUREG/CR-4551, Vol. 1, and in the user's manual for the PARTITION program.[2] This section details the partitioning process for source terms generated in the source term analysis for internal initiators.

The EH is based on converting the radionuclide release associated with a source term into an equivalent I-131 release and then estimating the number of early fatalities that would result from this equivalent I-131 release. This estimated number of early fatalities is the EH. The relationship between early fatalities and equivalent I-131 releases is shown in Figure B.4-1 of Appendix B and is based on site-specific MACCS calculations for different-sized releases of I-131.

The CH is based on an assumed linear relationship between cancer fatalities due to a radionuclide and the amount of that radionuclide released. Specifically, a site-specific MACCS calculation is performed for a fixed release of each of the 60 radionuclides included in the NUREG-1150 consequence calculations. The results of these calculations and the assumed linear relationship between the amount released and cancer fatalities for each radionuclide are then used to estimate the total number of chronic fatalities associated with a source term. This estimated number of chronic fatalities is the chronic health effect weight CH. The results of the MACCS calculations used in the determination of CH are shown in Table B.4-1 of Appendix B. Further, the input file for PARTITION containing the site-specific data used in the calculation of EH and CH is shown in Table B.4-2 of Appendix B.

The site-specific MACCS calculations that underlie the early and chronic health effect weights were performed with very conservative assumptions with respect to the energy and timing of the releases and also with respect to the emergency responses taken. As a result, these weights should be regarded as a measure of the potential of a source term to cause early and chronic fatalities rather than as an estimate of the fatalities that would actually result from a source term.

The partitioning process treats the cases for EH>0 and CH>0 and for EH=0 and CH>0 separately. Table 3.4-1 shows the division of the source terms into these two cases.

The case for EH>0 and CH>0 is treated first by PARTITION. As shown in Table 3.4-1, log CH ranges from -0.5459 to 5.1442, and log EH ranges from -0.5951 to 2.4375. Figure 3.4-1 shows a plot of the pairs (CH, EH) for the 46,714 source terms for which both EH and CH are nonzero. The partitioning process is based on laying a grid on the (CH, EH) space shown in Figure 3.4-1 and then pooling cells that have either a small frequency or contain a small number of source terms. Specifically, the grid is selected so that the ratio between the maximum and minimum value for CH in any cell and also the ratio between the maximum and minimum value for EH in any cell will be less than a specified value. In this analysis, the maximum allowable ratio was selected to be 4.05, which resulted in a loguniform division of the range of CH into 10 intervals and a similar division of the range of EH into five intervals. The result of placing the selected grid on the (CH, EH) space is also shown in Figure 3.4-1.

A summary of the partitioning process for EH>0 and CH>0 is given in Table 3.4-2. The table is divided into three parts. The first part is labeled "BEFORE PARTITIONING" and shows the distribution of the source terms before the partitioning process. As in Figure 3.4-1, the abscissa and ordinate correspond to CH and EH, respectively, with the ranges given in Table 3.4-1. The top plot shows the cell counts, and the bottom plot shows the fraction of the frequency in each cell. The second part of Table 3.4-2 is labeled "AFTER PARTITIONING" and shows the distribution of the source terms after the partitioning process. The partitioning process does not result in the loss of any source terms; rather, cells with a small number of source terms or a small frequency are pooled with other cells. Thus, the total number of source terms is not changed. The third part of this table is denoted "LABELING AFTER PARTITIONING" and shows the designators that will be used in the identification of source terms derived from the partitioning process.

A summary of the partitioning process for EH=0 and CH>0 is given in Table 3.4-3, which is structured analogously to Table 3.4-2 but has only one dimension instead of two. As indicated in Table 3.4-1, log(CH) ranges from -4.0011 to 3.7495. The cells shown in Table 3.4-3 are based on a loguniform division of the range of CH into eight intervals.

Table 3.4-1
Summary of Early and Chronic Health Effect Weights
for Internal Initiators

	Number of Source Terms	Percent of Total Frequency
EH>0 and CH>0	46714	12.75
EH=0 and CH>0	67757	87.25
EH=0 and CH=0	0	0.00
Total	114471	100.00

For EH>0 and CH>0, Range LOG10(CH) = -0.5459 to 5.1442
Range LOG10(EH) = -0.5951 to 2.4375

For EH=0 and CH>0, Range LOG10(CH) = -4.0011 to 3.7495

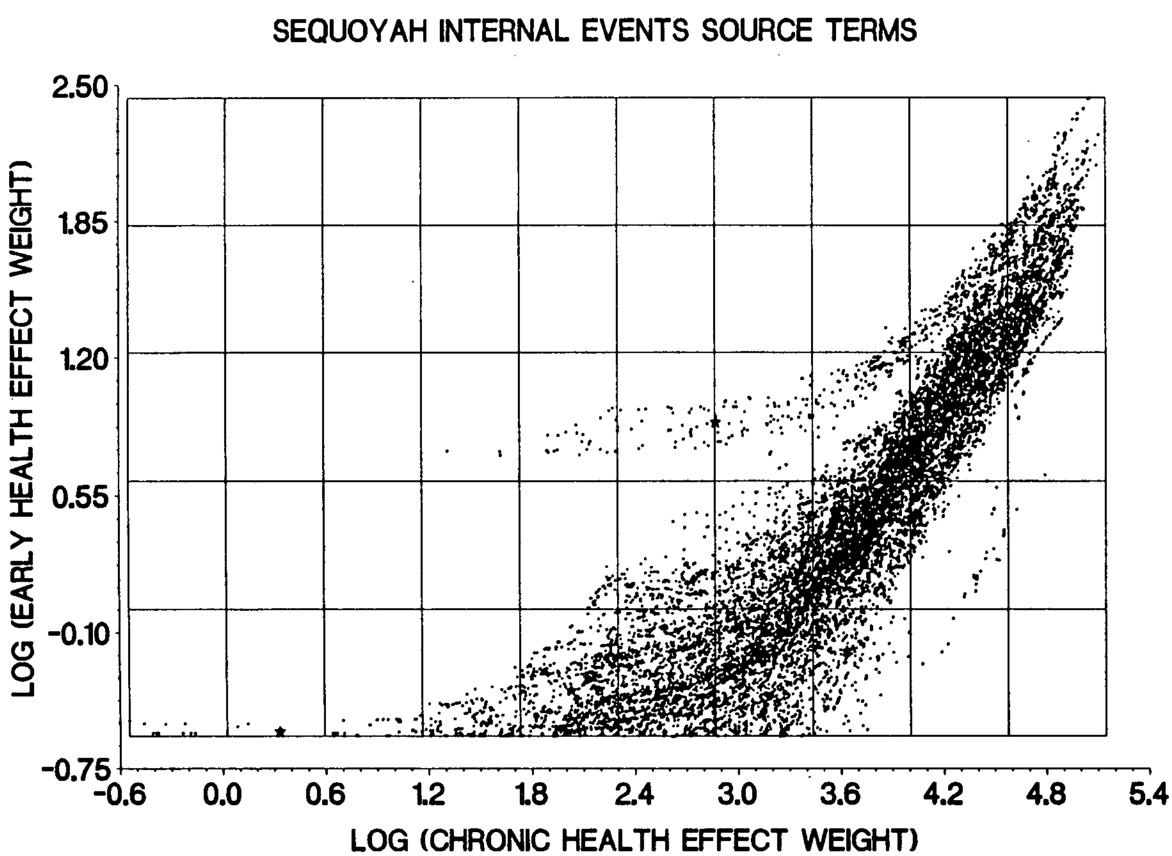


Figure 3.4-1. Distribution of Nonzero Early and Chronic Health Effect Weights for Internal Initiators

At this point, the result of partitioning is 18 groups of source terms as shown in Tables 3.4-2 and 3.4-3. These source term groups are now further subdivided on the basis of evacuation timing. Specifically, each group of source terms is subdivided into three subgroups:

Subgroup 1: Evacuation starts at least 30 min before the release begins;

Subgroup 2: Evacuation starts between 30 min before and 1 h after the release begins;

Subgroup 3: Evacuation starts more than 1 h after the release begins.

This sorting of source terms is based on the warning time and the release start time associated with a source term and on the site-specific evacuation delay time. By definition, the evacuation delay is the time interval between the time the warning is given and the time the evacuation actually begins. The evacuation delay time for Sequoyah is 2.3 h. Additional discussion of evacuation delay time is given in Volume 2, Part 7 of this report.[7]

Once the source term groups shown in Tables 3.4-2 and 3.4-3 are sorted into subgroups on the basis of evacuation timing, a frequency-weighted mean source term is calculated for each populated subgroup. In the consequence analysis, a full MACCS calculation is performed for the mean source term for each source term subgroup. The mean source terms obtained in this analysis are shown in Table 3.4-4. This table contains frequency-weighted mean source terms for both the source term groups and subgroups. In the table, SEQ-I and SEQI-J are used to label the mean source terms derived from source term groups and subgroups, respectively, where I designates the source term group and J designates the source term subgroup. It is the source terms for the subgroups, SEQ-I-J in Table 3.4-4, that are actually used for the risk calculations.

Although not parts of the source term definition, Table 3.4-4 also contains the mean frequency for the source term group, the conditional probability of the source term subgroups, and the mean value for the difference between the time at which release starts and the time at which evacuation starts (labeled dEVAC in the table). A positive value of dEVAC indicates that the evacuation starts before the release and a negative value of dEVAC indicates that the evacuation starts after the release. The mean frequency for a source term group is obtained by summing the frequencies of all source terms assigned to the group and then dividing by the sample size (200 in this analysis). The conditional probability of a subgroup is obtained by summing the frequencies of all source terms assigned to the subgroup and then dividing the resultant sum by the total frequency of all source terms in the associated source term group. Some source term subgroups are unpopulated; a mean source term does not appear for these subgroups in Table 3.4-4. To calculate the frequency-weighted mean source terms appearing in Table 3.4-4, each source term is weighted by the ratio between its frequency and the total frequency associated with the particular source term group or subgroup under consideration.

Source term groups SEQ-04 and SEQ-07 are dominated by Event V; Group SEQ-01 is dominated by early containment failures and "G" SGTRs; and Groups SEQ-16, SEQ-17, and SEQ-18 are dominated by late containment failures. The dominant accident is reflected in the mean source term for the group. For SEQ-04, Table 3.4-4 shows that almost all the probability is associated with the subgroup which has early release (at about 1 h), with evacuation starting after the release has commenced. The group with the highest release fractions, Group SEQ-14, is comprised of about two-thirds Event V source terms. About one-third of the source terms in Group SEQ-14 are from early containment failures and "G" SGTRs, and a small fraction come from "H" SGTRs. The frequency for this group, however, is fairly low; relatively few source terms fall in the grid represented by Group SEQ-14, and they are not exceptionally frequent. The most likely source term groups are SEQ-16, SEQ-17, and SEQ-18, which do not cause early fatalities and arise from accidents that do not result in bypass or early containment failure.

Table 3.4-2
Distribution of Source Terms with Nonzero Early Fatality and Chronic Fatality Weights for Internal Initiators

BEFORE PARTITIONING: CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 46714

	1	2	3	4	5	6	7	8	9	10
1									14	2399
2								165	5307	3813
3				40	247	306	590	3548	7354	166
4					76	470	2916	8341	795	1
5	38	4	66	267	1725	2581	4354	1124	7	

BEFORE PARTITIONING: PERCENT OF FREQUENCY CONTAINED IN EACH CELL

	1	2	3	4	5	6	7	8	9	10
1									0.00	1.90
2								1.39	7.34	13.68
3				0.11	0.06	0.81	0.96	5.18	19.66	0.10
4					0.01	0.29	4.16	14.01	2.53	0.00
5	0.73	0.00	0.09	0.30	4.95	7.75	12.94	1.04	0.00	

AFTER PARTITIONING: CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 46714

	1	2	3	4	5	6	7	8	9	10
1										2413
2								165	5307	3966
3							1091	3548	7367	
4							2941	8341	797	
5				375	1869	2976	4428	1130		

Table 3.4-2 (continued)

AFTER PARTITIONING: PERCENT OF FREQUENCY CONTAINED IN EACH CELL

	1	2	3	4	5	6	7	8	9	10
1										1.90
2								1.39	7.34	13.78
3							1.83	5.18	19.66	
4							4.16	14.01	2.53	
5				1.12	5.06	7.98	13.00	1.04		

LABELING AFTER PARTITIONING:

	1	2	3	4	5	6	7	8	9	10
1										SEQ-14
2								SEQ-07	SEQ-11	SEQ-15
3							SEQ-04	SEQ-08	SEQ-12	
4							SEQ-05	SEQ-09	SEQ-13	
5				SEQ-01	SEQ-02	SEQ-03	SEQ-06	SEQ-10		

Table 3.4-3
Distribution of Source Terms with Zero Early Fatality Weight and Nonzero Chronic Fatality Weight for Internal Initiators

BEFORE PARTITIONING: CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 67757

	1	2	3	4	5	6	7	8
1	850	3329	11263	16504	4875	16448	12474	2014

BEFORE PARTITIONING: PERCENT OF FREQUENCY CONTAINED IN EACH CELL

	1	2	3	4	5	6	7	8
1	2.35	13.75	30.92	30.01	3.50	10.60	8.15	0.72

AFTER PARTITIONING: CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 67757

	1	2	3	4	5	6	7	8
1			15442	18290		34025		

AFTER PARTITIONING: PERCENT OF FREQUENCY CONTAINED IN EACH CELL

	1	2	3	4	5	6	7	8
1			47.02	31.23		21.75		

LABELING AFTER PARTITIONING:

	1	2	3	4	5	6	7	8
1			SEQ-16	SEQ-17		SEQ-18		

Table 3.4-4
Mean Source Terms Resulting from Partitioning for Internal Initiators - Sequoyah

Source Term	Freq. (1/yr)	Cond. Prob.	Warn (s)	dEvac (s)	Elev (m)	Energy (W)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
SEQ-01	7.8E-08		2.2E+04	-2.6E+03	10.	2.6E+06	2.8E+04	3.0E+02	9.2E-01	3.9E-05	3.8E-05	4.4E-06	9.1E-07	1.5E-07	5.6E-08	2.5E-07	9.7E-07
						6.8E+05	8.5E+05	8.5E+05	5.2E-02	1.9E-03	4.8E-05	1.6E-05	1.0E-06	4.9E-08	1.1E-07	1.5E-07	9.9E-07
SEQ-01-1		0.000															
SEQ-01-2		1.000	2.2E+04	-2.6E+03	10.	2.6E+06	2.8E+04	3.0E+02	9.2E-01	3.9E-05	3.8E-05	4.4E-06	9.1E-07	1.5E-07	5.6E-08	2.5E-07	9.7E-07
						6.8E+05	8.5E+05	8.5E+05	5.2E-02	1.9E-03	4.8E-05	1.6E-05	1.0E-06	4.9E-08	1.1E-07	1.5E-07	9.9E-07
SEQ-01-3		0.000															
SEQ-02	3.5E-07		2.2E+04	9.1E+03	10.	8.6E+05	3.9E+04	2.1E+02	3.5E-01	2.5E-04	2.1E-04	4.0E-05	1.8E-06	1.1E-06	1.2E-07	3.3E-07	2.4E-06
						2.5E+06	8.1E+04	5.0E+04	6.5E-01	4.1E-02	1.1E-04	2.2E-04	4.7E-05	5.0E-06	5.7E-06	5.1E-06	3.8E-05
SEQ-02-1		0.619	2.2E+04	1.6E+04	10.	3.0E+03	4.7E+04	3.0E+00	2.9E-03	5.5E-08	2.2E-08	2.6E-08	6.6E-09	1.2E-09	3.0E-10	1.2E-09	7.0E-09
						3.8E+06	4.7E+04	4.1E+02	1.0E+00	5.9E-02	1.3E-04	2.6E-04	3.4E-05	7.6E-06	5.7E-06	3.7E-06	2.7E-05
SEQ-02-2		0.360	2.2E+04	-2.6E+03	10.	2.3E+06	2.8E+04	4.7E+02	9.2E-01	6.8E-04	5.9E-04	1.1E-04	4.9E-06	3.1E-06	3.4E-07	9.0E-07	6.8E-06
						3.2E+05	1.4E+05	1.4E+05	8.2E-02	1.3E-02	7.8E-05	1.7E-04	7.2E-05	8.2E-07	6.2E-06	7.6E-06	5.9E-05
SEQ-02-3		0.021	1.3E+03	-5.9E+03	0.	1.9E+06	3.7E+03	1.8E+03	1.0E+00	1.7E-05	2.4E-05	1.9E-06	1.1E-07	9.5E-08	6.3E-09	1.7E-08	1.6E-07
						1.7E+05	1.0E+04	2.2E+04	1.4E-04	5.8E-03	2.7E-04	2.7E-05	9.8E-06	9.7E-09	1.9E-07	1.5E-07	5.9E-06
SEQ-03	5.6E-07		2.2E+04	5.1E+03	10.	6.9E+06	3.6E+04	2.3E+02	5.5E-01	2.4E-03	2.0E-03	1.9E-04	1.6E-05	5.1E-06	9.3E-07	3.0E-06	2.1E-05
						2.8E+06	9.0E+04	6.7E+04	4.5E-01	3.6E-02	1.7E-03	1.1E-03	1.6E-04	4.9E-06	3.0E-05	2.3E-05	1.3E-04
SEQ-03-1		0.409	2.3E+04	1.6E+04	10.	1.9E+04	4.7E+04	1.9E+01	1.8E-02	4.6E-05	3.0E-05	3.6E-06	3.0E-08	1.5E-12	4.4E-13	4.7E-13	5.1E-08
						5.1E+06	4.7E+04	4.9E+02	9.8E-01	6.7E-02	2.4E-03	1.3E-03	1.4E-04	8.2E-06	4.4E-05	2.4E-05	1.2E-04
SEQ-03-2		0.591	2.2E+04	-2.6E+03	10.	1.2E+07	2.8E+04	3.8E+02	9.2E-01	4.1E-03	3.4E-03	3.3E-04	2.8E-05	8.6E-06	1.6E-06	5.1E-06	3.5E-05
						1.1E+06	1.2E+05	1.1E+05	7.7E-02	1.5E-02	1.2E-03	9.5E-04	1.7E-04	2.6E-06	2.0E-05	2.2E-05	1.4E-04
SEQ-03-3		0.000															
SEQ-04	1.3E-07		1.3E+03	-5.8E+03	0.	1.8E+06	3.7E+03	1.8E+03	9.8E-01	3.9E-03	3.8E-03	7.4E-04	1.8E-04	4.0E-05	9.2E-06	3.4E-05	2.3E-04
						1.7E+05	1.0E+04	2.2E+04	2.2E-02	5.2E-02	4.6E-03	3.8E-03	1.4E-04	3.3E-05	1.1E-05	9.8E-06	1.2E-04
SEQ-04-1		0.002	2.2E+04	1.6E+04	10.	3.1E+03	4.7E+04	3.1E+00	3.0E-03	2.4E-05	2.8E-05	5.6E-07	1.1E-08	3.6E-09	2.8E-10	3.8E-10	1.9E-08
						6.5E+05	4.7E+04	1.1E+04	1.0E+00	1.3E-02	8.6E-03	2.1E-01	8.9E-03	1.5E-02	1.7E-03	1.8E-03	8.5E-03
SEQ-04-2		0.000	1.3E+04	-1.1E+03	10.	1.0E+06	2.0E+04	1.0E+03	8.0E-01	5.6E-03	3.2E-03	8.2E-04	3.9E-07	4.1E-11	4.1E-11	4.3E-11	3.1E-05
						0.0E+00	2.1E+04	2.0E+02	2.0E-01	6.5E-02	4.0E-04	1.3E-04	1.7E-05	8.8E-07	3.0E-06	3.1E-06	1.8E-05
SEQ-04-3		0.998	1.3E+03	-5.9E+03	0.	1.9E+06	3.7E+03	1.8E+03	9.8E-01	3.9E-03	3.8E-03	7.4E-04	1.8E-04	4.0E-05	9.3E-06	3.4E-05	2.3E-04
						1.7E+05	1.0E+04	2.2E+04	2.0E-02	5.2E-02	4.6E-03	3.3E-03	1.2E-04	5.3E-07	6.9E-06	5.7E-06	1.0E-04
SEQ-05	2.9E-07		2.2E+04	3.8E+03	10.	6.1E+06	3.4E+04	6.2E+02	5.6E-01	1.1E-02	9.2E-03	6.1E-03	1.0E-03	1.7E-03	4.1E-04	4.5E-04	1.4E-03
						2.3E+06	1.9E+05	1.7E+05	3.9E-01	4.9E-02	1.1E-02	1.3E-02	1.4E-03	4.3E-04	2.2E-04	2.4E-04	1.2E-03
SEQ-05-1		0.328	2.2E+04	1.6E+04	10.	3.2E+01	4.7E+04	3.2E-02	3.0E-05	8.6E-07	6.6E-07	1.2E-07	1.2E-09	8.8E-11	3.0E-11	3.1E-11	2.6E-09
						5.1E+06	4.7E+04	2.1E+02	1.0E+00	8.3E-02	2.4E-02	2.5E-02	4.6E-04	3.4E-05	7.2E-05	7.3E-05	4.2E-04
SEQ-05-2		0.672	2.1E+04	-2.4E+03	10.	9.1E+06	2.7E+04	9.2E+02	8.3E-01	1.7E-02	1.4E-02	9.0E-03	1.5E-03	2.5E-03	6.1E-04	6.8E-04	2.1E-03
						9.3E+05	2.5E+05	2.5E+05	9.4E-02	3.3E-02	4.2E-03	6.7E-03	1.8E-03	6.2E-04	3.0E-04	3.2E-04	1.6E-03
SEQ-05-3		0.000															

Table 3.4-4 (continued)

Source Term	Freq. (1/yr)	Cond. Prob.	Warn (s)	dEvac (s)	Elev (m)	Energy (W)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
SEQ-06	9.1E-07		2.3E+04	4.5E+03	10.	2.9E+06	3.5E+04	5.9E+02	5.3E-01	9.6E-03	7.6E-03	1.3E-03	5.2E-05	4.0E-05	6.6E-06	9.6E-06	7.8E-05
						3.1E+06	2.2E+05	1.9E+05	4.3E-01	3.2E-02	5.4E-03	6.4E-03	3.0E-04	7.1E-06	1.7E-05	2.8E-05	2.8E-04
SEQ-06-1		0.391	2.4E+04	1.5E+04	10.	9.9E+04	4.7E+04	9.9E+01	9.1E-02	1.4E-03	9.1E-04	7.9E-05	1.2E-06	2.1E-07	1.9E-08	4.9E-08	2.7E-06
						6.8E+06	4.7E+04	1.3E+03	9.1E-01	6.4E-02	9.5E-03	7.6E-03	3.4E-04	1.4E-05	2.1E-05	4.8E-05	2.9E-04
SEQ-06-2		0.609	2.2E+04	-2.5E+03	10.	4.7E+06	2.8E+04	9.1E+02	8.1E-01	1.5E-02	1.2E-02	2.1E-03	8.4E-05	6.5E-05	1.1E-05	1.6E-05	1.3E-04
						6.5E+05	3.2E+05	3.2E+05	1.3E-01	1.2E-02	2.7E-03	5.6E-03	2.8E-04	2.5E-06	1.4E-05	1.6E-05	2.8E-04
SEQ-06-3		0.000															
SEQ-07	9.7E-08		1.3E+03	-5.9E+03	0.	1.9E+06	3.6E+03	1.8E+03	1.0E+00	4.7E-02	4.6E-02	4.2E-03	4.4E-04	1.6E-04	1.8E-05	6.1E-05	5.6E-04
						1.7E+05	1.0E+04	2.2E+04	2.3E-03	1.1E-01	3.9E-02	5.2E-02	1.3E-02	1.1E-04	1.8E-03	2.0E-03	1.0E-02
SEQ-07-1		0.000															
SEQ-07-2		0.000															
SEQ-07-3		1.000	1.3E+03	-5.9E+03	0.	1.9E+06	3.6E+03	1.8E+03	1.0E+00	4.7E-02	4.6E-02	4.2E-03	4.4E-04	1.6E-04	1.8E-05	6.1E-05	5.6E-04
						1.7E+05	1.0E+04	2.2E+04	2.3E-03	1.1E-01	3.9E-02	5.2E-02	1.3E-02	1.1E-04	1.8E-03	2.0E-03	1.0E-02
SEQ-08	3.6E-07		1.5E+04	-3.32E+03	7.	1.1E+07	2.0E+04	1.2E+03	9.1E-01	5.5E-02	5.1E-02	1.8E-02	2.8E-03	3.2E-03	7.2E-04	8.4E-04	3.9E-03
						8.6E+05	1.2E+05	1.2E+05	5.7E-02	4.7E-02	8.3E-03	2.6E-02	1.2E-02	8.1E-04	1.6E-03	1.6E-03	1.1E-02
SEQ-08-1		0.035	3.5E+04	7.7E+03	10.	9.0E+05	5.1E+04	9.0E+02	8.5E-01	4.8E-02	3.9E-02	1.2E-02	2.4E-04	2.1E-04	1.5E-05	3.7E-05	4.9E-04
						3.6E+06	5.1E+04	9.7E+03	1.5E-01	4.1E-02	8.5E-03	7.5E-02	7.3E-02	1.4E-03	1.0E-02	1.1E-02	5.9E-02
SEQ-08-2		0.619	2.1E+04	-2.4E+03	10.	1.6E+07	2.7E+04	8.7E+02	8.7E-01	6.5E-02	6.0E-02	2.7E-02	4.4E-03	5.1E-03	1.2E-03	1.3E-03	6.0E-03
						1.1E+06	1.8E+05	1.8E+05	7.9E-02	5.7E-02	1.2E-02	2.0E-02	5.8E-03	1.0E-03	6.0E-04	5.9E-04	5.0E-03
SEQ-08-3		0.346	1.3E+03	-5.9E+03	0.	2.0E+06	3.6E+03	1.8E+03	9.9E-01	3.8E-02	3.8E-02	3.1E-03	2.3E-04	8.2E-05	1.0E-05	3.3E-05	3.2E-04
						1.7E+05	1.0E+04	2.2E+04	8.5E-03	3.0E-02	1.8E-03	3.1E-02	1.8E-02	3.9E-04	2.6E-03	2.4E-03	1.6E-02
SEQ-09	9.8E-07		2.4E+04	5.0E+02	10.	7.1E+06	3.3E+04	7.5E+02	7.8E-01	5.4E-02	4.6E-02	1.7E-02	2.2E-03	3.4E-03	8.0E-04	9.9E-04	3.0E-03
						1.7E+06	3.8E+05	3.7E+05	1.1E-01	2.6E-02	8.2E-03	5.8E-03	1.1E-03	8.3E-05	1.0E-04	1.2E-04	8.5E-04
SEQ-09-1		0.245	3.2E+04	9.4E+03	10.	7.2E+05	5.0E+04	7.2E+02	5.6E-01	6.6E-02	5.3E-02	1.2E-02	1.3E-03	2.7E-04	9.9E-05	4.2E-04	1.5E-03
						3.4E+06	5.1E+04	1.2E+04	3.2E-01	4.4E-02	1.8E-02	7.3E-03	9.3E-04	7.3E-05	1.3E-04	2.0E-04	8.7E-04
SEQ-09-2		0.755	2.1E+04	-2.4E+03	10.	9.1E+06	2.7E+04	7.6E+02	8.5E-01	5.0E-02	4.4E-02	1.8E-02	2.4E-03	4.5E-03	1.0E-03	1.2E-03	3.4E-03
						1.2E+06	4.9E+05	4.9E+05	4.6E-02	2.1E-02	5.1E-03	5.4E-03	1.1E-03	8.6E-05	9.3E-05	9.7E-05	8.5E-04
SEQ-09-3		0.000															
SEQ-10	7.3E-08		2.8E+04	6.2E+03	10.	3.3E+06	4.2E+04	1.4E+03	6.0E-01	3.3E-02	2.5E-02	5.6E-03	1.0E-04	5.1E-05	3.1E-06	6.0E-06	1.9E-04
						4.1E+06	3.3E+05	3.0E+05	2.7E-01	4.2E-02	2.1E-02	1.1E-02	2.6E-04	2.9E-06	6.9E-06	4.6E-06	1.8E-04
SEQ-10-1		0.692	3.1E+04	1.0E+04	10.	6.6E+05	5.0E+04	6.6E+02	5.1E-01	4.0E-02	3.0E-02	6.3E-03	1.2E-04	6.2E-05	3.5E-06	6.9E-06	2.2E-04
						5.5E+06	5.0E+04	1.3E+04	3.7E-01	2.1E-02	1.4E-02	7.6E-03	3.8E-04	4.2E-06	9.9E-06	6.5E-06	2.6E-04
SEQ-10-2		0.308	2.0E+04	-2.3E+03	10.	9.2E+06	2.6E+04	3.1E+03	8.0E-01	1.7E-02	1.4E-02	4.0E-03	6.9E-05	2.8E-05	2.2E-06	4.1E-06	1.3E-04
						1.0E+06	9.5E+05	9.5E+05	2.9E-02	8.7E-02	3.6E-02	1.9E-02	7.3E-06	2.1E-07	2.3E-07	3.2E-07	6.3E-06
SEQ-10-3		0.000															

Table 3.4-4 (continued)

Source Term	Freq. (1/yr)	Cond. Prob.	Warn (s)	dEvac (s)	Elev (m)	Energy (W)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
SEQ-11	5.1E-07		1.2E+04	-3.34E+03	6.	8.2E+06	1.7E+04	1.5E+03	8.8E-01	2.0E-01	2.0E-01	8.1E-02	2.3E-02	6.2E-03	2.3E-03	1.2E-02	2.5E-02
						7.3E+05	1.6E+05	1.6E+05	4.5E-02	5.8E-02	2.4E-02	8.1E-02	3.9E-02	9.1E-04	5.0E-03	5.1E-03	3.2E-02
SEQ-11-1		0.021	3.5E+04	7.5E+03	10.	9.2E+05	5.1E+04	9.2E+02	8.7E-01	2.9E-01	2.7E-01	1.6E-01	7.1E-02	1.7E-02	3.3E-03	1.2E-02	8.0E-02
						2.9E+06	5.2E+04	1.6E+04	1.3E-01	3.8E-02	3.0E-02	3.1E-02	2.3E-02	2.3E-03	3.6E-03	4.6E-03	2.2E-02
SEQ-11-2		0.570	1.9E+04	-2.1E+03	10.	1.3E+07	2.5E+04	1.2E+03	8.1E-01	2.2E-01	2.2E-01	1.0E-01	8.2E-03	4.7E-03	9.3E-04	1.8E-03	1.0E-02
						1.0E+06	2.7E+05	2.7E+05	7.0E-02	6.5E-02	3.6E-02	1.1E-01	5.6E-02	1.3E-03	7.2E-03	7.3E-03	4.5E-02
SEQ-11-3		0.409	1.3E+03	-5.9E+03	0.	2.3E+06	3.7E+03	1.8E+03	1.0E+00	1.7E-01	1.7E-01	4.9E-02	4.1E-02	7.6E-03	4.2E-03	2.7E-02	4.2E-02
						1.7E+05	1.0E+04	2.2E+04	4.9E-03	5.0E-02	6.0E-03	4.9E-02	1.7E-02	2.8E-04	2.2E-03	2.2E-03	1.4E-02
SEQ-12	1.4E-06		2.4E+04	-6.2E+01	10.	8.8E+06	3.2E+04	8.9E+02	8.2E-01	1.5E-01	1.5E-01	5.5E-02	4.5E-03	4.0E-03	8.2E-04	1.2E-03	5.9E-03
						1.2E+06	3.1E+05	3.0E+05	9.7E-02	4.0E-02	2.3E-02	4.4E-02	1.3E-02	2.8E-04	1.1E-03	9.8E-04	1.1E-02
SEQ-12-1		0.221	3.4E+04	8.0E+03	10.	8.6E+05	5.0E+04	8.6E+02	7.7E-01	2.1E-01	1.8E-01	5.9E-02	6.8E-03	2.7E-03	4.2E-04	1.1E-03	8.3E-03
						1.8E+06	5.1E+04	1.4E+04	1.8E-01	5.2E-02	3.4E-02	1.8E-02	2.4E-03	2.3E-04	1.7E-04	3.3E-04	2.2E-03
SEQ-12-2		0.779	2.1E+04	-2.4E+03	10.	1.1E+07	2.7E+04	9.0E+02	8.3E-01	1.4E-01	1.4E-01	5.4E-02	3.8E-03	4.4E-03	9.4E-04	1.3E-03	5.2E-03
						9.9E+05	3.8E+05	3.8E+05	7.4E-02	3.7E-02	2.0E-02	5.1E-02	1.6E-02	2.9E-04	1.4E-03	1.2E-03	1.3E-02
SEQ-12-3		0.000	1.3E+03	-5.9E+03	0.	3.7E+06	3.7E+03	1.8E+03	1.0E+00	9.2E-02	1.1E-01	1.1E-02	4.1E-04	2.7E-04	1.6E-05	3.1E-05	7.3E-04
						1.7E+05	1.0E+04	2.2E+04	2.9E-03	1.8E-02	3.9E-04	4.2E-03	9.1E-06	1.7E-10	1.5E-09	1.5E-08	2.3E-05
SEQ-13	1.8E-07		3.0E+04	2.9E+03	10.	4.0E+06	4.1E+04	7.0E+02	8.5E-01	9.7E-02	9.1E-02	2.0E-02	6.3E-03	9.8E-04	2.8E-04	1.0E-03	6.9E-03
						9.3E+05	3.3E+05	3.1E+05	1.2E-01	3.0E-02	1.6E-02	7.8E-03	7.4E-04	5.1E-05	2.3E-05	6.6E-05	7.3E-04
SEQ-13-1		0.576	3.6E+04	7.0E+03	10.	9.7E+05	5.1E+04	9.7E+02	8.7E-01	1.1E-01	9.4E-02	3.3E-02	1.1E-02	1.6E-03	4.7E-04	1.8E-03	1.2E-02
						4.5E+05	5.2E+04	1.4E+04	9.3E-02	2.2E-02	1.0E-02	9.4E-03	1.2E-03	8.9E-05	4.0E-05	1.1E-04	1.2E-03
SEQ-13-2		0.424	2.2E+04	-2.6E+03	10.	8.2E+06	2.8E+04	3.4E+02	8.3E-01	7.9E-02	8.5E-02	2.8E-03	1.3E-04	1.2E-04	2.5E-05	3.5E-05	1.8E-04
						1.6E+06	7.1E+05	7.0E+05	1.7E-01	4.1E-02	2.3E-02	5.6E-03	5.5E-05	4.1E-07	7.9E-07	2.1E-06	5.3E-05
SEQ-13-3		0.000															
SEQ-14	1.3E-07		9.3E+03	-4.1E+03	4.	9.7E+06	1.4E+04	1.4E+03	9.7E-01	5.9E-01	5.8E-01	2.1E-01	1.1E-01	2.2E-02	1.1E-02	6.9E-02	1.1E-01
						5.3E+05	7.4E+04	7.7E+04	1.9E-02	3.0E-02	1.1E-02	2.1E-01	9.7E-02	1.9E-03	7.0E-03	9.3E-03	8.1E-02
SEQ-14-1		0.039	3.6E+04	6.7E+03	10.	1.0E+06	5.1E+04	1.0E+03	9.5E-01	6.2E-01	6.1E-01	4.1E-01	3.8E-01	6.5E-02	3.6E-02	2.1E-01	3.8E-01
						0.0E+00	5.2E+04	1.5E+04	4.8E-02	3.9E-02	3.1E-02	1.0E-01	9.3E-02	4.1E-03	1.1E-02	2.1E-02	7.9E-02
SEQ-14-2		0.359	2.0E+04	-2.2E+03	10.	2.1E+07	2.6E+04	8.1E+02	9.5E-01	7.7E-01	7.4E-01	3.2E-01	4.3E-02	1.3E-02	3.9E-03	2.3E-02	5.2E-02
						1.2E+06	1.8E+05	1.8E+05	1.8E-02	2.2E-02	2.0E-02	2.5E-01	1.4E-01	1.7E-03	1.3E-02	1.4E-02	1.1E-01
SEQ-14-3		0.602	1.2E+03	-5.9E+03	0.	3.5E+06	3.7E+03	1.8E+03	9.8E-01	4.8E-01	4.9E-01	1.2E-01	1.3E-01	2.4E-02	1.4E-02	8.6E-02	1.3E-01
						1.7E+05	1.0E+04	2.2E+04	1.8E-02	3.4E-02	4.6E-03	1.9E-01	7.4E-02	1.8E-03	3.2E-03	5.4E-03	6.3E-02
SEQ-15	9.6E-07		3.0E+04	3.2E+03	10.	8.8E+06	4.1E+04	8.2E+02	9.4E-01	4.8E-01	4.7E-01	1.5E-01	1.4E-02	6.8E-03	1.1E-03	2.5E-03	1.9E-02
						5.1E+05	7.6E+04	5.0E+04	4.7E-02	5.1E-02	3.0E-02	7.0E-02	1.2E-02	4.7E-04	1.0E-03	1.0E-03	1.1E-02
SEQ-15-1		0.610	3.6E+04	6.7E+03	10.	1.0E+06	5.1E+04	1.0E+03	9.4E-01	4.8E-01	4.5E-01	1.8E-01	1.9E-02	7.9E-03	1.2E-03	3.3E-03	2.7E-02
						5.1E+04	5.2E+04	1.4E+04	5.0E-02	3.3E-02	2.3E-02	2.5E-02	5.2E-03	5.9E-04	7.6E-04	9.7E-04	5.1E-03
SEQ-15-2		0.384	2.0E+04	-2.3E+03	10.	2.1E+07	2.6E+04	5.3E+02	9.4E-01	5.0E-01	5.0E-01	9.6E-02	4.3E-03	5.2E-03	1.0E-03	1.2E-03	6.5E-03
						1.3E+06	1.2E+05	1.1E+05	4.3E-02	7.8E-02	4.0E-02	1.4E-01	2.3E-02	2.8E-04	1.5E-03	1.2E-03	1.9E-02
SEQ-15-3		0.006	1.3E+03	-5.9E+03	0.	3.7E+06	3.7E+03	1.8E+03	9.8E-01	3.7E-01	3.8E-01	2.4E-02	1.4E-03	6.2E-04	6.9E-05	2.0E-04	2.2E-03
						1.7E+05	1.0E+04	2.2E+04	1.6E-02	4.0E-02	6.3E-03	1.4E-01	2.3E-02	2.9E-05	1.1E-03	1.3E-03	1.8E-02

Table 3.4-4 (continued)

Source Term	Freq. (1/yr)	Cond. Prob.	Warn (s)	dEvac (s)	Elev (m)	Energy (W)	Start (s)	Dur (s)	Release Fractions NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
SEQ-16	2.2E-05		2.2E+04	1.6E+04	0.	3.5E-03	4.7E+04	7.4E-04	1.9E-08	1.6E-15	1.2E-15	2.5E-16	2.8E-18	2.4E-22	1.9E-22	1.9E-22	4.6E-18
						1.1E-02	4.7E+04	8.6E+04	4.3E-03	1.3E-05	2.9E-09	2.1E-09	6.0E-10	1.9E-11	6.7E-11	6.5E-11	5.1E-10
SEQ-16-1		1.000	2.2E+04	1.6E+04	0.	0.0E+00	4.7E+04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
						0.0E+00	4.7E+04	8.6E+04	4.3E-03	1.3E-05	2.9E-09	2.1E-09	6.0E-10	1.9E-11	6.7E-11	6.5E-11	5.1E-10
SEQ-16-2		0.000	2.2E+04	-2.6E+03	10.	5.2E+04	2.8E+04	1.1E+04	2.7E-01	2.3E-08	1.8E-08	3.7E-09	4.1E-11	3.6E-15	2.8E-15	2.8E-15	6.7E-11
						1.6E+05	1.0E+06	1.0E+06	0.0E+00	7.3E-08	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
SEQ-16-3		0.000															
SEQ-17	1.5E-05		2.2E+04	1.6E+04	0.	7.5E+02	4.7E+04	7.7E+00	8.6E-04	9.5E-09	2.8E-09	2.5E-09	2.3E-10	5.9E-11	1.1E-11	4.2E-11	2.7E-10
						6.9E+04	5.0E+04	8.4E+04	3.7E-02	1.8E-04	7.3E-08	3.9E-08	4.4E-09	2.2E-10	4.8E-10	6.6E-10	3.8E-09
SEQ-17-1		0.999	2.2E+04	1.6E+04	0.	5.2E+02	4.7E+04	5.2E-01	4.4E-04	4.5E-09	2.4E-10	2.0E-09	1.8E-10	4.6E-11	8.3E-12	3.1E-11	2.1E-10
						6.9E+04	5.0E+04	8.4E+04	3.7E-02	1.8E-04	7.1E-08	3.9E-08	4.4E-09	2.2E-10	4.8E-10	6.6E-10	3.8E-09
SEQ-17-2		0.001	2.2E+04	-2.6E+03	10.	3.1E+05	2.8E+04	9.7E+03	5.7E-01	6.8E-06	3.5E-06	6.9E-07	6.5E-08	1.7E-08	3.9E-09	1.5E-08	8.1E-08
						3.2E+05	9.9E+05	9.9E+05	1.8E-01	8.5E-04	2.7E-06	4.9E-08	3.4E-09	3.0E-10	4.1E-10	6.8E-10	3.2E-09
SEQ-17-3		0.000															
SEQ-18	1.0E-05		2.2E+04	1.4E+04	10.	1.7E+05	4.5E+04	6.4E+02	1.0E-01	4.3E-04	3.2E-04	8.4E-05	1.8E-06	4.7E-07	9.4E-08	3.6E-07	2.8E-06
						2.3E+06	2.2E+05	1.1E+05	8.8E-01	2.5E-02	3.9E-04	2.1E-04	8.6E-06	8.6E-07	6.1E-07	5.8E-07	6.8E-06
SEQ-18-1		0.888	2.3E+04	1.6E+04	10.	1.8E+04	4.7E+04	1.8E+01	1.0E-02	1.8E-04	1.3E-04	1.4E-05	2.4E-07	6.8E-08	5.2E-09	1.3E-08	4.4E-07
						2.5E+06	1.2E+05	4.1E+03	9.8E-01	2.8E-02	3.1E-04	1.8E-04	9.5E-06	9.3E-07	6.6E-07	6.1E-07	7.5E-06
SEQ-18-2		0.112	2.2E+04	-2.5E+03	10.	1.4E+06	2.8E+04	5.6E+03	8.2E-01	2.4E-03	1.9E-03	6.4E-04	1.5E-05	3.7E-06	8.0E-07	3.1E-06	2.1E-05
						5.9E+05	9.7E+05	9.7E+05	9.7E-02	7.0E-03	1.1E-03	4.6E-04	1.5E-06	2.8E-07	1.7E-07	3.1E-07	1.4E-06
SEQ-18-3		0.000															

3.5 Insights from the Source Term Analysis

The range in the release fractions calculated for similar accidents is large--typically two orders of magnitude for the more volatile radionuclide classes and four orders of magnitude of more for the less volatile radionuclides. While iodine and cesium release fractions exceeding 0.10 are possible for many different types of accidents, they are most likely for bypass events. For containment bypass sequences, a large release is virtually assured because there are no mechanisms by which the releases can be mitigated. For accident sequences in which the containment is not bypassed but fails, the potential for mitigation of the releases exists, particularly for the late failures. The result is that the range of release fractions for non-bypass accidents with containment failures is extended beyond that for bypass accidents in the direction of lower releases.

The timing of evacuation relative to the release of the radionuclides is important for evaluating the early consequences of the releases. For Event V, evacuation starts more than 1 h after the release has begun. For containment failures at VB and SGTRs without stuck-open secondary SRVs, the evacuation occurs between 30 min before and 1 h after the release begins. For SGTRs with stuck-open secondary SRVs and late failures of containment, the evacuation occurs at times much greater than 30 min before the release begins.

3.6 References

1. H.-N Jow, J. L. Sprung, J. A. Rollistin, and D. I. Chanin, "MELCOR Accident Consequence Code System (MAACS): Model Description," NUREG/CR-4691, SAND86-1562, Volume 2, Sandia National Laboratories, February 1990.

2. R. L. Iman, J. C. Helton, and J. D. Johnson, "A User's Guide for PARTITION: A Program for Defining the Source Term/Consequence Analysis Interfaces in the NUREG-1150 Probabilistic Risk Assessments," NUREG/CR-5253, SAND88-2940, Sandia National Laboratories, May 1990.

4. CONSEQUENCE ANALYSIS

Offsite consequences were calculated with MACCS[1,2,3] for each of the source term groups defined in the partitioning process. This code has been used for some time and will not be described in detail. Although the variables thought to be the largest contributors to the uncertainty in risk were sampled from distributions in the accident frequency analysis, the accident progression analysis, and the source term analysis, there was no analogous treatment of uncertainties in the consequence analysis. Variability in the weather was fully accounted for, but the uncertainty in other parameters such as the dry deposition speed or the evacuation rate was not considered.

4.1 Description of the Consequence Analysis

Offsite consequences were calculated with MACCS for each of the source term groups defined in the partitioning process. MACCS tracks the dispersion of the radioactive material in the atmosphere from the plant and computes deposition on the ground. MACCS then calculates the effects of this radioactivity on the population and the environment. Doses and the ensuing health effects from 60 radionuclides are computed for the following pathways: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, inhalation of resuspended ground contamination, ingestion of contaminated water, and ingestion of contaminated food.

MACCS treats atmospheric dispersion by the use of multiple, straight-line Gaussian plumes. Each plume can have a different direction, duration, and initial radionuclide concentration. Crosswind dispersion is treated by a multi-step function. Dry deposition and wet deposition are treated as independent processes. The weather variability is treated by means of a stratified sampling process.

For early exposure, the following pathways are considered: immersion or cloudshine, inhalation from the plume, groundshine, deposition on the skin, and inhalation of resuspended ground contamination. Skin deposition and inhalation of resuspended ground contamination have generally not been considered in previous consequence models. For the long-term exposure, MACCS considers the following four pathways: groundshine, inhalation of resuspended ground contamination, ingestion of contaminated water, and ingestion of contaminated food. The direct exposure pathways (groundshine and inhalation of resuspended ground contamination) produce doses in the population living in the area surrounding the plant. The indirect exposure pathways (ingestion of contaminated water and food) produce doses in those who ingest food or water emanating from the area around the accident site. The contamination of water bodies is estimated for the washoff of land-deposited material as well as direct deposition. The food pathway model includes direct deposition onto crop and uptake from the soil. The health effects models link the dose received by an organ to predicted morbidity or mortality. The models used in MACCS calculate both short-term and long-term effects for a number of organs.

Both short-term and long-term mitigative measures are modeled in MACCS. Short-term actions include evacuation, sheltering, and emergency relocation out of the emergency planing zone. Long-term actions include later relocation and restrictions on land use and crop disposition. Relocation and land decontamination, interdiction, and condemnation are based on projected long-term doses from groundshine and inhalation of resuspended radioactivity. The disposal of agricultural products is based on the products' contamination levels and the removal of farmland from crop production is based on ground contamination criteria.

The MACCS consequence model calculates a large number of different consequence measures. Results for the following six consequence measures are given in this report: early fatalities, total latent cancer fatalities, population dose within 50 miles, population dose for the entire region, early fatality risk within 1 mile, and latent cancer fatality risk within 10 miles. These consequence measures are described in Table 4.1-1. For the analyses performed for NUREG-1150, 99.5 percent of the population evacuates and 0.5 percent of the population does not evacuate and continues normal activity. Details of the methods used to incorporate the consequence results for the source term groups into the integrated risk analysis are given in Volume 1 of this report.

4.2 MACCS Input for Sequoyah

The values of most MACCS input parameters (e.g., aerosol dry deposition velocity, health effects model parameter values, food pathway transfer factors) do not depend on site characteristics. For those parameters that depend on site characteristics (e.g., evacuation speed, shielding factors, farmland usage), the methods used to calculate the parameters are essentially the same for all sites. Because the methods used to develop input parameter values for the MACCS NUREG-1150 analyses and the parameter values developed using those methods are documented in Volume 2, Part 7 of this report, only a small portion of the MACCS input is presented here.

Table 4.2-1 lists the MACCS input parameters that are highly dependent upon site location and presents the values of these parameters used in the MACCS calculations for the Sequoyah site. The evacuation delay period begins when general emergency conditions occur and ends when the general public starts to evacuate. Nonfarm wealth includes personnel, business, and public property. The farmland fractions do not add to one because not all farmland is under cultivation. In addition to the site specific data presented in Table 4.2-1, the Sequoyah MACCS calculations used one year of meteorological data from the Sequoyah site and regional population data developed from the 1980 census tapes. The following table gives the population within certain distances of the plant as summarized from the MACCS demographic input.

Distance from Plant (km)	Distance from Plant (miles)	Population
1.6	1.0	213
4.8	3.0	2432
16.1	10.0	38,972
48.3	30.0	514,226
160.9	100.0	3,221,558
563.3	350.0	36,593,188
1609.3	1000.0	180,568,384

Table 4.2-2 lists the shielding parameters used in this analysis.

Table 4.1-1
Definition of Consequence Analysis Results

Variable	Definition
Early fatalities	Number of fatalities within 1 yr of the accident.
Total latent cancer fatalities	Number of latent cancer fatalities due to both early and chronic exposure.
Population dose within 50 miles	Population dose, expressed in effective dose equivalents for whole body exposure (person-rem) due to early and chronic exposure pathways within 50 miles of the reactor. Due to the nature of the chronic pathways models, the actual exposure due to food and water consumption may take place beyond 50 miles.
Population dose within entire region	Population dose, expressed in effective dose equivalents for whole body exposure (person-rem) due to early and chronic exposure pathways within the entire region.
Individual early fatality risk within one mile	The probability of dying within 1 yr for an individual within one mile of the exclusion boundary [i.e., Σ (ef/pop)p, where ef is the number of early fatalities, pop is the population size, p is the weather condition probability, and the summation is over all weather conditions].
Individual latent cancer risk within 10 miles	The probability of dying from cancer due to the accident for an individual within 10 miles of the plant [i.e., Σ (cf/pop)p, where cf is the number of cancer fatalities due to direct exposure in the resident population, pop is the population size, p is the weather condition probability, and the summation is over all weather conditions; chronic exposure does not include ingestion, but does include integrated groundshine and inhalation exposure from $t = 0$ to $t = \infty$].

Table 4.2-1
Site Specific Input Data for Sequoyah MACCS Calculations

Parameter	
Reactor Power Level (MWt)	3423
Containment Height (m)	40
Containment Width (m)	40
Exclusion Zone Distance (km)	0.585
Evacuation Delay (h)	2.3
Evacuation Speed (m/s)	1.8
Farmland Fractions by Crop Categories	
Pasture	0.69
Stored Forage	0.006
Grains	0.16
Green Leafy Vegetables	0.0007
Legumes and Seeds	0.15
Roots and Tubers	0.001
Other Food Crops	0.005
Non-Farm Wealth ($/person)	66,000
Farm Wealth	
Value ($/hectare)	1855
Fraction in Improvements	0.27

Table 4.2-2
Shielding Factors for Sequoyah MACCS Calculations

	Population Response		
Radiation Pathway	Evacuate	Normal Activity	Take Shelter
Internal Initiators			
Cloudshine	1.0	0.75	0.65
Groundshine	0.5	0.33	0.20
Inhalation	1.0	0.41	0.33
Skin	1.0	0.41	0.33

4.3 Results of MACCS Consequence Calculations

The results in this section are conditional on the occurrence of a release. That is, given that a release takes place, with release fractions and other characteristics as defined by one of the source term groups, then the consequences reported in this section are calculated. The tables and figures in this section contain no information about the frequency with which these consequences may be expected. Information about the frequencies of consequences of various magnitudes is contained in the risk results (Chapter 5).

4.3.1 Results for Internal Initiators

The integration of the NUREG-1150 probabilistic risk assessments uses the results of the MACCS consequence calculations in two forms. In the first form, a single mean (over weather variation) result is reported for each consequence measure. This produces an nSTG x nC matrix of mean consequence measures, where nSTG is the number of source term groups and nC is the number of consequence measures under consideration. For internal initiators at Sequoyah, nSTG = 55 and nC = 6. The resultant 55 x 6 matrix of mean consequence measures is shown in Table 4.3-1. The source terms that give rise to these mean consequence measures are given in Table 3.4-4. Some of the cases indicated in Table 3.4-4 have a zero frequency, and no consequence results are reported for these cases in Table 4.3-1. The mean consequence measures in Table 4.3-1 are used by PRAMIS[4] and RISQUE in the calculation of the mean risk results for internal initiators at Sequoyah. An early fatality consequence value less than 1.0 may be interpreted as the probability of obtaining one death. The population dose is the effective dose equivalent to the whole body for the population in the region indicated.

Table C.1-1 in Appendix C provides a breakdown of mean consequence results between individuals who evacuate, continue normal activities, and actively shelter; information on the division of results between early and chronic exposure is also given. In addition to the six consequence measures reported here, Table C.1-1 contains results for early injuries (prodromal vomiting), economic cost, and individual early fatality risk at 1 mile. Note that the individual early fatality risk at one mile is distinct from individual early fatality risk within one mile. The risk at one mile (listed only in Appendix C) is for a hypothetical individual at that distance. The risk within one mile (reported in the text) uses the actual residence distances for all people living within one mile of the plant. Only if there are no people living one mile of the plant is the calculation made assuming that a hypothetical person is located exactly one mile from the plant.

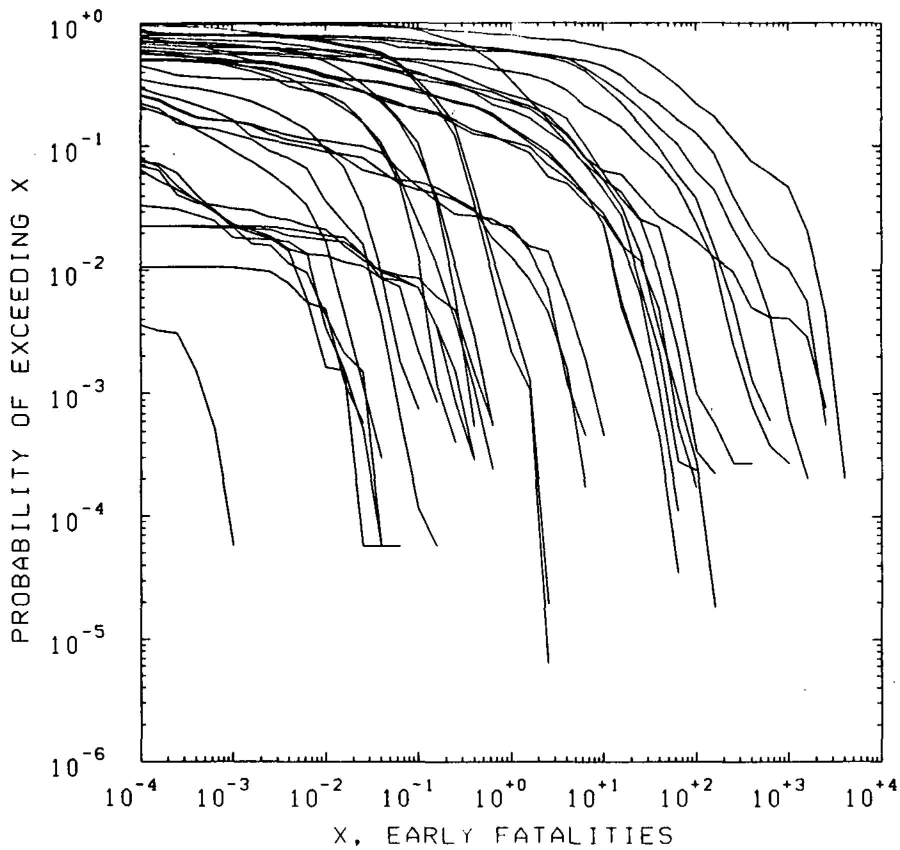
In the second form, a complementary cumulative distribution function (CCDF) is used for each consequence measure. Conditional on the occurrence of a source term, each of these CCDFs gives the probability that individual consequence values will be exceeded due to the uncertainty in the weather conditions at the time of an accident. These CCDFs are given in Figure 4.3-1. Each frame in this figure displays the CCDFs for a single consequence measure for all the subgroup source terms (SEQ-I-J) in Table 3.4-4 that have a nonzero frequency. The CCDFs were generated using the estimate

Table 4.3-1
Mean Consequence Results for Internal Initiators
(Population Doses in Sv)

Source Term Group	Early Fatalities	Total Lat. Cancer Fatalities	Pop. Dose Within 50 mi	Pop. Dose Entire Region	Individual Early Fat. Risk 0 - 1 mi	Individual Lat. Can. Fat. Risk 0 - 10 mi
SEQ-01-1	--	--	--	--	--	--
SEQ-01-2	1.73E-05	1.14E+01	3.19E+02	7.04E+02	4.35E-08	3.94E-05
SEQ-01-3	--	--	--	--	--	--
SEQ-02-1	2.82E-05	5.01E+01	1.26E+03	4.46E+03	7.10E-08	1.33E-05
SEQ-02-2	7.24E-05	6.09E+01	1.26E+03	3.78E+03	1.82E-07	8.12E-05
SEQ-02-3	8.15E-01	3.55E+01	1.06E+03	1.94E+03	1.37E-03	2.35E-04
SEQ-03-1	6.15E-05	2.41E+02	2.91E+03	1.51E+04	1.54E-07	3.15E-05
SEQ-03-2	0.00E+00	3.15E+02	3.17E+03	1.81E+04	0.00E+00	1.01E-04
SEQ-03-3	--	--	--	--	--	--
SEQ-04-1	2.87E-02	4.94E+02	7.71E+03	2.92E+04	5.60E-05	9.12E-05
SEQ-04-2	8.63E-01	1.91E+02	4.60E+03	1.22E+04	2.09E-03	2.90E-04
SEQ-04-3	8.41E-01	3.71E+02	6.32E+03	2.17E+04	1.42E-03	3.53E-04
SEQ-05-1	2.15E-04	9.01E+02	4.41E+03	5.20E+04	5.25E-07	5.25E-05
SEQ-05-2	2.12E-05	7.67E+02	6.42E+03	4.53E+04	5.35E-08	3.26E-04
SEQ-05-3	--	--	--	--	--	--
SEQ-06-1	4.97E-05	5.80E+02	3.79E+03	3.39E+04	1.25E-07	5.43E-05
SEQ-06-2	7.00E-07	6.77E+02	5.83E+03	3.85E+04	1.76E-09	1.91E-04
SEQ-06-3	--	--	--	--	--	--
SEQ-07-1	--	--	--	--	--	--
SEQ-07-2	--	--	--	--	--	--
SEQ-07-3	1.95E+00	1.69E+03	1.49E+04	9.93E+04	3.06E-03	7.04E-04
SEQ-08-1	2.20E-03	1.50E+03	1.30E+04	9.64E+04	5.40E-06	1.37E-04
SEQ-08-2	3.16E-04	1.89E+03	1.02E+04	1.12E+05	7.45E-07	4.86E-04
SEQ-08-3	1.62E+00	1.05E+03	1.07E+04	6.25E+04	2.61E-03	5.53E-04
SEQ-09-1	2.52E-03	1.61E+03	9.32E+03	9.27E+04	6.25E-06	1.47E-04
SEQ-09-2	1.89E-04	1.45E+03	8.32E+03	8.44E+04	4.65E-07	5.87E-04
SEQ-09-3	--	--	--	--	--	--

Table 4.3-1 (continued)

Source Term Group	Early Fatalities	Total Lat. Cancer Fatalities	Pop. Dose Within 50 mi	Pop. Dose Entire Region	Individual Early Fat. Risk 0 - 1 mi	Individual Lat. Can. Fat. Risk 0 - 10 mi
SEQ-10-1	5.50E-04	1.26E+03	7.20E+03	7.18E+04	1.39E-06	1.35E-04
SEQ-10-2	1.01E-06	1.24E+03	1.08E+04	7.11E+04	2.55E-09	3.06E-04
SEQ-10-3	--	--	--	--	--	--
SEQ-11-1	9.50E-02	3.16E+03	2.57E+04	1.87E+05	1.02E-04	6.38E-04
SEQ-11-2	2.28E-02	3.70E+03	1.97E+04	2.24E+05	2.56E-05	1.01E-03
SEQ-11-3	2.81E+01	2.72E+03	3.37E+04	1.52E+05	1.57E-02	7.38E-03
SEQ-12-1	2.91E-02	2.58E+03	1.63E+04	1.52E+05	5.35E-05	1.77E-04
SEQ-12-2	3.49E-03	3.08E+03	1.45E+04	1.80E+05	5.12E-06	8.66E-04
SEQ-12-3	2.50E+00	1.82E+03	1.09E+04	1.05E+05	3.54E-03	7.52E-04
SEQ-13-1	1.10E-02	1.62E+03	1.21E+04	9.54E+04	2.40E-05	1.95E-04
SEQ-13-2	1.89E-04	2.46E+03	1.16E+04	1.41E+05	4.63E-07	4.09E-04
SEQ-13-3	--	--	--	--	--	--
SEQ-14-1	1.29E+01	8.80E+03	1.13E+05	4.00E+05	1.43E-03	8.18E-03
SEQ-14-2	2.49E+00	6.96E+03	2.96E+04	4.18E+05	5.42E-04	3.07E-03
SEQ-14-3	1.41E+02	5.90E+03	8.20E+04	3.15E+05	2.92E-02	1.48E-02
SEQ-15-1	1.08E-01	3.45E+03	2.27E+04	2.04E+05	1.09E-04	4.76E-04
SEQ-15-2	1.98E-01	5.41E+03	2.10E+04	3.23E+05	1.54E-04	1.28E-03
SEQ-15-3	1.61E+01	3.50E+03	2.54E+04	2.09E+05	1.38E-02	2.00E-03
SEQ-16-1	0.00E+00	2.24E-02	1.38E+00	2.34E+00	0.00E+00	7.29E-09
SEQ-16-2	0.00E+00	6.02E-01	1.98E+01	3.21E+01	0.00E+00	3.03E-06
SEQ-16-3	--	--	--	--	--	--
SEQ-17-1	0.00E+00	2.35E-01	1.14E+01	2.41E+01	0.00E+00	1.06E-07
SEQ-17-2	0.00E+00	2.53E+00	8.09E+01	1.82E+02	0.00E+00	7.71E-06
SEQ-17-3	--	--	--	--	--	--
SEQ-18-1	0.00E+00	4.70E+01	1.06E+03	3.45E+03	0.00E+00	3.54E-05
SEQ-18-2	0.00E+00	1.84E+02	3.06E+03	1.05E+04	0.00E+00	1.34E-04
SEQ-18-3	--	--	--	--	--	--
SEQ-19	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00



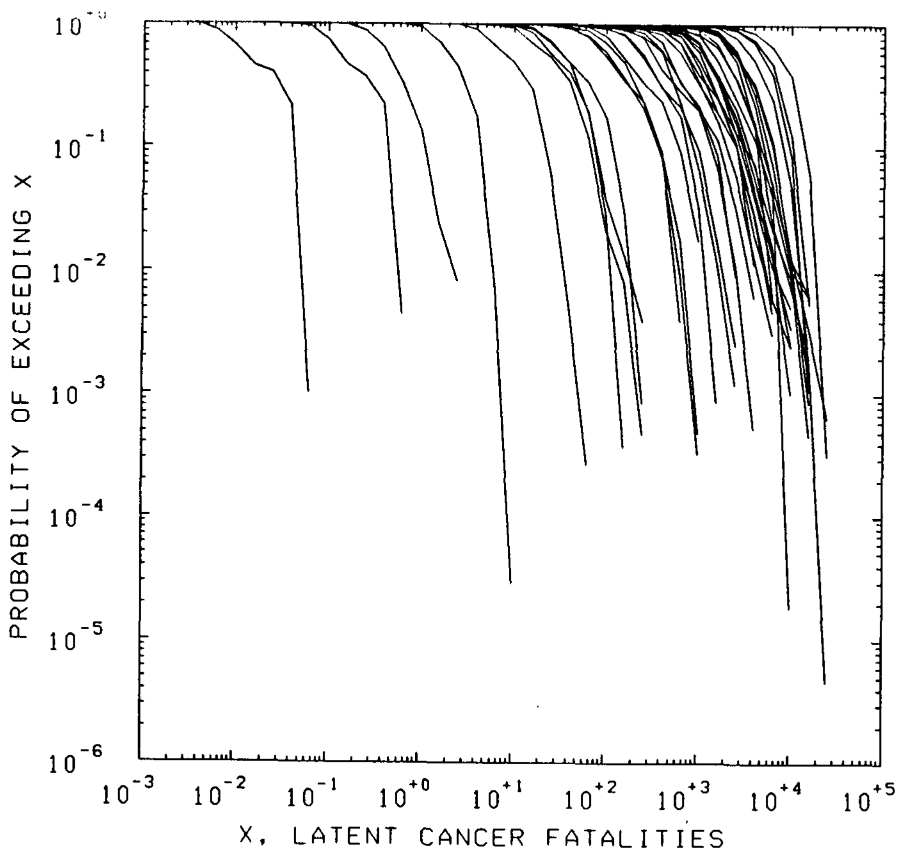
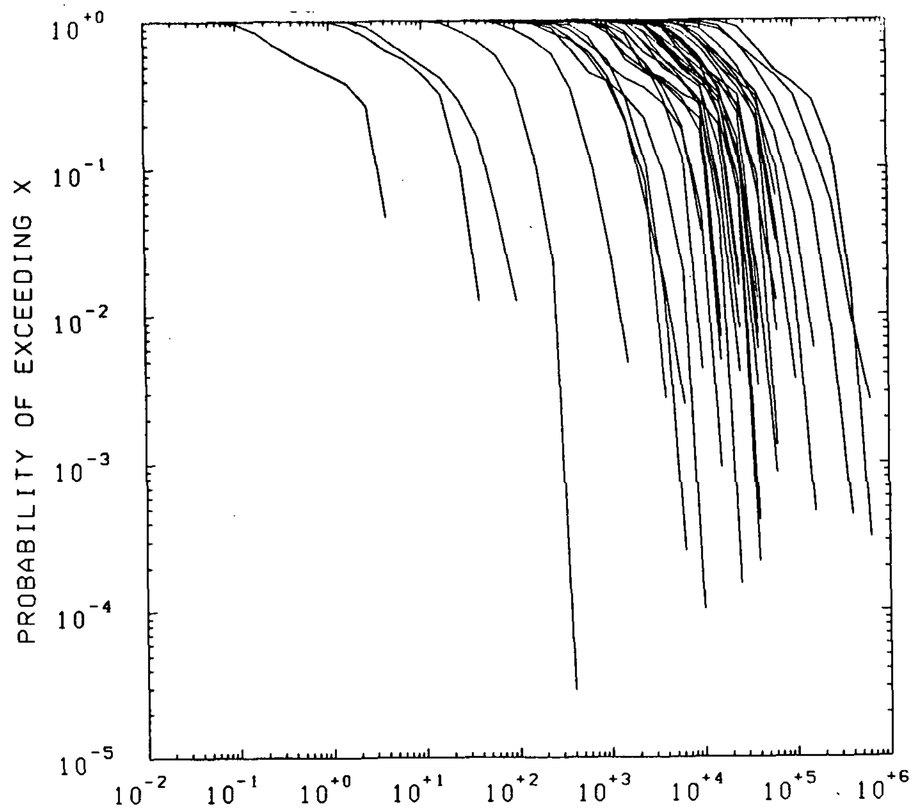


Figure 4.3-1. Consequences Conditional on Source Terms



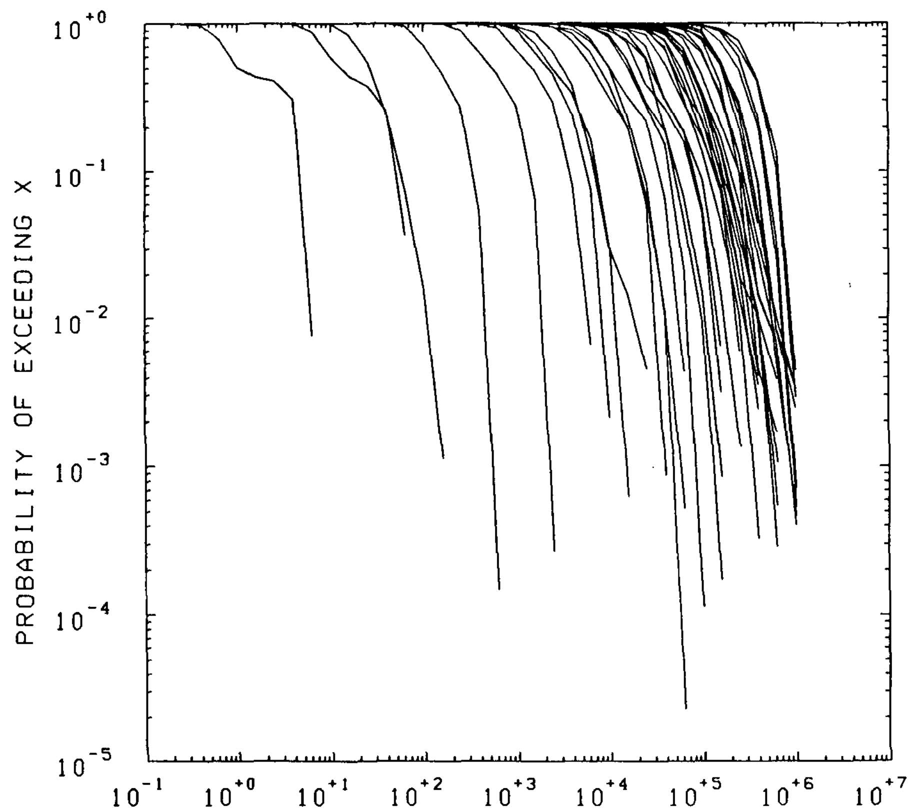
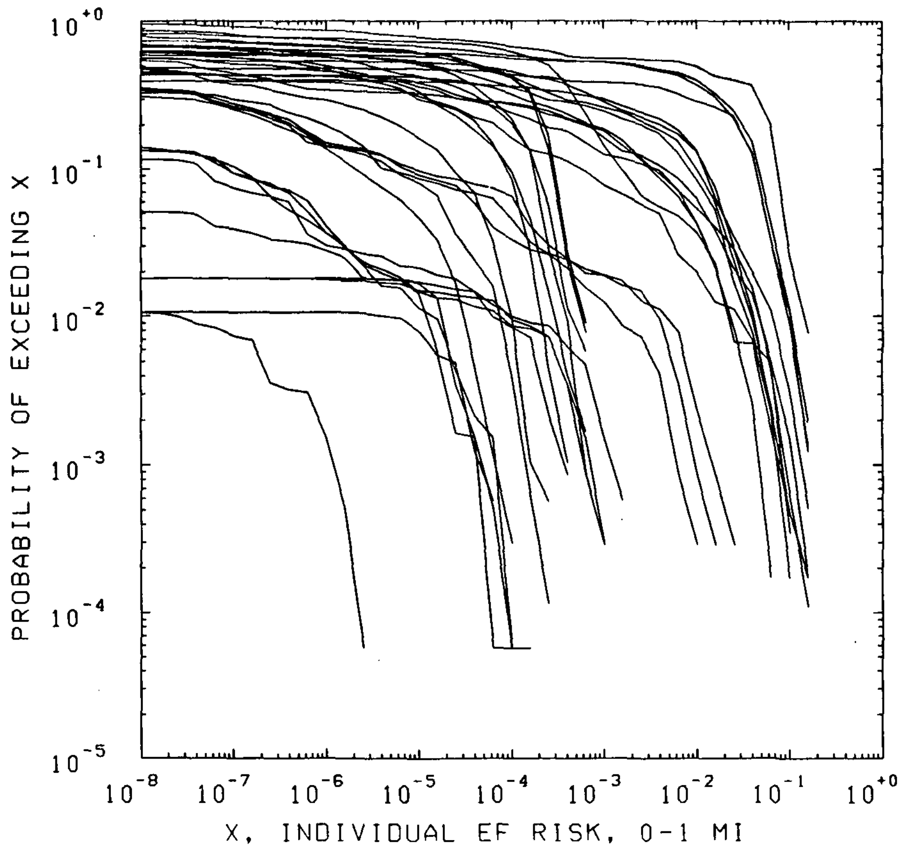


Figure 4.3-1. (continued)



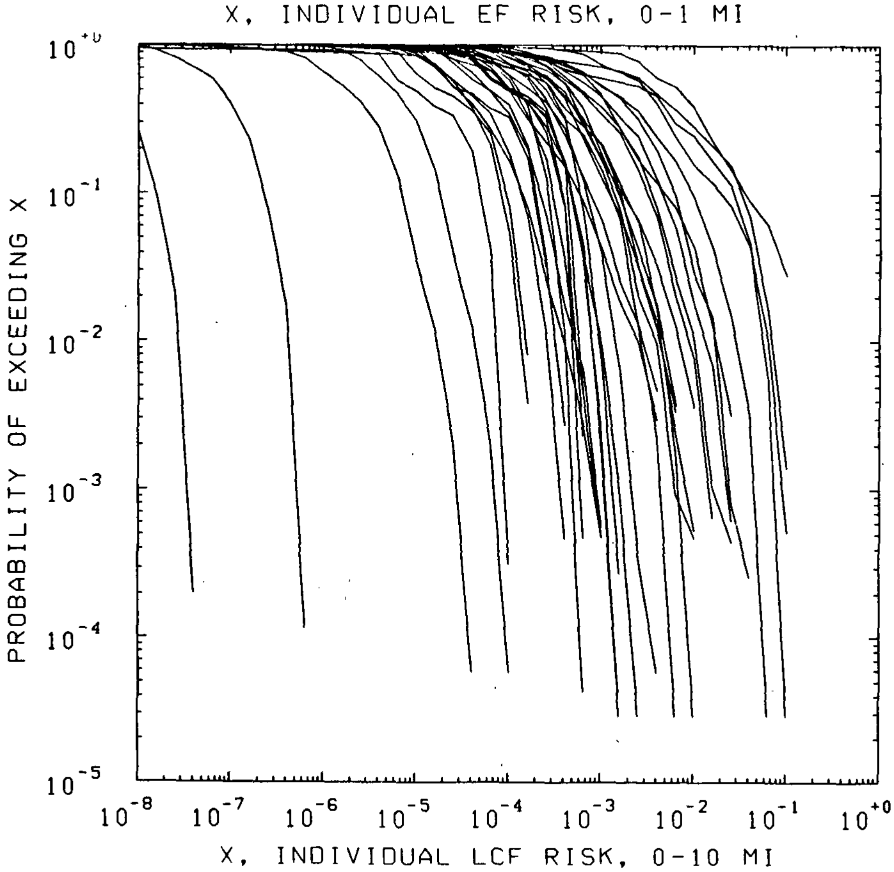


Figure 4.3-1. (continued)

that 99.5% of the population evacuates and 0.5% of the population continues normal activities. Each of the mean consequence results in Table 4.3-1 is the result of reducing one of the CCDFs in Figure 4.3-1 to a single number. The CCDFs in Figure 4.3-1 will subsequently be used to create CCDFs for risk, with the PRPOST code, which is described in Volume 1 of this report. The CCDFs for risk are presented in the next chapter; they relate consequence values with the frequency at which these values are exceeded.

4.4 References

1. D. I. Chanin, J. L. Sprung, L. T. Ritchie, and H.-N. Jow, "MELCOR Accident Consequence Code System (MACCS): User's Guide," NUREG/CR-4691, SAND86-1562, Volume 1, Sandia National Laboratories, February 1990.

2. H.-N. Jow, J. L. Sprung, J. A. Rollstin, and D. I. Chanin, "MELCOR Accident Consequence Code System (MACCS): Model Description," NUREG/CR-4691, SAND86-1562, Volume 2, Sandia National Laboratories, February 1990.

3. J. A. Rollstin, D. I. Chanin, and H.-N. Jow, "MELCOR Accident Consequence Code System (MACCS): Programmer's Reference Manual," NUREG/CR-4691, SAND86-1562, Volume 3, Sandia National Laboratories, February 1990.

4. R. L. Iman, J. D. Johnson, and J. C. Helton, "A User's Guide for the Probabilistic Risk Assessment Model Integration System (PRAMIS)," NUREG/CR-5262, SAND88-3093, Sandia National Laboratories, May 1989.

5. RISK RESULTS FOR SEQUOYAH

This section gives the results of the integrated risk analysis for the Sequoyah plant. Section 5.1 gives the risk results for internal initiators.

Risk is determined by bringing together the results of four constituent analyses: the accident frequency, accident progression, source term, and consequence analyses. The phrase, integrated risk analysis, is used to refer to the combined result when all four analyses are combined. The way in which these analyses contribute to risk analysis is summarized in Section 1.4 of this volume. More detail on the methods used in calculating risk can be found in Volume 1.

The figures in this section present only a very small portion of the total risk output available. Detailed listings of results are available on computer media by request.

5.1 Results for Internal Initiators

This section describes the results of the integrated risk analysis for internal initiators at the Sequoyah plant. Section 5.1.1 discusses basic risk results for internal initiators. Section 5.1.2 addresses the types of accidents and plant features that are important in determining the risk from internal initiators at Sequoyah. Finally, Section 5.1.3 gives the results of the regression analysis performed to determine the important contributors to the uncertainty in risk.

5.1.1 Risk Results

Figure 5.1-1 shows the basic results of the integrated risk analysis for internal initiators at Sequoyah. This figure shows the complementary cumulative distribution functions (CCDFs) for early fatalities, latent cancer fatalities, population dose within 50 miles, population dose within the entire region, individual risk of early fatality within one mile of the site boundary, and individual risk of latent cancer fatality within 10 miles.

The CCDFs display the relationship between the frequency of the consequence and the magnitude of the consequence. As there are 200 observations in the sample for Sequoyah, the complete set of risk results, at the most basic level, consists of 200 CCDFs for each consequence measure. Plots showing these 200 curves are contained in Appendix D; only four statistical measures of the 200 curves are shown in Figure 5.1-1. These measures are generated by analyzing the plots in the vertical direction. For each consequence value on the abscissa, there are 200 values of the exceedance frequency (one for each observation or sample element), and from these 200 values, the mean, median, 95th percentile, and 5th percentile values are calculated. When this is done for each value of the consequence measure, the curves in Figure 5.1-1 are obtained. Thus, Figure 5.1-1 gives the relationship between the magnitude of the consequence and the frequency at which the consequence is exceeded, as well as the variation in that

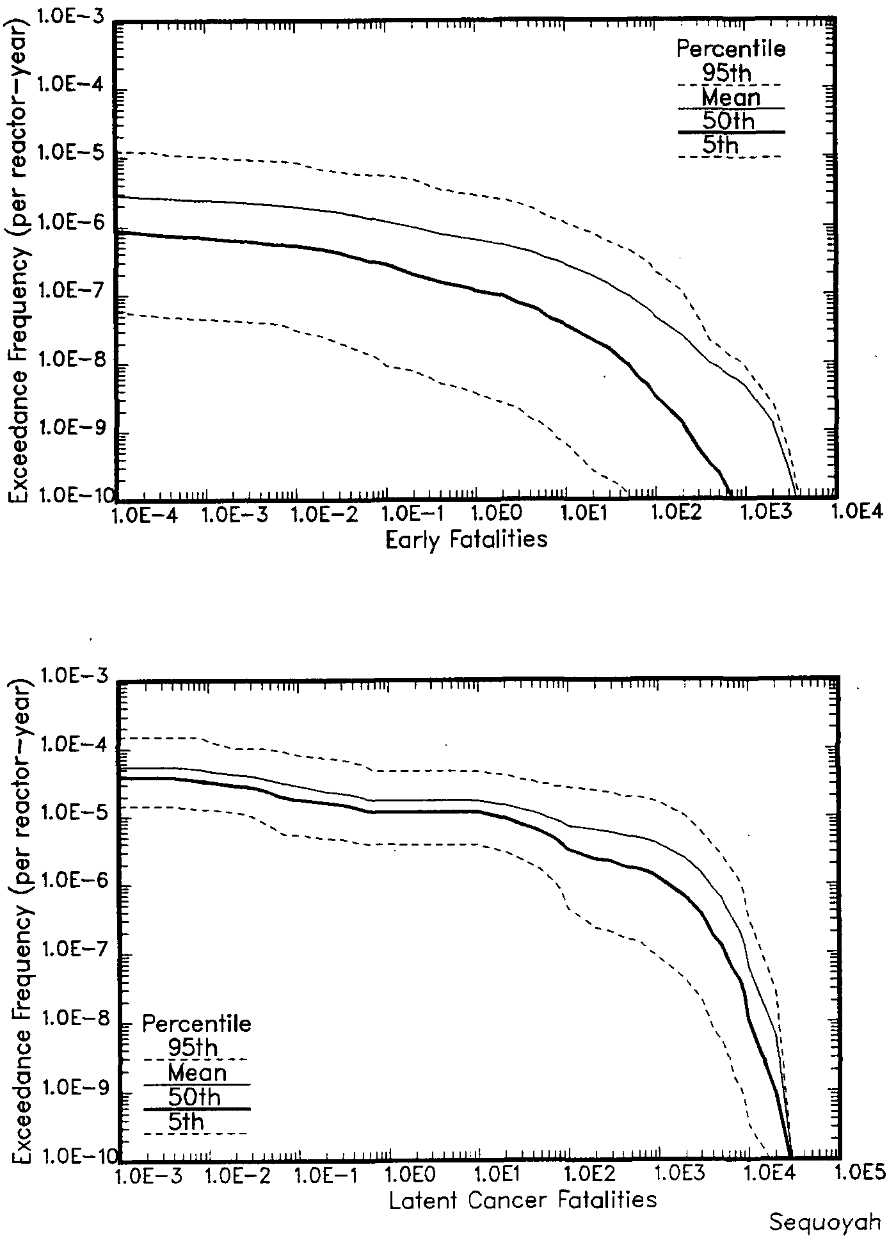
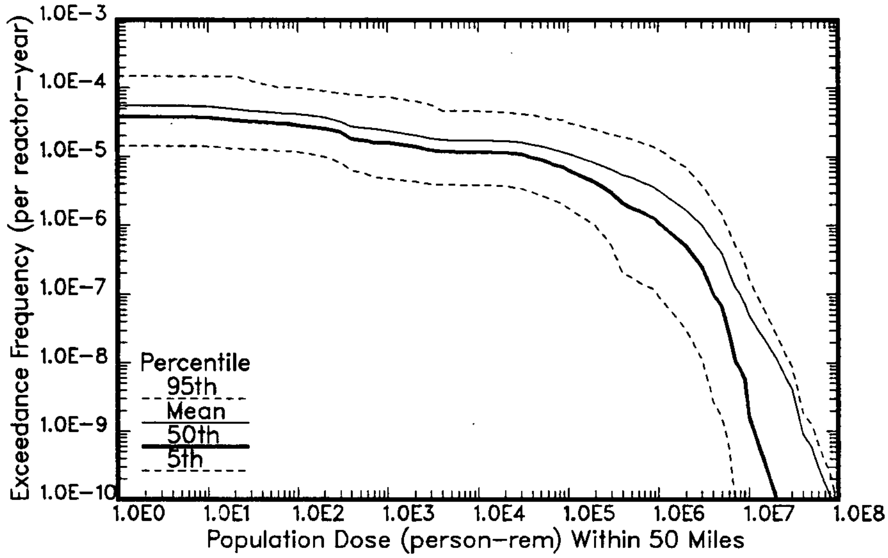


Figure 5.1-1. Exceedance Frequencies for Risk
(Sequoyah, All Internal Initiators)



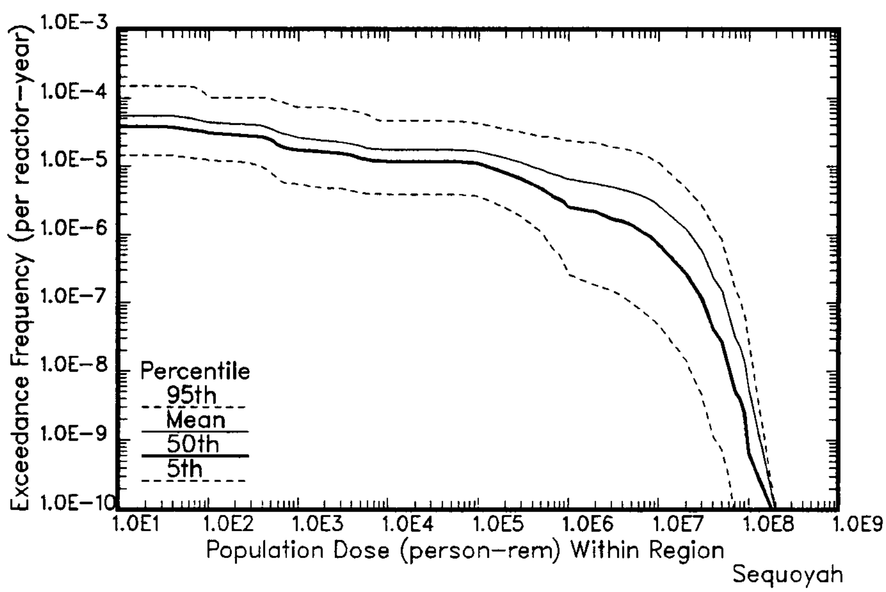
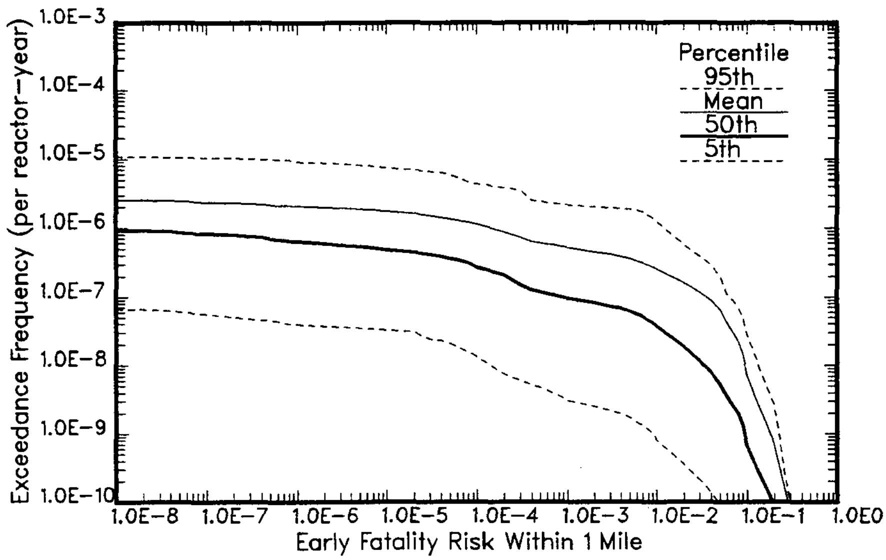


Figure 5.1-1. (continued)



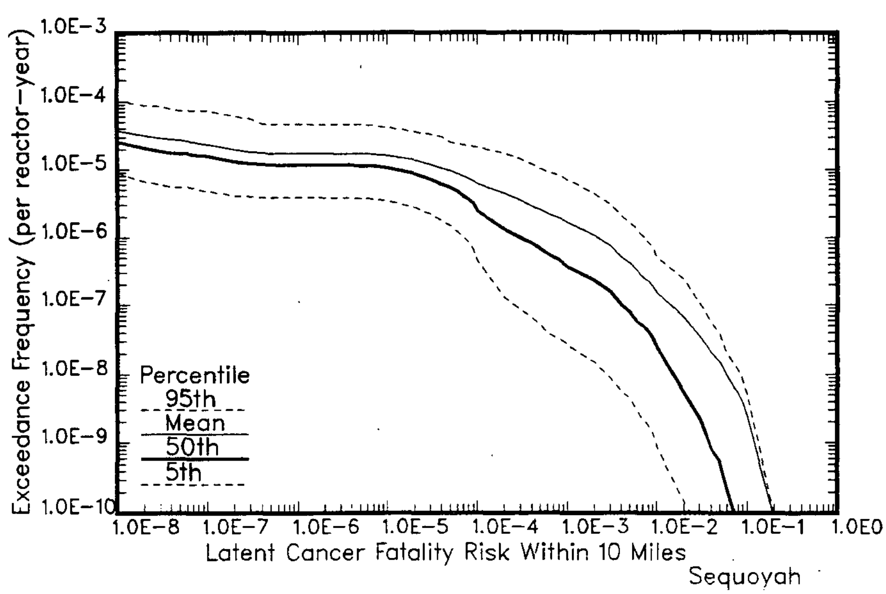


Figure 5.1-1. (continued)

relationship. The percentile and mean curves in Figure 5.1-1 and similar figures are only valid when read from the abscissa; that is, the percentiles and means do not apply for a given value of exceedance frequency.

Although the abscissa in the last two plots in Figure 5.1-1 is labeled "Risk," this reflects historical usage and is not really correct. The x-axis in these plots actually represents conditional probability: specifically, the probability that an individual, randomly located in the spatial interval according to the population distribution, will die if the accident occurs. The ordinate gives the frequency of an accident that produces a conditional probability that exceeds the value on the abscissa. The actual risk measure (i.e., product of the consequence and its associated frequency) does not result until the curves in the last two plots of Figure 5.1-1 are reduced to single values.

The curves for latent cancer fatalities in Figure 5.1-1 are relatively flat from about 0.6 to 10 fatalities. This means that latent cancer fatalities in this range are very unlikely. Any type of containment failure (CF) or bypass is likely to lead to more than 10 delayed fatalities; it is quite unlikely, however, that an accident will result in more than a few thousand delayed fatalities. If the containment does not fail, the eventual release of the noble gases (xenon and krypton) from the containment due to design basis leakage will probably cause less than 0.6 latent cancer fatalities.

The variation from the 5th to the 95th percentiles indicates the uncertainty in the risk estimates due to uncertainty in the basic parameters in the three sampled constituent analyses (the accident frequency, accident progression, and source term analyses). The variation along a curve in Figure 5.1-1 (or along one of the individual curves in Appendix D) is indicative of the variation in risk due to different types of accidents and due to different weather conditions at the time of the accident. Thus, the individual curves in Appendix D can be viewed as representing stochastic variability (i.e., the effects of probabilistic events in which it is possible for the accident to develop in more than one way), and the variability between curves can be seen as representing the effects of imprecisely known parameters and processes that are mostly nonstochastic. As the magnitude of the consequence measure increases, the mean curve typically approaches or exceeds the 95th percentile curve. This results when the mean is dominated by a few large observations, which often happens for large values of the consequences because only a few observations have nonzero exceedance frequencies for these large consequences. Figure 5.1-1 shows the following mean and median exceedance frequencies for fixed values of early fatalities (EFs) and latent cancer fatalities (LCFs):

	Exceedance Frequency (1/R-yr)	
Consequence	Mean	Median
1 EF	6E-7	1E-7
100 EF	5E-8	3E-9
100 LCF	7E-6	3E-6
10,000 LCF	6E-8	1E-8

Although the LCF values mentioned above may appear large, they must be considered in perspective; the calculated LCFs occur throughout the entire region and over several decades. Between 400,000 to 500,000 deaths due to cancer occur every year in the U.S. The population within 350 miles of the plant is about 37 million and within 1000 miles of the plant is about 180 million. When spread over two or three decades, even tens of thousands of additional LCFs are statistically indistinguishable from the general background morbidity due to malignant neoplasms in such a large population.

Although the CCDF for each observation conveys the most information about risk, a single number may be generated for each consequence measure for each observation. This value, denoted annual risk, is determined by summing the product of the frequencies and consequences for all the points used to construct the CCDF for each observation in the sample. The construction of annual risk has the effect of averaging over the different weather states and includes contributions from all the different types of accidents that can occur. Since the complete analysis consisted of a sample of 200 observations, there are 200 values of annual risk for each consequence measure. These 200 values may be ordered and plotted as histograms, as in Figure 5.1-2. The four statistical measures used above are shown on these plots and are also reported in Table 5.1-1. Note that considerable information has been lost in going from the CCDFs in Appendix D to the histograms of annual values in Figure 5.1-2; the relationship between the size of the consequence and its frequency has been sacrificed to obtain a single value for risk for each observation.

The plots in Figure 5.1-2 show the variation in the annual risk for six consequence measures. Where the mean is close to the 95th percentile, it may be inferred that a relatively small number of observations dominate the mean value. This is more likely to occur for the EF consequence measures than for the latent cancer fatality or population dose consequence measures due to the threshold effect for EFs. In essence, Figure 5.1-2 shows the probability density functions of the logarithms of the consequence measures. Equivalent density functions could be generated for the consequence measures themselves, but would appear quite different due to the change in scale. Another alternative, but equivalent display, for the results in Figure 5.1-2 would be to use cumulative distribution functions.

The safety goals are expressed in terms of mean individual fatality risks, which is really an individual's probability of becoming a casualty of a reactor accident in a given year. The individual Ef risk within one mile is the frequency (per year) that a person living within one mile of the site boundary will die within a year due to the accident. The entire population within one mile is considered to obtain an average value. The individual latent cancer fatality risk within 10 miles is the frequency (per year) that a person living within 10 miles of the plant will die many years later from cancer due to radiation exposure received from the accident. The entire population within 10 miles is considered to obtain an average value. A single value for individual fatality risk for each observation is obtained by reducing the CCDF for each observation to a single value. The density distribution of these 200 values is plotted in the last two frames of Figure 5.1-2. Although the values are really frequencies, they are so small that they are essentially probabilities that an individual will become a casualty of a reactor accident in a given year.

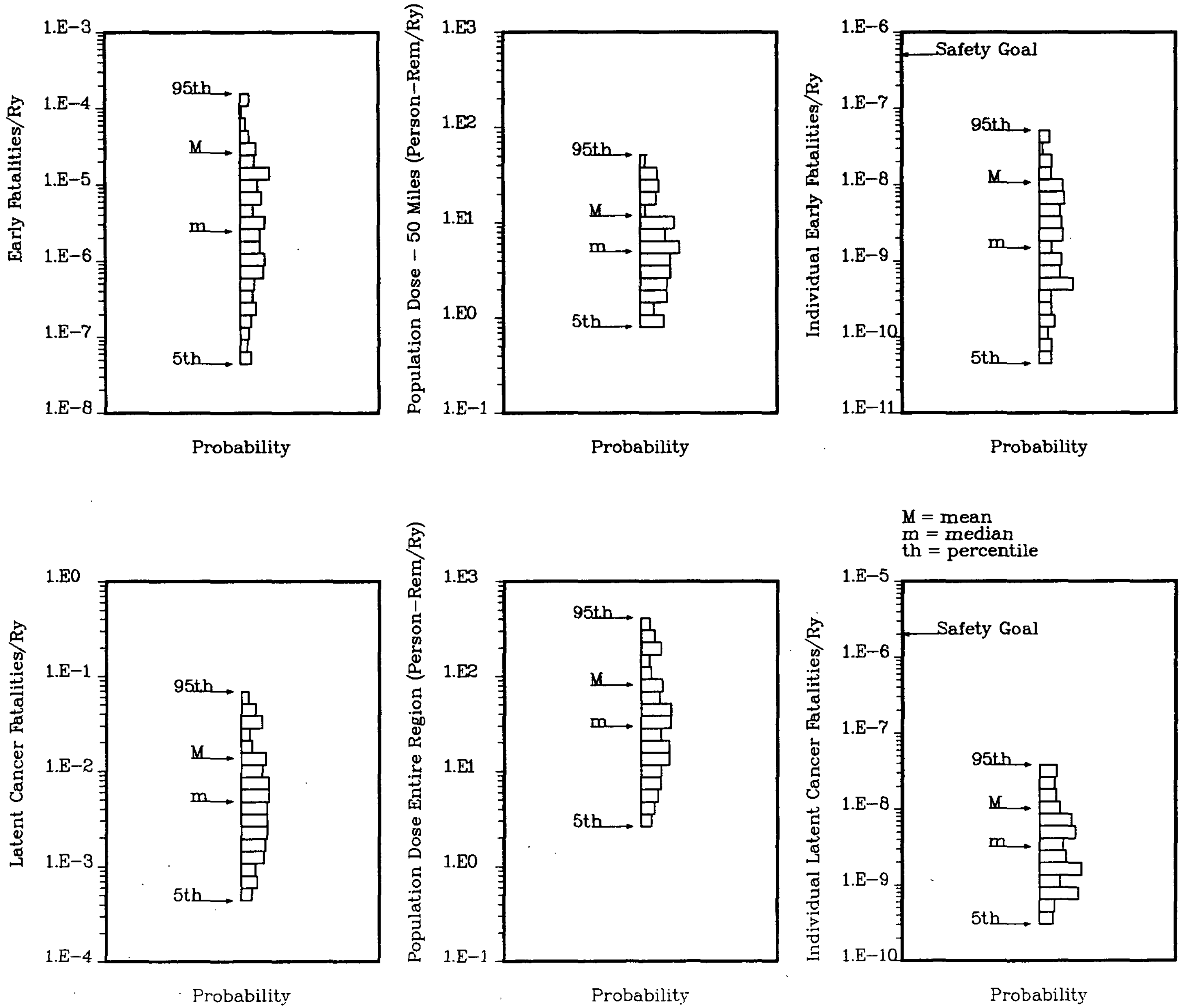


Figure 5.1-2. Distributions of Annual Risk (Sequoyah: All Internal Initiators)

The plots for individual risk in Figure 5.1-2 show that both risk distributions for Sequoyah fall well below the safety goal. A single measure of risk for the entire sample may be obtained by taking the average value from the histograms in Figure 5.1-2. This measure of risk is commonly called mean risk, although it is actually the average of the annual risk, or the mean value of the mean risk. The mean risk values for the six consequence measures reported here are displayed in Figure 5.1-2. The important contributors to mean risk are considered in Section 5.1.2.

The offsite risk at Sequoyah is relatively low with respect to the safety goals. There are several factors that lead to these low values for risk. The core damage frequency for Sequoyah is quite low, and the mean value is 5.6E-05. If core damage occurs, it is unlikely that the containment will fail, and if it does fail, there are several features of the Sequoyah plant that tend to reduce the source term and therefore the consequences.

A factor influencing the risk estimates is arresting the core damage process before vessel failure and achieving a safe, stable state, as at TMI-2. Obtaining sufficient ECCI after the onset of core damage may come about through the recovery of offsite power, or the depressurization of the RCS to the point that injection by systems operating at the onset of core damage commences. A significant fraction of the time, the accidents in the most likely three plant damage state (PDS) groups loss-of-coolant accidents (LOCAs), fast station blackout (SBO), and slow SBO, comprising about 89% of the mean core damage frequency (MCDF) result in arrest of the core damage process and no vessel breach (VB). If the vessel fails, it is likely that either the core debris released from the vessel will be cooled or, if core-concrete interaction (CCI) is initiated, it will occur under a pool of water.

The EF risk depends on both the magnitude of the release and on the timing of CF. If the containment fails early in the accident, or if the containment is bypassed, it is more likely that a portion of the population will be exposed to the release than if the containment fails after the nearby population has been evacuated. A large potential exists for CF at the time of VB at Sequoyah. Postulated pressure rises at vessel failure resulting from direct containment heating (DCH) coupled with hydrogen combustion can be high with respect to the predicted strength of the Sequoyah containment.

The DCH/hydrogen threat is reduced by two means. The first is when the cavity becomes deeply flooded; that is, the water level is above the bottom head of the vessel and can be up to the hot leg inlets on the vessel. Dispersal of debris from the cavity into the lower containment is therefore inhibited when the cavity is flooded to this level. The second is when mechanisms that lead to depressurization of the reactor coolant system (RCS) before failure of the vessel are considered. The RCS depressurization mechanisms included are temperature-induced (T-I) failure of the hot leg or surge line, power operated relief valves (PORVs) sticking open, T-I reactor coolant pumps (RCP) seal failure, T-I SGTR, and deliberate opening of the PORVs by the operators. Only the first three of these mechanisms were very effective in this analysis, but they were sufficient to ensure that only a small fraction of the accidents that were at full system pressure at the onset of core damage were still at that pressure at VB.

Reducing the RCS pressure at VB, of course, reduces the loads placed on the containment at VB, and thus reduces the probability of CF.

The LCFs are generally associated with the population that does not evacuate. Thus, this risk measure is not particularly sensitive to the timing of CF, but rather to whether the containment fails. Furthermore, because there is no threshold effect for LCFs, this consequence measure is not as sensitive to the magnitude of the release as is the EF risk. LCF risk is primarily dependent on frequency of Cf. Unlike EF risk, late CFs as well as EFs of the containment are important to the latent cancers.

There are several features of the Sequoyah plant that reduce the magnitude of the source term. In the majority of the accidents analyzed, the in-vessel releases experience decontamination by the ice condenser (IC). Many times if VB is predicted to occur, the CCI is either inhibited because a coolable debris bed is formed and the cavity water is replenished, or the release from the CCI is scrubbed by an overlying water pool. Operation of the containment spray system (CSS) also helps to mitigate the source term.

Table 5.1-1
Distributions for Annual Risk at Sequoyah Due to Internal Initiators
(All values per reactor-yr; population doses in person-rem)

Risk Measure	5th%tile	Median	Mean	95th%tile
Core Damage	1.5E-5	3.9E-5	5.6E-5	1.5E-4
EFs	4.7E-8	2.4E-6	2.6E-5	1.2E-4
LCFs	5.6E-4	4.8E-3	1.4E-2	5.3E-2
Population Dose 50 mi	8.7E-1	5.0E+0	1.2E+1	4.6E+1
Population Dose Entire Region	3.5E+0	2.9E+1	8.1E+1	3.1E+2
Ind. EF Risk, 0 - 1 mile	4.6E-11	1.5E-9	1.1E-8	4.3E-8
Ind. LCF Risk 0 - 10 miles	3.9E-10	3.2E-9	1.0E-8	3.5E-8

5.1.2 Contributors to Risk

There are two distinct ways to calculate contribution to risk, and to facilitate their definition, the following quantities are introduced:

rC_j = risk (units: consequences/reactor-yr) for consequence measure j,

rC_{ij} = value for rC_j obtained for observation i,

rC_{jk} = risk (units: consequences/reactor-yr) for consequence measure j due to PDS group k,

rC_{ijk} = value for rC_{jk} obtained for observation i, and

nLHS = number of observations in the Latin Hypercube Sample (LHS).

The notation here is similar to that in Section 1.4 The value of nLHS is 200 for Sequoyah. The risk rC_{ij} is the j^{th} element of the vector $\mathbf{rC}_i$ in Equation 1.9 of Section 1.4. The risk rC_{ijk} is the j^{th} element of the vector $\mathbf{rC}_i$ when the frequencies of all the PDS groups except group k in the vector $\mathbf{fPDS}_i$ are set to zero. The vector $\mathbf{fPDS}_i$ is equal to the product $\mathbf{fIE}_i$ $\mathbf{P}_i(\mathbf{IE}\rightarrow\mathbf{PDS})$.

The result of the first method for computing contribution to risk is denoted the fractional contribution to mean risk (FCMR). The contribution of PDS group k to the risk for consequence measure j, $FCMR_{jk}$, is defined as the ratio of the annual risk due to PDS group k to the total annual risk. That is, $FCMR_{jk}$ is defined by

$$FCMR_{jk} = E(rC_{jk})/E(rC_j),$$

where E(x) represents the annual value of x. Computationally, $FCMR_{jk}$ is found by use of the relation

$$FCMR_{jk} = [\Sigma\ rC_{ijk}/nLHS]/[\Sigma\ rC_{ij}/nLHS]$$

$$= \Sigma\ rC_{ijk}/\Sigma\ rC_{ij},$$

where the summations are from i = 1 to i = nLHS.

The result of the second method for computing contribution to risk is denoted the mean fractional contribution to risk (MFCR). The contribution of PDS group k to the risk for consequence measure j, $FCMR_{jk}$, is defined as the annual value of ratio of the risk due to PDS group k to the total risk. That is,

$$MFCR_{jk} = E(rC_{jk}/rC_j).$$

Computationally, $MFCR_{jk}$ is found by use of the relation

$$MFCR_{jk} = \Sigma(rC_{ijk}/rC_{ij})/nLHS,$$

where the summation again is from i = 1 to i = nLHS.

For FCMR, the averaging over the observations is done before the ratio of group risk to total risk is formed; for MFCR, the averaging over the observations is done after the ratio of group risk to total risk is formed.

Table 5.1-2 gives the values of FCMR and MFCR for the seven PDS groups. Not surprisingly, the two methods of calculating contribution to risk yield different values. Both methods of computing the contributions to risk are conceptually valid, so the conclusion is clear: contributors to mean risk can only be interpreted in a very broad sense. That is, it is valid to say that Event V is a major contributor to mean EF risk at Sequoyah. It is not valid to state that Event V contributes to 68% of the EF risk at Sequoyah.

Pie charts for both methods of computing the contribution to risk are shown in Figure 5.1-3 for EFs and for LCFs for the seven PDS groups. The variations between the two methods of computing contribution to risk are higher for EFs than for LCFs because of the threshold effect involved in determining the number of early fatalities. The differences are readily apparent when this method of displaying the results is used, and suggest the level of confidence that these results warrant.

The contributions of the summary accident progression bins (APBs) to mean risk can also be computed in two ways. Table 5.1-3 and Figure 5.1-4 display the results of these calculations.

To determine the reproducibility of the integrated risk analyses performed for NUREG-1150, a second sample was run through the entire integrated risk analyses for the Surry plant. The second sample is just as valid as the first sample, and differs from the first sample only in that a different random seed was used in the LHS program. Therefore, the differences in the results between the two samples indicate of the robustness of the analysis methods. In addition, a comparison of the two samples indicates which method of calculating the contribution to risk tends to be more stable. The results from the Surry analysis regarding second sample and a comparison of the two samples are presented in NUREG/CR-4551, Volume 3. Several insights gleaned from this comparison are summarized below. First, considering the EF and LCF risk distributions, the agreement between the two samples is remarkably good. This agreement indicates that the methods used for this integrated risk analysis are sound. Differences between the two samples can generally be found at the extremes of the distribution, which is not surprising since the extremes are determined by relatively few observations. Also, the variations between samples are higher for FCMR than for MFCR, indicating that MFCR is a more robust measure of the risk results than FCMR.

The FCMR measure of the contribution to mean risk tends to be less stable than the MFCR measure because often the annual risk for each observation is dominated by a few APBs that have both high frequency and high source terms, and the mean risk is dominated by a few observations that have very large values of annual risk. The bulk of the mean risk is contributed by about 10 to 20 observations. While the sample as a whole is reproducible, the 10 to 20 observations that control mean risk are generally not reproducible. Since it is the exact nature of these 10 or so

Table 5.1-2
Fractional PDS Contributions to Annual Risk at
Sequoyah Due to Internal Initiators

PDS Group	Method	Core Damage	EF	LCF	Population Dose 50 miles	Population Dose Region	Ind. EF Risk-1 mile	Ind. LCF Risk-10 mile
Slow SBO	FCMR	8.2	6.9	12.5	11.1	12.5	8.5	11.8
	MFCR	8.0	6.7	8.4	8.0	8.3	7.0	8.2
Fast SBO	FCMR	16.6	16.0	28.6	26.5	28.7	17.7	28.3
	MFCR	16.8	18.2	25.4	24.3	25.4	19.0	23.9
LOCAs	FCMR	63.1	1.7	14.2	18.6	14.6	3.2	14.9
	MFCR	60.2	13.0	20.9	28.1	22.1	12.8	25.7
Event V	FCMR	1.2	68.0	10.3	14.9	9.8	61.8	29.2
	MFCR	1.5	40.5	10.0	10.4	9.7	37.7	16.2
Transients	FCMR	4.2	0.1	0.5	0.5	0.5	0.2	0.5
	MFCR	5.7	1.3	1.4	1.3	1.4	1.4	1.7
ATWS	FCMR	3.7	1.9	3.8	3.7	3.8	2.2	4.1
	MFCR	4.3	6.8	5.7	5.3	5.6	7.2	7.5
SGTR	FCMR	3.1	5.3	30.1	24.7	30.1	6.4	11.3
	MFCR	3.6	13.5	28.1	22.6	27.5	14.9	16.9

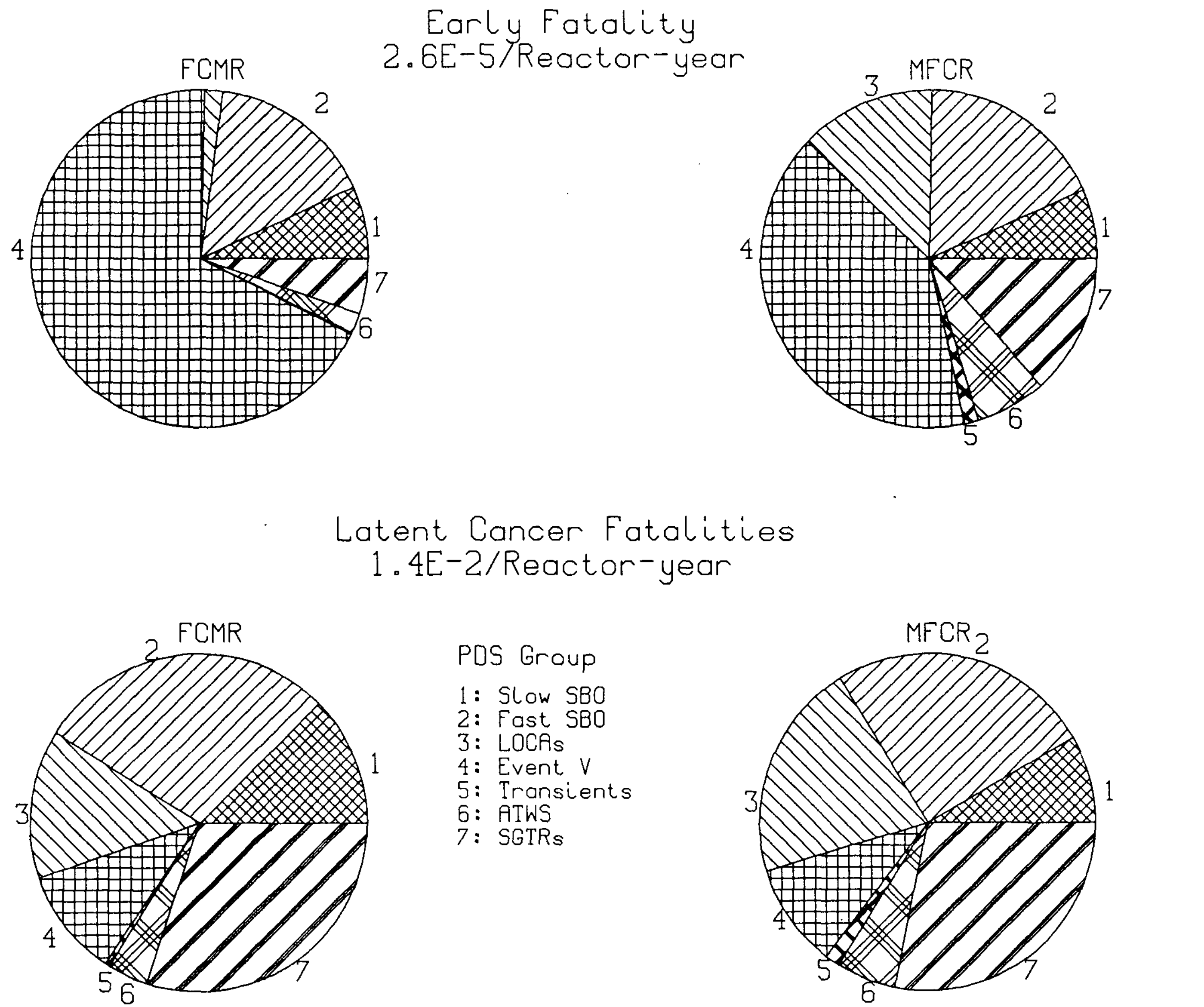


Figure 5.1-3. Fractional PDS Contributions to Annual Risk, Sequoyah (Internal Initiators)

Table 5.1-3
Fractional APB Contributions (%) to Annual
Risk at Sequoyah Due to Internal Initiators

Summary APB	Method	EFs	LCFs	Population Dose 50 miles	Population Dose Region	Ind. EF Risk-1 mile	Ind. LCF Risk-10 mile
VB, CF during core degradation	FCMR	1.6	4.4	3.7	4.3	2.3	4.3
	MFCR	8.5	5.6	4.1	5.4	7.8	6.0
VB, Alpha mode	FCMR	0.4	0.8	0.7	0.8	0.5	0.9
	MFCR	3.9	1.6	1.2	1.5	3.5	2.0
VB, CF at VB, RCS pressure >200 psia	FCMR	8.0	22.8	21.0	22.7	11.3	22.7
	MFCR	7.6	11.5	10.7	11.3	8.9	12.3
VB, CF at VB, RCS pressure <200 psia	FCMR	13.9	16.7	14.7	16.7	13.4	18.5
	MFCR	14.6	11.9	10.2	11.7	14.4	12.6
VB, late CF	FCMR	0.0	3.8	4.9	4.0	0.0	1.0
	MFCR	0.5	9.0	9.6	9.2	0.8	4.9
VB, very late CF, or BMT	FCMR	0.0	2.2	6.9	2.8	0.0	3.0
	MFCR	0.0	10.9	21.0	12.7	0.0	15.7
Bypass	FCMR	75.4	44.2	42.9	43.7	70.6	44.6
	MFCR	61.7	43.8	37.6	42.6	60.6	40.4
VB, No CF, No Bypass	FCMR	0.0	0.0	0.1	0.0	0.0	0.0
	MFCR	0.0	0.0	0.2	0.1	0.0	0.0
No VB, CF during core degradation	FCMR	0.8	5.2	5.0	5.1	1.8	5.1
	MFCR	3.3	5.6	5.2	5.5	4.1	6.1
No VB, No CF	FCMR	0.0	0.0	0.1	0.0	0.0	0.0
	MFCR	0.0	0.0	0.2	0.1	0.0	0.0

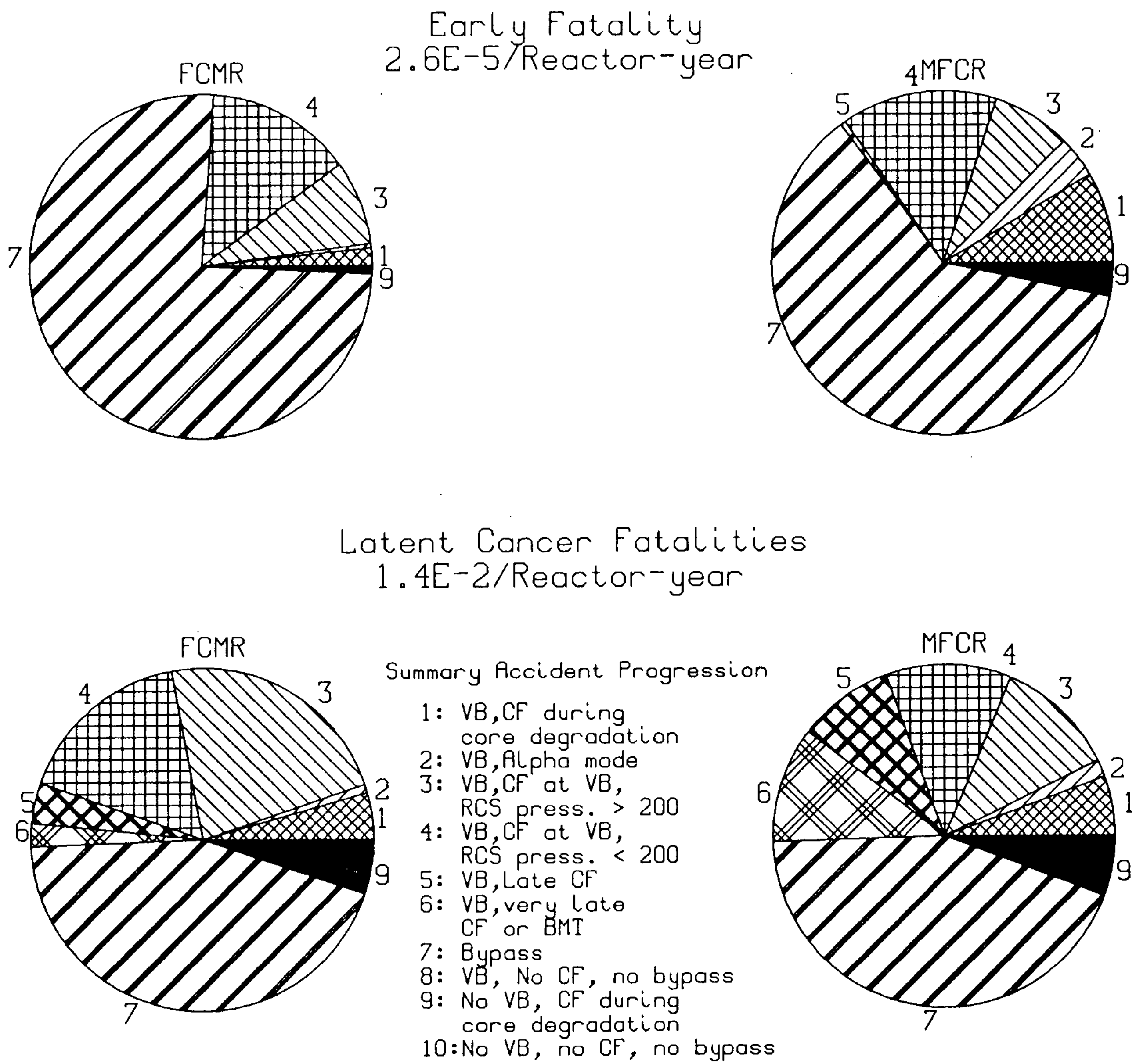


Figure 5.1-4. Fractional APB Contributions to Annual Risk, Sequoyah (Internal Initiators)

observations that determines the contributors to mean risk, it is not surprising that FCMR is not a robust measure of the entire risk analysis.

Both FCMR and MFCR are conceptually valid methods of computing the contributions to mean risk. However, given the overall structure of the PRAs performed for NUREG-1150, MFCR is the more appropriate measure. The analysis performed for each observation in the sample can be viewed as a complete PRA. In a single observation, each sampled variable has a fixed value representing one possible value for an imprecisely known quantity. Each observation yields an estimate for the ratio rC_{jk}/rC_j (the fractional contribution of PDS group k to the risk for consequence measure j) based on an internally consistent set of assumptions. Taken as a whole, the sample produces a distribution for fractional contributions to risk.

MFCR results from averaging over the sampled variables and is thus consistent with other annual values reported in this study. That is, for other quantities, a single value is obtained for each observation in the sample, and distributions and means are reported for these values. Thus, the calculation of MFCR is consistent with the manner in which mean risk values are calculated. The FMCR results are not consistent with this pattern of obtaining a complete result for each observation and then analyzing the distribution of results.

This is an appropriate place to remind the reader of a caveat made elsewhere in this report: a mean value is a summary measure and information is lost in generating it. Thus, considerable caution should be used in drawing conclusions solely from mean values. A mean is obtained by reducing an entire distribution to a single number.

Even though the measures for determining the contributors to mean risk are only approximate, the types of accidents that are the largest contributors to offsite risk at Sequoyah are clear. For the two consequence measures that depend on a large early release, EFs and individual risk of EF within one mile, Event V is the major contributor to mean risk, with the blackout sequences also playing an important role.

Although its overall frequency is low, Event V dominates the EF risks because a large unmitigated release occurs shortly after the accident begins. Evacuation occurs after the release has begun. One might expect that SGTR accidents would contribute to EF risks in a similar fashion. However the SGTR accidents that lead to large releases, the "H" SGTRs with stuck-open secondary SRVs, are very lengthy accidents. Therefore, although the releases from the "H" SGTR accidents are large, they occur after the evacuation is complete and cause relatively few early fatalities.

The SBOs are also significant contributors to EF risks. The blackout accidents are responsible for a large part of the early CFs. By referring to Table 5.1-3, it can be seen that the fourth bin involving containment failure at VB (CF at VB) with the RCS at low pressure (failure due mainly to hydrogen burns) is the dominant bin contributing to the early fatality risks. The third bin, which involves failure of the containment when the vessel is breached at high pressure, also contributes, but as discussed in Subsection 5.1.1, although the potential for CF at VB with high RCS pressure is quite high, the actual probability is lower due to deep-flooding of the cavity, core damage arrest before VB, and RCS depressurization. Also, for the nonblackout accidents in which CF occurs at VB with the RCS at high pressure, there is more mitigation of the releases due to the operation of sprays, etc.

It might be expected that early CF would contribute about the same to risk as Event V because, given core damage, the frequency of early CF is about the same as the frequency of Event V. When comparing to Event V, however, the evacuation for the early CFs for SBOs occurs earlier with respect to the timing of the releases. The early CFs usually involve energetic releases due to the dominance of rupture failures of containment. This results in lofting of the plume above the population, thus reducing the EF risks and increasing slightly the LCF risks. The energy associated with Event V releases is much lower.

LCFs and population dose depend primarily on the total amount of radioactivity released. Thus, unlike EF risk, the timing of CF is not particularly important for the remaining four consequence measures: population dose within 50 miles, population dose within the entire region, LCFs, and individual risk of LCF within 10 miles. The LCF risk and population dose are dominated by SBO, SGTRs, and LOCAs. For SBOs and LOCAs, the early failures of containment dominate the contributions, with less contribution from the late CF. The later failures of containment involve more time for natural deposition mechanisms and mitigation mechanisms such as sprays to reduce the releases to the environment.

Most of the contribution from SGTRs to LCFs and population dose comes from the "H" SGTRs (secondary SRVs stuck open). Although the "H" SGTR accident is unlikely (MCDF about 1.3E-6/R-yr), there is a direct open path from the reactor vessel to the environment throughout the accident. SGTRs were not considered as initiators in the previous version of this analysis[2], so the "H" SGTRs are "new" accidents for the NUREG-1150 pressurized water reactor (PWR) analyses. Thus, their importance to the latent cancer fatality risk was unrecognized at the time the expert panel on source term issues was meeting. After the contribution of the "H" SGTRs was evident, an ad hoc expert panel was convened to consider releases from "H" SGTR accidents (see NUREG-4551 Volume 2, Part 6). This panel concluded that there would be few effective removal mechanisms operating in the release path through the steam generator (SG) and the secondary system safety valves. Thus, the release fractions are high for this accident. Since the onset of core damage occurs about 10 h after the start of the accident for "H" SGTRs, the evacuation is complete before the releases commence; thus, "H" SGTRs are not significant contributors to the EF risk. However, the "H" SGTR accidents significantly contribute to LCF risk and population dose.

The ninth bin that involves accidents in which the vessel does not fail but the containment fails during core degradation (CD) or the containment is not isolated at the uncovering of top of active fuel (UTAF) makes a minor contribution to the EF risk, and a somewhat greater contribution to the LCF risk. It must be remembered that although the vessel does not fail in these accidents, compromise of the containment pressure boundary will allow a portion of the in-vessel releases to escape into the environment. The combination of the threshold effect associated with EFs with the fact that the releases associated with this bin are fairly small results in few EFs. For latent cancers, on the other hand, there is no threshold effect, resulting in higher values for latent cancers.

5.1.3 Contributors to Uncertainty

Figure 5.1-1 provides information on the frequency at which values for individual consequence measures will be exceeded. Specifically, mean, median, 5th percentile, and 95th percentile values are shown for these exceedance frequencies. Thus, Figure 5.1-1 can be viewed as presenting uncertainty analysis results for the risk at Sequoyah due to internal initiators. The 200 underlying exceedance frequency curves (CCDFs) for Figure 5.1-1 are contained in Appendix D.

As the curves in Figure 5.1-1 and in Appendix D show, there is significant uncertainty in the frequency at which a given consequence value will be exceeded. Due to the complexity of the underlying analysis and the concurrent variation of a large number of variables within this analysis, it is difficult to ascertain the cause of this uncertainty on the basis of a simple inspection of the results. However, numerical sensitivity analysis techniques provide a systematic way of investigating the observed variation in exceedance frequencies.

This section presents the results of using regression-based sensitivity analysis techniques to examine the variability in the consequences of internally initiated accidents at Sequoyah. The dependent variable is the risk (units: consequences/year) for each consequence measure. For a given observation in the sample, this variable is obtained by multiplying the each consequence value by its frequency and then summing these products. This variable can be viewed as the result of reducing each of the curves in Figure D.1 to a single number.

The uncertainty analysis techniques used in this study can be viewed as creating a mapping from analysis input to analysis results. The variables sampled in the generation of this mapping are presented in Tables 2.2-5, 2.3-2, and 3.2-2. These variables are the independent variables in the sensitivity studies presented in this section. Variables that are correlated to each other are treated as a single variable in sensitivity analysis. For example, in Table 2.3-2, the variables RCP-SL-P2 through RCP-SL-P4 are all correlated, and therefore, in the sensitivity analysis, they are treated as a single variable (i.e., RCP-SL-P).

Regression-based sensitivity analysis results for EFs and LCFs for all internally initiated events are presented in Table 5.1-4. This table contains the results of performing a stepwise regression on these two measures of risk. The results for individual risk of EF within 1 mile are similar to the results for EFs. The results for population dose within 50 miles, and within the entire region, and individual risk of LCF within 10 miles are similar to the results for LCF. Therefore, these data are not presented here. The statistical package SAS[1] was used to perform the regression.

For EFs and LCFs, Table 5.1-4 lists the variables in the order that they entered the regression analysis, gives the sign (i.e., positive or negative) on regression coefficients for the variables in the final regression model and shows the R^2 values that result with the entry of successive variables into the model. The tendency of a dependent variable to increase with an independent variable is indicated by a positive regression coefficient, and the tendency of a dependent variable to decrease when an independent variable increases is indicated by a negative regression coefficient.

The regression analysis for EFs accounts for about 50% of the observed variability. The independent variables that account for this variability determine the frequency and the magnitude of an early release. The regression analysis for LCF is somewhat less successful, as it is able to account for only 30% of the variability. The independent variables that account for this variability are predominantly those variables that determine the frequencies of the accident.

Because the regression results for all internal events do not account for much of the variability, the same type of stepwise regression analysis was performed for each PDS group. The results from the regression performed for the EFs and LCFs for each PDS group are presented in Tables 5.1-5 through 5.1-11. The most robust results are exhibited for the bypass accidents, PDS Groups 4 and 7, and to a lesser degree, for the ATWS accidents, PDS Group 6. For PDS Group 4, Event V, more than 95% of the variability is explained for both early fatality and latent cancer fatality risks. At least 90% is accounted for by the initiating event frequency of check valve failure in one of the LPIS trains, V-TRAIN. Most of the remaining variability for both risk measures involves the probability that the releases are scrubbed by fire sprays, V-SPRAYS, as well as the decontamination factor associated with the sprays, VDF.

For PDS Group 7, SGTRs, about 80% of the variables for both risk measures is explained: the variables involved include the release fraction from the vessel to the environment, FISGFOSG; the initiating event frequency for SGTRs, IE-SGTR; and the fraction of the fission products released from the core to the vessel, FCOR.

The bypass accidents lend themselves best to analysis with a linear regression model, because the risks are directly related to a product of several variables. For example, for Event V, the risks are directly related to V-TRAIN * FVES * FCOR, and for SGTR, the risks are directly related to IE-SGTR * FCOR * FISGFOSG.

For PDS Group 6, ATWS, much of the risk is associated with the PDS that involves an SGTR. For this group, 65% of the variability is explained for early fatalities, and 86% for latent cancers. The variables involved include the same as mentioned for SGTR, as well as the probability of failure of automatic insertion of control rods, AU-SCRAM, and the probability of failure to effect manual scram due to operator error, MN-SCRAM.

For PDS Groups 1, 2, 3, and 5, the SBO, LOCA and Transient PDS Groups, less than 60% of the variability is explained for both early fatalities and latent cancer fatalities. The models involved with these PDS groups are more complex and nonlinear than for the bypass accidents, and different variables come into play for different degrees of risk measures. Some of the variables that are involved with explaining the variability in the early and latent cancer fatality risks for these PDS Groups include: the CF pressure, CF-PRES; the pressure rise in containment at VB, DP1-VB; the fraction of core that is involved in HPME, FR-HPME; and the decontamination factor for the ice condenser, DF-IC.

When the signs of the regression coefficients are noted, it is seen that most are positive; that is, an increase in the variable tends to increase the consequence. The variables that show negative signs are CF pressure, CF-PRES; probability that the PORVs will stick open, PORV-STK; probability that the releases from Event V, V-SPRAYS are scrubbed; and probability that a T-I RCP seal failure will occur after UTAF, RCP-SL-P. Obviously, increasing the failure pressure of the containment, as well as increasing the probability that the V releases are scrubbed will decrease the consequences. Increase in the other two variables decreases the amount of vessel failures at high pressure, and thus, the CFs at VB as well as the consequences are decreased. The accident frequency variable, RCP-SL-F, that represents the probability of a T-I RCP seal failure before UTAF has a positive sign associated with it because it is related to the accident initiation frequency.

Table 5.1-4
Summary of Regression Analyses for
Annual Risk at Sequoyah for Internal Initiators

	Early Fatalities			Latent Cancer Fatalities		
Step	VAR[a]	RC[b]	R^2[c]	VAR	RC	R^2
1	V-TRAIN	Pos.	0.26	IE-SGTR	Pos.	0.10
2	FVES	Pos.	0.30	CF-PRES	Neg.	0.15
3	RCP-SL-P	Neg.	0.33	DP1-VB	Pos.	0.21
4	CF-PRES	Neg.	0.36	V-TRAIN	Pos.	0.25
5	DP1-VB	Pos.	0.39	SRV-DRPZ	Pos.	0.28
6	FCONV	Pos.	0.41			
7	FISGFOSG	Pos.	0.43			
8	DFIC	Pos.	0.46			
9	FCCI	Pos.	0.48			

[a] Variables listed in the order that they entered the regression analysis.

[b] Sign (positive or negative) on the regression coefficients (RCs) in final regression model.
Pos: Increase in independent variable increases dependent variable.
Neg: Increase in independent variable decreases dependent variable.

[c] R^2 values with the entry of successive variables into the regression model.

Table 5.1-5
Summary of Regression Analyses for
Annual Risk at Sequoyah for PDS Group 1: Slow SBO

	Early Fatalities			Latent Cancer Fatalities		
Step	VAR[a]	RC[b]	R^{2c}	VAR	RC	R^2
1	CF-PRES	Neg.	0.0668	DG-FSTRT	Pos.	0.1227
2	DP1-VB	Pos.	0.1365	CF-PRES	Neg.	0.2075
3	H2-INV	Pos.	0.2009	AC-UNIT2	Pos.	0.2829
4	FR-HPME	Pos.	0.2367	IE-LOSP	Pos.	0.3338
5	DG-FSTRT	Pos.	0.2671	RCP-SL-F	Pos.	0.3869
6	BETA2-DG	Pos.	0.2956	H2-INV	Pos.	0.4305
7	DFIC	Pos.	0.3236	DP1-VB	Pos.	0.4602
8				DG-FRUN6	Pos.	0.4832
9				H2-EXV	Pos.	0.5052
10				FR-HPME	Pos.	0.5234
11				BETA2-DG	Pos.	0.5406

[a] Variables listed in the order that they entered the regression analysis.

[b] Sign (positive or negative) on the RCs in final regression model.
Pos: Increase in independent variable increases dependent variable.
Neg: Increase in independent variable decreases dependent variable.

[c] R^2 values with the entry of successive variables into the regression model.

Table 5.1-6
Summary of Regression Analyses for
Annual Risk at Sequoyah for PDS Group 2: Fast SBO

	Early Fatalities			Latent Cancer Fatalities		
Step	VAR[a]	RC[b]	R^2[c]	VAR	RC	R^2
1	CF-PRES	Neg.	0.0669	DG-FSTRT	Pos.	0.1216
2	DG-FSTRT	Pos.	0.1065	CF-PRES	Neg.	0.1913
3	TDP-FSTR	Pos.	0.1456	IE-LOSP	Pos.	0.2586
4	RCP-SL-P	Neg.	0.1845	TDP-FSTR	Pos.	0.3101
5	H2-EXV	Pos.	0.2284	H2-EXV	Pos.	0.3440
6	DP1-VB	Pos.	0.2722	DG-FRUN6	Pos.	0.3778
7	RCP-SL-F	Pos.	0.3053	DP1-VB	Pos.	0.4083
8				RCP-SL-F	Pos.	0.4338
9				RCP-SL-P	Neg.	0.4557
10				HE-XTIE	Pos.	0.4780
11				BETA2-DG	Pos.	0.5042

[a] Variables listed in the order that they entered the regression analysis.

[b] Sign (positive or negative) on the RCs in final regression model.
Pos: Increase in independent variable increases dependent variable.
Neg: Increase in independent variable decreases dependent variable.

[c] R^2 values with the entry of successive variables into the regression model.

Table 5.1-7
Summary of Regression Analyses for
Annual Risk at Sequoyah for PDS Group 3: LOCAs

	Early Fatalities			Latent Cancer Fatalities		
Step	VAR[a]	RC[b]	R^2[c]	VAR	RC	R^2
1	FR-HPME	Pos.	0.0671	HE-FCV	Pos.	0.1345
2	VL-CCI	Pos.	0.1218	MOV-FOPN	Pos.	0.1797
3	CF-PRES	Neg.	0.1617	CF-PRES	Neg.	0.2133
4	DFIC	Pos.	0.1986	VB-ALPHA	Pos.	0.2415
5	VB-ALPHA	Pos.	0.2393	DP1-VB	Pos.	0.2678
6	FCONV	Pos.	0.2808			
7	MOV-FOPN	Pos.	0.3058			
8	AFW-STMB	Pos.	0.3301			

[a] Variables listed in the order that they entered the regression analysis.

[b] Sign (positive or negative) on the RCs in final regression model.
Pos: Increase in independent variable increases dependent variable.
Neg: Increase in independent variable decreases dependent variable.

[c] R^2 values with the entry of successive variables into the regression model.

Table 5.1-8
Summary of Regression Analyses for
Annual Risk at Sequoyah for PDS Group 4: Event V

	Early Fatalities			Latent Cancer Fatalities		
Step	VAR[a]	RC[b]	R^2 [c]	VAR	RC	R^2
1	V-TRAIN	Pos.	0.8959	V-TRAIN	Pos.	0.9651
2	V-SPRAYS	Neg.	0.9132	VDF	Pos.	0.9787
3	FCONC	Pos.	0.9285	V-SPRAYS	Neg.	0.9835
4	VDF	Pos.	0.9440			
5	FCONV	Pos.	0.9537			
6	FVES	Pos.	0.9634			

[a] Variables listed in the order that they entered the regression analysis.

[b] Sign (positive or negative) on the RCs in final regression model.
Pos: Increase in independent variable increases dependent variable.
Neg: Increase in independent variable decreases dependent variable.

[c] R^2 values with the entry of successive variables into the regression model.

Table 5.1-9
Summary of Regression Analyses for
Annual Risk at Sequoyah for PDS Group 5: Transients

	Early Fatalities			Latent Cancer Fatalities		
Step	VAR[a]	RC[b]	R^2[c]	VAR	RC	R^2
1	H2-INV	Pos.	0.1052	H2-INV	Pos.	0.1724
2	FR-HPME	Pos.	0.1530	BETA8AOV	Pos	0.2384
3	FCOR	Pos.	0.1966	HE-FDBLD	Pos.	0.3013
4	CF-PRES	Neg.	0.2392	PORV-STK	Neg.	0.3487
5	DP1-VB	Pos.	0.2734	MDP-FSTR	Pos.	0.3951
6	PORV-STK	Neg.	0.3048	FR-HPME	Pos.	0.4342
7				CNT-ISO	Pos.	0.4640
8				FCOR	Pos.	0.4887
9				IE-LMFWS	Pos.	0.5103
10				CF-PRES	Neg.	0.5304

[a] Variables listed in the order that they entered the regression analysis.

[b] Sign (positive or negative) on the RCs in final regression model.
Pos: Increase in independent variable increases dependent variable.
Neg: Increase in independent variable decreases dependent variable.

[c] R^2 values with the entry of successive variables into the regression model.

Table 5.1-10
Summary of Regression Analyses for
Annual Risk at Sequoyah for PDS Group 6: ATWS

	Early Fatalities			Latent Cancer Fatalities		
Step	VAR[a]	RC[b]	R^{2c}	VAR	RC	R^2
1	FISGFOSG	Pos.	0.2728	IE-SGTR	Pos.	0.3016
2	FCOR	Pos.	0.4498	AU-SCRAM	Pos.	0.5718
3	IE-SGTR	Pos.	0.5483	MN-SCRAM	Pos.	0.7317
4	AU-SCRAM	Pos.	0.6201	FISGFOSG	Pos.	0.7875
5	MN-SCRAM	Pos.	0.6554	FCOR	Pos.	0.8261
6				VB-ALPHA	Pos.	0.8432
7				UNFV-MOD	Pos.	0.8556
8				H2-INV	Pos.	0.8631

[a] Variables listed in the order that they entered the regression analysis.

[b] Sign (positive or negative) on the RCs in final regression model.
Pos: Increase in independent variable increases dependent variable.
Neg: Increase in independent variable decreases dependent variable.

[c] R^2 values with the entry of successive variables into the regression model.

Table 5.1-11
Summary of Regression Analyses for
Annual Risk at Sequoyah for PDS Group 7: SGTRs

	Early Fatalities			Latent Cancer Fatalities		
Step	VAR[a]	RC[b]	R^{2c}	VAR	RC	R^2
1	FISGFOSG	Pos.	0.3946	IE-SGTR	Pos.	0.4033
2	FCOR	Pos.	0.6178	FISGFOSG	Pos.	0.5708
3	IE-SGTR	Pos.	0.7507	SRV-DPRZ	Pos.	0.6459
4	MFW-FRST	Pos.	0.7671	FCOR	Pos.	0.7110
5	CF-PRES	Neg.	0.7799	HE-DPRSG	Pos.	0.7574
6	MDP-FSTR	Pos.	0.7906	MS-LIAS	Pos.	0.7723
7				MFW-FRST	Pos.	0.7823
8				MDP-FSTR	Pos.	0.7923
9				CF-PRES	Pos.	0.8010

[a] Variables listed in the order that they entered the regression analysis.

[b] Sign (positive or negative) on the RCs in final regression model.
Pos: Increase in independent variable increases dependent variable.
Neg: Increase in independent variable decreases dependent variable.

[c] R^2 values with the entry of successive variables into the regression model.

5.2 References

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2. USNRC, "Severe Accident Risks: An Assessment of Five Nuclear Power Plants," U. S. Nuclear Regulatory Commission, NUREG-1150, June 1989.

6. INSIGHTS AND CONCLUSIONS

Core Damage Arrest. The inclusion of the possibility of arresting the core degradation (CD) process before vessel failure is an important feature of this analysis. For internal initiators, there is a good chance that non-bypass accidents will be arrested before vessel failure. This may be due to the recovery of offsite power (ROSP) or the reduction of reactor coolant system (RCS) pressure to the point where an operable system can inject. The arrest of core damage before vessel breach (VB) plays an important part in reducing the risk due to the most frequent types of internal accidents: loss-of-coolant accidents (LOCAs) and station blackouts (SBOs).

Depressurization of the RCS. Depressurization of the RCS before the vessel fails is important in reducing the loads placed upon the containment at VB and in arresting core damage before VB. For accidents in which the RCS is at the power-operated relief valve (PORV) setpoint pressure during CD, the effective mechanisms for pressure reduction are temperature-induced (T-I) failure of the hot leg or surge line, T-I failure of the RCP seals, and the sticking open of the PORVs. All of these mechanisms are inadvertent and beyond the control of the operators. The apparent beneficial effects of reducing the pressure in the RCS when lower head failure is imminent indicate that further investigation of depressurization may be warranted. The dependency of the probability of containment failure (CF) on RCS pressure boundary failures that occur at unpredictable locations and at unpredictable times is somewhat unsettling. Studies of the effects of increasing PORV capacity, providing the means to open the PORVs in blackout situations, and changing the procedures to remove restrictive conditions on deliberate RCS pressure reduction might prove rewarding in decreasing the probability of early CF at pressurized water reactors (PWRs). Depressurization may involve the loss of considerable inventory from the RCS. Any studies undertaken should consider possible drawbacks as well as benefits.

Containment Failure. If a core damage accident proceeds to the point where the lower head of the reactor vessel fails, the containment is not likely to fail at this time. This is partially due to the depressurization of the RCS before vessel failure, partially due to deep-flooding of the reactor cavity, which inhibits dispersal of core debris from the cavity in high pressure accidents, and partially due to the strength of the Sequoyah containment relative to the loads expected. Hydrogen burns before VB for the SBO accidents and hydrogen burn/direct containment heating (DCH) events are the factors that lead to early CFs when they do occur. Early CFs contribute significantly to the risks that depend on a large early release (early fatalities (EFs)) and are major contributors to the risks that are functions of the total release (latent cancer fatalities (LCFs) and population dose). For SBOs, late failures occur from hydrogen burns upon power recovery during core-concrete interaction (CCI). Very late failures that are many hours after VB depend upon the availability of containment heat removal (CHR). If CHR is recovered within a day or so, basemat melt-through is the most probable failure mode. If CHR is not recovered, an overpressure failure within a day or two after the start of the accident is the likely mode.

Bypass Accidents. Bypass accidents are major contributors to the risks that depend on a large early release as well as those that are functions of the total release. Event V is the accident most likely to result in a large, early release for internal initiators. Steam generator tube ruptures (SGTRs) are also important contributors to large releases, but most of the large releases due to SGTRs occur many hours after the start of the accident, and thus they contribute significantly to the risks that depend on the total release. The most important SGTRs are those in which the SRVs on the secondary system stick open. Although the bypass accidents are not the most frequent types of internal accidents, the somewhat low probability of CF (especially early CF) for the non-bypass accidents results in the large contributions of the bypass accidents to risk.

Fission Product Releases. There is considerable uncertainty in the release fractions for all types of accidents. There are several features of the Sequoyah plant that tend to mitigate the release. First, the in-vessel releases are generally directed to the ice condenser where they experience some decontamination. If the sprays are operating, the radionuclides will also contribute to the decontamination of the releases. The reactor cavity pool also offers a mechanism for reducing the release of radionuclides from CCI. The largest releases tend to occur when the containment is bypassed or when early failure of containment involving catastrophic rupture occurs. Catastrophic rupture is assumed to cause bypass of the ice condenser and failure of the containment sprays.

Uncertainty in Risk. Considerable uncertainty is associated with the risk estimates produced in this analysis. The largest contributors to the uncertainty in EFs and LCFs for the bypass sequences are the variability in frequencies of the initiating events and the uncertainty in some of the parameters that determine the magnitude of the fission product release to the environment. For non-bypass accidents, the variability in frequencies of the initiating events and the uncertainty in the accident progression parameters and probabilities contribute to the uncertainty in latent cancers. The contribution to the uncertainty in EFs for non-bypass accidents arises from variability in all the constituent analyses that were incorporated into the uncertainty analysis: initiating events, accident progression, and fission product release.

Comparison with the Safety Goals. For both the individual risk of EF within one mile of the site boundary and the individual risk of LCF within 10 miles, the mean annual risk and even the 95th percentile for annual risk fall more than an order of magnitude below the safety goals. Indeed, even the maximum of the 200 values that make up the annual risk distributions falls well below the safety goals.

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11. ABSTRACT (200 words or less)

In support of the U.S. Nuclear Regulatory Commission's assessment of the risk from severe accidents at commercial nuclear power plants in the U.S. reported in NUREG-1150, the Severe Accident Risk Reduction Program has completed a revised calculation of the risk to the general public from severe accidents at the Sequoyah Power Station, Unit 1. This power plant, located in southeastern Tennessee, is operated by the Tennessee Valley Authority.

The emphasis in this risk analysis was not on determining a "so-called" point estimate of risk. Rather it was to determine the distribution of risk, and to discover the uncertainties that account for the breadth of this distribution. Off-site risks from initiating events internal to the power station were assessed.

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