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# Evaluation of Severe Accident Risks: Grand Gulf, Unit 1

Main Report

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Prepared by  
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S. J. Higgins, C. N. Amos, A. W. Shiver

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## Main Report

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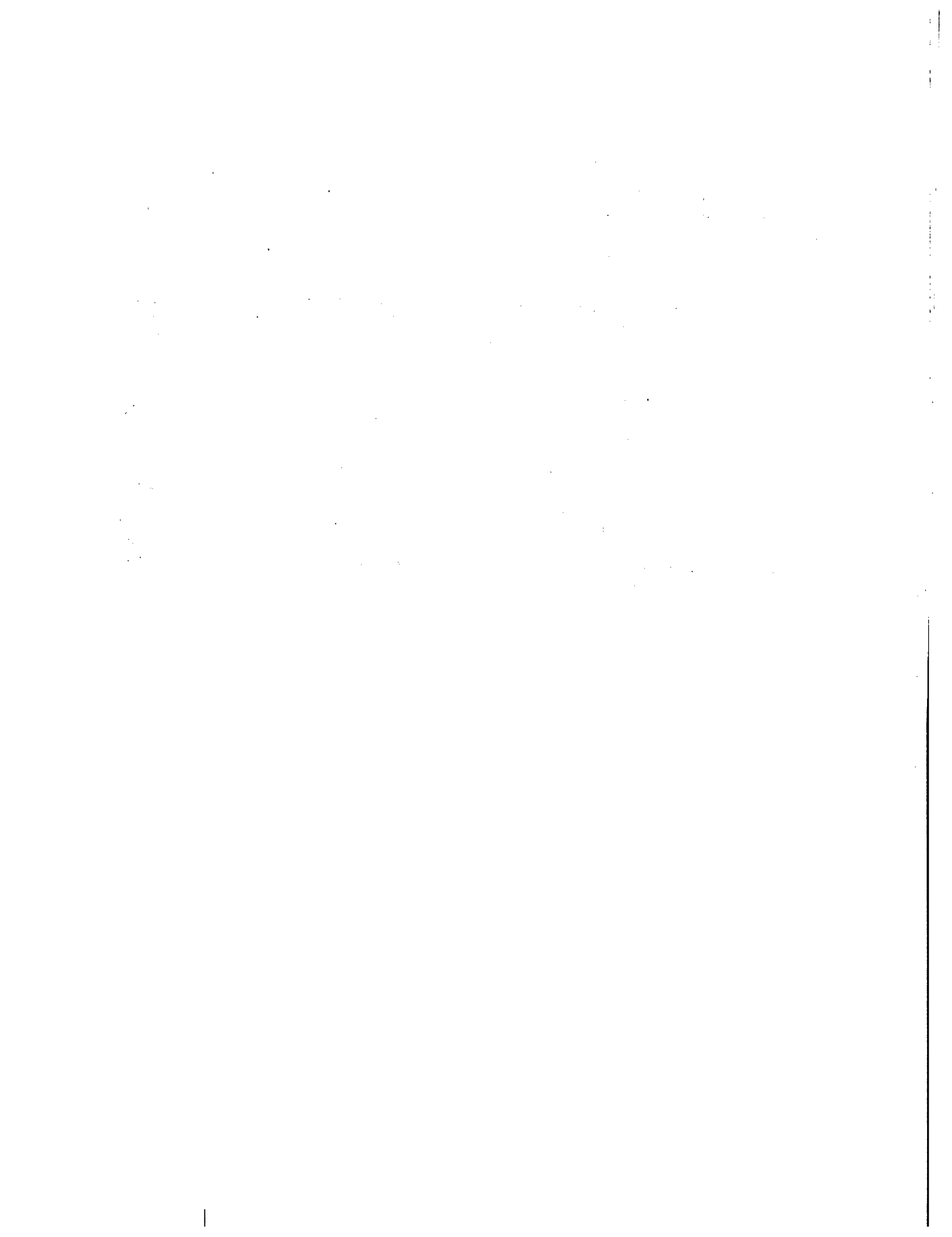
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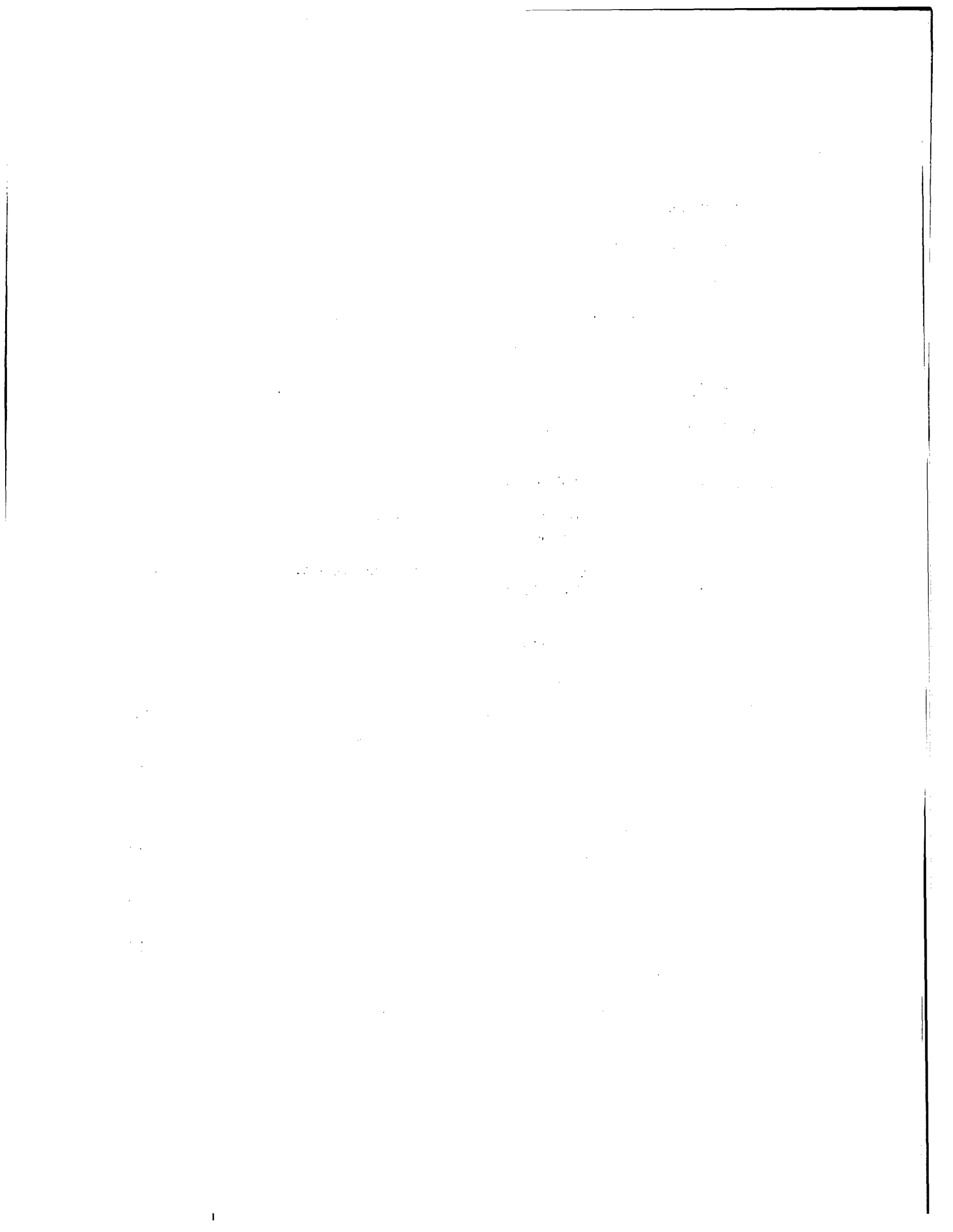
## ABSTRACT

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 severe accidents at the Grand Gulf Nuclear Station, Unit 1. This power plant, located in Port Gibson, Mississippi, is operated by the System Energy Resources, Inc. (SERI).

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, both with respect to the safety goals and to the other plants analyzed in NUREG-1150. The offsite risk is dominated by station blackout type accidents (loss of all ac power) in which core damage occurs shortly after the initiation of the accident. The low values for risk can be attributed to the low core damage frequency, the good emergency response, and plant features that reduce the potential source term. Given that core damage occurs, it appears quite likely that the containment will fail during the accident. Hydrogen combustion events are the dominant causes of containment failure. Considerable uncertainty is associated with the risk estimates produced in this analysis.





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## 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 and for the revised draft of NUREG-1150 are given in Table 1. They were produced by the three interfacing programs at Sandia National Laboratories (SNL) that performed the work: the Accident Sequence Evaluation Program (ASEP), the Severe Accident Risk Reduction Program (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 utilized 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 Grand Gulf analysis is contained in Volume 6 of NUREG/CR-4551. Note that the Grand Gulf plant was also treated in Volume 4 of the original Draft for Comment version of 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 the External Event Core Damage Frequency Analysis for NUREG-1150," M. P. Bohn and J. A. Lambright, Sandia National Laboratories, Albuquerque, NM, December 1990

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, 1989.



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

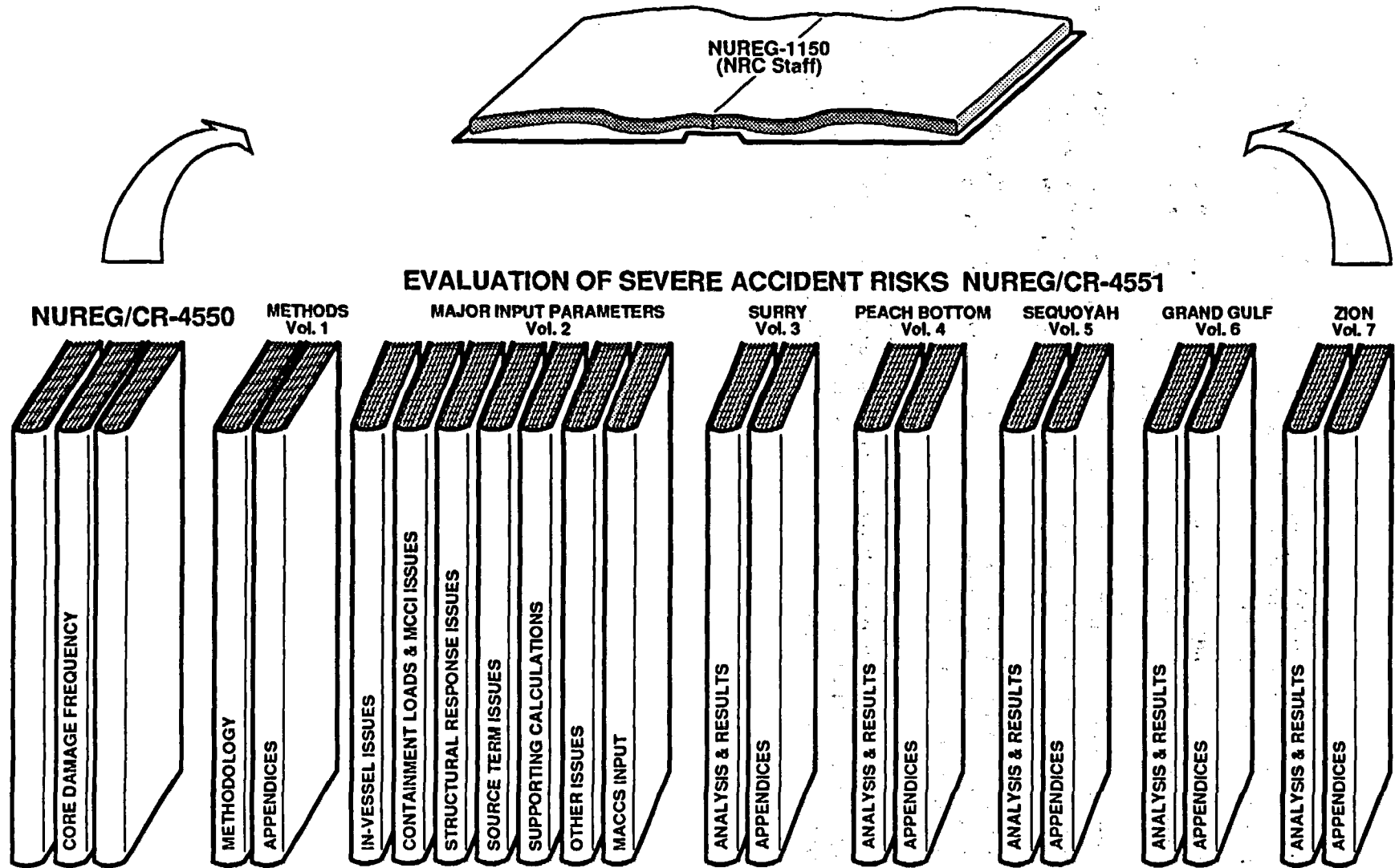
NUREG/CR-4624, BMI-2139, R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios," Volumes I-V, Battelle's Columbus Division, Columbus, OH, 1986.

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, 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.

# SUPPORT DOCUMENTS TO NUREG-1150



AX

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

Table 1. NUREG-1150 Analysis Documentation

<u>Original Documentation</u>			
NUREG/CR-4550		NUREG/CR-4551	NUREG/CR-4700
Analysis of Core Damage Frequency From Internal Events		Evaluation of Severe Accident Risks and the Potential for Risk Reduction	Containment Event Analysis for Potential Severe Accidents
Vol.	1 Methodology	Vol.	1 Surry Unit 1
	2 Summary (Not Published)		2 Sequoyah Unit 1
	3 Surry Unit 1		3 Peach Bottom Unit 2
	4 Peach Bottom Unit 2		4 Grand Gulf Unit 1
	5 Sequoyah Unit 1		
	6 Grand Gulf Unit 1		
	7 Zion Unit 1		
<u>Revised Documentation</u>			
NUREG/CR-4550, Rev. 1, Analysis of Core Damage Frequency		NUREG/CR-4551, Rev. 1, Eval. of Severe Accident Risks	
Vol.	1 Methodology	Vol.	1 Part 1, Methodology; Part 2, Appendices
	2 Part 1 Expert Judgment Elicit. Expert Panel		2 Part 1 In-Vessel Issues
	Part 2 Expert Judgment Elicit. Project Staff		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 Unit 1 Internal Events	3	Part 1 Surry Analysis and Results
	Part 2 Surry Unit 1 Internal Events App.		Part 2 Surry Appendices
	Part 3 Surry External Events		
4	Part 1 Peach Bottom Unit 2 Internal Events	4	Part 1 Peach Bottom Analysis and Results
	Part 2 Peach Bottom Unit 2 Int. Events App.		Part 2 Peach Bottom Appendices
	Part 3 Peach Bottom Unit 2 External Events		
5	Part 1 Sequoyah Unit 1 Internal Events	5	Part 1 Sequoyah Analysis and Results
	Part 2 Sequoyah Unit 1 Internal Events App.		Part 2 Sequoyah Appendices
6	Part 1 Grand Gulf Unit 1 Internal Events	6	Part 1 Grand Gulf Analysis and Results
	Part 2 Grand Gulf Unit 1 Internal Events App.		Part 2 Grand Gulf Appendices
7	Zion Unit 1 Internal Events	7	Part 1 Zion Analysis and Results
			Part 2 Appendices

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## ACRONYMS AND INITIALISMS

ADS automatic depressurization system  
APB accident progression bin  
APET accident progression event tree  
ASEP accident sequence evaluation program  
ATWS anticipated transient without scram

BAF bottom of active fuel  
BNL Brookhaven National Laboratory  
BWR boiling water reactor

CCF common cause failure  
CCI core-concrete interaction  
CCDF complementary cumulative distribution function  
CDF cumulative distribution function  
CF containment failure  
CFW chronic fatality weight  
CS containment spray system  
CST condensate storage tank

DCH direct containment heating  
DG diesel generator

ECCS emergency core cooling system(s)  
EF early fatalities  
EFW early fatality weight  
EOP emergency operating procedures  
EPRI Electric Power Research Institute  
EVSE ex-vessel steam explosion

FSAR final safety analysis report  
FWS firewater system

HEP human error probability  
HIS hydrogen ignition system  
HPCS high pressure core spray  
HPME high pressure melt ejection  
HRA human reliability analysis

INEL Idaho National Engineering Laboratory

LCF latent cancer fatalities  
LHS Latin Hypercube Sampling  
LOCA loss-of-coolant accident  
LOSP loss of offsite power  
LPCI low pressure coolant injection  
LPCS low pressure core spray  
LTSB long-term station blackout  
LWR light water reactor

MACCS MELCOR Accident Consequence Code System  
MCDF mean core damage frequency

MDP motor-driven pump  
 MOV motor-operated valve  
 MSIV main steam isolation valve  
  
 NRC Nuclear Regulatory Commission  
  
 PCS power conversion system  
 PDS plant damage state  
 PRA probabilistic risk analysis  
 PRUEP Phenomenology and Risk Uncertainty Evaluation Program  
 PWR pressurized water reactor  
  
 RCIC reactor core isolation cooling  
 RCS reactor coolant system  
 RHR residual heat removal  
 RPS reactor protection system  
 RSS Reactor Safety Study  
 RPV reactor pressure vessel  
  
 SAIC Science Applications International Corporation  
 SAROS Safety and Reliability Optimization Services, Inc.  
 SARRP Severe Accident Risk Reduction Program  
 SBO station blackout  
 SERG steam explosion review group  
 SERI System Energy Resources, Inc.  
 SLC standby liquid control  
 SNL Sandia National Laboratories  
 SORV stuck-open relief valve  
 SPC suppression pool cooling  
 SPMU suppression pool makeup  
 SRV safety relief valve  
 SSW standby service water  
 STSB short-term station blackout  
  
 TDP turbine-driven pump  
 TEMAC Top Event Matrix Analysis Code  
  
 VB vessel breach



## 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<sup>1</sup>, 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 Grand Gulf Nuclear Station, Unit 1. The first step is described in NUREG/CR-4550.<sup>2</sup>

### S.2 Overview of Grand Gulf Nuclear Station, Unit 1

The Grand Gulf Nuclear Station, Unit 1 is operated by System Energy Resources Inc. (SERI) and is located on the east bank of the Mississippi river in southwestern Mississippi, about six miles northwest of Port Gibson, Mississippi. The nearest large city is Jackson, Mississippi, approximately 55 miles to the northeast of the plant.



The nuclear reactor of Grand Gulf Unit 1 is a 3833 Mwt BWR-6 boiling water reactor (BWR) designed and supplied by General Electric Company. Unit 1, constructed by Bechtel Corporation, began commercial operation in July 1985.

Table S.1 summarizes the design features of the plant that are relevant to severe accidents. As is evident from this table, there is considerable redundancy and diversity of coolant injection and heat removal features at Grand Gulf. Grand Gulf has a Mark III BWR containment. The containment is a steel-lined reinforced concrete structure. In the Mark III design the reactor pressure vessel is housed in the drywell, which is in turn completely enclosed in the containment structure. The drywell and the containment communicate through passive vents in the suppression pool. Although the free volume of the containment is comparable with a large PWR containment, the design pressure of the Grand Gulf containment is fairly low (15 psig).

### S.3 Description of the Integrated Risk Analysis

Risk is determined by combining the results of four constituent analyses: the 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 Grand Gulf consisted of 250 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. This figure shows, in the boxes, the computer codes utilized. 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 PDS 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 Grand Gulf 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.

Table S.1  
Design Features Relevant to Severe Accidents  
Grand Gulf Unit 1

**Coolant Injection Systems**

- High Pressure Core Spray System (HPCS)**  
One train, one MDP\*  
Dedicated diesel generator
- Reactor Core Isolation Cooling System (RCIC)**  
One train, one TDP\*
- Low-Pressure Core Spray System (LPCS)**  
One train, one MDP\*
- Low-Pressure Coolant Injection System (LPCI)**  
Three trains, three MDP\*
- Backup Coolant Injection Systems**  
Standby service water system  
Firewater system  
Condensate system
- Automatic Depressurization System (ADS)**  
Eight relief valves  
Requires dc power

**Heat Removal Systems**

- Residual Heat Removal System**  
Suppression pool cooling mode:  
Removes decay heat from suppression pool--  
two trains, two MDP\*
- Shutdown Cooling System:**  
Removes decay heat during accidents in which  
reactor vessel integrity maintained and reactor  
pressure vessel (RPV) is at low pressure--  
two trains, two MDP\*
- Containment Spray System:**  
Suppression pressure in containment--  
two trains, two MDP\*

**Reactivity Control**

- Control Rods**
- Standby Liquid Control System**

Table S.1 (continued)

<b>Emergency Electrical Power</b>	<p><b>Electrical Power (ac)</b> Two diesel generators (DGs) HPCS diesel generator has crossties</p> <p><b>Electrical Power (dc)</b> 12-hour station batteries</p>
<b>Containment Structure</b>	<p><b>BWR Mark III</b> Reinforced concrete structure with steel liner Design pressure of 15 psig Volume is 1.67 million ft<sup>3</sup>. Free volume of 1.4 million ft<sup>3</sup></p>
<b>Drywell Structure</b>	<p>Completely enclosed within containment structure Communicates with wetwell through horizontal vents Internal design pressure of 30 psid Free volume of 270,000 ft<sup>3</sup></p>
<b>Reactor Pedestal Cavity</b>	<p>Cylindrical cavity located directly below RPV Water on drywell floor will drain into the cavity Volume of the cavity is large enough to contain any core debris released from the vessel</p>
<b>Containment Systems</b>	<p><b>Hydrogen Igniter System (HIS)</b> Prevents the buildup of large quantities of hydrogen in the containment --requires ac power</p> <p><b>Containment Venting</b> Used when suppression pool cooling and containment sprays have failed to reduce primary containment pressure--requires ac power</p>

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\*MDP - motor-driven pump  
TDP - turbine-driven pump

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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

S.S

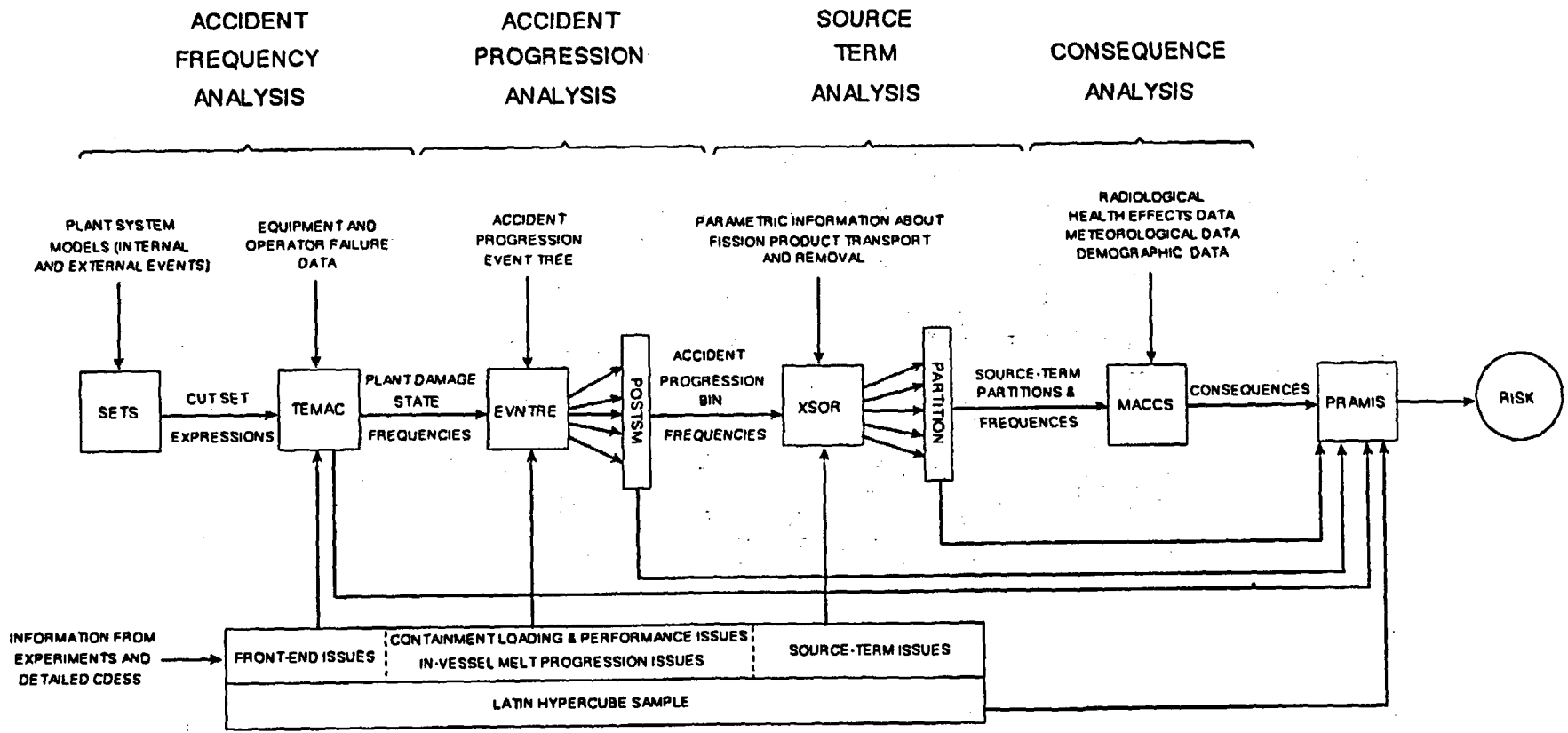


Figure S.1. Overview of Integrated Plant Analysis in NUREG-1150.

progression analysis is a probability for each APB, conditional on the occurrence of a PDS, for each observation in the sample.

A source term is calculated for each APB with a non-zero conditional probability for each observation in the sample by GGSOR, a fast-running parametric computer code. GGSOR is not a detailed mechanistic model; it is not designed to model the fission product transport, physics, and chemistry from first principles. Instead, GGSOR integrates the results of many detailed codes and the conclusions of many experts. Most of the parameters used to calculate fission product release fractions in GGSOR are sampled from distributions provided by an expert panel. Because of the large number of APBs, use of a fast-executing code like GGSOR 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 have 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 are included in 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 every 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 every PDS group and 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 Grand Gulf is documented elsewhere.<sup>2</sup> 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 Grand Gulf for the twelve internally initiated PDSs. The four summary measures are the mean and the 5th, 50th (median) and 95th percentiles.

PDSs 1, 2, 3, and 7 involve station blackout scenarios in which coolant injection is lost early such that core damage occurs in the short term with the RPV at high pressure. For PDSs 1, 2, and 3, offsite power is recoverable and the operators can depressurize the RPV. For PDSs 2 and 3 heat removal via the containment sprays is failed and not recoverable. For PDSs 1, 2, and 3 the core damage process may be arrested before the vessel fails if offsite power is recovered and coolant injection is restored to

Table S.2  
Grand Gulf Core Damage Frequencies  
Internal Initiators

PDS	Core Damage Frequency (1/R-yr)				‡ Mean TCD
	5‡	Median	Mean	95‡	Frequency
PDS-1	2.6E-08	5.1E-07	3.2E-06	1.1E-05	79
PDS-2	6.4E-11	2.1E-09	4.6E-08	1.9E-07	1
PDS-3	1.3E-09	3.4E-08	1.5E-07	6.7E-07	4
PDS-4	5.3E-11	2.3E-09	3.7E-08	1.6E-07	1
PDS-5	7.4E-13	3.2E-11	2.3E-09	3.0E-09	<<1
PDS-6	1.4E-12	1.3E-10	1.4E-09	7.2E-09	<<1
PDS-7	2.8E-08	2.4E-07	4.2E-07	1.6E-06	11
PDS-8	2.6E-10	8.4E-09	6.3E-08	2.7E-07	2
PDS-9	3.2E-10	7.9E-09	5.0E-08	1.9E-07	1
PDS-10	3.9E-10	8.9E-09	6.2E-08	2.3E-07	2
PDS-11	3.1E-11	1.2E-09	1.8E-08	5.3E-08	<1
PDS-12	4.9E-12	6.8E-11	2.9E-10	1.2E-09	<<1
Total	1.8E-07	1.1E-06	4.1E-06	1.4E-05	

the core. PDS 7 is different from the first three PDS in that both ac and dc power are lost and cannot be recovered. Except for the unlikely event that a safety relief valve (SRV) sticks open and depressurizes the RPV which then allows the fire water system to be used as a backup source of coolant injection, accidents that progress from this PDS always proceed to vessel failure. The PDS group that includes these four PDSs is referred to as the short-term station blackout (STSB) or STSB group.

PDSs 4, 5, 6, and 8 involve station blackout scenarios in which coolant injection is lost late such that core damage occurs in the long term. For PDSs 4, 5, and 6 core damage occurs with the RPV at low pressure and offsite power is recoverable. For PDSs 5 and 6 heat removal via the containment sprays is failed and not recoverable. For PDSs 4, 5, and 6 the core damage process may be arrested before the vessel fails if offsite power is recovered and coolant injection is restored to the core. PDS 8 is

different from the other 3 PDS in that both ac and dc power are lost and cannot be recovered. Thus, for accidents that progress from this PDS, the vessel always fails at high pressure. The PDS group that includes these four PDSs is referred to as the long-term station blackout (LTSB) or LTSB group.

PDSs 9 and 10 involve anticipated transient without scram (ATWS) scenarios. For PDS 9 coolant injection is lost early such that core damage occurs in the short term whereas for PDS 10 injection is lost late such that core damage occurs in the long term. For both PDSs, core damage occurs because the operators fail to depressurize the vessel to allow low pressure injection systems to cool the core. If the operators correct this error sufficiently early in the accident, the core damage process can be arrested before the vessel fails. The PDS group that includes these two PDSs is referred to as the ATWS group.

PDSs 11 and 12 involve transient scenarios where the power conversion system (PCS) is lost (i.e., T2). For PDS 11 coolant injection is lost early such that core damage occurs in the short term whereas for PDS 12 injection is lost late such that core damage occurs in the long term. For both PDSs core damage occurs because the operators fail to depressurize the vessel to allow low pressure injection systems to cool the core. If the operators correct this error sufficiently early in the accident, the core damage process can be arrested before the vessel fails. In both PDSs heat removal via the containment sprays is possible. The PDS group that includes these two PDSs is referred to as the transient or T2 group.

## 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 such as 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 the various events that may lead to the release of fission products due to the accident. The Grand Gulf APET consists of 125 questions, most of which have more than two branches. Four 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 RPV allows injection by a low pressure injection system that previously could not function with the RPV at high pressure. Containment failure is considered before vessel breach, around the time of vessel breach, and late in the accident. The dominant events that can cause containment failure are hydrogen combustion events (both deflagrations and detonations) and the accumulation of steam and/or noncondensibles in the containment.

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 accident progression bins (APBs).

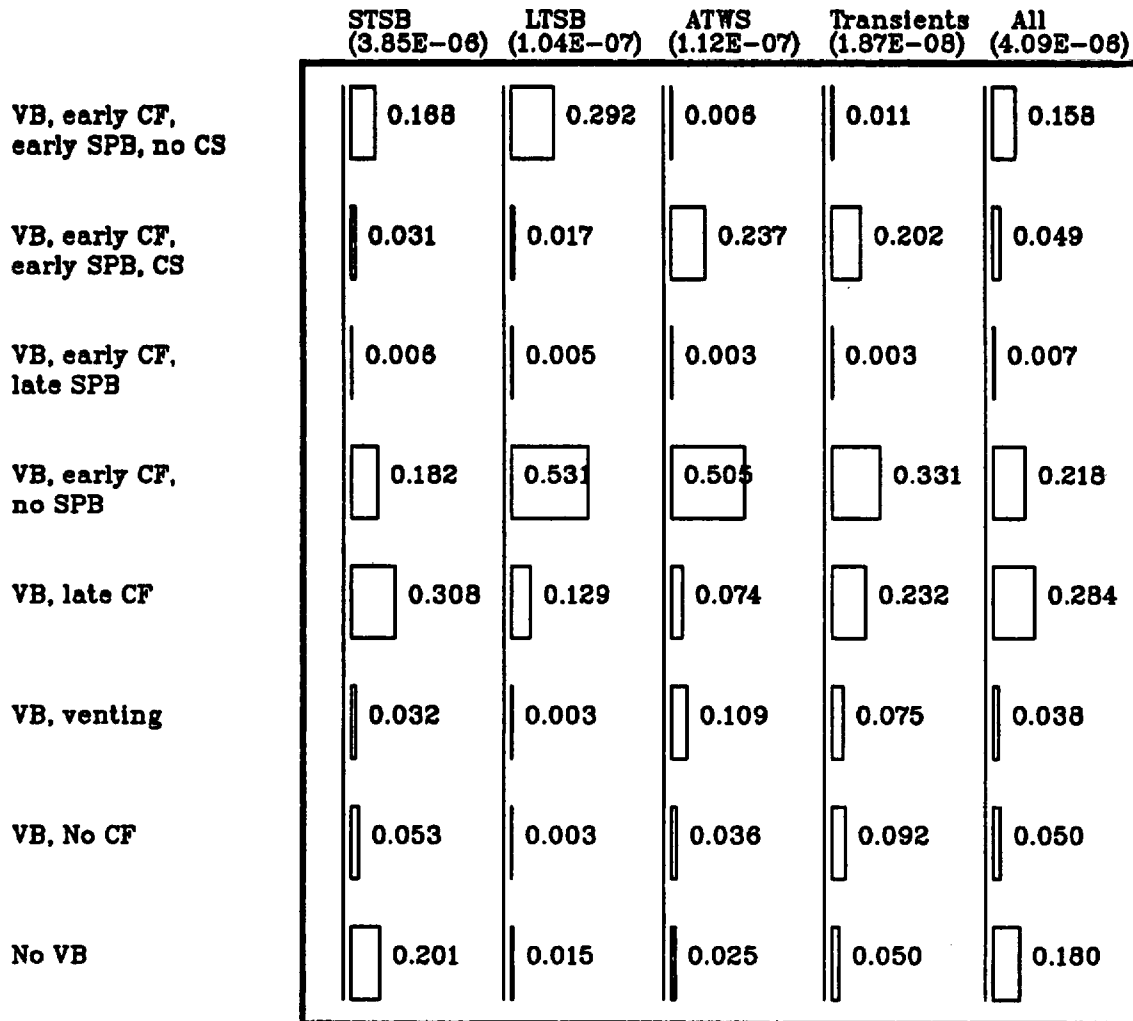
### S.5.2 Results of the Accident Progression Analysis

Results of the accident progression analysis for internal initiators at Grand Gulf are summarized in Figures S.2, S.3, and S.4. Figure S.2 shows the mean distribution among the summary accident progression bins 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 (CF) for a given PDS group are displayed in Figure S.4. Early CF any time before vessel breach, at vessel breach, or shortly following vessel breach.

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. Because roughly 90% of the total mean core damage frequency is attributed to the short-term station blackout (SBO) summary PDS group, the results presented in the frequency weighted average column are heavily influenced by the short-term SBO results. If the accident proceeds to core damage, containment failure during the accident is a likely outcome. The mean conditional probability of early containment failure is approximately 0.50 and half of this mean value is associated with accidents that also involve some bypass of the suppression pool (i.e., drywell failure).

If the accident proceeds to vessel breach and the containment does not fail early, there is still a fairly high probability that the containment will fail late in the accident. Events that can fail the containment late in the accident are hydrogen burns and the accumulation of noncondensibles and steam in the containment. In the SBO PDSs ac power may not be available late in the accident and, thus, the containment sprays will not be available to condense the steam. Furthermore, even if the sprays are available, the accumulation of noncondensibles generated at vessel breach and during CGI may still fail the containment.





CF = Containment Failure  
 CS = Containment Sprays  
 CV = Containment Venting  
 SPB = Suppression Pool Bypass  
 VB = Vessel Breach

Grand Gulf

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

Containment venting is not a likely outcome in this analysis. There are several reasons for this result. First, the dominant PDSs are the short-term SBOs. In these PDSs the suppression pool remains subcooled during core damage and, therefore, the containment is not pressurized by the accumulation of steam. During core damage and after vessel breach a significant quantity of radionuclides will be released to the containment. After vessel breach it is unlikely that the operator will vent these releases to the outside environment.

The results of this analysis indicate that there is a high likelihood that the reactor cavity will contain water at vessel breach. With respect to containment integrity and radionuclide release, this situation has both disadvantages and advantages. The presence of water allows for the

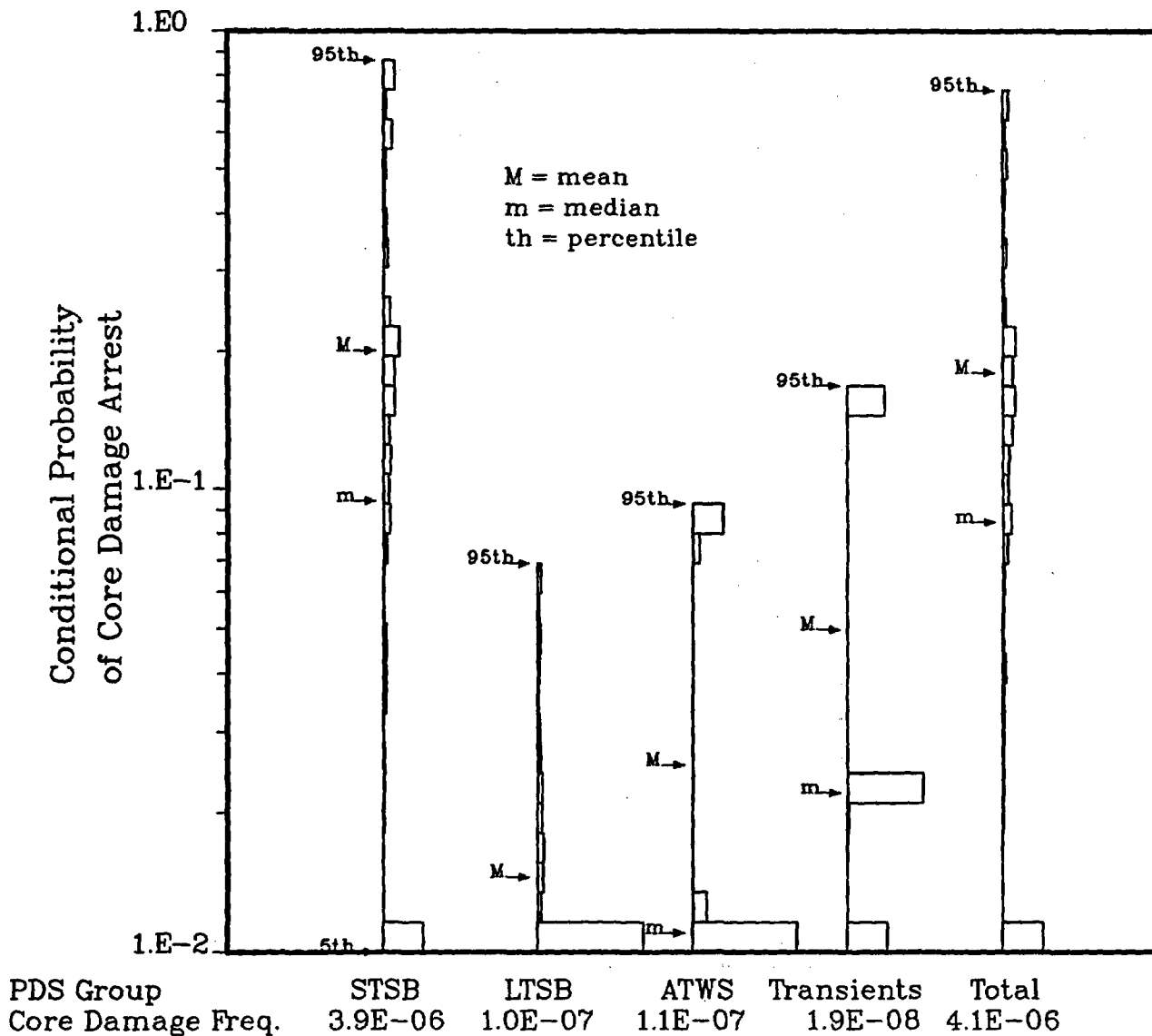


Figure S.3. Probability of Core Damage Arrest.

possibility of ex-vessel steam explosions which can indirectly threaten the integrity of the drywell through the failure of the reactor pedestal. An ex-vessel steam explosion also contributes to radionuclide release at vessel breach. On the other hand, this water also contributes to the high 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.

Core Damage Arrest. For the short-term SBO group the probability of core damage arrest is driven by the likelihood that ac power is recovered early in the accident. Injection to the RPV generally follows ac power recovery. Although the mean probability of recovering ac power is high (0.60) for most of the short-term SBO PDSs, there are several factors that tend to reduce the probability of core damage arrest. First, restoration of coolant injection to the RPV does not guarantee that the vessel will not



As with the short-term SBO group, the probability of core damage arrest for the long-term SBO group is also driven by the likelihood that ac power is recovered. The probability of core damage arrest for the long-term SBO group, however, is significantly lower than the corresponding value for the short-term SBO group. Two factors are responsible for most of this difference. First, the probability of recovering ac power during a long term SBO is considerably lower than the probability of recovering ac power during a short term SBO. Second, in PDS 8, which accounts for approximately half of this group's mean frequency, ac power cannot be recovered and the accident always proceeds to vessel breach.

For both the ATWS PDS group and the T2 PDS group, the probability that core damage is not arrested is driven by operator errors. In these PDSs low pressure injection systems are available; however, the operator fails to depressurize the RPV.

It must be remembered that core damage arrest does not necessarily mean that there will be no radionuclide releases during the accident. 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. This radionuclide release is generally small, however, because in the majority of the cases in which vessel breach is averted these releases are scrubbed as they pass through the suppression pool. Furthermore, if the vessel does not fail, there are no ex-vessel releases (e.g., CCI releases).

Early Containment Failure. The early fatality risk depends strongly on the probability of early CF. Early CF includes both failures that occur before vessel breach and during the time period around vessel breach. The Grand Gulf containment is a fairly weak structure when considering the loads that can potentially occur during the course of the accident. The design pressure is only 15 psig and the assessed mean failure pressure is 55 psig. Because of its low failure pressure, the Grand Gulf containment is not only susceptible 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 vessel breach, should it occur, is a key factor that affects the probability of containment failure. In a BWR core there is a large inventory of zirconium. Large amounts of hydrogen are produced from the oxidation of this metal during the core damage process. If the HIS is not operating, which is the case in a SBO, the hydrogen will accumulate in the containment. For accidents in which the suppression pool is subcooled, the steam released from the RPV is condensed in the pool. The lack of steam in the containment atmosphere in combination with the large amount of hydrogen released during the core degradation process allows mixtures to form that have a high hydrogen concentration. Subsequent ignition of this hydrogen by either random sources or by the recovery of ac power can result in loads that not only can threaten the containment but also can pose a significant challenge to the drywell structure.

Figure S.4 shows the probability distribution for early CF at Grand Gulf. The probability distributions displayed in this figure are for accidents that proceed to vessel breach and are conditional on core damage.

The weakness of the containment, relative to the loads that are imposed on it, is reflected in the relatively high containment failure probabilities. Hydrogen combustion events are the dominant events that cause early CF in the short-term SBO and T2 PDS groups. The mean probability of early containment failure for these two groups is roughly 0.5. The majority of these failures are caused by hydrogen deflagrations rather than by detonations. In both of these summary PDS groups the suppression pool is subcooled before vessel breach and, therefore, there is no significant accumulation of steam in the containment. This virtually eliminates the possibility of early CF from slow pressurization events (e.g., accumulation of steam). Because the HIS is not available during a short-term SBO, severe hydrogen combustion events before vessel breach are possible. In the short-term SBO PDS group, about half of the mean probability is associated with CFs that occur before vessel breach and the other half with failures that occur shortly after vessel breach. In the T2 PDS group, on the other hand, almost all of the early CFs occur at the time of vessel breach. For accidents in the T2 group, it is likely that the operator turned on the HIS before core damage and, therefore, the hydrogen generated before vessel breach is usually burned such that the resulting load is benign. The rapid combustion of hydrogen generated at vessel breach, however, can still lead to early CF.

For the long-term SBO PDS group, the mean conditional probability of early CF is 0.85. Less than half of these early CFs are caused by hydrogen combustion events. In this summary PDS group the suppression pool is saturated and the containment is pressurized by the accumulation of steam that is generated by the hot pool. In most of these accidents hydrogen burns before vessel breach are not possible because the containment is steam-inert. Approximately two thirds of this mean probability results from early CFs that occur before vessel breach and the preponderance of these CFs are caused by pressurization events associated with the accumulation of steam in the containment. There are a few cases, however, in which the containment sprays are recovered before vessel breach and a combustible mixture is formed by the condensation of the steam. Subsequent ignition of this mixture can result in containment failure. The remaining third of the mean probability results from early CFs that occur at vessel breach and the vast majority of these failures are caused by hydrogen combustion events.

For the ATWS PDS group, the mean conditional probability of early CF is 0.76. Similar to the long-term SBO group, less than half of the early CF probability associated with the ATWS group is caused by hydrogen combustion events. This PDS group consists of both a long-term PDS and a short-term PDS. In the long-term PDS the suppression pool is saturated and either the operators vent the containment or the containment fails before vessel breach from the accumulation of steam in the containment. This PDS is responsible for a little more than half of this group's mean frequency. In the short-term PDS, on the the other hand, almost all of the early CF

probability is associated with failures that occur at the time of vessel breach. The pool is subcooled in the short-term PDS. Although combustible mixtures can form in the containment before vessel breach in this PDS, the HIS is typically on during core damage and, therefore, the hydrogen generated before vessel breach is usually burned such that the resulting load is benign.

Early Drywell Failure. Early drywell failure is an important attribute of the accident progression because failure of the drywell establishes a pathway for radionuclides in the drywell to bypass the suppression pool. Because accidents that result in early drywell failure coincident with early containment failure are generally the dominant risk contributors, it is appropriate to discuss the events that can lead to early drywell failure.

Before vessel breach the only significant event that causes drywell failure is hydrogen combustion. Slow pressurization events associated with the accumulation of steam in the containment are not a threat to the drywell structure. For the short-term SBO PDS group, most of the failures are caused by deflagrations. A relatively small fraction of these failures is caused by detonations. The mean probability of drywell failure before vessel breach is considerably less for the other PDS groups. There are several reasons for the lower failure probability in these groups. In the long-term SBO PDS group the containment is frequently steam-inert during this stage of the accident. In the ATWS PDS group, the containment is steam inert in some of the cases and in many of the other cases the HIS is operating during core damage. In the T2 PDS group, the HIS is also generally operating during the core damage process.

For drywell failures that occur at vessel breach, loads accompanying vessel breach are responsible for the majority of these failures. These quasi-static loads, which were provided by the Containment Loads Expert Panel, include contributions from: DCH, ex-vessel steam explosions, hydrogen burns, and RPV blow down. At vessel breach these events pressurize the drywell volume before the suppression pool vents clear. Nearly half of the drywell failures that occur at vessel breach are caused by these loads. In addition to directly pressurizing the drywell volume, these loads can also pressurize the reactor cavity and fail the pedestal. The loss of reactor support can induce drywell failure. Roughly a quarter of the drywell failures that occur at vessel breach can be attributed to failure of the reactor pedestal.

## S.6 Source Term Analysis

### S.6.1 Description of the Source Term Analysis

The source term for a given bin consists of the release fractions for the nine radionuclide classes for the early release and for the late release, and additional information about the timing of the releases, the energy associated with the releases, and the height of the releases. This source term comprises the information required for the calculation of 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: GGSOR. 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, GGSOR 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 be able to compute source terms for the numerous accident progression bins and the 250 observations that result from the sampling approach used in NUREG-1150.

GGSOR is a fast-running, parametric computer code used to calculate the source terms for each APB for each observation for Grand Gulf. As there are typically about three hundred bins for each observation, and 250 observations in the sample, the need for a source term calculation method that requires few computer resources for one evaluation is obvious. GGSOR 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 are used to 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 GGSOR 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 75,000 source terms computed for the Grand Gulf 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 accident progression bin is assigned to a source term group. In the risk computations, each accident progression bin 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 Grand Gulf are considered together, the plots shown in Figure S.5 are obtained. These plots show four statistical measures of the 250 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 and cesium is on the order of  $10^{-6}$ /year and for tellurium and strontium it is on the order of  $10^{-7}$ /year. The mean frequency of exceeding a release fraction of 0.01 for the La radionuclide class is on the order of  $10^{-8}$ /year.

## S.7 Consequence Analysis

### S.7.1 Description of the Consequence Analysis

Offsite consequences are calculated with the MELCOR Accident Consequence Code System (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.



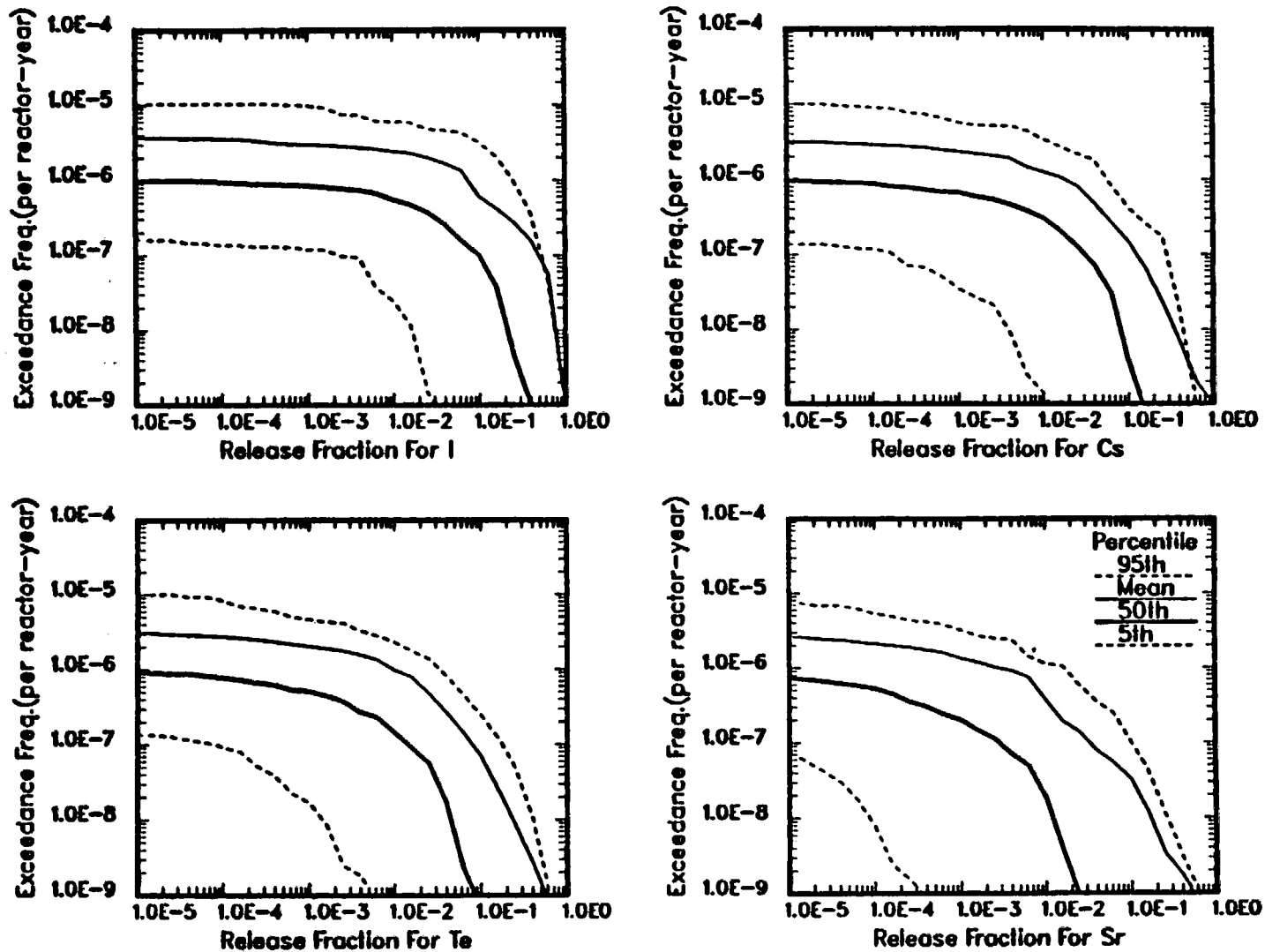


Figure S.5. Exceedance Frequencies for Release Fraction for Grand Gulf: All Internal Initiators.

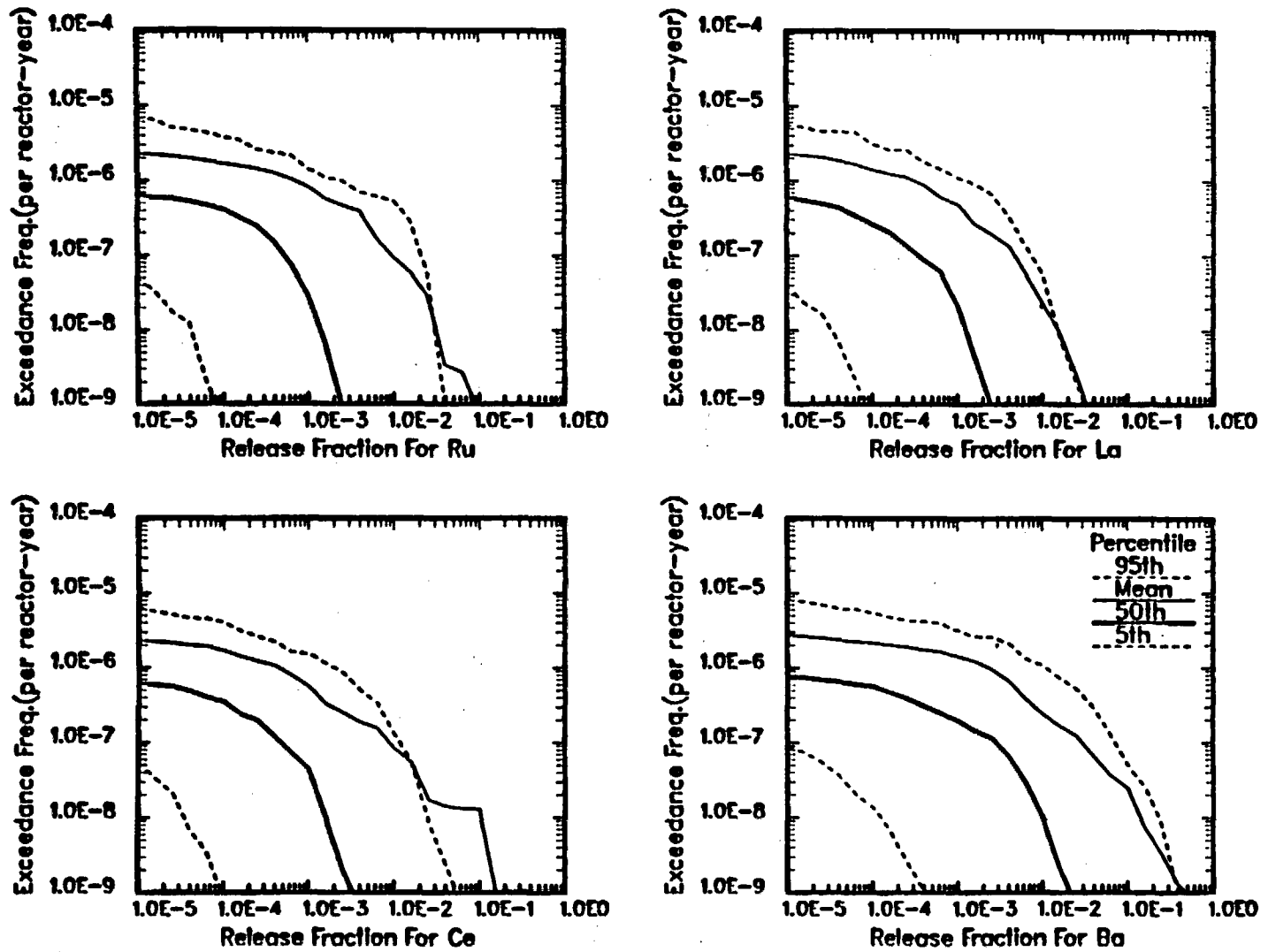


Figure S.5. (continued)

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 out of the emergency planning zone. Long-term actions include relocation and restrictions on land use and crops. Relocation and land decontamination, 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 ground 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 one 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 Grand Gulf, the evacuation delay time between warning and the beginning of evacuation is 1.25 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 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 1.25-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

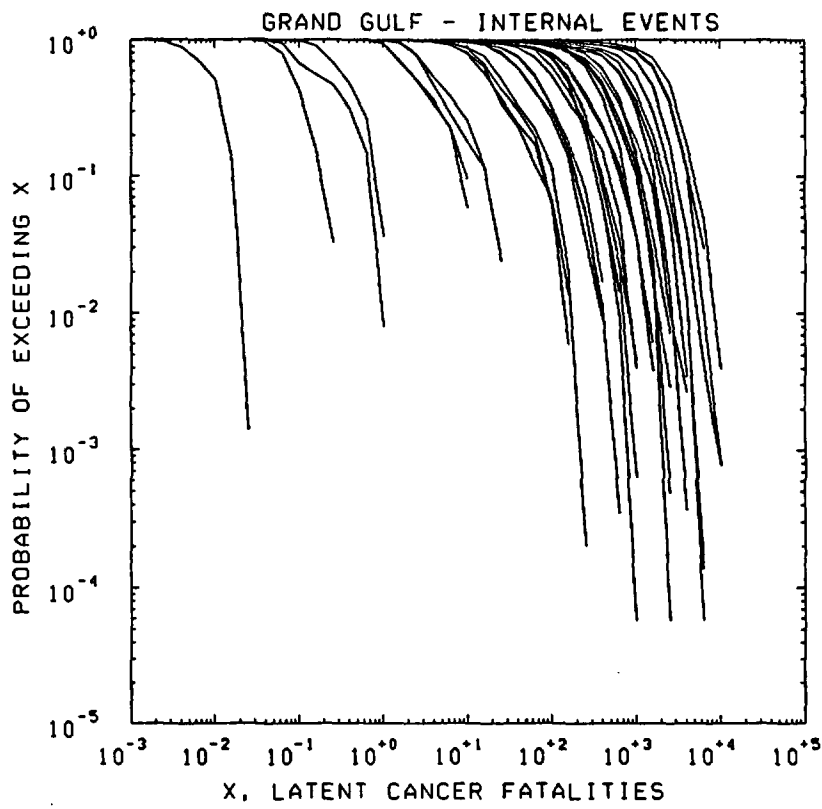
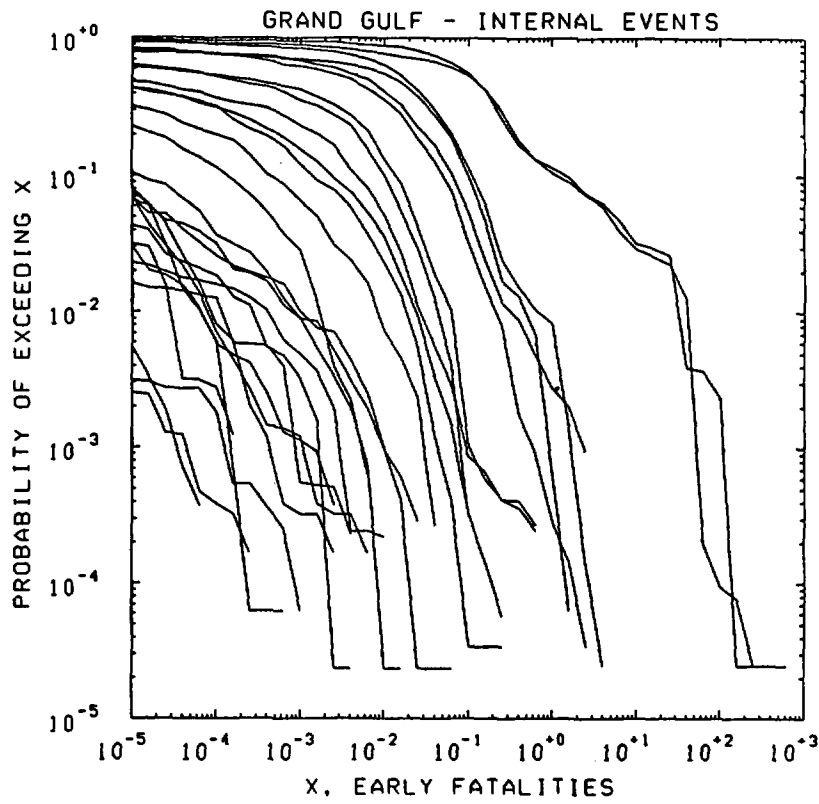


Figure S.6. Consequences Conditional on Source Terms.  
Grand Gulf: Internal Initiators.

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 existing 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 between source term groups that is due to the size and timing of the release as well.

## S.8 Integrated Risk Analysis

### S.8.1 Determination of Risk

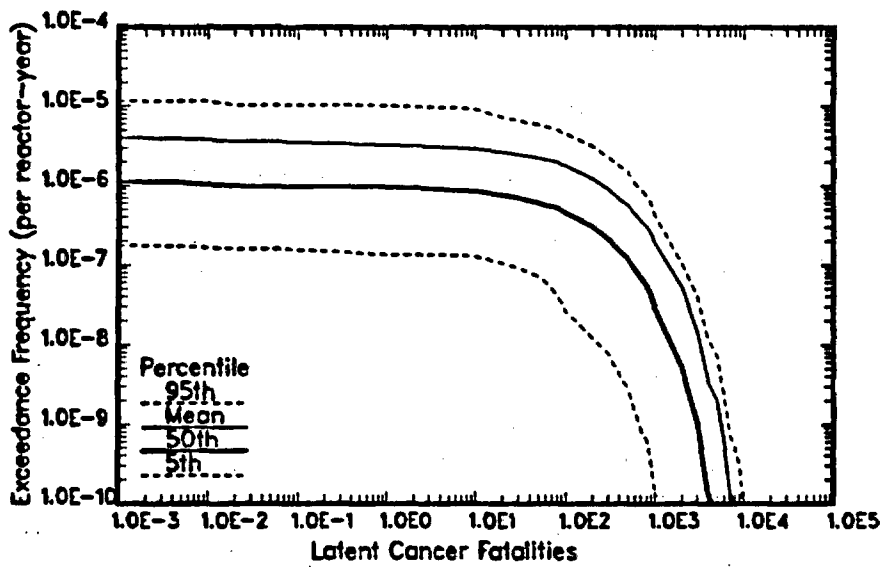
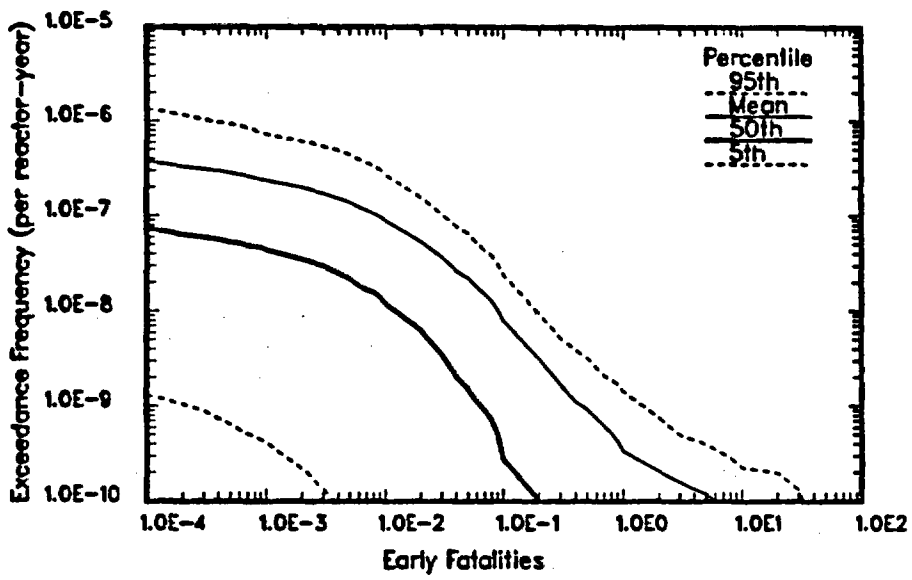
Risk is determined by bringing together the results of the four constituent analyses: the accident frequency analysis, the accident progression analysis, the source term analysis, and the consequence analysis. This process is described in general terms in Section S.2 of this summary, and in mathematical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

A source term is calculated for each APB for each observation; 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

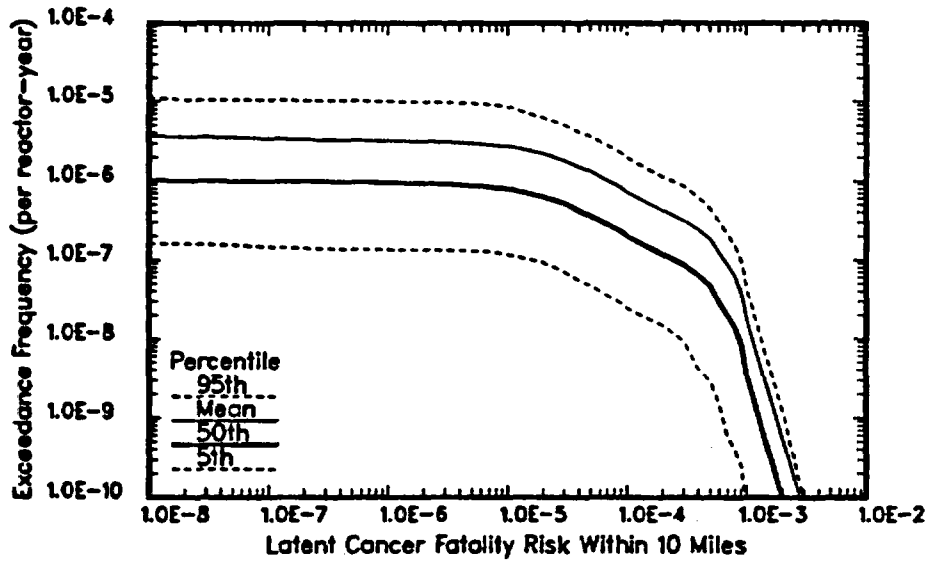
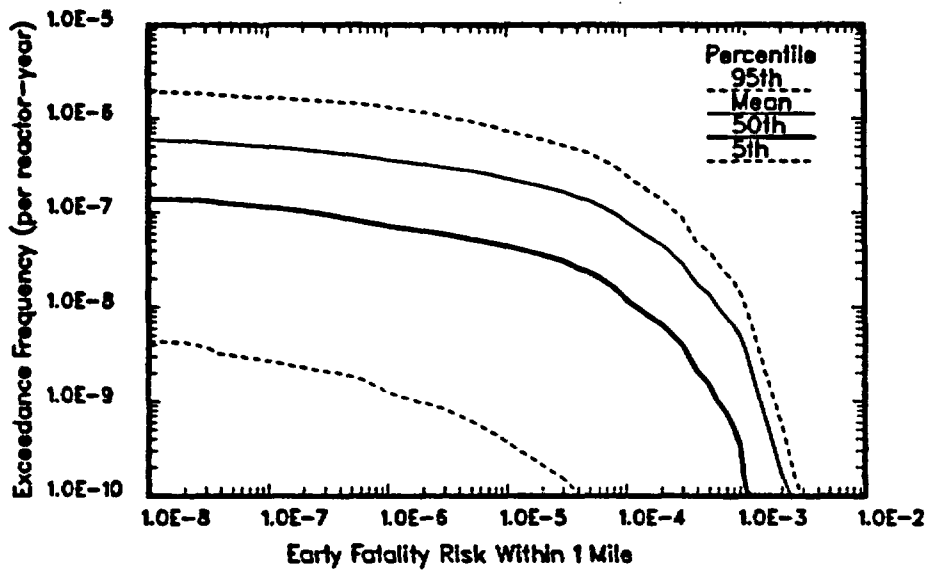
Measures of Risk. Figure S.7 shows the basic results of the integrated risk analysis for internal initiators at Grand Gulf. This figure shows four statistical measures of the families of complementary cumulative distribution functions (CCDFs) for early fatalities, latent cancer fatalities, individual risk of early fatality within one mile of the site boundary, and individual risk of latent cancer fatality within 10 miles of the plant. The CCDFs display the relationship between the frequency of the consequence and the magnitude of the consequence. As there are 250 observations in the sample for Grand Gulf, the actual risk results at the most basic level are 250 CCDFs for each consequence measure. Figure S.7 displays the 5th percentile, median, mean, and 95th percentile for these 250 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



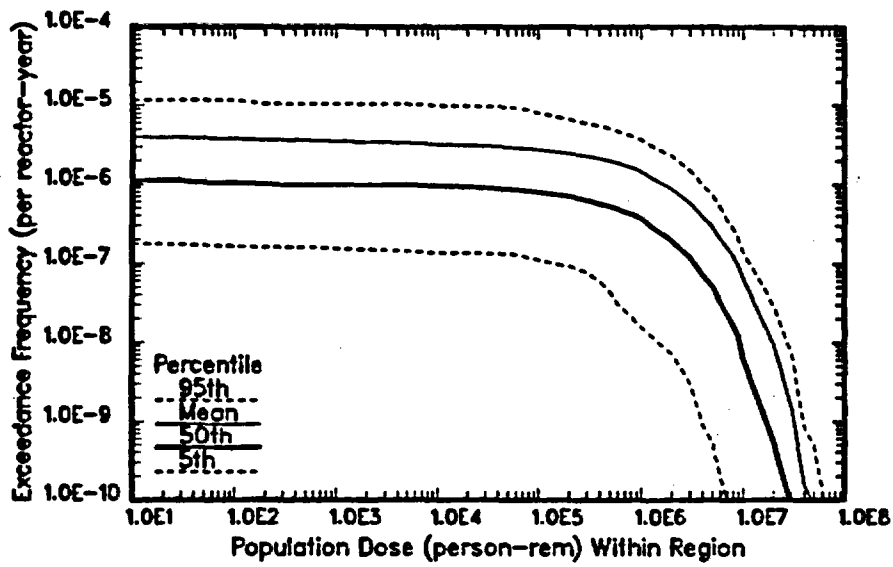
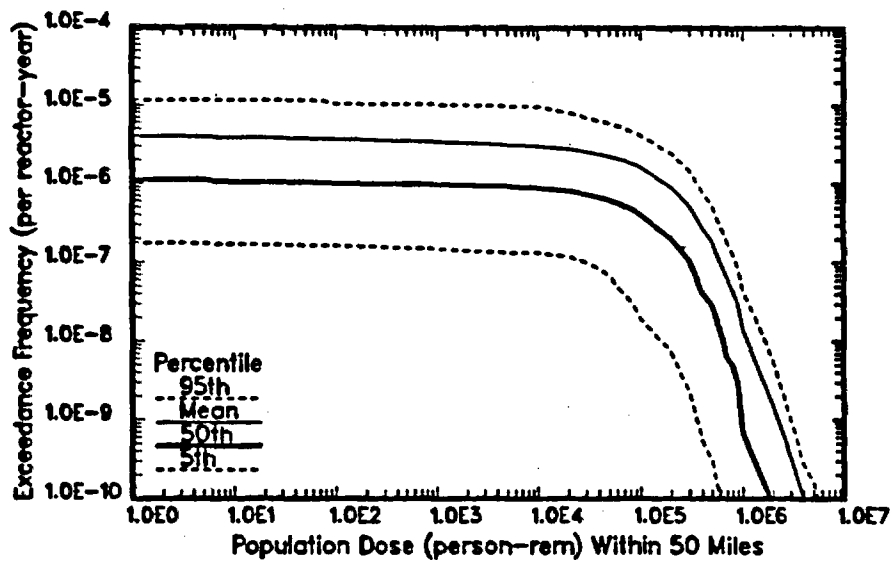
GRAND GULF BaseCase

Figure S.7. Exceedance Frequencies for Risk.  
 Grand Gulf: All Internal Initiators.



GRAND GULF BaseCase

Figure S.7. (continued)



GRAND GULF BaseCase

Figure S.7. (continued)



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.

Although the CCDFs convey the most information about the offsite risk, summary measures are also useful. Such a summary value, denoted 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 250 observations, there are 250 values of annual risk for each consequence measure.

These 250 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 Grand Gulf falls below the safety goals and the means are also 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. 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.

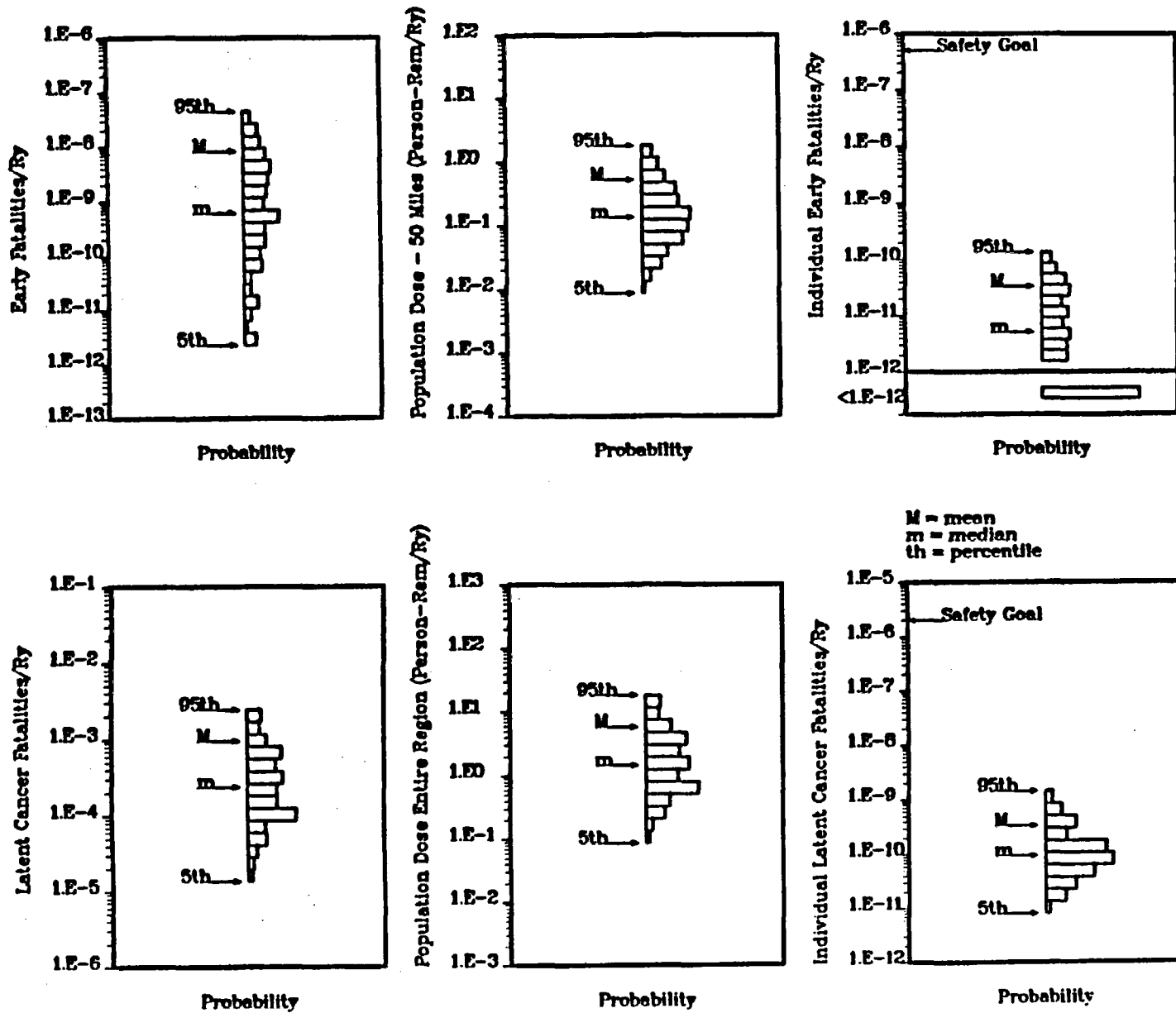


Figure S.8. Distributions of Annual Risk. Grand Gulf: All Internal Initiators.

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 four summary PDS groups.

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

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

<u>PDS Group</u>	<u>FCMR</u>	<u>MFCR</u>
Fast SBO	93.2	84.1
Slow SBO	4.7	6.5
ATWS	2.0	7.9
T2 Trans.	0.2	1.5

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

<u>PDS Group</u>	<u>FCMR</u>	<u>MFCR</u>
Fast SBO	91.3	85.3
Slow SBO	4.8	5.0
ATWS	3.5	8.2
T2 Trans.	0.4	1.5

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Pie charts for the contributions of the summary 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 the contributions of the summary APBs to mean risk. Not surprisingly, the two methods of calculating contribution to risk yield different values. Because both methods of computing the contributions to risk are conceptually valid, 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 the short-term SBO groups is the major contributor to mean early fatality risk at Grand Gulf. It is not valid to state that the short-term SBO group contributes 93.2% of the early fatality risk at Grand Gulf.

Although the exact values are different for each method, the basic conclusions that can be drawn from these results are the same. For all of the consequence measures, the mean risk is dominated by the short-term SBO PDS group. This group is the dominant contributor to the core damage frequency and because ac power is not initially available in these PDSs, there is a significant probability that these accidents will involve early

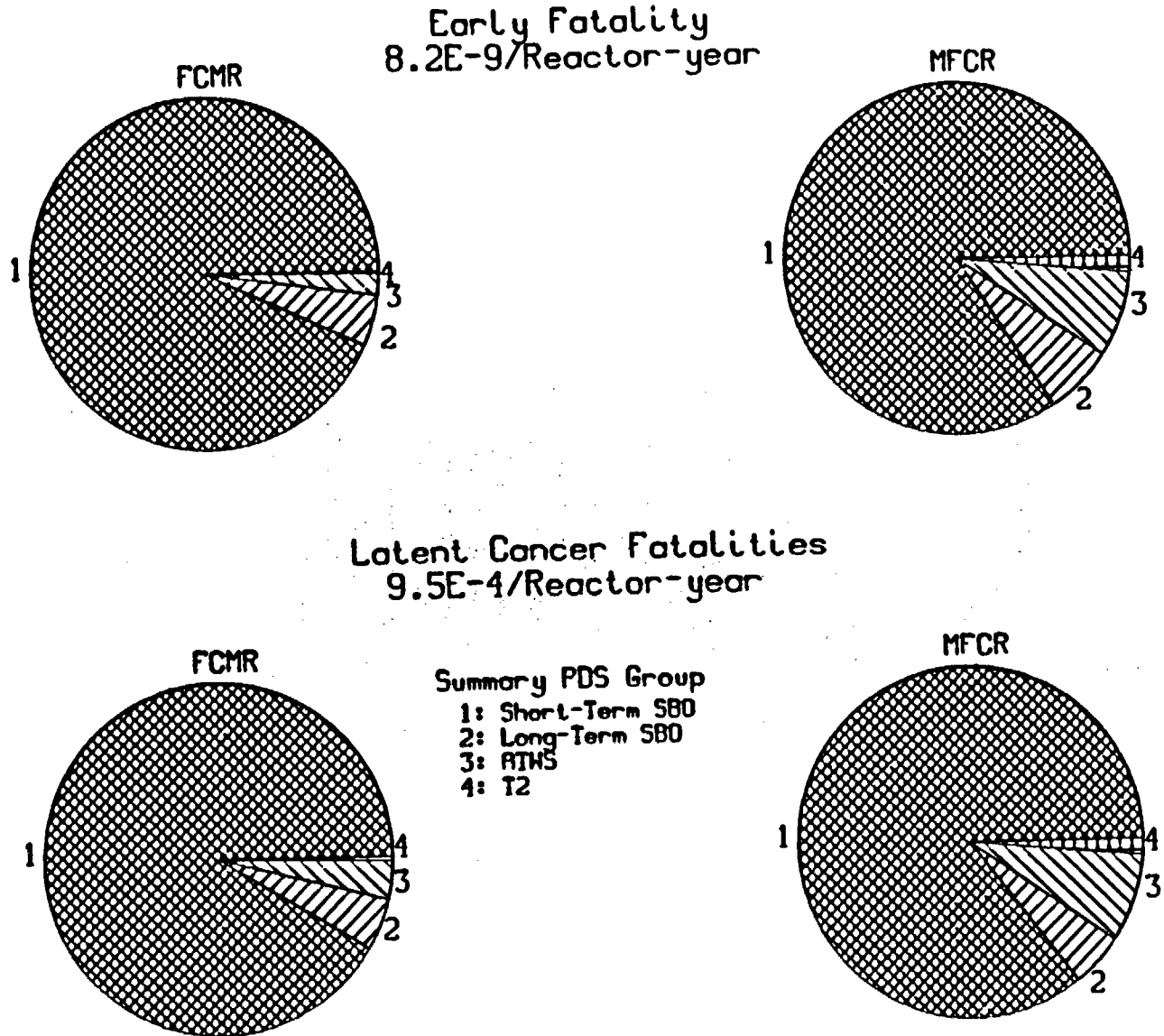


Figure S.9. Fractional PDS Contributions to Annual Risk. Grand Gulf: All Internal Initiators.  
(MFCR = Mean Fractional Contribution to Risk; FCMR = Fractional Contribution to Mean Risk).

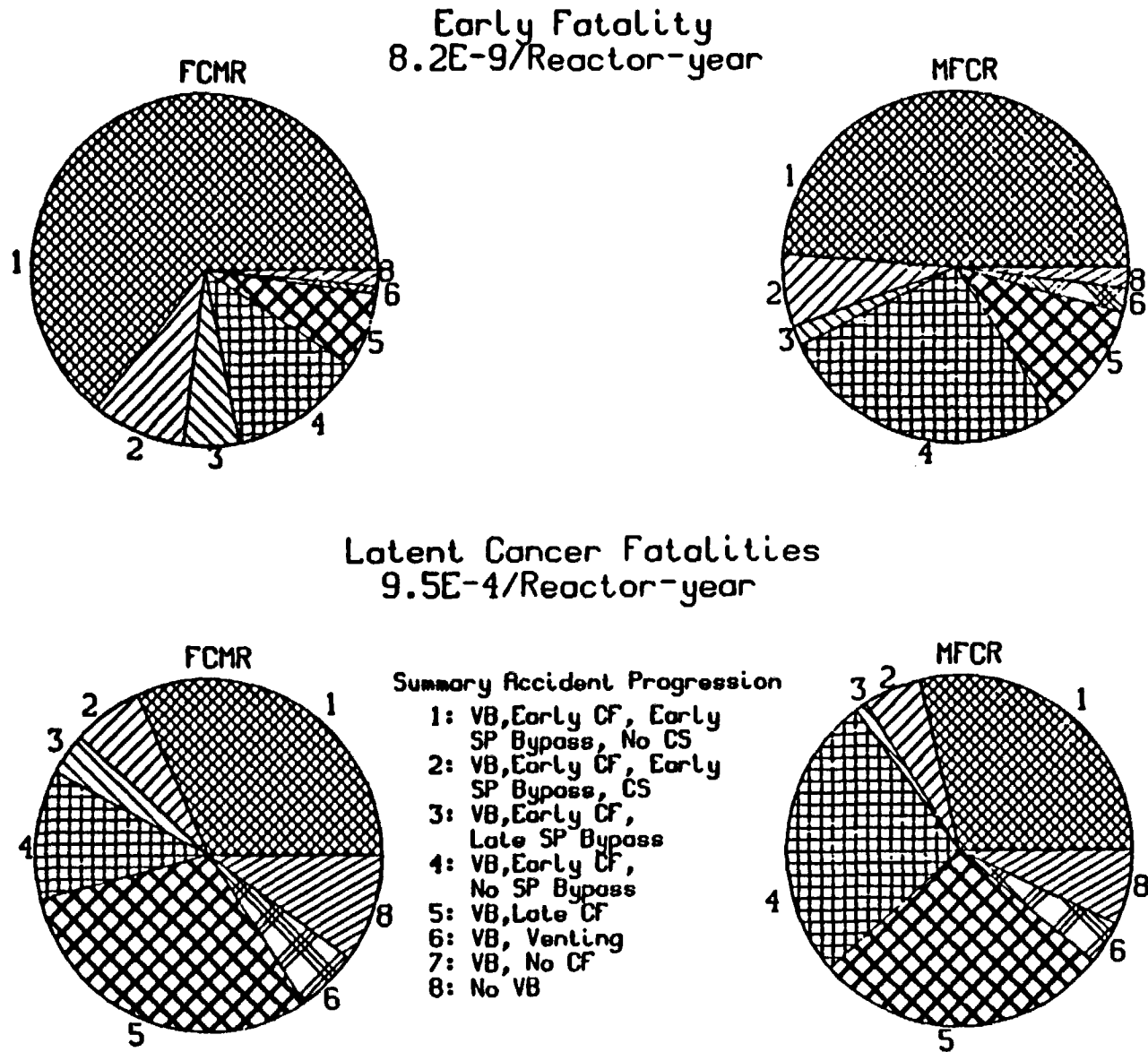


Figure S.10. Fractional ABP Contributions to Annual Risk. Grand Gulf: All Internal Initiators. (MFCR = Mean Fractional Contribution to Risk; FCMR = Fractional Contribution to Mean Risk).

containment failure and vessel breach. Thus, these accidents are not only the most frequent but they also involve accidents that can potentially result in a large early release. The long-term SBO group and the ATWS group contribute considerably less to these risk measures and the T2 group is a very minor contributor.

For early fatalities, which depend on a large early release, the risk is dominated by accidents that progress to vessel breach and that involve early containment failures. Accidents in which the containment fails late are much less significant. In Figure S.10 the first bin (vessel breach, Early CF, Early SP Bypass, No CS) is the dominant contributor to these risk measures because the containment fails early and the releases at vessel breach and after vessel breach are not scrubbed by either the pool or the containment sprays. Although the fourth bin in Figure S.10 (vessel breach, Early CF, No SP Bypass) does not involve drywell failure, its contribution to early fatality risk is higher than the second bin (vessel breach, Early CF, Early SP Bypass, CS Avail.) in which the drywell fails early in the accident. The reason for this is that the mean probability of the fourth bin is roughly four times the mean probability of the second bin. Thus, although the fourth bin does not involve drywell failure, the probability of this bin coupled with the fact that the containment fails early is sufficient to make this bin a significant contributor to early fatality risk.

Latent cancer fatalities 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. If the suppression pool is bypassed there is a greater likelihood that the release will be large. Thus, accidents in which some of the releases are not scrubbed by either the pool or the sprays tend to contribute more to latent cancer fatality risk than accidents in which the drywell remains intact. It is for this reason that the first bin in Figure S.10 (vessel breach, Early CF, Early SP Bypass, No CS) is the dominant contributor to the latent cancer fatality risk.

The bin that involves accidents in which the vessel does not fail makes a minor contribution to the early fatality risk; however, it makes a noticeable contribution to the latent cancer fatality risk. It must be remembered that although the vessel does not fail in these accidents, the containment can still fail early in these accidents from the combustion of hydrogen in the wetwell. Early failure of the containment will allow a portion of the in-vessel releases to escape into the environment. The combination of the threshold effect associated with early fatalities with the fact that the releases associated with this bin are fairly small results in few early fatalities. For latent cancers, on the other hand, there is no threshold effect. Thus any releases that are not trapped by the suppression pool or removed by the containment sprays can contribute to the latent cancer risk.

#### S.8.4 Important Contributors to the Uncertainty in Risk

The important contributors to the uncertainty in internally initiated risk are determined by performing regression-based sensitivity analyses for the mean values for risk. The regression analyses for early fatalities and individual risk of early fatality within 1 mile only account for about 45% 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 analyses for the other four consequence measures are somewhat more successful as they are able to account for about 60% of the variability. The independent variables that account for this variability are predominantly those variables that determine the frequencies of the accident.

#### S.9 Insights and Conclusions

Core Damage Arrest. For the dominant summary PDS group, short-term SBO, there is a significant probability that the core damage process will be arrested and vessel failure will be averted. For the accidents in which the vessel does not fail, there are no ex-vessel fission product releases (e.g., DCH or CCI). Furthermore, loads accompanying vessel breach, which pose a significant challenge to both the drywell and the containment, are avoided. The conditional probability of core damage arrest in the short-term SBO PDS group is driven by the ac power recovery probability. In the other summary PDS groups (i.e., long-term SBO, ATWS, and T2) it is unlikely that core damage process will be arrested. The core damage arrest probability for the long-term SBO group is low because the probability of recovering ac power early in the accident is fairly low for this PDS group. In the ATWS and T2 PDS groups the low values for core damage arrest are attributed to fairly high likelihood that the operators fail to depressurize the RPV to allow coolant injection to be restored to the core.

Containment Failure. Given that core damage occurs, it is likely that the containment will fail during the course of the accident. Furthermore, for the dominant PDS summary group, short-term SBO, there is a substantial probability that the containment will fail early in the accident. Hydrogen combustion events are the dominant events that cause early CF in the short-term SBO and T2 PDS groups. The combination of a relatively weak containment, the copious production of hydrogen during core damage, and the unavailability of the HIS during a SBO leads to a high conditional probability of containment failure. For these two groups, the mean probability of early containment failure is approximately 0.5. In the short-term SBO group about half of the early CFs occur before vessel breach and the other half occur shortly after vessel breach. In the T2 PDS group the vast majority of the early containment failures occur around the time of vessel breach. For both the long-term SBO PDS group and the ATWS PDS group, hydrogen combustion events and pressurization of the containment from the accumulation of steam contribute to their high conditional probabilities of early containment failure.

Drywell Failure. Early drywell failure is an important attribute of the accident progression because failure of the drywell establishes a pathway

for radionuclides in the drywell to bypass the suppression pool. The suppression pool offers an important mechanism for reducing the source term. Accidents that result in early drywell failure coincident with early containment failure are generally the dominant contributors to risk. Of the accidents that result in early containment failure, roughly half of them also involve early drywell failure. Early drywell failures include failures that occur before vessel breach and failures that occur at vessel breach. Only the short-term SBO PDS group has significant probability of drywell failure before vessel breach. The vast majority of these drywell failures are caused by hydrogen combustion events. All of the PDS groups have a significant probability of drywell failure at the time of vessel breach. The majority of these failures are caused by loads accompanying vessel breach. These quasi-static loads include contributions from DCH, ex-vessel steam explosions, hydrogen burns and RPV blow down.

Fission Product Releases. There is considerable uncertainty in the release fractions for all types of accidents. There are several features of the Grand Gulf plant that tend to mitigate the release. First, the in-vessel releases are generally directed to the suppression pool where they are subjected to the pool decontamination factor. Provided the drywell has not failed, the radionuclides released into the drywell will also pass through the pool. Although generally not as effective as the suppression pool, the containment sprays and the reactor cavity pool also offer a mechanisms for reducing the release of radionuclides from the containment when the suppression pool has been bypassed. The largest releases tend to occur when the suppression pool is bypassed and the containment sprays are not operating.

Risk. The offsite risk from internal initiating events was found to be quite low, both with respect to the safety goals and to the other plants analyzed in NUREG-1150. The offsite risk is dominated by short-term SBO PDSs. The long-term SBO group and the ATWS group contribute considerably less to these risk measures and the T2 group is a very minor contributor. The low values for risk can be attributed to the low core damage frequency, the good emergency response, and plant features that reduce the potential source term.

Uncertainty in Risk. Considerable uncertainty is associated with the risk estimates produced in this analysis. The largest contributors to this uncertainty are the uncertainties in the parameters that determine the frequency of core damage and the uncertainty in some of the parameters that determine the magnitude of the fission product release to the environment. Propagation of the uncertainties in the accident frequency, accident progression, and source term analyses through to risk allows the uncertainty to be quantitatively calculated and displayed.

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 95th percentile value for annual risk falls nearly three orders of magnitude below the safety goals. Furthermore, for both of these risk measures, the maximum of the 250 values that make up the annual risk distributions also falls well below the safety goal.



## References

1. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants", Second Draft for Peer Review, NUREG-1150, June 1989.
2. M. T. Drouin, J. L. LaChance, B. J. Shapiro, S. Miller, and T. A. Wheeler, "Analysis of Core Damage Frequency: Grand Gulf, Unit 1 Internal Events," NUREG/CR-4550, SAND86-2084, Vol. 6, Rev. 1, Sandia National Laboratories, September 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<sup>1</sup> 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 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 in terms of health effects and financial impact.
5. Risk integration: the combination of the outputs of the previous tasks into an overall expression of risk.
6. Uncertainty analysis: the determination of which uncertainties in the preceding analyses 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 analyses reported here start with the onset of core damage and conclude with an integrated estimate of overall risk and uncertainty in risk. This volume, Volume 6, describes these analyses, the inputs utilized in them, and the results obtained, for Grand Gulf Nuclear Station, Unit 1. The methods utilized 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).<sup>2</sup> 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"<sup>1</sup> 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 analysis of Zion was performed by Idaho National Engineering Laboratory (INEL) and Brookhaven National Laboratory (BNL).

The following are overall objectives of the NUREG-1150 program.

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";<sup>2</sup>
  - 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 Grand Gulf Nuclear Station, Unit 1

The subject of the analyses reported in this volume is the Grand Gulf Nuclear Station, Unit 1. It is operated by System Energy Resources Inc. (SERI) and is located on the east bank of the Mississippi river in southwestern Mississippi, about 6 miles northwest of Port Gibson, Mississippi. The nearest large city is Jackson, Mississippi, approximately 55 miles to the northeast of the plant.

The nuclear reactor of Grand Gulf Unit 1 is a 3833 MWt BWR-6 boiling water reactor (BWR) designed and supplied by General Electric Company. Unit 1, constructed by Bechtel Corporation, began commercial operation in July 1985.

Grand Gulf has three diesel generators (DGs) that are used to supply emergency ac power in the event that offsite power from the grid is lost. One of these DGs is dedicated to the high pressure core spray injection system (HPCS); the other two DGs supply ac power to two trains of emergency systems. In the event of an accident there are several systems that can supply coolant injection to the core. Two systems are available to provide high pressure coolant injection: the high pressure core spray system (HPCS) and the reactor core isolation cooling system (RCIC). HPCS has a motor-driven pump and can supply injection when the vessel pressure is either high or low. RCIC, on the other hand, uses a turbine-driven pump and can only be used when the vessel pressure is high. Both the low pressure core spray system (LPCS) and the low pressure coolant injection system (LPCI) can provide coolant injection to the reactor vessel during accidents in which the system pressure is low. Both systems use motor-driven pumps. LPCS has one train whereas LPCI consists of three trains. Additional systems that can be used as backup sources of coolant injection are the standby service water crosstie system, firewater system, control rod drive system, and the condensate system. To allow low pressure injection systems to supply coolant to the vessel, the automatic depressurization system (ADS) is used to depressurize the reactor vessel. This system uses eight relief valves to direct the vessel steam to the suppression pool.

The Grand Gulf containment is a Mark III BWR containment. The containment is a steel-lined reinforced concrete structure. In the Mark III design the reactor pressure vessel (RPV) is housed in the drywell which is in turn completely enclosed in the containment structure. The drywell and the containment communicate through passive vents in the suppression pool. Figure 1.1 shows a section through the Grand Gulf containment. During an accident, steam from the vessel is directed through the safety/relief valves and is discharged through a sparger into the suppression pool. The steam is condensed in the pool and any noncondensable gases pass through the pool into the containment atmosphere. Similarly, any steam and noncondensable gases released into the drywell are vented into the suppression pool. The design pressure of the Grand Gulf containment is

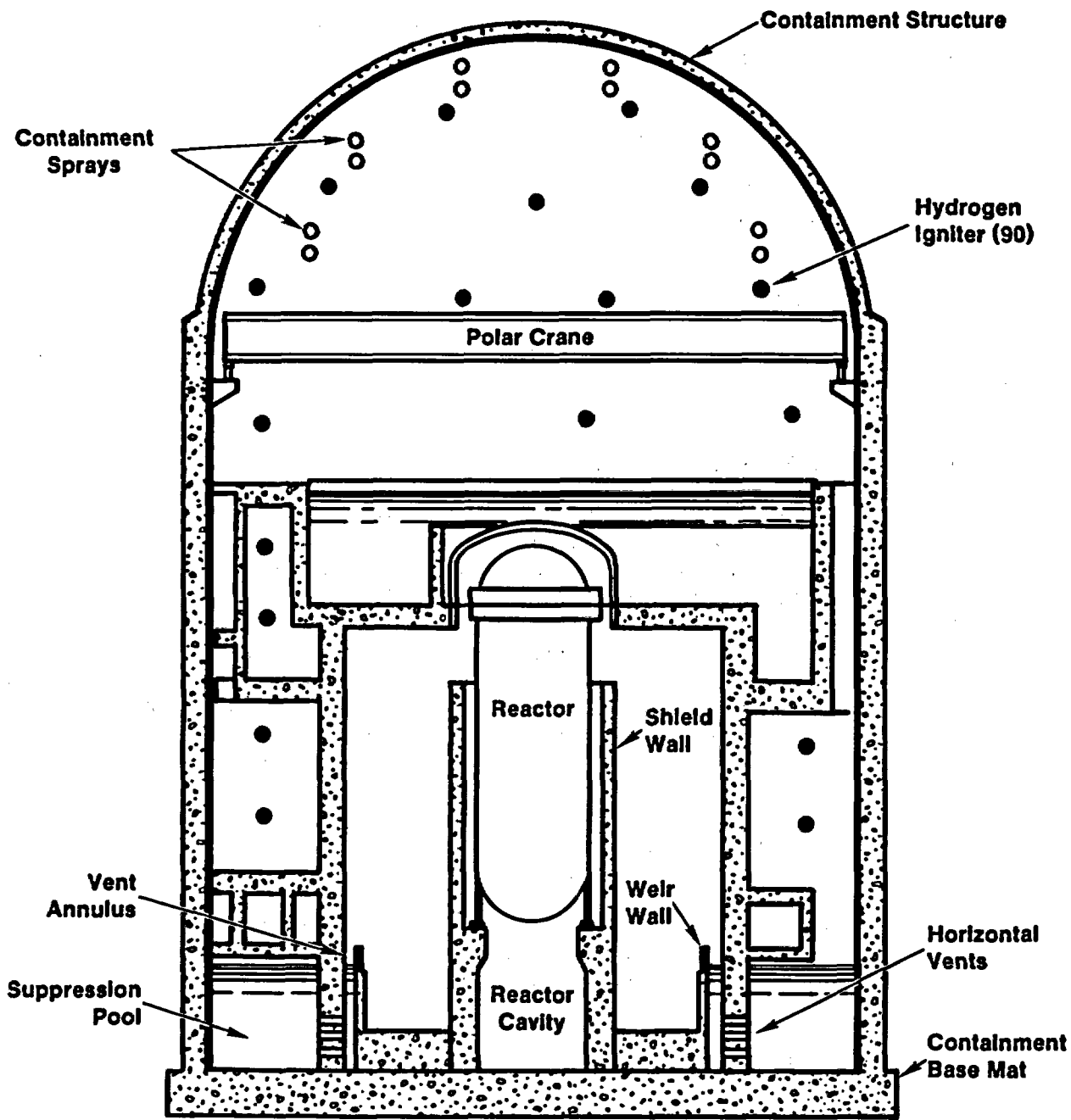


Figure 1.1. Section of Grand Gulf Containment.

15 psig (103 kPa). Although the design pressure is fairly low, the volume of the containment is comparable with a large PWR containment (1.67 million cubic feet).

To suppress the pressure in the containment during an accident, two trains of containment sprays are located in the Grand Gulf containment. The containment spray system is one mode of the residual heat removal system (RHR). In the event that the RHR system fails to suppress the pressure in the containment, the containment can be vented.

To reduce the potential of a severe hydrogen combustion event during an accident, the containment has a hydrogen ignition system (HIS). This system is designed to prevent the buildup of large quantities of hydrogen inside the containment. Igniters are located throughout the containment and drywell volumes.

Section 2.1 of this volume contains more detail on the plant's features important to the progression of the accident and to the containment's performance.

### 1.3 Changes Since the Draft Report

The Grand Gulf analyses for the February 1987 draft of NUREG-1150 were presented in Volume 4 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 completely 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 completely 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 thirty. Separate panels of experts were convened for In-Vessel Processes, Containment Loads, Containment Structural Response, Molten Core-Containment Interactions, 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 Grand Gulf 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 Grand Gulf APET, the user function is called to: determine the containment baseline pressure during the various time periods; compute the amount of hydrogen released to the containment at the time of vessel breach and during CCI; compute the concentration and the flammability of the atmosphere in the containment and drywell during the various time periods; calculate the pressure rise due to hydrogen burns; determine whether the containment fails and the mode of failure; determine whether the drywell 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 ignition probability, detonation probability 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. In this analysis, drywell failure from deflagrations is also considered. In addition to combustion events, another major change in the APET is the section that addresses vessel breach. In-vessel steam explosions are now addressed in the tree. Furthermore, the tree was modified to incorporate new information supplied by the Containment Loads Expert Panel on loads accompanying vessel breach. Pressurization of the drywell and pressurization the reactor cavity from events at vessel breach are considered. Failure of the reactor pedestal at vessel breach was not included in the previous analysis.

Because of changes in the accident progression analysis and the source term analysis, the definitions of bins used to group the results from the accident progression analysis have also changed.

Source Term Analysis. While the basic parametric approach used in the original version of GGSOR, the code used to compute source terms, has been retained in the present version of GGSOR, the code has been completely rewritten with a different orientation.

The current version of GGSOR is quite different. First, it is not tied to the source term code package (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 utilized the results of one or more codes in deriving their distributions, but GGSOR itself merely combines the parameters defined by the expert panel.

Finally, a new method to group the source terms computed by GGSOR has been devised. A source term is calculated for each accident progression bin 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 utilized 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 version 1 does not differ so markedly from that for the previous version of NUREG-1150 as does 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",<sup>3</sup> whereas version 1.5 of MACCS uses the lung data from Revision 1 (1989) of this report.<sup>4</sup> 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 analysis of the Grand Gulf plant for NUREG-1150 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 in itself. By taking care to carefully define 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 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. In order to capture the uncertainties, the first three of the four constituent analyses attempt to utilize 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<sup>5,6</sup> 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.

1.9

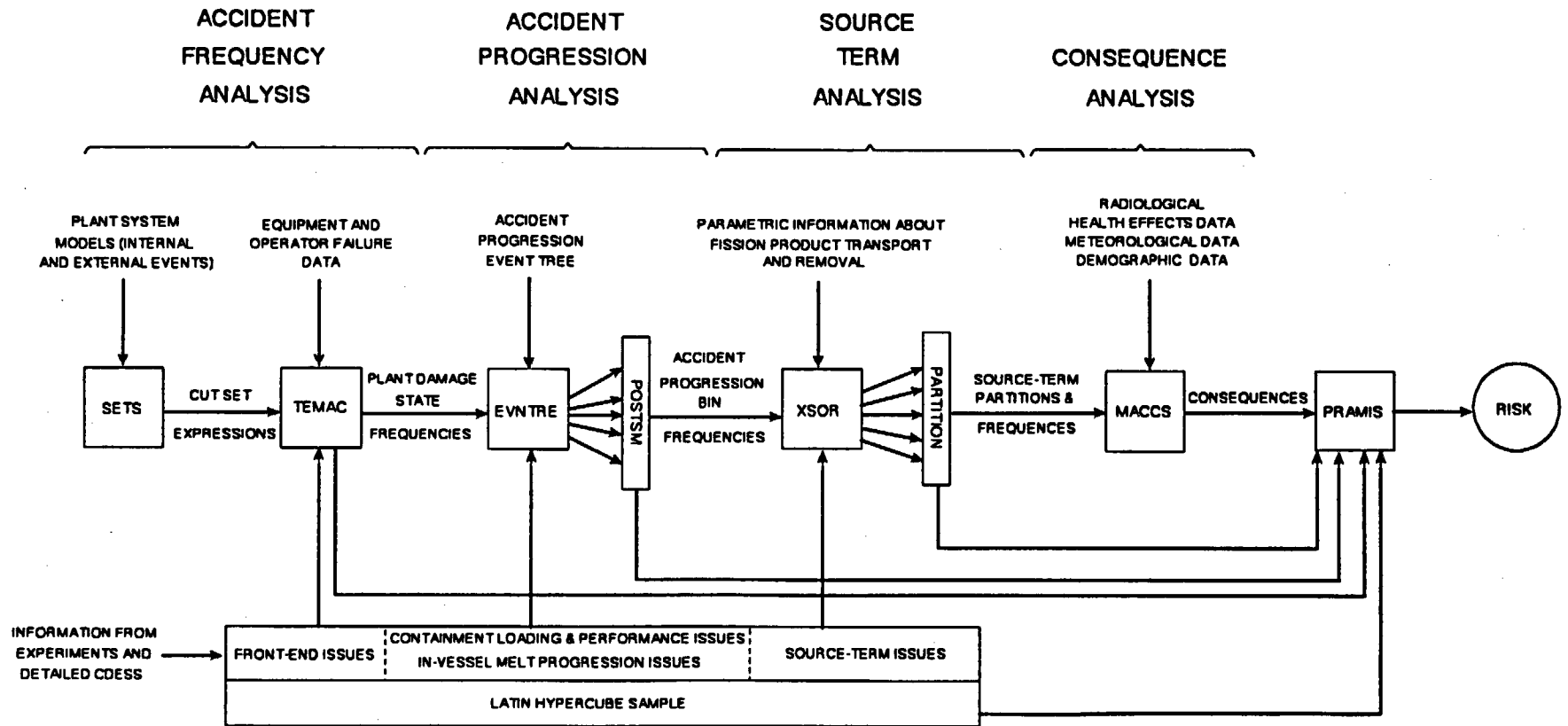


Figure 1.2. Overview of Integrated Plant Analysis in NUREG-1150.

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<sup>7</sup> is used to combine 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.<sup>8,9,10,11,12</sup> 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<sup>13,14</sup> 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

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

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

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

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

and

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

The elements  $pPDS_{ij}$  of  $P(IE \rightarrow PDS)$  are conditional probabilities: given that initiating event  $i$  has occurred,  $pPDS_{ij}$  is the probability that plant damage state  $j$  will also occur. The elements of  $P(IE \rightarrow PDS)$  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  $P(IE \rightarrow PDS)$  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.<sup>15</sup> The definition of each PDS provides enough information to define the initial conditions for the accident progression event tree (APET) analysis. Due to the large number of questions in the Grand Gulf 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 accident progression bins, 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 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

$$f_{APB} = f_{PDS} P(PDS \rightarrow APB), \quad (\text{Eq. 1.2})$$

where  $f_{PDS}$  is the vector of frequencies for the PDSs defined in Eq. 1.1,  $f_{APB}$  is the vector of frequencies for the accident progression bins, and  $P(PDS \rightarrow APB)$  is the matrix of transition probabilities from PDSs to accident progression bins. Specifically:

$$f_{APB} = [f_{APB_1}, \dots, f_{APB_{n_{APB}}}],$$

$f_{APB_k}$  = frequency ( $\text{yr}^{-1}$ ) for accident progression bin  $k$ ,

$n_{APB}$  = number of accident progression bins,

$$P(PDS \rightarrow APB) = \begin{bmatrix} p_{APB_{11}} & \dots & p_{APB_{1,n_{APB}}} \\ \vdots & & \vdots \\ p_{APB_{n_{PDS},1}} & \dots & p_{APB_{n_{PDS},n_{APB}}} \end{bmatrix}$$

and

$p_{APB_{jk}}$  = probability that plant damage state  $j$  will lead to accident progression bin  $k$ .

The properties of  $f_{PDS}$  are given in conjunction with Eq. 1.1. The elements  $p_{APB_{jk}}$  of  $P(PDS \rightarrow APB)$  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 GGSOR. GGSOR is not a detailed mechanistic model and is not designed to simulate the fission product transport, physics, and chemistry from first principles. Instead, GGSOR 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,<sup>16,17</sup> MELCOR,<sup>18</sup> and MAAP.<sup>19</sup> Most of the parameters utilized calculating the fission product release fractions in GGSOR are sampled from distributions provided by an expert panel. Because of the large number of APBs, use of fast-executing code like GGSOR 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,<sup>20,21,22</sup> 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.<sup>23</sup>

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

$$fSTG = fAPB P(APB \rightarrow STG) \quad (Eq. 1.3)$$

where fAPB is the vector of frequencies for the accident progression bins 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 accident progression bins to source term groups. Specifically,

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

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

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

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

and

$p_{STG_{k,l}}$  = probability that accident progression bin  $k$  will be assigned to source term group  $l$ .

$$= \begin{cases} 1 & \text{if accident progression bin } k \text{ is} \\ & \text{assigned to source term group } l \\ 0 & \text{otherwise.} \end{cases}$$

The properties of  $f_{APB}$  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

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

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,

$$rC = \{rC_1, \dots, rC_{nC}\},$$

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

$nC$  = number of consequence measures,

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

and

$cSTG_{lm}$  = 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  $cSTG_{lm}$  of cSTG are determined from consequence calculations with MACCS for individual source term groups.

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

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

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<sup>24</sup> 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 (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 utilizes both mean risk and CCDFs to report the risk results.

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

$$X_1, X_2, \dots, X_{nV} \quad \text{(Eq. 1.6)}$$

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

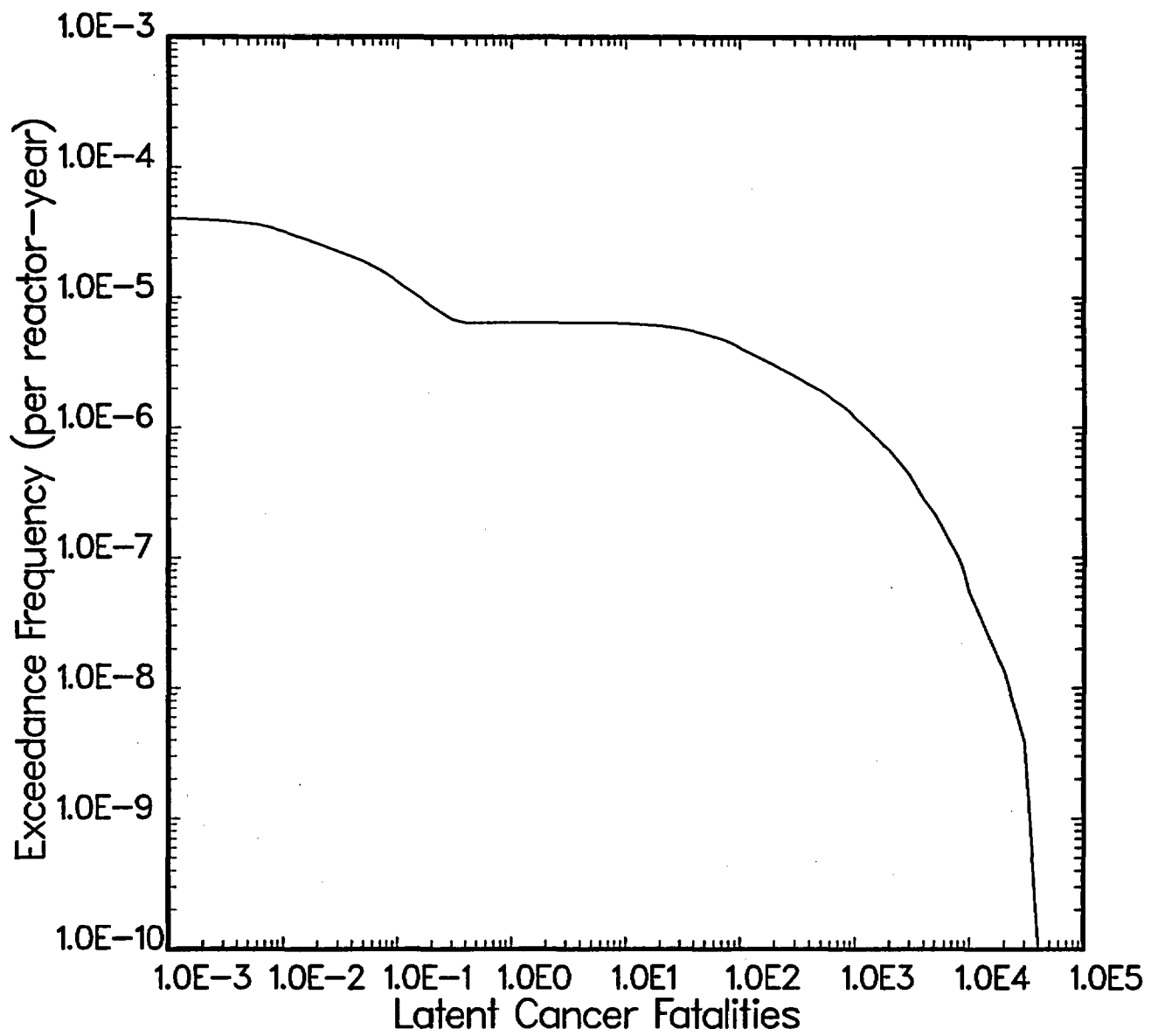


Figure 1.3. Example Risk CCDF.



techniques<sup>25</sup> (also in Vol. 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

$$D_1, D_2, \dots, D_{nV}, \quad (\text{Eq. 1.7})$$

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

From these distributions, a stratified Monte Carlo technique, Latin Hypercube Sampling,<sup>26,27</sup> 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

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

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

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

where the subscript  $i$  is used to denote the evaluation of the expression in Eq. 1.5 with the  $i^{\text{th}}$  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  $P(IE \rightarrow PDS)$ ,  $P(PDS \rightarrow APB)$  and  $P(APB \rightarrow STG)$  are based on results obtained with TEMAC, EVNTRE and GGSOR, 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. The first volume 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. The second volume 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 core-containment interactions, 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 presents analogous results for Zion. The Zion analyses were performed by BNL.

This volume of NUREG/CR-4551, Volume 6, presents risk and constituent analysis results for Unit 1 of the Grand Gulf Nuclear Station, operated by the System Energy Resources Inc.. Part 1 of this volume presents the analysis and the results in some detail; Part 2 consists of appendices, which contain further detail. Following a summary and an introduction, Chapter 2 of this volume presents the results of the accident progression analysis for internal initiating events. Chapter 3 presents the result 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 Grand Gulf, and Chapter 6 contains the insights and conclusions of the complete analysis.

## 1.6 References

1. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, June 1989.
2. U.S. Nuclear Regulatory Commission, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), 1975.
3. J. S. Evans et al., "Health Effects Models for Nuclear Power Plant Accident Consequence Analysis," NUREG/CR-4214, SAND85-7185, Sandia National Laboratories, August 1986.
4. J. S. Evans et al., "Health Effects Models for Nuclear Power Plant Accident Consequence Analysis," NUREG/CR-4214, Revision 1, SAND85-7185, Sandia National Laboratories, and Harvard University, Cambridge, MA, (Part I published January 1990; Part II published May 1989).
5. S. Kaplan, "Matrix Theory Formalism for Event Tree Analysis: Application to Nuclear-Risk Analysis," Risk Analysis, 2, pp. 9-18, 1982.
6. D. C. Bley, S. Kaplan, and B. J. Garrick, "Assembling and Decomposing PRA Results: A Matrix Formalism," in Proceedings of the International Meeting on Thermal Nuclear Reactor Safety, NUREG/CP-0027, Vol. 1, pp. 173-182, U. S. Nuclear Regulatory Commission, Washington, D.C., 1982.
7. R. B. Worrell, "SETS Reference Manual," NUREG/CR-4213, SAND83-2675, Sandia National Laboratories, May 1985.
8. R. C. Bertucio and A. J. Julius, "Analysis of Core Damage Frequency: Surry, Unit 1 Internal Events," NUREG/CR-4550, Vol. 3, SAND86-2084, Sandia National Laboratories, April 1990.
9. R. C. Bertucio and S. R. Brown, "Analysis of Core Damage Frequency: Sequoyah, Unit 1 Internal Events," NUREG/CR-4550, Vol. 5, SAND86-2084, Sandia National Laboratories, 1989.
10. A. M. Kolaczowski et al., "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events," NUREG/CR-4550, Vol. 4, SAND86-2084, Sandia National Laboratories, April 1989.
11. M. T. Drouin et al., "Analysis of Core Damage Frequency: Grand Gulf, Unit 1 Internal Events," NUREG/CR-4550, Vol. 6, SAND86-2084, Sandia National Laboratories, September 1989.
12. M. B. Sattison and K. W. Hall, "Analysis of Core Damage Frequency: Zion, Unit 1 Internal Events," NUREG/CR-4550, Vol. 7, EGG-2554, EG&G Idaho Inc., (Idaho National Engineering Laboratory), May 1990.

13. R. L. Iman, "A Matrix-Based Approach to Uncertainty and Sensitivity Analysis for Fault Trees," Risk Analysis, 7, pp. 21-33, 1987.
14. R. L. Iman and M. J. Shortencarier, "A User's Guide for the Top Event Matrix Analysis Code (TEMAC)," NUREG/CR-4598, SAND86-0960, Sandia National Laboratories, April 1986.
15. J. M. Griesmeyer and L. N. Smith, "A Reference Manual for the Event Progression Analysis Code (EVNTRE)," NUREG/CR-5174, SAND88-1607, Sandia National Laboratories, September 1989.
16. R. S. Denning, J. A. Gieseke, P. Cybulskis, K. W. Lee, H. Jordan, L. A. Curtis, R. F. Kelly, V. Kogan, and P. M. Schumacher, "Radionuclide Calculations for Selected Severe Accident Scenarios," NUREG/CR-4624, BMI-2139, Vols. 1-5, Battelle Columbus Division, 1986.
17. M. T. Leonard et al., "Supplemental Radionuclide Release Calculations for Selected Severe Accident Scenarios," NUREG/CR-5062, BMI-2160, Battelle Columbus Division, 1988.
18. S. E. Dingman et al., "MELCOR Analyses for Accident Progression Issues," NUREG/CR-5331, SAND89-0072, Sandia National Laboratories, December 1990.
19. Industry Degraded Core Rulemaking Program, "Modular Accident Analysis Program (MAAP) User's Manual," IDCOR Technical Report on subtasks 16.2 and 16.3, Fausts & Associates, Inc., for the Atomic Industrial Forum, Bethesda, MD, 1987.
20. 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, Vol. 1, Sandia National Laboratories, February 1990.
21. H.-N. Jow, J. L. Sprung, J. A. Rollstin, L. T. Ritchie, and D. I. Chanin, "MELCOR Accident Consequence Code System (MACCS): Model Description," NUREG/CR-4691, SAND86-1562, Vol. 2, Sandia National Laboratories, February 1990.
22. 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, Vol. 3, Sandia National Laboratories, February 1990.
23. R. L. Iman, J. C. Helton, and J. D. Johnson, "PARTITION: A Program Defining the Source Term/Consequence Analysis Interface in the NUREG-1150 Probabilistic Risk Assessments User's Guide," NUREG/CR-5253, SAND88-2940, Sandia National Laboratories, May 1989.
24. R. L. Iman, J. D. Johnson, and J. C. Helton, "PRAMIS: Probabilistic Risk Assessment Model Integration System User's Guide," NUREG/CR-5262, SAND88-3093, Sandia National Laboratories, May 1990.

25. S. C. Hora and R. L. Iman, "Expert Opinion in Risk Analysis - The NUREG-1150 Methodology," Nuclear Science and Engineering, 102: pp. 323-331 (1989).
26. M. J. McKay, W. J. Conover, and R. J. Beckman, "A Comparison of Three Methods for Selecting Values of Input Variables in the Analysis of Output from a Computer Code," Technometrics, 21, 239-245, 1979.
27. R. L. Iman and M. J. Shortencarier, "A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models," NUREG/CR-3624, SAND83-2365, Sandia National Laboratories, March 1984.

## 2. ANALYSIS OF THE ACCIDENT PROGRESSION

This chapter describes the analysis of the progression of the accident, starting from significant core uncovering (i.e., 2 ft above the bottom of the active fuel [BAF] with imminent re-flooding of the core not expected) and continuing for about 24 h or until the bulk of the radioactive material that is going to be released has been released. 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 of this report. 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 their 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 Plant Features Important to the Accident Progression at Grand Gulf

The entire Grand Gulf 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 Drywell Structure and Suppression Pool;
- The Reactor Pedestal Cavity;
- The Hydrogen Ignition System (HIS);
- The Containment Heat Removal System; and
- The Automatic Depressurization System (ADS).

The Grand Gulf Containment Structure. Grand Gulf has a Mark III containment. The Grand Gulf containment is a reinforced concrete structure with a steel liner. An important feature of the the Mark III containment is its large free volume (1,400,000 ft<sup>3</sup>) which allows it to have a low design pressure (15 psig). The assessed mean failure pressure of the containment is 55 psig. Because of its large volume, the Grand Gulf containment is not inerted. Thus, during accidents in which the HIS is not available, combustible hydrogen mixtures can be present in the containment.

The Drywell Structure and Suppression Pool. The Grand Gulf drywell houses the reactor pressure vessel (RPV) and is completely surrounded by the containment structure. The drywell structure is a reinforced concrete structure and has a design pressure of 30 psid. The free volume of the drywell is 270,000 ft<sup>3</sup>. The assessed mean failure pressure of the drywell structure is 85 psid.

The drywell volume communicates to the containment volume through the vapor suppression pool. Passive vents allow the passage of steam and air into the wetwell after first passing through the pool which provides the condensing action. The RPV safety/relief line (including those associated with the ADS) discharge through spargers into the suppression pool, which again provides condensation of any steam releases. Thus, in-vessel releases are first passed through the pool before being released to the wetwell air space. The steam is condensed in the pool and the noncondensibles (i.e., hydrogen) are passed to the wetwell air space. Similarly, releases accompanying vessel breach are directed to the suppression pool (assuming the drywell structure is intact) before being released into the containment. This process reduces the pressure in the containment; however, it also allows combustible mixtures of hydrogen and air to accumulate in the containment. The HIS is designed to burn this hydrogen at low concentrations so that the accompanying containment pressurization is negligible.

The Reactor Pedestal Cavity. The reactor pedestal cavity is located directly below the RPV. The upper section of the cavity is formed by the 5.75 ft thick pedestal wall and the lower section of the cavity is recessed into the drywell floor. The pedestal cavity is essentially a right cylinder with a diameter of 21.17 ft and a depth of approximately 28 ft. The upper section of the cavity contains the control rod drive (CRD) housings. The major pedestal penetrations are the CRD piping penetrations at the top of the pedestal and the CRD removal opening which is a 3 ft by 7 ft doorway and is located 9.5 ft above the cavity floor. The cavity can contain all of the core debris released at the time of vessel breach. Thus, direct attack of the drywell wall by core debris is not an issue at Grand Gulf as it is for the Mark I containments. When the drywell is completely flooded a water depth of 22.8 ft can be established in the cavity. There are two pathways by which water in the drywell can enter the reactor cavity. The first pathway is through the drywell floor drains. There are four 4-inch drains in the drywell floor that connect to the equipment drain sump in the pedestal. The second pathway is through a door (3 ft by 7 ft) in the pedestal located 3'-4" above the drywell floor. The potential for large amounts of water to be in the cavity has two major implications. First, when core debris is released from the vessel at the time of vessel breach the potential exist for large fuel-coolant interactions (FCIs) to occur if the cavity is full of water. These FCIs can fail the drywell directly from quasi-static pressure loads or can fail the RPV pedestal, which can then lead to drywell failure (e.g., penetration failure). On the other hand, a large amount of water in the cavity can cool the core debris that is released from the reactor vessel and thus mitigate the releases associated with core-concrete interactions (CCIs).

Hydrogen Ignition System (HIS). The Grand Gulf containment has an HIS. Igniters are located throughout the containment and drywell volumes. The function of the HIS is to prevent the buildup of large quantities of hydrogen inside the containment during accident conditions. This is accomplished by igniting, via a spark, small amounts of hydrogen before it has had a chance to accumulate. The HIS consists of 90 General Motors AC Division glow plugs (Model 7G), 45 powered by each ac power division. The HIS is manually actuated. The glow plugs would not perform their function without ac power. Thus, the HIS will not be available either during a station blackout or if the operators fail to actuate the system if ac power is available.

Containment Heat Removal Systems. Suppression pool cooling (SPC) and the containment spray system (CS) are two modes of the residual heat removal (RHR) system. The RHR system is a two-train system with motor-operated valves and pumps. Both trains have two heat exchangers in series downstream of the pump. In either the SPC or the CS modes of operation, the RHR system can remove heat from the suppression pool by passing water from the pool through heat exchangers (with service water on the shell side). In the CS mode, water is sprayed into the containment. The SPC system is manually initiated and controlled. The CS system, on the other hand, is automatically initiated and controlled. Both the SPC and the CS modes of RHR require ac power and are, therefore, unavailable during a station blackout.

The Automatic Depressurization System. The Automatic Depressurization System (ADS) is designed to depressurize the reactor vessel to a pressure at which the low pressure injection systems can inject coolant to the reactor vessel. The ADS consists of eight relief valves capable of being manually opened. For the system to be automatically initiated a low pressure injection pump must be running. Thus, the ADS will not be automatically initiated during a station blackout. The operator can also manually initiate the ADS, or he may depressurize the reactor vessel using the 12 safety relief valves (SRVs) that are not connected to the ADS logic. Each valve discharges into the suppression pool. The ADS valves are located in the drywell and pressures of approximately 100 psi will prevent opening the ADS valves. The assessed containment failure pressure at the 99th percentile is only 97 psig and, thus, failure of the ADS because of high pressure is not considered in this study. The ADS does, however, require dc power. Therefore, the RPV can not be depressurized in sequences that involve failure of dc power.

## 2.2 Interface with the Core Damage Frequency Analysis

### 2.2.1 Definition of PDSs

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 after core damage has begun. For Grand Gulf, the PDS is



denoted by a 12-letter indicator that defines six 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 6.<sup>1</sup> The methods used in the accident frequency analysis are presented in NUREG/CR-4550, Volume 1.<sup>2</sup>

Table 2.2-1 lists the six characteristics used to define the PDSs. Under each characteristic are given the possible values for that characteristic. For example, the first characteristic denotes the initiating event and the status of ac and dc power at the time core damage begins (assumed to be when the water level is 2 ft above the BAF). Table 2.2-1 shows that there are four possibilities for this characteristic: B1 for a station blackout with offsite power not recoverable because there is no emergency dc power; B2 for a station blackout with offsite power recoverable; T2 for loss of power conversion system (PCS) transient; and TC for an anticipated transient without scram (ATWS).

The first characteristic denotes the initiating event and the status of ac and dc power. The station blackouts are separated based on the availability of dc power. The loss of PCS transient and the ATWS event have both onsite and offsite power.

The second characteristic denotes the reactor vessel pressure at the time of core damage. The reactor pressure can be either high or low. High pressure is defined as system pressure (approximately 1040 psig). Low pressure is defined as being less than 200 psia.

The third characteristic denotes the type of coolant injection that is available or recoverable. This characteristic indicates if the coolant injection system is a high pressure system or a low pressure system. The availability of the firewater system and the condensate system are also indicated because these systems require operator actions to align these systems.

The fourth characteristic denotes the availability of the containment spray (CS) mode of RHR. In this analysis, the RHR heat exchangers are always available when the containment sprays are available. Therefore, there are no scenarios that involve spraying hot water because the heat exchangers are not available.

The fifth characteristic denotes the availability of the containment venting system, the standby gas treatment system, containment isolation system and the hydrogen ignition system. All of these systems require ac power and, therefore, their availability is directly related to the availability of ac power.

The sixth characteristic denotes the time of core damage. Two times are considered in this analysis: core damage occurs in the short term (at  $\approx 1$  h), and core damage occurs in the long term (at  $\geq 12$  h). When the core damage occurs in the short term the accident is referred to as a short term or fast accident (e.g., short term station blackout or fast station blackout). Similarly, when core damage occurs in the long term the accident is referred to as a long term or slow accident (e.g., slow TC).

Table 2.2-1  
Grand Gulf PDS Characteristics

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1. What is the initiating event and what is the status of ac and dc power?
  - B1 - Station blackout transient has occurred. Offsite power is not recoverable because there is no emergency dc power.
  - B2 - Station blackout transient has occurred. Offsite power is recoverable.
  - T2 - Loss of PCS transient has occurred. Offsite or onsite power is available.
  - TC - ATWS has occurred. Offsite or onsite power is available.
  
2. Is the reactor vessel at high or low pressure?
  - P1 - The reactor vessel is at high pressure at the onset of core damage and depressurization is not possible.
  - P2 - The reactor vessel is at high pressure at the onset of core damage because the operator failed to depressurize; depressurization is possible.
  - P3 - The reactor vessel could be at high pressure at the onset of core damage. The operator depressurizing the vessel (which is possible) was not included in the model.
  - P4 - The reactor vessel is at low pressure.
  
3. What type of coolant injection is available or recoverable?
  - I1 - Injection to the reactor vessel is not available after the onset of core damage.
  - I2 - Injection with the firewater system is available before and after the onset of core damage.
  - I3 - Injection with the condensate system is recoverable with the restoration of offsite power.
  - I4 - Injection with the low pressure systems (LPCS and coolant injection) is recoverable with the restoration of offsite power.
  - I5 - Injection with both the high and low pressure systems is recoverable with the restoration of offsite power.

Table 2.2-1 (Continued)

- 
- I6 - Injection with the high pressure systems (reactor core isolation cooling [RCIC] and CRD) and the low pressure systems (LPCS and coolant injection) is recoverable with the restoration of offsite power.
4. Is Containment Spray (CS) mode of RHR available or recoverable?
- H1 - CS is not available at the onset of core damage, neither is it recoverable.
- H2 - At least one train of CS is recoverable with the restoration of offsite power.
- H3 - At least one train of CS is available at the onset of core damage.
5. Are the following systems available: venting, SBT, CI, and H<sub>2</sub>I?
- M1 - Miscellaneous systems (venting, SBT, CI, and H<sub>2</sub>I) are not available at the onset of core damage.
- M2 - Miscellaneous systems (venting, SBT, CI, and H<sub>2</sub>I) are recoverable with the restoration of offsite power.
- M3 - Miscellaneous systems (venting, SBT, CI, and H<sub>2</sub>I) are available at the onset of core damage.
6. When does core damage occur?
- ST - Core damage occurs in the short term (at  $\approx 1$  h).
- LT - Core damage occurs in the long term (at  $\geq 12$  h).
-

### 2.2.2 PDS Frequencies

In this subsection the 12 PDSs are described and their core damage frequencies are presented. The accident frequency analysis for internal initiators was performed with more observations per sample than were the accident progression analysis and the subsequent analyses. Since the samples were different in the random seed as well as the number of observations, the core damage frequencies differ slightly as is to be expected. The PDSs used in the Grand Gulf accident progression, source term, and risk integration analyses are presented in Table 2.2-2. The mean core damage frequencies presented in this table are based on a sample size of 250. The core damage frequency distributions for the 12 PDSs, based on a sample size of 250, are presented in Figure 2.2-1.

The accident frequency analysis reports the PDS frequencies based on a sample size of 1000 (see Section 5 of NUREG/CR-4550, Vol. 6, Part 1).<sup>1</sup> 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 250 observations for the integrated risk 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 utilized for the stand-alone accident frequency analysis and the 250-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 5.3-1 of NUREG/CR-4550, Volume 6, Part 1. Samples containing 250 observations are used for the integrated risk analysis at Grand Gulf. The 5th percentile, median, mean, and 95th percentile core damage frequencies for this sample are shown on the second line of Table 2.2-3 for each PDS.

The differences between distributions for core damage frequency for the two samples are within the statistical variation to be expected. Note that the fractional contributions of each PDS to the TMCD in Table 2.2-2 are slightly different from those in Table 2.2-3. This is due to the fact that the PDS fractional contributions in Table 2.2-2 are based on the sample of 250 observations, and the contributions in Table 2.2-3 are based on the sample of 1000 observations.

The remaining portion of this subsection describes the essential characteristics of each of the 12 PDSs. The descriptions of the PDSs were extracted from Chapter 5 of NUREG/CR-4550, Volume 6, Part 1.<sup>1</sup>

Table 2.2-2  
PDS Core Damage Frequencies for Grand Gulf

<u>PDS Number</u>	<u>PDS Name</u>	<u>Mean CD Frequency (1/yr)</u>	<u>PDS % TPCD Freq.</u>	<u>PDS Descriptor</u>
1	Fast Blackout	3.2E-06	79.2	B2-P3-I5-H2-M2-ST
2	Fast Blackout	4.6E-08	1.1	B2-P3-I5-H1-M2-ST
3	Fast Blackout	1.5E-07	3.7	B2-P3-I3-H1-M2-ST
4	Slow Blackout	3.7E-08	0.9	B2-P4-I5-H2-M2-LT
5	Slow Blackout	2.3E-09	<<1	B2-P4-I5-H1-M2-LT
6	Slow Blackout	1.4E-09	<<1	B2-P4-I2-H1-M2-LT
7	Fast Blackout	4.2E-07	10.3	B1-P1-I1-H1-M1-ST
8	Slow Blackout	6.3E-08	1.5	B1-P1-I1-H1-M1-LT
9	Fast ATWS	5.0E-08	1.2	TC-P2-I6-H3-M3-ST
10	Slow ATWS	6.2E-08	1.5	TC-P2-I4-H3-M3-LT
11	Fast T2	1.8E-08	0.4	T2-P2-I5-H3-M3-LT
12	Slow T2	2.9E-10	<<1	T2-P2-I5-H3-M3-LT

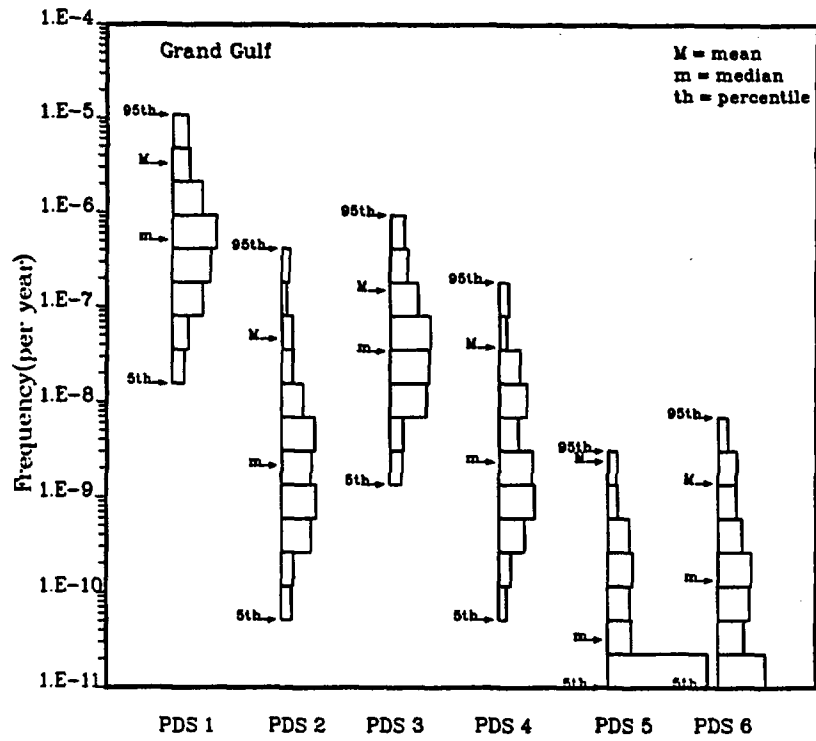


Figure 2.2-1. Core Damage Frequency Distributions.

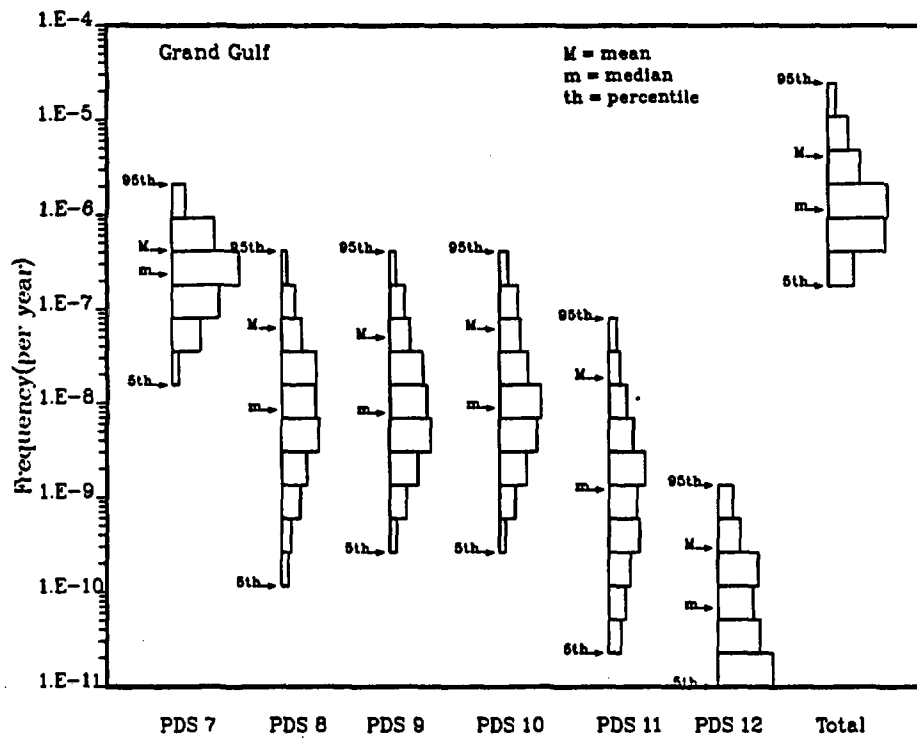


Figure 2.2-1 (continued).

Table 2.2-3  
Plant Damage State Comparison for Grand Gulf

PDS	LHS Sample Size <sup>(1)</sup>	Core Damage Frequency (1/yr)				% TMCD Freq. <sup>(2)</sup>
		5%	Median	Mean	95%	
PDS1	1000	2.5E-08	5.0E-07	3.2E-06	9.6E-06	79
Fast SBO	250	2.6E-08	5.1E-07	3.2E-06	1.1E-05	
PDS2	1000	6.9E-11	2.3E-09	4.8E-08	9.4E-08	1
Fast SBO	250	6.4E-11	2.1E-09	4.6E-08	1.9E-07	
PDS3	1000	1.5E-09	3.1E-08	1.8E-07	5.8E-07	4
Fast SBO	250	1.3E-09	3.4E-08	1.5E-07	6.7E-07	
PDS4	1000	6.4E-11	2.7E-09	3.9E-08	1.0E-07	1
Slow SBO	250	5.3E-11	2.3E-09	3.7E-08	1.6E-07	
PDS5	1000	5.5E-13	3.6E-11	1.3E-09	2.7E-09	<<1
Slow SBO	250	7.4E-13	3.2E-11	2.3E-09	3.0E-09	
PDS6	1000	2.4E-12	1.4E-10	2.0E-09	5.8E-09	<<1
Slow SBO	250	1.4E-12	1.3E-10	1.4E-09	7.2E-09	
PDS7	1000	2.9E-08	2.3E-07	4.3E-07	1.4E-06	11
Fast SBO	250	2.8E-08	2.4E-07	4.2E-07	1.6E-06	
PDS8	1000	3.0E-10	9.2E-09	6.6E-08	2.0E-07	2
Slow SBO	250	2.6E-10	8.4E-09	6.3E-08	2.7E-07	
PDS9	1000	3.9E-10	8.9E-09	5.0E-08	2.3E-07	1
Fast ATWS	250	3.2E-10	7.9E-09	5.0E-08	1.9E-07	
PDS10	1000	4.9E-10	1.0E-08	6.3E-08	2.8E-07	2
Slow ATWS	250	3.9E-10	8.9E-09	6.2E-08	2.3E-07	
PDS11	1000	2.5E-11	1.3E-09	1.2E-08	4.4E-08	<1
Fast T2	250	3.1E-11	1.2E-09	1.8E-08	5.3E-08	
PDS12	1000	1.6E-13	9.9E-12	2.7E-10	8.3E-10	<<1
Slow T2	250	4.9E-12	6.8E-11	2.9E-10	1.2E-09	
Total	1000	1.7E-07	1.2E-06	4.0E-06	1.2E-05	
	250	1.8E-07	1.1E-06	4.1E-06	1.4E-05	

Notes:

(1)The Accident Frequency Analysis used a LHS sample size of 1000  
The Accident Progression Analysis used a LHS sample size of 250

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

PDS-1 (B2-P3-I5-H2-M2-ST). This PDS involves station blackout scenarios where loss of offsite power (LOSP) is recoverable (B2). Coolant injection is lost early such that core damage occurs in the short term (ST) and with the vessel at high pressure (P3) because depressurization did not have an effect in the prevention of core damage. If offsite power is restored then the following functions are available: either high pressure injection or low pressure injection or both (I5), heat removal via the sprays (H2), and the miscellaneous systems--venting, standby gas treatment (SBGT), containment isolation (CI), hydrogen ignition (H2I) (M2).

This PDS also includes cut sets with either one or two stuck-open relief valves (SORVs). With the restoration of offsite power, the following coolant injection systems are recoverable: HPCS, condensate, Low Pressure Coolant Injection (LPCI) and Low Pressure Core Spray (LPCS). In some cases, HPCS and LPCS are recoverable, but only for around 12 h; they are then lost on room heatup. The firewater system is available in every cut set. For those cut sets with two SORVs, the RCIC system is available but is not sufficient to prevent core damage.

PDS-2 (B2-P3-I5-H1-M2-ST). This PDS involves station blackout scenarios where LOSP is recoverable (B2). Coolant injection is lost early so that core damage occurs in the short term (ST) and with the vessel at high pressure (P3) because depressurization did not have an effect in the prevention of core damage. If offsite power is restored then the following functions are available: either high pressure injection or low pressure injection or both (I5), and the miscellaneous systems--venting, SBGT, CI, H2I (M2). Heat removal via the sprays is not available with the recovery of offsite power (H1).

This PDS also includes cut sets with either one or two SORVs. With the restoration of offsite power, the following coolant injection systems are recoverable: HPCS and condensate. In some cases, LPCS is recoverable, but only for approximately 12 h, at which time they fail as a result of room heatup. The Firewater system is available in every cut set. For those cut sets with two SORVs, the RCIC system is available but is not sufficient to prevent core damage.

PDS-3 (B2-P3-I3-H1-M2-ST). This PDS involves station blackout scenarios where LOSP is recoverable (B2). Coolant injection is lost early so that core damage occurs in the short term (ST) and with the vessel at high pressure (P3) because depressurization did not have an effect in the prevention of core damage. If offsite power is restored then the following functions are available: low pressure injection only with condensate (I3) and the miscellaneous systems--venting, SBGT, CI, H2I (M2). Heat removal via the sprays is not available with the restoration of offsite power (H1).

This PDS also includes cut sets with either one or two SORVs. With the restoration of offsite power, the following coolant injection system is recoverable: condensate. HPCS and LPCS are available with the recovery of offsite power, but only for approximately 12 h, at which time they fail as a result of room heatup. The Firewater system is available in every cut set. For those cut sets with two SORVs, the RCIC system is available but is not sufficient to prevent core damage.



PDS-4 (B2-P4-I5-H2-M2-LT). This PDS involves station blackout scenarios where LOSP is recoverable (B2). Coolant injection is lost late so that core damage occurs in the long term (LT) and with the vessel at low pressure (P4). If offsite power is restored then the following functions are available: either high pressure injection or low pressure injection or both (I5), heat removal via the sprays (H2), and the miscellaneous systems--venting, SBT, CI, H2I (M2).

With the restoration of offsite power, the following coolant injection systems are recoverable: HPCS, condensate, LPCI and LPCS. In some cases, HPCS and LPCS are recoverable, but only for approximately 12 h, at which time they fail as a result of room heatup. The Firewater system is available in every cut set.

PDS-5 (B2-P4-I5-H1-M2-LT). This PDS involves station blackout scenarios in which LOSP is recoverable (B2). Coolant injection is lost late so that core damage occurs in the long term (LT) and with the vessel at low pressure (P4). If offsite power is restored, then the functions of high pressure injection or low pressure injection or both (I5) are available, as well as the miscellaneous systems of venting, SBT, CI, and H2I (M2). Heat removal via the sprays is not available with the restoration of offsite power (H1).

There are some cut sets in which heat removal sprays are available with offsite power restoration, but these have negligible contribution and were not removed.

PDS-6 (B2-P4-I2-H1-M2-LT). This PDS involves station blackout scenarios where LOSP is recoverable (B2). Coolant injection is lost late so that core damage occurs in the long term (LT) and with the vessel at low pressure (P4). Firewater is recoverable (I2). If offsite power is restored, then the following functions are available: the miscellaneous systems--venting, SBT, CI, H2I (M2). Heat removal via the sprays is not available with the restoration of offsite power (H1).

HPCS is available with the restoration of offsite power, but only for around 12 h; it is then lost on room heatup.

PDS-7 (B1-P1-I1-H1-M1-ST). This PDS involves station blackout (without any dc power) scenarios where LOSP is not recoverable (B1). Coolant injection is lost early so that core damage occurs in the short term (ST) and with the vessel at high pressure (P1) and depressurization is not possible. Since offsite power is not recoverable, the functions of injection [I1], heat removal [H1], and those of the miscellaneous systems [M1], are not available.

PDS-8 (B1-P1-I1-H1-M1-LT). This PDS involves station blackout (without any dc power) scenarios where LOSP is not recoverable (B1). Coolant injection is lost late so that core damage occurs in the long term (LT) and with the vessel at high pressure (P1), and depressurization is not possible. Since offsite power is not recoverable, functions (i.e., injection [I1], heat removal [H1] and the miscellaneous systems [M1]) are not available.

PDS-9 (TC-P2-I6-H3-M3-ST). This PDS involves ATWS transient scenarios (TC). Coolant injection is lost early so that core damage occurs in the short term (ST) and with the vessel at high pressure because the operator failed to depressurize (P2). High pressure injection with RCIC is available (I6). Heat removal via the sprays is available (H3) and the miscellaneous systems (i.e., venting, SBT, CI and H2I) are available (M3).

PDS-10 (TC-P2-I4-H3-M3-LT). This PDS involves ATWS transient scenarios (TC). Coolant injection is lost late such that core damage occurs in the long term (LT) and with the vessel at high pressure because the operator failed to depressurize (P2). Low pressure injection is recoverable with reactor depressurization (I4). Heat removal via the sprays is available (H3) and the miscellaneous systems (i.e., venting, SBT, CI and H2I) are available (M3).

PDS-11 (T2-P2-I5-H3-M3-ST). This PDS involves transient scenarios where the PCS is lost (T2). Coolant injection is lost early so that core damage occurs in the short term (ST) and with the vessel at high pressure because the operator failed to depressurize (P2). Both high pressure and low pressure are recoverable (I5) since the failures involved operator failures. Heat removal via the sprays is available (H3) and the miscellaneous systems (i.e., venting, SBT, CI and H2I) are available (M3).

PDS-12 (T2-P2-I5-H3-M3-LT). This PDS involves transient scenarios where the PCS is lost (T2). Coolant injection is lost late so that core damage occurs in the long term (ST) and with the vessel at high pressure because the operator failed to depressurize (P2). Both high and low pressure are recoverable (I5) since the failures involved operator failures. Heat removal via the sprays is available (H3) and the miscellaneous systems (i.e., venting, SBT, CI and H2I) are available (M3).

### 2.2.3 High-Level Grouping of PDSs

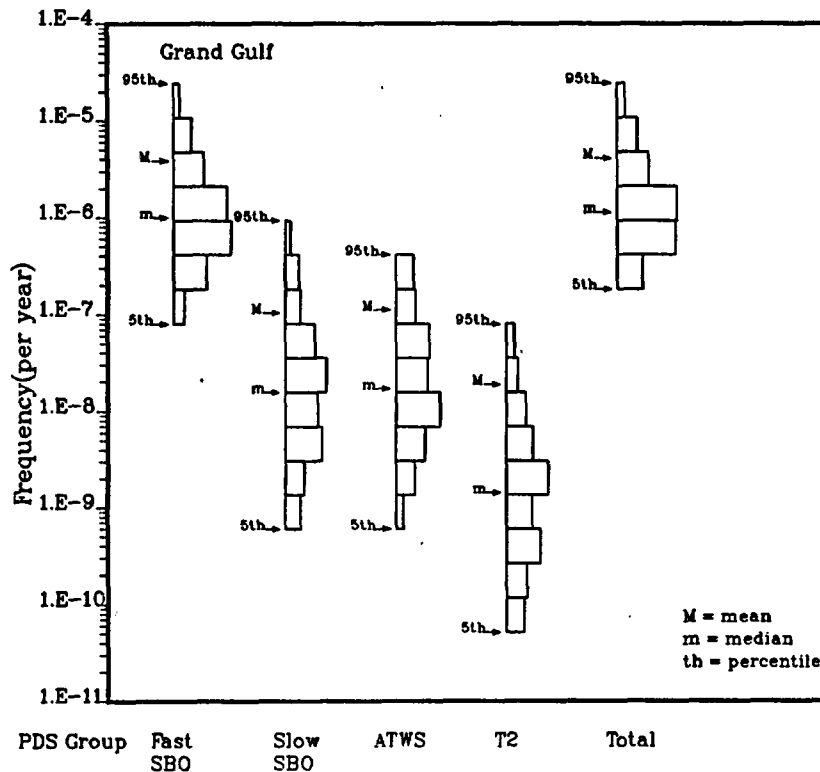
To provide simpler, more easily understood summaries for NUREG-1150, the 12 PDSs described above were further condensed into the following four groups:

1. Short Term Station Blackout
2. Long Term Station Blackout
3. ATWS
4. T2 Transients

These four groups are denoted summary PDS groups or collapsed PDS groups. The mapping from the 12 groups described in the previous section into the four 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. The core damage frequency distributions for the four summary PDS groups are presented in Figure 2.2-2.

**Table 2.2-4**  
**Relationship between PDSs and Summary Groups**

<u>Summary Group</u>	<u>% TMCDF</u>	<u>PDS Groups</u>	<u>% TMCDF</u>
1. Fast SBO	95	1. Fast Blackout	79
		2. Fast Blackout	1
		3. Fast Blackout	4
		7. Fast Blackout	11
2. Slow SBO	3	4. Slow Blackout	1
		5. Slow Blackout	<<1
		6. Slow Blackout	<<1
		8. Slow Blackout	2
3. ATWS	3	9. Fast ATWS	1
		10. Slow ATWS	2
4. Transients	<1	11. Fast Transient	<1
		12. Slow Transient	<<1



**Figure 2.2-2. Core Damage Frequency Distributions for the Summary PDS Groups. Grand Gulf: Internal Initiators.**

#### 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. 6, Part 1.)<sup>1</sup> 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.

The first column in Table 2.2-5 contains the variable name which is an eight character identifier. Where these differ from the identifiers used in the fault trees, these identifiers 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. 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. 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 Grand Gulf accident frequency analysis report (NUREG/CR-4550, Vol. 6).<sup>1</sup> This report also gives the source or the derivation of the distributions for all these variables.

#### 2.3 Description of the APET

This section describes the APET that is used to perform the accident progression analysis for Grand Gulf. 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.<sup>3</sup>

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 Grand Gulf APET, which can be evaluated relatively 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 Grand Gulf APET. Details, including a complete listing of the APET and a discussion of each question, may 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 Grand Gulf.

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

<u>Variable</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
MOV-FOP	1.5E-05 0.085	Lognormal Mean=0.003	None		Probability of failure to open (per demand) for motor operated valves (generic).
MOV-MAIN	4.0E-6 0.023	Lognormal Mean=7.8E-4	None		Probability that the motor operated valve is out for maintenance (per demand) (generic). [MOV-MA]
MDP-FSTR	1.5E-5 0.085	Lognormal Mean=0.003	None		Probability of failure to start (per demand) for motor-driven pumps (generic). [MDP-FS]
MDP-FRUN	3.6E-6 0.020	Lognormal Mean=7.2E-4	None		Probability of failure to run (per demand) for motor-driven pumps (generic). [MDP-FR]
MDP-MAIN	9.9E-6 0.057	Lognormal Mean=1.9E-3	None		Probability that the motor-driven pump is out for maintenance (per demand) (generic). [MDP-MA]
TDP-FSTR	1.5E-4 0.85	Lognormal Mean=0.029	None		Probability of failure to start (per demand) for turbine-driven pump (RCIC) (generic). [TDP-FS]
TDP-FRUN	0.012 1.0	Max Entropy Mean =0.12	None		Probability for failure to run (per demand) for turbine-driven pump (RCIC) (generic). [TDP-FR]
DDP-FRUN	9.4E-5 0.54	Lognormal Mean=0.019	None		Probability for failure to run (per demand) for diesel-driven pump (FWS) (generic). [DDP-FR]

Table 2.2-5 (continued)

<u>Variable</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
DGN-FSTR	0.003 0.19	Lognormal Mean=0.030	None		Probability that a diesel generator fails to start (per demand) (generic). [DGN-FS]
DGN-FRUN	7.9E-5 0.45	Lognormal Mean=0.016	None		Probability that a diesel generator fails to run (per demand) (generic). [DGN-FR]
DGN-MAIN	3.0E-5 0.17	Lognormal Mean=0.006	None		Probability that a diesel generator is out for maintenance (per demand) (generic). [DGN-MA]
BAT-FDP	1.0E-4 0.006	Lognormal Mean=0.001	None		Probability that a battery fails to deliver power (per demand) (generic). [BAT-LP]
SSW-MAIN	2.4E-6 0.079	Lognormal Mean=0.0017	None		Probability to fail to restore the SSW train after maintenance (HRA). [SSW-XHE-RE-TAB2,4]
RCIC-DEP	0.0041 0.41	Max Entropy Mean=.041	None		Probability to fail to depressurize the RPV via the RCIC steam line after 12 h (HRA). [RA-RCICDEP-12HR]
MC-DPRES	3.3E-7 0.0019	Lognormal Mean=6.9E-5	None		Probability of common cause miscalibration of drywell pressure sensors (HRA). [CCF-MC]
BETA-2DG	0.0039 0.24	Lognormal Mean=3.8E-2	None		Beta factor for common cause failure of two diesel generators (generic).
BETA-BAT	4.1E-4 0.025	Lognormal Mean=0.004	None		Beta factor for common cause failure of three batteries (generic). [BETA-3BAT]

2.17

Table 2.2-5 (continued)

<u>Variable</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
BETA-SSW	0.0014 0.088	Lognormal Mean=0.014	None		Beta factor for common cause failure of three service water system motor driven pumps (generic). [BETA-3SSW]
IE-T2	0.16 10.0	Lognormal Mean=1.6	None		Initiating event: frequency (1/yr) of a transient with loss of PCS.
RA-INJ-1	1.7E-6 0.0099	Lognormal Mean=3.4E-4	None		Probability to fail to manually actuate injection within one hour after an auto-actuation failure (HRA).
FWSACT12	1.5E-4 0.85	Lognormal Mean=0.029	None		Probability to fail to manually align and actuate the FWS after 12 hours (HRA). [RA-FWSACT-12HR]
RA-PCS-1	0.01 1.0	Max Entropy Mean=0.1	None		Probability to fail to recover PCS within one hour (generic).
IE-TC	0.72 45	Lognormal Mean=7.2	None		Initiating event: frequency (1/yr) of a transient (combination of T1, T2, and T3s).
F-RPS	5.0E-8 2.8E-3	Lognormal Mean=9.9E-6	None		Probability of mechanical failure of the reactor protection system. [CM]
F-ADS	0.0125 1.0	Max Entropy Mean=0.125	None		Probability that the operator fails to depressurize the RPV during an ATWS. [ADS-XHE]

Table 2.2-5 (continued)

<u>Variable</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
IE-LOSP	6.3E-5 0.58	LOSP Data Mean=0.1	None		Initiating event: frequency (1/yr) of LOSP. [IE-T1]
AC-ST-FE	0.086 0.35	LOSP Data Mean=0.19	Rank 1	AC-LT-n AC-ST-n	Probability of failure to restore AC power within 1 h. [RA-LOSP-1HR]
AC-LT-FE	3.5E-4 0.10	LOSP Data Mean=0.015	Rank 1	AC-LT-n AC-ST-n	Probability of failure to restore AC power within 12 h. [RA-LOSP-12HR]



### 2.3.1 Overview of the APET

The APET for Grand Gulf considers the progression of the accident from the time core damage is imminent (i.e., 2 ft above the BAF) through the CCI. Although the CCI may progress at ever slower rates for days, the end of this analysis has been arbitrarily set at 24 h. Except in very unusual accidents, almost all of the fission products that are going to be released from the containment will have been released by 24 h after the initiator.

This event tree is based on the Grand Gulf containment arrangement, systems, and procedures. In addition, emphasis was placed on modeling the accident progression for the dominant PDSs presented in the accident frequency analysis described in NUREG/CR-4550, Vol. 6, Part 1.<sup>1</sup>

Table 2.3.1 lists the 125 questions in the Grand Gulf APET. In this APET four time periods are considered. To facilitate understanding of the APET and referencing between questions, each branch or 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:

**E1 Initial** Questions 1 through 22 determine the conditions at the beginning of the accident (i.e., before core damage).

**E2 Early** Questions 23 through 57 address the progression of the accident from the beginning of core damage to just before vessel breach. Questions in this time period consider the status of various systems (coolant injection, ac power, HIS, etc.), the molecular composition of the containment and drywell atmosphere, hydrogen burn phenomena (e.g., ignition and loads), and the containment and drywell structural response to containment loads.

Questions 53 through 57 establish the conditions in the containment and drywell just before vessel breach. These questions determine the amount of water in the reactor cavity, the containment pressure, and whether the drywell atmosphere is combustible.

**I Intermediate** Questions 58 through 98 determine the progression of the accident from immediately before vessel breach to the time of significant CCI. The potential for core damage arrest (i.e., no vessel breach) is addressed in this time period. The majority of these questions address the loads accompanying vessel breach and the containment and drywell structural response to these loads. Hydrogen combustion is considered both at the time of vessel breach and during the time period before significant CCI begins. Hydrogen phenomena associated with the hydrogen produced during CCI is addressed in the next time period.

Questions 96 through 98 establish the conditions in the containment and drywell for the next time period. These questions determine the containment pressure and the amount of water in the reactor pedestal cavity.

L Late

Questions 99 through 125 determine the progression of the accident during the CCI. Containment failures from hydrogen combustion and late overpressure (i.e., from steam and noncondensibles) are addressed in this time period. Similarly, drywell failures from hydrogen combustion and reactor pedestal failure (caused by concrete erosion) are also considered during this time period.

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

This APET does not contain any questions to resolve core-vulnerable sequences. These are PDSs which have failure of containment heat removal only. The continual deposition of decay heat in the containment by operation of the emergency core cooling system (ECCS) in the recirculation mode is predicted to lead to eventual containment failure in many hours or a few days. Containment failure, in turn, may lead to ECCS failure. This is not the case in this study. In the accident frequency analysis it was determined that deformation of injection lines does not occur, and since the systems that take suction from the suppression pool can pump saturated water and, thus, continue to operate, loss of injection does not occur as a result of containment failure. Thus, there are no core-vulnerable sequences.

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, and the general types of calculations performed by the user function are described below. The user function is called to:

1. Determine the containment baseline pressure during the various time periods;
2. Compute the amount of hydrogen released to the containment at the time of vessel breach and during CCI;
3. Compute the concentration and the flammability of the atmosphere in the containment and drywell during the various time periods;
4. Calculate the pressure rise due to hydrogen burns;
5. Determine whether the containment fails and the mode of failure;
6. Determine whether the drywell fails and the mode of failure.

### 2.3.2 Overview of the APET Quantification

This section summarizes the ways in which the questions in the Grand Gulf APET were quantified and discusses these methods briefly. A detailed discussion of each question may be found in Appendix A.1.1.

In addition to the number and name of the question, Table 2.3-1 indicates if the question was sampled, and how the question was evaluated or quantified. In the sampling column, an entry of P indicates that a parameter is sampled from a distribution. 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. The difference 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) of this report.

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.

The number of questions associated with each type of quantification is summarized in Table 2.3-2.

In some cases, a question may have been quantified by more than one source. If this is the case, the entry under Quantification in Table 2.3-1 represents the major contributor to the quantification. For example, Question 70, which addresses the loads accompanying vessel breach, was quantified by the Containment Loads Expert Panel and by the project staff. The majority of cases were quantified by the expert panel. There were several cases, however, which the Expert Panel felt were not important. These cases were quantified internally by the project staff. However, because the majority of the cases were quantified by the Expert Panel, the entry in Table 2.3-1 for Question 70 indicates that this question was quantified by the Containment Loads Expert Panel.

Table 2.3-1  
Questions in the Grand Gulf APET

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
1.	What is the initiating event?		PDS
2.	Is there a Station Blackout?		PDS
3.	Is dc Power not available?		PDS
4.	Do one or more S/RVs fail to reclose?	SF	PDS
5.	Does HPCS fail to inject?		PDS
6.	Does RCIC fail to inject initially?		PDS
7.	Does the CRD hydraulic system fail to inject?		PDS
8.	Does the condensate system fail?		PDS
9.	Do the LPCS and LPCI systems fail?		PDS
10.	Does RHR fail (heat exchangers not available)?		PDS
11.	Does the service water system or cross-tie to LPCI fail?		PDS
12.	Does the fire protection system cross-tie to LPCI fail?		PDS
13.	Are the containment (wetwell) sprays failed?		PDS
14.	What is the status of vessel depressurization?		PDS
15.	When does core damage occur?		PDS
16.	What is the level of pre-existing leakage or isolation failure?		AcFrqAn
17.	What is the level of pre-existing suppression pool bypass?		AcFrqAn
18.	What is the structural capacity of the containment?	P	Struct

Table 2.3-1 (continued)

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
19.	What is the structural capacity of the drywell?	P	Struct
20.	What type of sequence is this (summary of plant damage)?		Summary
21.	Do the operators turn on the HIS before core damage (CD)?		AcFrqAn
22.	Is the containment not vented before CD?		AcFrqAn
23.	Does (do) any SRV tailpipe vacuum breaker(s) stick wide open?	SF	Internal
24.	Does ac power remain lost during core degradation?	SF	ROSP
25.	Is dc power available during core degradation?		AcFrqAn
26.	What is the RPV pressure during core degradation?		AcFrqAn
27.	What is the status of the HIS before vessel breach (VB)?		AcFrqAn
28.	Is RPV injection restored during core degradation?		AcFrqAn
29.	Is the core in a critical configuration following injection recovery?		Internal
30.	What is the status of containment sprays?		Internal
31.	What amount of oxygen is in the wetwell during CD?		Internal
32.	What amount of oxygen is in the drywell during CD?		Internal
33.	What amount of steam is present in the containment at core damage?		Internal

Table 2.3-1 (continued)

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
34.	What amount of steam is present in the drywell at core damage?		Internal
35.	Total amount of H <sub>2</sub> released in-vessel during CD?	P	In-Vessel
36.	What is the level of in-vessel zirconium oxidation?		Summary
37.	What is the containment pressure during CD?		UFUN-Int
38.	What is the level of containment leakage due to slow pressurization before VB?	ZO	UFUN-Int
39.	What is the maximum hydrogen concentration in the wetwell before VB?		UFUN-Int
40.	To what level is the wetwell inert during CD?		UFUN-Int
41.	Do diffusion flames consume the hydrogen released before VB?	SF	Internal
42.	What is the maximum hydrogen concentration in the drywell before VB?		UFUN-Int
43.	Do deflagrations occur in the wetwell prior to VB?	SF	Loads
44.	Is there a detonation in the wetwell prior to VB?	SF	Loads
45.	What is the level of containment impulse load before vessel breach?		Summary
46.	With what efficiency is H <sub>2</sub> burned prior to VB?	P	Loads
47.	What is the peak pressure in containment from a hydrogen burn?		UFUN-Int

Table 2.3-1 (continued)

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
48.	What is the level of drywell leakage induced by an early detonation in containment?	ZO	UFUN-Str
49.	What is the level of containment leakage induced by an early detonation?	ZO	UFUN-Str
50.	What is the level of containment leakage before vessel breach?	ZO	UFUN-Int
51.	What is the level of drywell leakage induced by containment pressurization?	ZO	UFUN-Int
52.	What is the level of suppression pool bypass following early combustion events?	ZO	Internal
53.	Has the upper pool dumped?		Summary
54.	Is there water in the reactor cavity?	ZO	Internal
55.	What is the containment pressure before VB?		UFUN-Int
56.	To what level is the DW steam inert at VB?		UFUN-Int
57.	Is there sufficient H <sub>2</sub> for combustion/detonation in the DW before VB?		UFUN-Int
58.	Does an Alpha Mode Event fail both the vessel and the containment?	SF	Note 1
59.	What fraction of the core participates in core slump?		Internal
60.	Is there a large in-vessel steam explosion?		Internal
61.	What fraction of the core debris would be mobile at vessel breach?	ZO	Internal

Table 2.3-1 (continued)

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
62.	Does a large in-vessel steam explosion fail the vessel?	Z0	Internal
63.	What is the mode of vessel breach?	Z0	Internal
64.	Does high pressure melt ejection occur?	Z0	Internal
65.	Does a detonation occur in the DW at VB?		Summary
66.	Does a deflagration occur in the DW at VB?		Summary
67.	Does a large ex-vessel steam explosion occur?		Internal
68.	What amount of H <sub>2</sub> is released at vessel breach?	P	In-Vessel
69.	How much hydrogen is released at vessel breach?		UFUN-Int
70.	What is the peak drywell/wetwell pressure difference resulting from VB?	P	Loads
71.	What is the peak pedestal pressure at VB?	P	Loads
72.	Is the impulse loading to the drywell at VB sufficient to cause failure?	Z0	UFUN-Str
73.	Is drywell pressurization at VB sufficient to cause failure?	Z0	UFUN-Int
74.	Does the RPV pedestal fail due to pressurization at vessel breach?	P	Internal
75.	Does the RPV pedestal fail from an ex-vessel steam explosion (impulse loading)?	SF	Internal
76.	Does the RPV pedestal failure induce drywell failure?	Z0	Struct



Table 2.3-1 (continued)

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
77.	What is the pressure in the containment at VB prior to a hydrogen burn?	P	Internal
78.	What is the concentration of hydrogen in containment immediately after VB?		UFUN-Int
79.	Is ac power not recovered following vessel breach?	SF	ROSP
80.	Is dc power available following vessel breach?		AcFrqAn
81.	What is the status of containment sprays following vessel breach?	ZO	Internal
82.	To what level is the wetwell inert after VB?		UFUN-Int
83.	Is there sufficient oxygen in the containment to support combustion		UFUN-Int
84.	Does ignition occur in the containment at VB?	SF	Loads
85.	Does ignition occur in the containment following vessel breach?	SF	Internal
86.	Is there a detonation in the wetwell following VB?	SF	Internal
87.	What is the level of containment impulse load following vessel breach?		Summary
88.	With what efficiency is H <sub>2</sub> burned following VB?	P	Internal
89.	What would be the peak pressure in containment from a hydrogen burn at VB?		UFUN-Int
90.	What is the level of containment pressurization at vessel breach?		UFUN-Int

Table 2.3-1 (continued)

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
91.	What is the level of drywell leakage induced by a detonation in containment at VB?	ZO	UFUN-Str
92.	What is the level of containment leakage induced by a detonation at VB?	ZO	UFUN-Str
93.	What is the level of containment leakage following vessel breach?	ZO	UFUN-Int
94.	What is the level of drywell leakage induced by containment pressurization?	ZO	UFUN-Int
95.	What is the level of suppression pool bypass following VB?	ZO	Internal
96.	What is the containment pressure after VB?		UFUN-Int
97.	Is water not supplied to the debris late?	ZO	Internal
98.	Is there water in the reactor cavity after VB?		Internal
99.	What is the nature of the CCI?		Internal
100.	What fraction of core not participating in HPME participates in CCI?	P	Internal
101.	How much H <sub>2</sub> (& equivalent CO) and CO <sub>2</sub> are produced during CCI?		UFUN-Int
102.	What is the level of zirconium oxidation in the pedestal before CCI?		Summary
103.	Is the containment not vented following VB?		Internal
104.	Is ac power not recovered late in the accident?	SF	ROSP
105.	Is dc power available late in the accident?		AcFrqAn

Table 2.3-1 (continued)

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
106.	What is the late status of containment sprays?	ZO	Internal
107.	What is the late concentration of combustible gases in the containment?		UFUN-Int
108.	To what level is the wetwell inert after VB?		UFUN-Int
109.	Is there sufficient oxygen in the containment to support late combustion?		UFUN-Int
110.	Does ignition occur late in the containment?	SF	Internal
111.	Is there a detonation in the wetwell following VB?		Internal
112.	What is the late level of containment impulse load?		Internal
113.	What is the late gas combustion efficiency?		Internal
114.	What is be the peak pressure in containment from a late hydrogen burn?		UFUN-Int
115.	What is the level of drywell leakage induced by a late detonation in containment?	ZO	UFUN-Str
116.	What is the level of containment leakage induced by a late detonation?	ZO	UFUN-Str
117.	What is the level of containment leakage induced by late combustion events?	ZO	UFUN-Int
118.	What is the level of drywell leakage induced by late combustion?	ZO	UFUN-Int
119.	Is the containment not vented late?		Internal

Table 2.3-1 (Continued)

<u>Question Number</u>	<u>Question</u>	<u>Sampling</u>	<u>Quantification</u>
120.	How much concrete must be eroded to cause pedestal failure?	P	Struct
121.	At what time does pedestal failure occur?	P	MCCI
122.	What is the level of late suppression pool bypass?	ZO	Struct
123.	What is the late containment pressure due to non-condensibles or steam?	P	Internal
124.	Does containment failure occur late due to non-condensibles or steam?	ZO	UFUN-Int
125.	What is the long-term level of containment leakage?		Summary

Notes to Table 2.3-1

Note 1. The Alpha mode of vessel and containment failure was previously considered by the Steam Explosion Review Group. The distribution used in this analysis is based on information contained in the report generated by this group. See the discussion of Question 58 in Appendix A.1.1.

Key to Abbreviations and Initialisms in Table 2.3-1

AcFrqAn	The quantification was performed by the Accident Frequency Analysis project staff.
Internal	The quantification 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 Expert Panel on In-Vessel Issues.
Loads	This question was quantified by sampling an aggregate distribution provided by the Expert Panel on Containment Loads Issues.
MCCI	This question was quantified by sampling an aggregate distribution provided by the Expert Panel on Molten Core/Containment Interaction Issues.

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

P	A value, sampled from a distribution, is assigned to a parameter.
PDS	The quantification follows directly the definition of the PDS.
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 from a aggregate distribution provided by the Expert Panel on Structural Issues.
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-Str	This question is quantified by the execution of a module in the User Function subroutine, using distributions from the Structural Expert Panel
UFUN-Int	This question is quantified by the execution of a module in the User Function subroutine using models and data generated by the project staff.
ZO	Zero-One sampling: the branch probabilities are either 0.0 or 1.0.

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Table 2.3-2  
Grand Gulf APET Quantification Summary

<u>Type of Quantification</u>	<u>Number of Questions</u>	<u>Comments</u>
PDS	15	Determined by the PDS.
AcFrqAn	10	Determined by the Accident Frequency Analysis.
Internal	37	Quantified internally in this analysis.
Summary	9	The branch taken at this question follows directly from the branches taken at previous questions.
ROSP	3	This question was quantified by sampling a distribution derived from the offsite power recovery data for the plant.
UFUN-Str	7	Calculated in the User Function using distributions from the Structural Expert Panel.
UFUN-Int	29	Calculated in the User Function using models and data generated by the project staff.
In-Vessel	2	Distributions from the In-Vessel Expert Panel.
Loads	6	Distributions from the Containment Loads Expert Panel
MCCI	1	Distributions from the Molten Core-Containment Interaction Panel.
Struct	5	Distributions from the Structural Expert Panel.
Other Expert	1	See Note 1, Table 2.3-1.

### 2.3.3 Variables Sampled for the Accident Progression Analysis

About 186 variables were sampled for the accident progression analysis. That is, every time the APET was evaluated by EVNTRE, the original values of about 186 variables were replaced with values selected for the particular observation under consideration. These values were selected by the Latin Hypercube Sampling (LHS) program from distributions that were defined before the APET was evaluated. Many of these distributions were determined by expert panels. Table 2.3-3 lists the variables in the APET which were sampled for the accident progression analysis. Some of them are

branch fractions; the others are parameter values for use in calculations performed while the APET is being evaluated.

In Table 2.3-3, the first column gives the variable abbreviation or identifier, and the question (and case if appropriate) in which the variable is used. Where several variables are correlated, they are treated as one variable in the sensitivity analysis (see section 5.1.4), but are different variables as far as the accident progression analysis and sampling process are concerned.

The second column gives the range of the distribution for the variable. The minimum and maximum values of the distribution are listed in this column. 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 2.3-3 indicates the type of distribution used. For uniform distributions from 0.0 to 1.0, the mean is obvious and so is not listed. Otherwise, the mean 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 by some method other than the formal expert elicitation process. (None of the distributions obtained by aggregating the conclusions of experts can be described succinctly in words. Plots of the aggregate expert distributions are contained in Volume 2 of this report. A listing of the input to the LHS program that contains many of these 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 2.3-3 show whether the variable is correlated with any other variable. "Rank 1" indicates a rank correlation of 1.0. An "n" is used to indicate any integer. In the entry for H2INVES1, H2INVESn in the "Correl. with" column indicates that H2INVES1 is correlated with H2INVES2, H2INVES3, . . . ., and H2INVES6. For further information on each of the variables listed in the table, see the detailed discussion of the indicated APET question in Appendix A.

Table 2.3-3  
Variables Sampled in the Accident Progression Analysis

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
PCFail Q18 C1 Q123 C3	195 755	Internal Mean = 383	Rank 1	IPDWF EPDWF	Containment failure pressure (kPa).
IPDWF Q19 C1	260 963	Internal Mean = 588	Rank 1	PCFail EPDWF	Drywell failure pressure (kPa) when the pressure loading is inside the drywell.
EPDWF Q19 C1	260 963	Internal Mean = 588	Rank 1	PCFail IPDWF	Drywell failure pressure (kPa) when the pressure loading is outside the drywell (i.e., in the wetwell).
CFRan Q18 C1	0.0 1.0	Uniform	Rank 1	DWFRan	Random number used to determine the mode of containment failure (Quasi-static loads).
DWFRan Q19 C1	0.0 1.0	Uniform	Rank 1	CFRan	Random number used to determine the mode of drywell failure (Quasi-static loads).
IMPCF Q18 C1	0.0 102.5	Experts Mean = 19.5	Rank 1	IMPDWF	The failure impulse (kPa-s) of the containment.
IMPDWF Q19 C1	2.5 125	Experts Mean = 33.	Rank 1	IMPCF	The failure impulse (kPa-s) of the drywell.
IMRanC Q18 C1	0.0 1.0	Uniform	Rank 1	IMRanD	Random number used to determine the mode of containment failure (Impulse loads).
IMRanD Q19 C1	0.0 1.0	Uniform	Rank 1	IMRanC	Random number used to determine the mode of drywell failure (Impulse loads).

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Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
SRVBkr1 Q23 C2	0.01 0.50	Uniform	None		The failure probability of a SRV tailpipe vacuum breaker (RPV at high pressure).
SRVBkr2 Q23 C4	0.01 0.10	Uniform	None		The failure probability of a SRV tailpipe vacuum breaker (either ATWS or RPV at low pressure).
H2INVES1 Q35 C1	0.0 955	Experts Mean = 222	Rank 1	H2INVESn	The amount of hydrogen (kg-moles) produced in-vessel during an ATWS in which coolant injection is restored to the RPV.
H2INVES2 Q35 C2	0.0 1267	Experts Mean = 461	Rank 1	H2INVESn	The amount of hydrogen (kg-moles) produced in-vessel during an ATWS in which coolant injection is not restored to the RPV.
H2INVES3 Q35 C3	0.0 1042	Experts Mean = 333	Rank 1	H2INVESn	The amount of H <sub>2</sub> (kg-moles) produced in-vessel. The RPV is at high pressure and coolant is restored to the RPV. The PDS is not an ATWS.
H2INVES4 Q35 C4	0.0 903	Experts Mean = 283	Rank 1	H2INVESn	The amount of H <sub>2</sub> (kg-moles) produced in-vessel. The RPV is at low pressure and coolant is restored to the RPV. The PDS is not an ATWS.
H2INVES5 Q35 C5	36.4 1251	Experts Mean = 450	Rank 1	H2INVESn	The amount of H <sub>2</sub> (kg-moles) produced in-vessel. The RPV is at high pressure and coolant is not restored to the RPV. The PDS is not an ATWS.

Table 2.3-3 (continued)

<u>Var. Ques. &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
H2INVES6 Q35 C6	0.0 1285	Experts Mean = 466	Rank 1	H2INVESn	The amount of H <sub>2</sub> (kg-moles) produced in-vessel. The RPV is at low pressure and coolant is not restored to the RPV. The PDS is not an ATWS.
Dif-nSB Q41 C3	0.5 1.0	Uniform	None		The probability that the H <sub>2</sub> in the burns as a diffusion flame when the PDS is not a station blackout and the HIS is off.
Dif-SB1 Q41 C4	0.0 0.17	Internal Mean = .12	Rank 1	Dif-SBn	The probability that the H <sub>2</sub> in the containment burns as diffusion flame during a SB in which ac power is recovered and the HIS is on.
Dif-SB2 Q41 C5	0.0 0.085	Internal Mean = .06	Rank 1	Dif-SBn	The probability that the H <sub>2</sub> in the containment burns as diffusion flame during a SB in which ac power is recovered and the HIS is off.
DflgBVB1 Q43 C4	0.0 0.72	Experts Mean = .18	Rank 1	DflgBVBn	Probability of hydrogen ignition before VB. The RPV is at high pressure, there is no ac power and H <sub>2</sub> < 4%.
DflgBVB2 Q43 C5	0.0 0.74	Experts Mean = .23	Rank 1	DflgBVBn	Probability of hydrogen ignition before VB. The RPV is at high pressure and there is no ac power and 4% < H <sub>2</sub> < 8%.
DflgBVB3 Q43 C6 Q85 C6	0.0 0.72	Experts Mean = .21	Rank 1	DflgBVBn	Probability of hydrogen ignition. The RPV is at low pressure and there is no ac power and 4% < H <sub>2</sub> < 8%.
DflgBVB4 Q43 C7	0.0 0.75	Experts Mean = .28	Rank 1	DflgBVBn	Probability of hydrogen ignition before VB. The RPV is at high pressure and there is no ac power and 8% < H <sub>2</sub> < 12%.

Table 2.3-3 (continued)

Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
DflgBVB5 Q43 C8 Q85 C5	0.0 0.75	Experts Mean = .28	Rank 1	DflgBVBn	Probability of hydrogen ignition. The RPV is at low pressure and there is no ac power and $8\% < H_2 < 12\%$ .
DflgBVB6 Q43 C9	0.0 0.75	Experts Mean = .39	Rank 1	DflgBVBn	Probability of hydrogen ignition before VB. The RPV is at high pressure, there is no ac power and $12\% < H_2 < 16\%$ .
DflgBVB7 Q43 C10 Q85 C4	0.0 0.75	Experts Mean = .38	Rank 1	DflgBVBn	Probability of hydrogen ignition. The RPV is at low pressure and there is no ac power and $12\% < H_2 < 16\%$ .
DflgBVB8 Q43 C11	0.0 0.75	Experts Mean = .50	Rank 1	DflgBVBn	Probability of hydrogen ignition before VB. The RPV is at high pressure and there is no ac power and $H_2 > 16\%$ .
DflgBVB9 Q43 C12 Q85 C3	0.0 0.75	Experts Mean = .49	Rank 1	DflgBVBn	Probability of hydrogen ignition. The RPV is at low pressure and there is no ac power and $H_2 > 16\%$ .
DtonBVB1 Q44 C2 Q86 C4	0.0 .66	Experts Mean = .22	Rank 1	DtonBVBn	Probability that the hydrogen detonates given that the $H_2$ has ignited, the steam concentration is high, and $12\% < H_2 < 16\%$ .
DtonBVB2 Q44 C4 Q86 C6,8	0.0 .75	Experts Mean = .25	Rank 1	DtonBVBn	Probability that the hydrogen detonates given that the $H_2$ has ignited, the steam concentration is high, and $H_2 > 16\%$ .
DtonBVB3 Q44 C5 Q86 C7	0.0 .65	Experts Mean = .26	Rank 1	DtonBVBn	Probability that the hydrogen detonates given that the $H_2$ has ignited, the steam concentration is low, and $16\% < H_2 < 20\%$ .

Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
DtonBVB4 Q44 C7 Q86 C9	0.0 .70	Experts Mean = .45	Rank 1	DtonBVBn	Probability that the hydrogen detonates given that the H <sub>2</sub> has ignited, the steam concentration is low, and H <sub>2</sub> > 20%.
ImpLoad1 Q44 C2 Q86 C4,6,8	2.8 12.3	Experts Mean = 5.8	Rank 1	ImpLoadn	The impulse load (kPa-s) on the drywell from a detonation when 12% < H <sub>2</sub> < 16%.
ImpLoad2 Q44 C5 Q86 C2,3,7,9	0.0 63.0	Experts Mean = 12.4	Rank 1	ImpLoadn	The impulse load (kPa-s) on the drywell from a detonation when H <sub>2</sub> > 16%.
EffBrnP1 Q46 C2	0.0 0.24	Experts Mean = .079	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is high and H <sub>2</sub> < 4%.
EffBrnP2 Q46 C4 Q88 C1	0.0 0.64	Experts Mean = .28	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is high and 4% < H <sub>2</sub> < 8%.
EffBrnP3 Q46 C5 Q88 C2	0.0 0.65	Experts Mean = .28	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is low and 4% < H <sub>2</sub> < 8%.
EffBrnP4 Q46 C6 Q88 C3	0.0 0.93	Experts Mean = .46	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is high and 8% < H <sub>2</sub> < 12%.
EffBrnP5 Q46 C7 Q88 C4	0.0 0.83	Experts Mean = .57	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is low and 8% < H <sub>2</sub> < 12%.

Table 2.3-3 (continued)

Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
EffBrnP6 Q46 C8 Q88 C5	0.0 0.84	Experts Mean = .48	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is high and $12\% < H_2 < 16\%$ .
EffBrnP7 Q46 C9 Q88 C6	0.0 0.94	Experts Mean = .73	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is low and $12\% < H_2 < 16\%$ .
2.40 EffBrnP8 Q46 C10 Q88 C7	0.0 0.78	Experts Mean = .49	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is high and $H_2 > 16\%$ .
EffBrnP9 Q46 C11 Q88 C8	0.0 1.0	Experts Mean = .75	Rank 1	EffBrnPn	The effective burn efficiency in the wetwell given that the steam concentration is low and $H_2 > 16\%$ .
BrnCmpl1 Q46 C2 Q88 C1,2	0.0 0.69	Experts Mean = .27	Rank 1	BrnCmpln	The actual burn completeness in the wetwell given that $H_2 < 8\%$ .
BrnCmpl2 Q46 C6 Q88 C3,4	0.37 0.89	Experts Mean = .74	Rank 1	BrnCmpln	The actual burn completeness in the wetwell given that $8\% < H_2 < 12\%$ .
BrnCmpl3 Q46 C8 Q88 C5,6	0.53 1.0	Experts Mean = .88	Rank 1	BrnCmpln	The actual burn completeness in the wetwell given that $12\% < H_2 < 16\%$ .
BrnCmpl4 Q46 C10 Q88 C7,8	0.59 1.0	Experts Mean = .93	Rank 1	BrnCmpln	The actual burn completeness in the wetwell given that $H_2 > 16\%$ .

Table 2.3-3 (continued)

Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
DWVacBkr Q52 C2 Q95 C2,3 Q122 C3,5	Zero One	Fail 0.05	None		The probability that the drywell vacuum breaker will fail to reclose after a hydrogen burn in the the wetwell (ac power must be available).
DWFldDif Q54 C4	Zero One	Fld 0.45 Wet 0.45 Dry 0.10	None		The probability that a hydrogen burn (diffusion flame) pushes suppression pool water in the drywell
DWFldH <sub>2</sub> 1 Q54 C5	Zero One	Fld 0.50 Wet 0.50	Rank 1	DWFldH <sub>2</sub> n	The probability that the accumulation of H <sub>2</sub> in the wetwell pushes pool water into the drywell given that the upper pool has dumped.
DWFldH <sub>2</sub> 2 Q54 C8	Zero One	Wet 0.50 Dry 0.50	Rank 1	DWFldH <sub>2</sub> n	The probability that the accumulation of H <sub>2</sub> in the wetwell pushes pool water into the drywell given that the upper pool has not dumped.
ALPHA1 Q58 C2	0.0 1.0	Experts Mean = .01	Rank 1	ALPHAn	Probability that an Alpha mode event occurs, given that the RPV is at low pressure.
ALPHA2 Q58 C1	0.0 0.1	Experts Mean = .001	Rank 1	ALPHAn	Probability that an Alpha mode event occurs, given that the RPV is at high pressure.
LiqVB1 Q61 C1	Zero One	HiLiq 0.025 LoLiq 0.975	None		Probability that there is a large amount of molten core debris (HiLiq) at VB given that coolant injection is being supplied to the RPV.

Table 2.3-3 (continued)

Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
LiqVB2 Q61 C2	Zero One	HiLiq 0.10 LoLiq 0.90	None		Probability that there is a large amount of molten core debris (HiLiq) at VB given that coolant is not being supplied to the RPV.
F-RPV-SE Q62 C2	Zero One	BtHd 0.2 LgBrch 0.2 SmBrch 0.3 nFail 0.3	None		The probability that an in-vessel steam explosion will fail the RPV
2.42 F-RPV1 Q63 C5	Zero One	BtHd 0.124 LgBrch 0.005 SmBrch 0.371 nFail 0.500	Rank 1	F-RPVn	The probability that the RPV will fail given that a large amount of the core is molten and coolant is being injected into the RPV.
F-RPV2 Q63 C6,C7 C9,C10	Zero One	BtHd 0.249 LgBrch 0.005 SmBrch 0.746 nFail 0.000	Rank 1	F-RPVn	The probability that the RPV will fail given that there is no coolant injection.
F-RPV3 Q63 C8	Zero One	BtHd 0.062 LgBrch 0.005 SmBrch 0.188 nFail 0.745	Rank 1	F-RPVn	The probability that the RPV will fail given that a small amount of the core is molten and coolant is being injected into the RPV.
HPME Q64 C2	Zero One	HPME 0.8	None		The probability of an HPME event given that the RPV fails at high pressure.
H <sub>2</sub> AVB1 Q68 C2	0.0 781	Experts Mean = 61	Rank 1	H <sub>2</sub> AVBn H <sub>2</sub> INVEsn	The amount of H <sub>2</sub> (kg-moles) produced at VB during an ATWS in which coolant injection is restored to the RPV.

Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
H <sub>2</sub> AVB2 Q68 C3	0.0 642	Experts Mean = 89	Rank 1	H <sub>2</sub> AVBn H <sub>2</sub> INVESn	The amount of hydrogen (kg-moles) produced at vessel breach during an ATWS in which coolant injection is not restored to the RPV.
H <sub>2</sub> AVB3 Q68 C4	0.0 260	Experts Mean = 53	Rank 1	H <sub>2</sub> AVBn H <sub>2</sub> INVESn	The amount of H <sub>2</sub> (kg-moles) produced at VB. The PDS is not an ATWS, the RPV is pressurized, and coolant is restored to the RPV during CD.
H <sub>2</sub> AVB4 Q68 C5	0.0 156	Experts Mean = 27	Rank 1	H <sub>2</sub> AVBn H <sub>2</sub> INVESn	The amount of H <sub>2</sub> (kg-moles) produced at VB. The RPV is at low pressure and coolant is restored to the RPV. The PDS is not an ATWS.
H <sub>2</sub> AVB5 Q68 C6	0.0 625	Experts Mean = 234	Rank 1	H <sub>2</sub> AVBn H <sub>2</sub> INVESn	The amount of H <sub>2</sub> (kg-moles) produced at VB. The RPV is at high pressure and coolant is not restored to the RPV. The PDS is not an ATWS.
H <sub>2</sub> AVB6 Q68 C7	0.0 417	Experts Mean = 62	Rank 1	H <sub>2</sub> AVBn H <sub>2</sub> INVESn	The amount of H <sub>2</sub> (kg-moles) produced at VB. The PDS is a not an ATWS, the RPV is pressurized, and coolant was restored to the RPV during CD.
DWFPVB1 Q70 C2	0.0 2000	Experts Mean = 434	Rank 1	DWFPVB2,5,6 CP-VB1	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at high pressure into a wet cavity (Expert



Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
DWPVB2 Q70 C3	0.0 2000	Experts Mean = 332	Rank 1	DWPVB1,5,6 CP-VB1	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-hC).
DWPVB3 Q70 C4	33. 950	Experts Mean = 392	Rank 1	DWPVB4,7,8 CP-VB2	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-HC).
DWPVB4 Q70 C5	20. 531	Experts Mean = 242	Rank 1	DWPVB3,7,8 CP-VB2	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-hC).
DWPVB5 Q70 C6	0.0 2000	Experts Mean = 425	Rank 1	DWPVB1,2,6 CP-VB1	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-Hc).
DWPVB6 Q70 C7	0.0 2000	Experts Mean = 311	Rank 1	DWPVB1,2,5 CP-VB1	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-hc).
DWPVB7 Q70 C8	33. 850	Experts Mean = 336	Rank 1	DWPVB4,5,8 CP-VB2	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-Hc).

Table 2.3-3 (continued)

Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
DWPVB8 Q70 C9	20. 531	Experts Mean = 222	Rank 1	DWPVB4, 5, 7 CP-VB2	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-hc).
DWPVB9 Q70 C10	0.0 2000	Experts Mean = 295	Rank 1	DWPVB9-12 CP-VB3	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-HC).
2.45 DWPVB10 Q70 C11	0.0 2000	Experts Mean = 242	Rank 1	DWPVB9-12 CP-VB3	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-hC).
DWPVB11 Q70 C12	0.0 2000	Experts Mean = 290	Rank 1	DWPVB9-12 CP-VB3	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-Hc).
DWPVB12 Q70 C13	0.0 2000	Experts Mean = 239	Rank 1	DWPVB9-12 CP-VB3	The peak drywell/wetwell pressure differential (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-hc).
PedVB1 Q71 C2	550 8370	Experts Mean = 3580	Rank 1	PedVB2, 5, 6	The peak pedestal cavity pressure (kPa) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-HC).
PedVB2 Q71 C3	468 8370	Experts Mean = 2780	Rank 1	PedVB1, 5, 6	The peak pedestal cavity pressure (kPa) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-hC).

Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
PedVB3 Q71 C4	385 6000	Experts Mean = 3080	Rank 1	PedVB4,7,8	The peak pedestal cavity pressure (kPa) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-HC).
PedVB4 Q71 C5	0 4980	Experts Mean = 1720	Rank 1	PedVB5,7,8	The peak pedestal cavity pressure (kPa) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-hC).
PedVB5 Q71 C6	440 6700	Experts Mean = 3250	Rank 1	PedVB1,2,6	The peak pedestal cavity pressure (kPa) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-Hc).
PedVB6 Q71 C7	374 5690	Experts Mean = 2170	Rank 1	PedVB1,2,5	The peak pedestal cavity pressure (kPa) at VB. RPV fails at high pressure into a wet cavity (Expert Case 1-hc).
PedVB7 Q71 C8	308 6000	Experts Mean = 2850	Rank 1	PedVB5,6,8	The peak pedestal cavity pressure (kPa) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-Hc).
PedVB8 Q71 C9	262 3990	Experts Mean = 1430	Rank 1	PedVB5,6,7	The peak pedestal cavity pressure (kPa) at VB. RPV fails at high pressure into a dry cavity (Expert Case 2-hc).
PedVB9 Q71 C10 Q71 C12	200 4200	Experts Mean = 1120	Rank 1	PedVBn n=11-17	The peak pedestal cavity pressure (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Cases 3-OHC and 3-oHC).
PedVB10 Q71 C11	138 2400	Experts Mean = 734	Rank 1	PedVBn n=11-17	The peak pedestal cavity pressure (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-OhC).

Table 2.3-3 (continued)

Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
PedVB11 Q71 C13	69 2400	Experts Mean = 557	Rank 1	PedVBn n=11-17	The peak pedestal cavity pressure (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-ohC).
PedVB12 Q71 C14	100 4200	Experts Mean = 1000	Rank 1	PedVBn n=11-17	The peak pedestal cavity pressure (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-OHc).
PedVB13 Q71 C15 Q71 C16	100 2100	Experts Mean = 606	Rank 1	PedVBn n=11-17	The peak pedestal cavity pressure (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Cases 3-Ohc and 3-ohC).
PedVB14 Q71 C17	69 1600	Experts Mean = 436	Rank 1	PedVBn n=11-17	The peak pedestal cavity pressure (kPa) at VB. RPV fails at low pressure into a wet cavity (Expert Case 3-ohc).
PedFail Q74 C1	900 1700	Uniform	None		Pedestal failure pressure (kPa)
PedExSE Q75 C1	0.0 1.0	Uniform	None		The probability that the reactor pedestal fails from an ExSE given that an ExSE occurs at VB.
CP-VB1 Q77 C2	3.35 227	Internal Mean = 50	Rank 1	DWFPVBn n=1,2,5,6	Wetwell pressure rise (kPa) at VB prior to a burn. RPV fails at high pressure into a wet cavity; the suppression pool is bypassed at VB.
CP-VB2 Q77 C3	4.36 92.5	Internal Mean = 41	Rank 1	DWFPVBn n=3,4,7,8	Wetwell pressure rise (kPa) at VB prior to a burn. RPV fails at high pressure into a dry cavity; the suppression pool is bypassed at VB.

Table 2.3-3 (continued)

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Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
CP-VB3 Q77 C4	2.36 227	Internal Mean = 35	Rank 1	DWPVBn n=10-13	Wetwell pressure rise (kPa) at VB prior to a burn. RPV fails at low pressure into a wet cavity; the suppression pool is bypassed at VB.
CP-VB4 Q77 C6	0.0 113.5	Uniform	None		Wetwell pressure rise (kPa) at VB prior to a burn. Either HPME or ExSE occurs at Q77 C7 VB; the suppression pool is not bypassed at VB.
CSFail1 Q81 C2	Zero One	Fail 0.50 Recvy 0.50	Rank 1	CSFail2	Probability that the energetic events that ruptured the containment at VB also fail the CS (ac power is not available).
CSFail2 Q81 C4	Zero One	Fail 0.50 Oper. 0.50	Rank 1	CSFail1	Probability that the energetic events that ruptured the containment at VB also fail the CS (CS were operating before VB).
CSFail3 Q81 C6 Q106 C6	Zero One	Fail 0.50 Avail 0.45 Oper. 0.05	None		Probability that the energetic events that ruptured the containment at VB also fail the CS (ac power is recovered following VB).
IgnAVB1 Q84 C3	0.1 0.92	Expert Mean = .63	Rank 1	IgnAVBn	The H <sub>2</sub> ignition probability at VB given that the RPV fails at high pressure or there is an ExSE and H <sub>2</sub> > 16%.
IgnAVB2 Q84 C4	0.04 0.87	Expert Mean = .56	Rank 1	IgnAVBn	The H <sub>2</sub> ignition probability at VB given that the RPV fails at high pressure or there is an ExSE and 12% < H <sub>2</sub> < 16%.

Table 2.3-3 (continued)

Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
IgnAVB3 Q84 C5	0.02 0.67	Expert Mean = .43	Rank 1	IgnAVBn	The H <sub>2</sub> ignition probability at VB given that the RPV fails at high pressure or there is an ExSE and 8% < H <sub>2</sub> < 12%.
IgnAVB4 Q84 C6	0.0 0.6	Expert Mean = .29	Rank 1	IgnAVBn	The H <sub>2</sub> ignition probability at VB given that the RPV fails at high pressure or there is an ExSE and 4% < H <sub>2</sub> < 8%.
IgnAVB5 Q84 C7	0.0 0.035	Expert Mean = -0.005	Rank 1	IgnAVBn	The H <sub>2</sub> ignition probability at VB given that the RPV fails at high pressure or there is an ExSE and H <sub>2</sub> < 4%.
DW-Ped-F Q76 C2 Q122 C2,5	Zero One	Fail 0.175	None		The probability that pedestal failure induces drywell failure given that the pedestal fails.
LDBWat1 Q97 C2	Zero One	noWat 0.50 LgWat 0.25 SmWat 0.25	None		The probability that a coolant injection system supplies water to the debris after VB given that ac power is not available.
LDBWat2 Q97 C4	Zero One	noWat 0.33 LgWat 0.33 SmWat 0.33	None		The probability that a coolant injection system supplies water to the debris after VB given that an injection system was working before VB.
LDBWat3 Q97 C5	Zero One	noWat 0.50 LgWat 0.25 SmWat 0.25	None		The probability that a coolant injection system supplies water to the debris after VB given that there was no injection before VB.

Table 2.3-3 (continued)

Variable Question & Case	Range	Distribution	Correlation	Correlated With	Description
CD-CCI1 Q100 C2	0.6 1.0	Uniform	None		The fraction of core debris that participates in CCI; given that a large amount of core debris participates in an ExSE.
CD-CCI2 Q100 C3	0.9 1.0	Uniform	None		The fraction of core debris that participates in CCI; given that a small amount of core debris participates in an ExSE.
L-CIgn1 Q110 C4	0.0 0.75	Expert Mean = 0.51	Rank 1	L-CIgnn	The H <sub>2</sub> ignition probability late in the accident given that there is no ac power and H <sub>2</sub> > 16%.
L-CIgn2 Q110 C5	0.0 0.75	Expert Mean = 0.42	Rank 1	L-CIgnn	The H <sub>2</sub> ignition probability late in the accident; there is no ac power and 12% < H <sub>2</sub> < 16%.
L-CIgn3 Q110 C6	0.0 0.75	Expert Mean = 0.33	Rank 1	L-CIgnn	The H <sub>2</sub> ignition probability late in the accident; there is no ac power and 8% < H <sub>2</sub> < 12%.
L-CIgn4 Q110 C7	0.0 0.75	Expert Mean = 0.29	Rank 1	L-CIgnn	The H <sub>2</sub> ignition probability late in the accident; there is no ac power and 4% < H <sub>2</sub> < 8%.
ConErPed Q120 C1	0.3 2.1	Expert Mean = 1.1	None		The depth (M) of concrete erosion that will fail the reactor pedestal.
PedFlG1 Q121 C3	0.0 0.53	Expert Mean = 0.19	Rank 1	PedFnCn	The depth of concrete eroded (M) in 1 h during CCI--Expert Group 1.

Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
PedF1G2 Q121 C5	0.0 0.39	Expert Mean = 0.14	Rank 1	PedFnCn	The depth of concrete eroded (M) in 1 h during CCI--Expert Group 2.
PedF1G3 Q121 C4	0.0 0.53	Expert Mean = 0.16	Rank 1	PedFnCn	The depth of concrete eroded (M) in 1 h during CCI--Expert Group 3.
PedF1G4 Q121 C9	0.023 0.43	Expert Mean = 0.20	Rank 1	PedFnCn	The depth of concrete eroded (M) in 1 h during CCI--Expert Group 4.
PedF1G5 Q121 C8	0.023 0.61	Expert Mean = 0.26	Rank 1	PedFnCn	The depth of concrete eroded (M) in 1 h during CCI--Expert Group 5.
PedF1G6 Q121 C6	0.023 0.60	Expert Mean = 0.20	Rank 1	PedFnCn	The depth of concrete eroded (M) in 1 h during CCI--Expert Group 6.
PedF1G7 Q121 C7	0.023 0.61	Expert Mean = 0.26	Rank 1	PedFnCn	The depth of concrete eroded (M) in 1 h during CCI--Expert Group 7.
PedF3G1 Q121 C3	0.0 0.75	Expert Mean = 0.32	Rank 1	PedFnCn	The depth of concrete eroded (M) in 3 h during CCI--Expert Group 1.
PedF3G2 Q121 C5	0.0 0.68	Expert Mean = 0.26	Rank 1	PedFnCn	The depth of concrete eroded (M) in 3 h during CCI--Expert Group 2.
PedF3G3 Q121 C4	0.0 0.75	Expert Mean = 0.29	Rank 1	PedFnCn	The depth of concrete eroded (M) in 3 h during CCI--Expert Group 3.
PedF3G4 Q121 C9	0.075 0.85	Expert Mean = 0.40	Rank 1	PedFnCn	The depth of concrete eroded (M) in 3 h during CCI--Expert Group 4.
PedF3G5 Q121 C8	0.075 0.85	Expert Mean = 0.47	Rank 1	PedFnCn	The depth of concrete eroded (M) in 3 h during CCI--Expert Group 5.



Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
PedF3G6 Q121 C6	0.075 0.85	Expert Mean = 0.41	Rank 1	PedFnCn	The depth of concrete eroded (M) in 3 h during CCI--Expert Group 6.
PedF3G7 Q121 C7	0.075 0.85	Expert Mean = 0.47	Rank 1	PedFnCn	The depth of concrete eroded (M) in 3 h during CCI--Expert Group 7.
PedF6G1 Q121 C3	0.15 1.26	Expert Mean = 0.55	Rank 1	PedFnCn	The depth of concrete eroded (M) in 6 h during CCI--Expert Group 1.
PedF6G2 Q121 C5	0.15 1.26	Expert Mean = 0.49	Rank 1	PedFnCn	The depth of concrete eroded (M) in 6 during CCI--Expert Group 2.
PedF6G3 Q121 C4	0.15 1.26	Expert Mean = 0.52	Rank 1	PedFnCn	The depth of concrete eroded (M) in 6 h during CCI--Expert Group 3.
PedF6G4 Q121 C9	0.23 1.26	Expert Mean = 0.62	Rank 1	PedFnCn	The depth of concrete eroded (M) in 6 h during CCI--Expert Group 4.
PedF6G5 Q121 C8	0.28 1.26	Expert Mean = 0.71	Rank 1	PedFnCn	The depth of concrete eroded (M) in 6 h during CCI--Expert Group 5.
PedF6G6 Q121 C6	0.23 1.26	Expert Mean = 0.66	Rank 1	PedFnCn	The depth of concrete eroded (M) in 6 h during CCI--Expert Group 6.
PedF6G7 Q121 C7	0.28 1.26	Expert Mean = 0.73	Rank 1	PedFnCn	The depth of concrete eroded (M) in 6 h during CCI--Expert Group 7.
PedF10G1 Q121 C3	0.36 1.41	Expert Mean = 0.83	Rank 1	PedFnCn	The depth of concrete eroded (M) in 10 h during CCI--Expert Group 1.
PedF10G2 Q121 C5	0.25 1.41	Expert Mean = 0.74	Rank 1	PedFnCn	The depth of concrete eroded (M) in 10 h during CCI--Expert Group 2.

Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
PedF10G3 Q121 C4	0.25 1.41	Expert Mean = 0.79	Rank 1	PedFnCn	The depth of concrete eroded (M) in 10 h during CCI--Expert Group 3.
PedF10G4 Q121 C9	0.30 1.57	Expert Mean = 0.82	Rank 1	PedFnCn	The depth of concrete eroded (M) in 10 h during CCI--Expert Group 4.
PedF10G5 Q121 C8	0.37 1.57	Expert Mean = 0.92	Rank 1	PedFnCn	The depth of concrete eroded (M) in 10 h during CCI--Expert Group 5.
PedF10G6 Q121 C6	0.29 1.57	Expert Mean = 0.83	Rank 1	PedFnCn	The depth of concrete eroded (M) in 10 h during CCI--Expert Group 6.
PedF10G7 Q121 C7	0.37 1.57	Expert Mean = 0.92	Rank 1	PedFnCn	The depth of concrete eroded (M) in 10 h during CCI--Expert Group 7.
Lt-Pres Q123 C2	250 550	Uniform	None		The pressure (kPa) in the containment late in the accident due to noncondensibles.
AC-LT-CD Q24 C2	0.00 0.31	Internal Mean = .19	Rank 1	AC-ST-n AC-LT-n	The probability that ac power is recovered before VB during a long-term SB given that it was not available at CD.
AC-ST-CD Q24 C3	0.39 0.82	Internal Mean = .62	Rank 1	AC-ST-n AC-LT-n	The probability that ac power is recovered before VB during a short-term SB given that it was not available at CD.
AC-LT-VB Q79 C3	0.00 0.23	Internal Mean = .10	Rank 1	AC-ST-n AC-LT-n	The probability that ac power is recovered after VB during a long-term SB given that it was not available before VB.

Table 2.3-3 (continued)

<u>Variable Question &amp; Case</u>	<u>Range</u>	<u>Distribution</u>	<u>Correlation</u>	<u>Correlated With</u>	<u>Description</u>
AC-ST-VB Q79 C4	0.14 0.58	Internal Mean = .38	Rank 1	AC-ST-n AC-LT-n	The probability that ac power is recovered after VB during a short-term SB given that it was not available before VB.
AC-LT-LT Q104 C3	0.00 0.19	Internal Mean = .09	Rank 1	AC-ST-n AC-LT-n	The probability that ac power is recovered late in the accident during a long-term SB given that it was not available after VB.
AC-ST-LT Q104 C4	0.38 0.87	Internal Mean = .77	Rank 1	AC-ST-n AC-LT-n	The probability that ac power is recovered late in the accident during a short-term SB given that it was not available after VB.

## 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 Accident Progression Bin. This bin describes the evaluation in enough detail that a source term (release of radionuclides) can be calculated for it. The accident progression bins 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 thirteen characteristics or quantities which define a certain feature of the evaluation of the APET. Section 2.4.1 describes the 13 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 reduced set of 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 Grand Gulf utilizes 13 characteristics. That is, there are 13 types of information required to define a path through the APET. A bin is defined by specifying a letter for each of the 13 characteristics, where each letter for each characteristic has a meaning defined below. For a characteristic, the possible states are termed attributes. The Grand Gulf binning characteristics are:

<u>Characteristic</u>	<u>Abbreviation</u>	<u>Description</u>
1	ASeq	Type of Accident Sequence
2	ZrOxid	Fraction of zirconium oxidized in-vessel
3	VB	Vessel Condition at Vessel Breach
4	DCH-SE	Fraction of core participating in direct containment heating or steam explosions
5	SPB-L	The mode and timing of suppression pool bypass
6	CLeak-L	The mode and timing of containment failure
7	Sprays	Period in which containment sprays operate
8	MCCI	Type of CCI

<u>Characteristic</u>	<u>Abbreviation</u>	<u>Description</u>
9	SRVBkr	Occurrence of a stuck open SRV tailpipe vacuum breaker
10	CF-BVB	Events causing containment failure before vessel breach
11	CF-VB	Events causing containment failure at vessel breach
12	DF-BVB	Events causing drywell failure before vessel breach
13	DF-VB	Events causing drywell failure at vessel breach

Most of this information, organized in this manner, is needed by GGSOR to calculate the fission product source terms. Characteristics 10 through 13 are not used by GGSOR, but have been retained because they provide useful information on the types of events that cause containment and drywell failures.

A description of each attribute for each characteristic is presented in Table 2.4-1. The remainder of this section consists of a brief description of each characteristic and an explanation of an example bin.

Characteristic 1 addresses the type of accident sequence that has occurred. Six attributes are defined. The attributes are based on the initiating event and the time at which core damage occurs. The initiating events include station blackout, loss of PCS transient, and ATWS. For each initiating event there are two times at which core damage occurs: short-term and long-term.

Characteristic 2 addresses the fraction of in-vessel zirconium that oxidized before vessel breach. There are two possible values for this characteristic: low and high. The demarcation point between the two ranges is 21%.

Characteristic 3 addresses the RPV pressure before vessel breach and the availability of coolant injection at vessel breach; there are five possibilities, including no vessel breach. The RPV can either be at high or low pressure before vessel breach. High pressure is system pressure (i.e., approximately 1000 psia) and low pressure is less than 200 psia. There are two possibilities for coolant injection: coolant is being injected into the RPV at or immediately after vessel breach, or coolant is not being injected into the RPV at or immediately after vessel breach.

Characteristic 4 addresses the fraction of core participating in DCH or an ex-vessel steam explosion. There are five attributes associated with this characteristic. There are two levels for DCH: low (10% of the core) and high (40% of the core). Similarly, there are two levels for steam explosions: low (5% of the core) and high (20% of the core). The fifth attribute is for the case when there are no DCH events or ex-vessel steam explosions. If a DCH event and a steam explosion both occur during an accident the attribute associated with the DCH event is assigned to this characteristic. The reason for this is that more radionuclides are released during a DCH event than are released from a steam explosion.

Characteristic 5 addresses the amount of suppression pool bypass and time that pool bypass occurs. There are eight attributes. The bypass can be either nominal (no change), small (leak), or large (rupture). Three time periods are addressed: early, intermediate, and late.

Characteristic 6 addresses the size of hole that results from containment failure and the time period in which the containment failed. There are nine attributes. The hole size can be either small (leak) or large (rupture). Three time periods are addressed: early, intermediate, and late. Containment venting before vessel breach and after vessel breach are also addressed by this characteristic. The last attribute is no containment failure.

Characteristic 7 addresses the period in which containment sprays operate; there are four attributes. Two time periods are addressed: early and intermediate. The intermediate time period does not include sprays operation in the late time period. If the sprays come on during the late time period it is because AC power was previously unavailable and it was restored during the late time period. By this time the majority of the fission products have already been released and the sprays are no longer effective in scrubbing the radionuclides. Sprays that operate only in the late period are grouped with the cases in which the sprays never operate. The four possibilities are: no containment sprays, only early containment sprays, only intermediate containment sprays, and early and intermediate containment sprays.

Characteristic 8 addresses CCI. There are five attributes, including no CCI releases. The first four attributes are concerned with the amount of water in the reactor pedestal cavity. The cavity can be dry, wet, or flooded. If the core debris is initially coolable but there is not a replenishable water supply, then delayed CCI occurs. That is, the core debris is initially cooled until all the water is boiled from the cavity. After the water is boiled away, CCI begins. It is estimated that CCI will be delayed for 3 h for this case.

Characteristic 9 addresses the occurrence of a stuck-open SRV tailpipe vacuum breaker. There are two possibilities: the vacuum breaker is stuck open or the vacuum breaker is closed.

Characteristic 10 addresses the events that can cause containment failure before vessel breach. There are eight attributes. The containment failure

can be caused by a slow pressurization event, a hydrogen deflagration, or a hydrogen detonation. The failure size can be either a leak or a rupture. In addition, venting can cause a breach in the containment boundary. The last attribute is no containment failure before vessel breach.

Characteristic 11 addresses the events that can cause containment failure at vessel breach. There are eight attributes. Containment failure during this time period can be caused by an alpha mode event, a hydrogen deflagration or a hydrogen detonation. The failure size can be either a leak or a rupture (except for the alpha mode event which is always considered as a rupture).

Characteristic 12 addresses the events that can cause drywell failure before vessel breach. There are five attributes. Drywell failure can be caused by hydrogen deflagrations or detonations. The failure size can be either a leak or a rupture. The last attribute is no drywell failure before vessel breach.

Characteristic 13 addresses the events that can cause drywell failure at the time of vessel breach. There are twelve attributes. Drywell failure can be caused by an alpha mode event, a hydrogen deflagration, a hydrogen detonation, or by quasi-static loads accompanying vessel breach. The failure size can be either a leak or a rupture (except for the alpha mode event which is always considered as a rupture). In addition, reactor pedestal failure can in some instances lead to drywell failure (e.g., movement of RPV causes a penetration failure). Pedestal failure during this time period are caused by either loads accompanying vessel breach or by dynamic loads associated with ex-vessel steam explosions. Drywell failures that are induced by pedestal failures are always assumed to be ruptures.

A typical bin might be ABBDAACCBGEEL which, using the information presented in Table 2.4-1, is:

A	Fst-SB	Accident sequence is a short-term station blackout
B	LoZrOx	A small fraction of the zirconium was oxidized in-vessel
B	LoP-nLPI	The RPV was at low pressure before vessel breach and there was no injection to the RPV after vessel breach
D	LoEXSE	A small fraction of the core participated in an ex-vessel steam explosion; there was no DCH event
A	SPBEOLO	There was no suppression pool bypass
A	CE-Lk	The containment failed early from the development of a leak
C	LCS	Spray operation was recovered after vessel breach
C	FLDCCI	CCI proceeded in a flooded reactor cavity
B	cSRVBkr	A SRV tailpipe vacuum breaker did not stick open
G	CL-DEF	A deflagration caused a leak in the containment before vessel breach
E	E-Leak	The containment failed from the development of leak before vessel breach
E	nDFail	The drywell did not fail before vessel breach
L	nIDWF	The drywell did not fail at the time of vessel breach

**Table 2.4-1**  
**Description of Accident Progression Bin Characteristics**

<u>Attribute</u>	<u>Mnemonic</u>	<u>Description</u>
<b>Characteristic 1: Type of Accident Sequence</b>		
A	Fst-SB	Short-term station blackout
B	Slw-SB	Long-term station blackout
C	Fst-T2	Short-term loss of PCS transient
D	Slw-T2	Long-term loss of PCS transient
E	Fst-TC	Short-term ATWS
F	Slw-TC	Long-term ATWS
<b>Characteristic 2: Fraction of Zirconium Oxidized In-Vessel</b>		
A	HiZrOx	High: Greater than 21 % of the In-Vessel zirconium has been oxidized before vessel breach
B	LoZrOx	Low: Less than 21% of the In-Vessel zirconium has been oxidized before vessel breach
<b>Characteristic 3: Vessel Condition at Vessel Breach</b>		
A	HiP-nLPI	RPV is at high pressure and there is no coolant injection after vessel breach
B	LoP-nLPI	RPV is at low pressure and there is no coolant injection after vessel breach
C	HiP-LPI	RPV is at high pressure and coolant is being injected after vessel breach
D	LoP-LPI	RPV is at low pressure and coolant is being injected after vessel breach
E	nVB	There is no vessel breach (i.e., core damage arrest)



Table 2.4-1 (continued)

<u>Attribute</u>	<u>Mnemonic</u>	<u>Description</u>
<b>Characteristic 4: Fraction of Core Participating in DCH or Steam Explosions</b>		
A	HiDCH	40% of the core participates in DCH
B	LoDCH	10% of the core participates in DCH
C	HiEXSE	40% of the core participates in ex-vessel steam explosions
D	LoEXSE	10% of the core participates in ex-vessel steam explosions
E	nDCH-SE	There are no DCH or steam explosions events
<b>Characteristic 5: Mode and Timing of Suppression Pool Bypass</b>		
A	SPBE0L0	Nominal leakage
B	SPBE0I3	Early nominal, intermediate rupture
C	SPBE0L2	Early nominal, late leakage
D	SPBE0L3	Early nominal, late rupture
E	SPBE2L2	Early leakage
F	SPBE2I3	Early leakage, intermediate rupture
G	SPBE2L3	Early leakage, late rupture
H	SPBE3L3	Early rupture
<b>Characteristic 6: Mode and Timing of Containment Failure</b>		
A	CE-Lk	Leak before vessel breach (VB)
B	CE-Rpt	Rupture before vessel breach
C	CE-VENT	Containment vented before vessel breach
D	CVB-LK	Leak at vessel breach
E	CVB-Rpt	Rupture at vessel breach
F	CL-Lk	Late Leak

Table 2.4-1 (continued)

<u>Attribute</u>	<u>Mnemonic</u>	<u>Description</u>
<b>Characteristic 6 (continued)</b>		
G	CL-Rpt	Late Rupture
H	CL-VENT	Containment vented late
I	CnFail	No containment failure
<b>Characteristic 7: Period in which Containment Sprays Operate</b>		
A	noCS	The containment sprays do no operate during the accident
B	ECSnoL	The sprays only operate before vessel breach (VB)
C	LCS	The sprays only operate after vessel breach
D	ECS	The sprays both before vessel breach and after vessel breach
<b>Characteristic 8: Type of Core-Concrete Interactions (CCI)</b>		
A	DryCCI	CCI occurs in a dry reactor pedestal cavity
B	WetCCI	CCI occurs in wet cavity
C	FLDCCI	CCI occurs in a flooded cavity
D	DlyCCI	CCI releases are delayed
E	noCCI	There are no CCI releases
<b>Characteristic 9: Occurrence of a Stuck Open SRV Tailpipe Vacuum Breaker</b>		
A	oSRVBkr	An SRV tailpipe vacuum breaker sticks open during core damage
B	cSRVBkr	There are no stuck open tailpipe vacuum breakers

Table 2.4-1 (continued)

<u>Attribute</u>	<u>Mnemonic</u>	<u>Description</u>
<b>Characteristic 10: Events Causing Containment Failure Before Vessel Breach</b>		
A	E-VENT	The containment was vented before core degradation (considered as a large hole).
B	CR-SP	The containment failed from either an isolation failure or from a slow pressurization event (i.e., steam buildup) which led to the development of a large hole or rupture; nominal hole size is 7 ft <sup>2</sup> .
C	CR-DET	The containment failed from a hydrogen detonation which led to the development of a large hole or rupture; nominal hole size is 7 ft <sup>2</sup> .
D	CR-DEF	The containment failed from a hydrogen deflagration which led to the development of a large hole or rupture; nominal hole size is 7 ft <sup>2</sup> .
E	CL-SP	The containment failed from either an isolation failure or from a slow pressurization event (i.e., steam buildup) which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
F	CL-DET	The containment failed from a hydrogen detonation which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
G	CL-DEF	The containment failed from a hydrogen deflagration which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
H	nCFail	The containment did not fail before vessel breach.

Table 2.4-1 (continued)

<u>Attribute</u>	<u>Mnemonic</u>	<u>Description</u>
<b>Characteristic 11: Events Causing Containment Failure at Vessel Breach</b>		
A	ERupt	The containment failed by the development of a large hole before vessel breach.
B	ALPHA	The containment failed from an alpha mode event which led to the development of a large hole or rupture; nominal hole size is 7 ft <sup>2</sup> .
C	IR-Det	The containment failed from a hydrogen detonation which led to the development of a large hole or rupture; nominal hole size is 7 ft <sup>2</sup> .
D	IR-Def	The containment failed from a hydrogen deflagration which led to the development of a large hole or rupture; nominal hole size is 7 ft <sup>2</sup> .
E	E-Leak	The containment failed by the development of a small hole before vessel breach.
F	IL-Det	The containment failed from a hydrogen detonation which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
G	IL-Def	The containment failed from a hydrogen deflagration which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
H	nICFail	The containment did not fail before or at the time of vessel breach.
<b>Characteristic 12: Events Causing Drywell Failure Before Vessel Breach</b>		
A	DR-Det	The drywell failed from a hydrogen detonation which led to the development of a large hole or rupture; nominal hole size is 1.0 ft <sup>2</sup> .

Table 2.4-1 (continued)

<u>Attribute</u>	<u>Mnemonic</u>	<u>Description</u>
B	DR-Def	The drywell failed from a hydrogen deflagration which led to the development of a large hole or rupture; nominal hole size is 1.0 ft <sup>2</sup> .
C	DL-Det	The drywell failed from a hydrogen detonation which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
D	DL-Def	The drywell failed from a hydrogen deflagration which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
E	nDFail	The drywell did not fail before vessel breach.

**Characteristic 13: Events Causing Drywell Failure at Vessel Breach**

A	EDWRpt	The drywell failed by the development of a large hole before vessel breach.
B	ALPHA	The drywell failed from an alpha mode event which led to the development of a large hole or rupture; nominal hole size is 1.0 ft <sup>2</sup> .
C	R-DWOP	The drywell failed from loads accompanying vessel breach which led to the development of a large hole or rupture; nominal hole size is 1.0 ft <sup>2</sup> .
D	R-PedP	The drywell failure was induced by the reactor pedestal failure which led to the development of a large hole or rupture; nominal hole size is 1.0 ft <sup>2</sup> . The pedestal failed from loads accompanying vessel breach.

Table 2.4-1 (continued)

<u>Attribute</u>	<u>Mnemonic</u>	<u>Description</u>
<b>Characteristic 13 (Continued)</b>		
E	R-PedSE	The drywell failure was induced by the reactor pedestal failure which led to the development of a large hole or rupture; nominal hole size is 1.0 ft <sup>2</sup> . The pedestal failed from dynamic loads associated with an ex-vessel steam explosion in the reactor cavity.
F	DR-Det	The drywell failed from a hydrogen detonation which led to the development of a large hole or rupture; nominal hole size is 1.0 ft <sup>2</sup> .
G	DR-Def	The drywell failed from a hydrogen deflagration which led to the development of a large hole or rupture; nominal hole size is 1.0 ft <sup>2</sup> .
H	EDWLk	The drywell failed by the development of a small hole before vessel breach.
I	LDWOP	The drywell failed from loads accompanying vessel breach which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
J	DL-Det	The drywell failed from a hydrogen detonation which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
K	DL-Def	The drywell failed from a hydrogen deflagration which led to the development of a small hole or leak; nominal hole size is 0.1 ft <sup>2</sup> .
L	nIDWF	The drywell did not fail before or at the time of vessel breach.

### 2.4.2 Rebinning

The binning scheme used for evaluating the APET does not exactly match the input information required by GGSOR. 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 GGSOR 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.

In the rebinning for Grand Gulf, there are no changes for characteristics 1 through 9. That is, for these nine characteristics, the information produced by the APET is exactly that used by GGSOR. The last four characteristics, 10, 11, 12, and 13, provide additional information on the types of events that caused containment and drywell failure. This additional information is not used by GGSOR and, therefore, has been deleted in the rebinning process.

Thus, the rebinning process converts the example bin, ABBDAACCBGEEL, to:

A	Fst-SB	Accident sequence is a short-term station blackout
B	LoZrOx	A small fraction of the zirconium was oxidized in-vessel
B	LoP-nLPI	The RPV was at low pressure before vessel breach and there was no injection to the RPV after vessel breach
D	LoEXSE	A small fraction of the core participated in an ex-vessel steam explosion; there was no DCH event
A	SPBEOLO	There was no suppression pool bypass
A	CE-Lk	The containment failed early from the development of a leak
C	LCS	Spray operation was recovered after vessel breach
C	FLDCCI	CCI proceeded in a flooded reactor cavity
B	cSRVBkr	A SRV tailpipe vacuum breaker did not stick open

### 2.4.3 Summary Bins for Presentation

For presentation purposes in NUREG-1150,<sup>4</sup> a set of "summary" bins has been adopted. Instead of the 13 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 eight summary bins for Grand Gulf are:

vessel breach, Early CF, Early SP Bypass, CS Not Available  
VB, Early CF, Early SP Bypass, CS Available  
VB, Early CF, Late SP Bypass  
VB, Early CF, No SP Bypass  
VB, Late CF  
VB, Vent  
VB, No CF  
No VB

In the summary binning scheme there are essentially four characteristics: vessel breach, containment failure, suppression pool bypass, and containment spray operation. Each of these characteristics and their associated attributes are defined in Table 2.4-2.

The summary bins are listed roughly in decreasing order of the severity of the resulting source term. The eight summary bins may now be defined as follows:

vessel breach, Early CF, Early SP Bypass, CS Not Available

Vessel breach occurs and both the containment and the drywell have failed either before or at the time of vessel breach. The containment sprays do not operate before or at the time of vessel breach.

vessel breach, Early CF, Early SP Bypass, CS Available

Vessel breach occurs and both the containment and the drywell fail either before or at the time of vessel breach. In this bin, however, the containment sprays do operate before or at the time of vessel breach.

vessel breach, Early CF, Late SP Bypass

Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail until the late time period and, thus, both the in-vessel releases and the releases associated with vessel breach are scrubbed by the suppression pool. Therefore, the availability of containment sprays during the time period that the suppression pool is not bypassed is not very important and, thus, the CS characteristic has been dropped.

vessel breach, Early CF, No SP Bypass

Vessel breach occurs and the containment fails either before or at the time of vessel breach. The drywell does not fail and, therefore, all of the radionuclide releases pass through the suppression pool. Because the pool has not been bypassed, the availability of the sprays is not very important and, thus, the CS characteristic has been dropped.



**vessel breach, Late CF**

Vessel breach occurs, however, the containment does not fail until the late time period. If the containment did not fail early it is unlikely that the drywell will fail early. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped.

**vessel breach, Vent**

This summary bin represents the case in which vessel breach occurs and the containment was vented during any of the time periods in the accident.

**VB, No CF**

Vessel breach occurs but there is no containment failure and any releases associated with normal containment leakage are minor. Thus, the suppression pool bypass characteristic and the containment spray characteristic have been dropped. The risk associated with this bin will be negligible.

**No vessel breach**

Vessel breach is averted. Thus, there are no releases associated with vessel breach and there are no CCI releases. It must be remembered, however, that the containment can fail even if vessel breach is averted. Thus, the potential exists for some of the in-vessel releases to be released to the environment. It follows that there will be some risk associated with this bin.

Table 2.4-2  
Description of Summary Accident Progression Bin Characteristics

<u>Attribute</u>	<u>Description</u>
<b>Characteristic 1: Vessel Breach (VB)</b>	
VB	Vessel breach occurs
No VB	Vessel breach does not occur.
<b>Characteristic 2: Containment Failure Time (CF)</b>	
Early CF	The containment fails either before or at the time of vessel breach from the development of a leak or a rupture.
Late CF	The containment fails during the late time period from the development of either a leak or a rupture.

Table 2.4-2 (Continued)

Attribute	Description
<b>Characteristic 2 (Continued)</b>	
Vent	The containment is vented during any of the time periods.
No CF	The containment does not fail.
<b>Characteristic 3: Suppression Pool (SP) Bypass</b>	
Early SP Bypass	The drywell fails either before or at the time of vessel breach from the development of a leak or a rupture.
Late SP Bypass	The drywell fails during the late time period from the development of either a leak or a rupture.
No SP Bypass	The drywell does not fail.
<b>Characteristic 4: Containment Spray (CS) Operation</b>	
CS Not Avail.	The containment sprays do not operate during the early or intermediate time periods.
CS Available	The containment sprays operate during either the early time period, the intermediate time period, or during both time periods.

## 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 accident progression bins (APBs), the discussion is primarily in terms of APBs. Some intermediate results are also presented.

Section 2.5.1 presents the results for the internal initiators. External events (fire and seismic) were not considered in the Grand Gulf 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 1: Short-Term SBO. This PDS involves station blackout scenarios where loss of offsite power (LOSP) is recoverable. Coolant injection is lost early so that core damage occurs in the short term and with the vessel at high pressure. If offsite power is restored, then the following functions are available: either high pressure injection or low pressure injection or both, heat removal via the sprays, and the miscellaneous systems-venting, standby gas treatment (SBGT), containment isolation (CI), hydrogen ignition (H<sub>2</sub>I). In addition, the firewater system is available. This PDS also includes cut sets with either one or two stuck open SRVs.

Table 2.5-1 lists the five most probable APBs for this PDS, the five most probable APBs that have vessel breach, and the five most probable APBs that have containment failure (CF). 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 250 observations that form the sample. If Bin A occurred with a probability of 0.004 for each observation, its probability would be 0.004 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 249 observations, its probability would also be 0.004. The remaining eight columns explain 8 of the 9 characteristics in the APB indicator for the rebinned results. The first characteristic, the accident sequence indicator (ASeq), has been omitted since this is defined by the PDS. 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 3837 source term bins for this PDS. To capture 95% of the probability, 1812 bins are required. The five most probable bins capture only 13% of the probability.

Table 2.5-1  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 1; Short-Term SBO

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Five Most Probable Bins*</b>										
1	ABBDDGCCB	0.032	LoZrOx	LoPnLPI	LoExSE	SPBEOL3	CL-Rpt	LCS	FLDCCI	cSRVBkr
2	ABEEAICEB	0.029	LoZrOx	nVB	nDCH-SE	SPBEOL0	CnFail	LCS	noCCI	cSRVBkr
3	ABEEAGCEB	0.027	LoZrOx	nVB	nDCH-SE	SPBEOL0	CL-Rpt	LCS	noCCI	cSRVBkr
4	ABEEAFCEB	0.026	LoZrOx	nVB	nDCH-SE	SPBEOL0	CL-Lk	LCS	noCCI	cSRVBkr
5	ABEEAHCEB	0.019	LoZrOx	nVB	nDCH-SE	SPBEOL0	CL-Vent	LCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have VB*</b>										
1	ABBDDGCCB	0.032	LoZrOx	LoPnLPI	LoExSE	SPBEOL3	CL-Rpt	LCS	FLDCCI	cSRVBkr
9	ABDDDGCCB	0.012	LoZrOx	LoP-LPI	LoExSE	SPBEOL3	CL-Rpt	LCS	FLCCI	cSRVBkr
12	ABBDDGACB	0.010	LoZrOx	LoPnLPI	LoExSE	SPBEOL3	CL-Rpt	noCS	FLCCI	cSRVBkr
13	ABBDDGCCA	0.010	LoZrOx	LoPnLPI	LoExSE	SPBEOL3	CL-Rpt	LCS	FLCCI	oSRVBkr
14	ABDDAICEB	0.008	LoZrOx	LoPnLPI	LoExSE	SPBEOL0	CnFail	LCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF*</b>										
7	AAEEABAEB	0.013	HiZrOx	nVB	nDCH-SE	SPBEOL0	CE-Rpt	noCS	noCCI	cSRVBkr
10	AAEEBAEB	0.011	HiZrOx	nVB	nDCH-SE	SPBE2L2	CE-Rpt	noCS	noCCI	cSRVBkr
15	AAEEAACEB	0.008	HiZrOx	nVB	nDCH-SE	SPBEOL0	CE-Lk	LCS	noCCI	cSRVBkr
18	AAEEHBAEB	0.007	HiZrOx	nVB	nDCH-SE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
31	AABDABACB	0.004	HiZrOx	LoPnLPI	LoEXSE	SPBEOL0	CE-Rpt	noCS	FLDCCI	cSRVBkr

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

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

Four of the five most probable bins have no vessel breach and in all of these five bins the containment either fails late or does not fail. The five most probable bins with vessel breach all occur with the RPV at low pressure and again, the containment either fails late or does not fail. The last part of Table 2.5-1 shows the five most probable APBs with early CF. (Early CF means CF before, at, or immediately after vessel breach.) In four of these five bins vessel breach does not occur. In the one bin that vessel breach occurs the drywell does not fail and, therefore, all of the releases pass through the suppression pool.

For this PDS the probability of recovering offsite electrical power before vessel breach (i.e., in the early time period) is 0.62. The probability of recovering coolant injection before vessel breach is 0.87 which includes the recovery of injection systems when ac power is recovered and the use of the firewater system for those accidents in which ac power is not recovered. If coolant injection is restored to the RPV, it is possible to arrest the core damage process and avoid vessel breach. For this PDS the probability that vessel breach is averted is 0.32. The probability that the containment fails early, with early defined as before or around the time of vessel breach, is 0.36.

2.5.1.2 Results for PDS 2: Short-Term SBO. PDS 2 is the same as PDS 1 except that heat removal via the sprays is not available with the recovery of offsite power.

Table 2.5-2 lists the five most probable APBs for this PDS, the five most probable APBs that have vessel breach, and the five most probable APBs that have early containment failure (CF). Evaluation of the APET produced 2571 source term bins for this PDS. To capture 95% of the probability, 1066 bins are required. The five most probable bins capture only 16% of the probability. In four of the five most probable bins, vessel breach is averted. In the bin that has vessel breach, the containment fails in the late time period. In all of the five most probable bins that have vessel breach the containment either fails in the late time period or does not fail. Similarly, in all of the five most probable bins that have early containment failure vessel breach is averted. Only two of the five most probable bins that have early containment failure have coincident drywell failure. Furthermore, in these two bins vessel breach is averted and there are no stuck open SRV tailpipe vacuum breakers. Thus, there are only in-vessel releases and these pass through the suppression pool.

The probability that offsite electrical power is recovered before vessel breach is 0.62. For this PDS the probability that coolant injection is recovered and vessel breach is averted is 0.32. The probability that the containment will fail early is 0.36.

2.5.1.3 Results for PDS 3: Short-Term SBO. PDS 3 is the same as PDS 1 except that heat removal via the sprays is not available with the recovery of offsite power and the only injection system that is available with the recovery of offsite power is the condensate system.

Table 2.5-2  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 2; Short-Term SBO

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Five Most Probable Bins*</b>										
1	ABDDGACB	0.043	LoZrOx	LoPnLPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	cSRVBkr
2	ABEEAIAEB	0.035	LoZrOx	nVB	nDCH-SE	SPBE0L0	CnFail	noCS	noCCI	cSRVBkr
3	ABEEAGAEB	0.034	LoZrOx	nVB	nDCH-SE	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
4	ABEEAFAEB	0.032	LoZrOx	nVB	nDCH-SE	SPBE0L0	CL-Lk	noCS	noCCI	cSRVBkr
5	ABEEAHAEB	0.021	LoZrOx	nVB	nDCH-SE	SPBE0L0	CL-VENT	noCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have VB*</b>										
1	ABDDGACB	0.043	LoZrOx	LoPnLPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	cSRVBkr
7	ABDDGACB	0.015	LoZrOx	LoP-LPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	cSRVBkr
10	ABDDGACA	0.012	LoZrOx	LoPnLPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	oSRVBkr
13	ABDAIAEB	0.010	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CnFail	noCS	noCCI	cSRVBkr
14	ABDAFAEB	0.010	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Lk	noCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF*</b>										
8	AAEEABAEB	0.014	HiZrOx	nVB	nDCH-SE	SPBE0L0	CE-Rpt	noCS	noCCI	cSRVBkr
12	AAEEBAEB	0.012	HiZrOx	nVB	nDCH-SE	SPBE2L2	CE-Rpt	noCS	noCCI	cSRVBkr
17	AAEEHBAEB	0.009	HiZrOx	nVB	nDCH-SE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
18	AAEEAAAEB	0.009	HiZrOx	nVB	nDCH-SE	SPBE0L0	CE-Lk	noCS	noCCI	cSRVBkr
30	ABEEAAAEB	0.005	LoZrOx	nVB	nDCH-SE	SPBE0L0	CE-Lk	noCS	noCCI	cSRVBkr

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

2.73

Table 2.5-3 lists the five most probable APBs for this PDS, the five most probable APBs that have vessel breach, and the five most probable APBs that have early containment failure (CF). Evaluation of the APET produced 2669 source term bins for this PDS. To capture 95% of the probability, 1216 bins are required. The five most probable bins capture only 12% of the probability.

In three of the five most probable bins vessel breach is averted and in the two bins that have vessel breach the containment fails in the late time period. In four of the five most probable bins that have vessel breach the containment either fails in the late time period or does not fail. Only two of the five most probable bins that have early containment failure have coincident drywell failure. Furthermore, in these two bins vessel breach is averted and there are no stuck open SRV tailpipe vacuum breakers. Thus, there are only in-vessel releases and these pass through the suppression pool.

The probability that offsite electrical power is recovered before vessel breach is 0.62. For this PDS the probability that coolant injection is recovered and vessel breach is averted is 0.21. The probability that the containment fails early is 0.44. The early containment failure probability is lower for PDS 1 than it is for this PDS because PDS 1 has a higher probability that vessel breach will be averted.

2.5.1.4 Results for PDS 4: Long-Term SBO. This PDS involves station blackout scenarios where LOSP is recoverable. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at low pressure. If offsite power is restored, then the following functions are available: either high pressure injection or low pressure injection or both, heat removal via the sprays, and the miscellaneous systems--venting, SBT, CI, H<sub>2</sub>I. In addition, the firewater system is recoverable.

Table 2.5-4 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. Evaluation of the APET produced 2353 source term bins for this PDS. To capture 95% of the probability, 842 bins are required. The ten most probable bins capture 23% of the probability.

In all of the ten most probable bins vessel breach occurs, the RPV is at low pressure, and an ex-vessel steam explosion, which involves a small amount of the core, occurs at vessel breach. Containment sprays are not available in any of the ten most probable bins. In all of the five most probable bins that have early containment failure and early suppression pool bypass vessel breach occurs with the RPV at low pressure followed by an ex-vessel steam explosion. There are no stuck open tailpipe vacuum breakers in these five bins so all of the in-vessel releases pass through the suppression pool. However, because there is early drywell failure, the ex-vessel releases will bypass the suppression pool. Although sprays are not available in these five bins, CCI either proceeds in a flooded cavity (3 bins) and, therefore, the CCI releases are scrubbed, or the core debris is cooled and there are no CCI releases (2 bins).

Table 2.5-3  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 3: Short-Term SBO

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Five Most Probable Bins*</b>										
1	ABDDGACB	0.041	LoZrOx	LoPnLPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	cSRVBkr
2	ABEEAGAEB	0.024	LoZrOx	nVB	nDCH-SE	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
3	ABEEAIAEB	0.022	LoZrOx	nVB	nDCH-SE	SPBE0L0	CnFail	noCS	noCCI	cSRVBkr
4	ABEEAFAEB	0.020	LoZrOx	nVB	nDCH-SE	SPBE0L0	CL-Lk	noCS	noCCI	cSRVBkr
5	ABDDGACB	0.014	LoZrOx	LoP-LPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	cSRVBkr
<b>Five Most Probable Bins that have VB*</b>										
1	ABDDGACB	0.041	LoZrOx	LoPnLPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	cSRVBkr
5	ABDDGACB	0.014	LoZrOx	LoP-LPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	cSRVBkr
6	ABABAEAE	0.013	LoZrOx	HiPnLPI	LoDCH	SPBE0L0	CVB-Rpt	noCS	noCCI	cSRVBkr
8	ABDDGACA	0.013	LoZrOx	LoPnLPI	LoEXSE	SPBE0L3	CL-Rpt	noCS	FLDCCI	cSRVBkr
10	ABBDAIAEB	0.010	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CnFail	noCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF*</b>										
6	ABABAEAE	0.013	LoZrOx	HiPnLPI	LoDCH	SPBE0L0	CVB-Rpt	noCS	noCCI	cSRVBkr
16	AAEEABAEB	0.007	HiZrOx	nVB	nDCH-SE	SPBE0L0	CE-Rpt	noCS	noCCI	cSRVBkr
18	AAEEBAEB	0.007	HiZrOx	nVB	nDCH-SE	SPBE2L2	CE-Rpt	noCS	noCCI	cSRVBkr
20	ABABBEAEB	0.007	LoZrOx	HiPnLPI	LoDCH	SPBE0L3	CVB-Rpt	noCS	noCCI	cSRVBkr
22	AAEEHBAEB	0.006	HiZrOx	nVB	nDCH-SE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr

\* 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 2.5-4  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 4: Long-Term SBO

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Ten Most Probable Bins*</b>										
1	BABDAGACB	0.032	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	FLDCCI	cSRVBkr
2	BABDAEACB	0.031	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	FLDCCI	cSRVBkr
3	BABDHBACB	0.026	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	FLDCCI	cSRVBkr
4	BBBDAGACB	0.026	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	FLDCCI	cSRVBkr
5	BABDAEAEb	0.014	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	noCCI	cSRVBkr
6	BABDBEACB	0.020	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr
7	BABDAGAEb	0.020	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
8	BABDHBAEb	0.018	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
9	BBBDAEACB	0.016	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	FLDCCI	cSRVBkr
10	BBBDAGAEb	0.015	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*</b>										
3	BABDHBACB	0.026	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	FLDCCI	cSRVBkr
6	BABDBEACB	0.020	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr
8	BABDHBAEb	0.018	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
11	BABDBEAEb	0.013	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	noCCI	cSRVBkr
12	BBBDBEACB	0.012	LoZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr

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

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

Because this is a slow SBO (i.e., core damage occurs  $\geq 12$  h) this PDS has a much lower probability of recovering offsite power than did the fast SBO in which core damage occurs in approximately 1 h. The probability that offsite electrical power is recovered before vessel breach is 0.19. For this PDS the probability that coolant injection is recovered and vessel breach is averted is only 0.05. The probability that the containment fails early is 0.65.

2.5.1.5 Results for PDS 5: Long-Term SBO. PDS 5 is the same as PDS 4 except that heat removal via the sprays is not available with the recovery of offsite power. However, because there is a low probability of recovering offsite power in this PDS this difference is not very important.

Table 2.5-5 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. Evaluation of the APET produced 1468 source term bins for this PDS. To capture 95% of the probability, 482 bins are required. The 10 most probable bins capture 26% of the probability.

In all of the 10 most probable bins in which vessel breach occurs, the RPV is at low pressure, and an ex-vessel steam explosion, which involves a small amount of the core, occurs at vessel breach. Containment sprays are not available in any of the 10 most probable bins. In all of the five most probable bins that have early containment failure and early suppression pool bypass vessel breach occurs with the RPV at low pressure followed by an ex-vessel steam explosion. There are no stuck open tailpipe vacuum breakers in these five bins so all of the in-vessel releases pass through the suppression pool. However, because there is early drywell failure, the ex-vessel releases will bypass the suppression pool. Although sprays are not available in these five bins, CCI either proceeds in a flooded cavity (three bins) and, therefore, the CCI releases are scrubbed, or the core debris is cooled and there are no CCI releases (two bins).

The probability that offsite electrical power is recovered before vessel breach is 0.19. For this PDS the probability that coolant injection is recovered and vessel breach is averted is only 0.05. The probability that the containment fails early is 0.64.

2.5.1.6 Results for PDS 6: Long-Term SBO. PDS 6 is the same as PDS 4 except that heat removal via the sprays is not available with the recovery of offsite power and the only injection system that is recoverable is the firewater system. However, because the operators did not use the firewater system during the many hours before core damage it is assumed that there is a negligible probability that they will use this system during core damage. Thus, there is no coolant injection to the RPV.

Table 2.5-6 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. Evaluation of the APET produced 1127 source term bins for this PDS. To capture 95% of the probability, 356 bins are required. The 10 most probable bins capture 31% of the probability.

Table 2.5-5  
 Results of the Accident Progression Analysis for Grand Gulf  
 Internal Initiators: PDS 5: Long-Term SBO

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOx</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Ten Most Probable Bins*</b>										
1	BABDAGACB	0.036	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	FLDCCI	cSRVBkr
2	BABDAEACB	0.034	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	FLDCCI	cSRVBkr
3	BABDHBACB	0.030	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	FLDCCI	cSRVBkr
4	BBBDAGACB	0.029	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	FLDCCI	cSRVBkr
5	BBBDAEACB	0.027	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	FLDCCI	cSRVBkr
6	BABDAEAEB	0.024	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	noCCI	cSRVBkr
7	BABDAGAEB	0.022	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
8	BABDBEACB	0.022	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr
9	BABDHBAEB	0.020	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
10	BBBDAGAEB	0.020	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*</b>										
3	BABDHBACB	0.030	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	FLDCCI	cSRVBkr
8	BABDBEACB	0.022	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr
9	BABDHBAEB	0.020	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
12	BBBDBEACB	0.015	LoZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr
13	BABDBEAEB	0.015	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	noCCI	cSRVBkr

\* 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 2.5-6  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 6: Long-Term SBO

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Ten Most Probable Bins*</b>										
1	BABDHBACB	0.044	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	FLDCCI	cSRVBkr
2	BABDAGACB	0.041	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	FLDCCI	cSRVBkr
3	BABDAEACB	0.037	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	FLDCCI	cSRVBkr
4	BBBDAGACB	0.033	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	FLDCCI	cSRVBkr
5	BABDHBAEB	0.029	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
6	BABDAEAEB	0.026	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	noCCI	cSRVBkr
7	BABDAGAEB	0.026	HiZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
8	BBBDAEACB	0.026	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CVB-Rpt	noCS	FLDCCI	cSRVBkr
9	BABDBEACB	0.024	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr
10	BBBDAGAEB	0.020	LoZrOx	LoPnLPI	LoEXSE	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*</b>										
1	BABDHBACB	0.044	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	FLDCCI	cSRVBkr
5	BABDHBAEB	0.029	HiZrOx	LoPnLPI	LoEXSE	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
9	BABDBEACB	0.024	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr
12	BBBDBEACB	0.016	LoZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	FLDCCI	cSRVBkr
13	BABDBEAEB	0.016	HiZrOx	LoPnLPI	LoEXSE	SPBE0I3	CVB-Rpt	noCS	noCCI	cSRVBkr

\* 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 all of the 10 most probable bins vessel breach occurs with the RPV at low pressure followed by an ex-vessel steam explosion that involves a small fraction of the core. The containment sprays do not operate during the accident but because there are no stuck open SRV tailpipe vacuum breakers all of the in-vessel releases are still scrubbed by the suppression pool. In all of the 10 most probable bins the core debris released from the vessel is cooled and there are no CCI releases.

The probability that offsite electrical power is recovered before vessel breach is 0.19. However, because there is no coolant injection to the vessel the probability of vessel breach is 1.0. The probability that the containment fails early is 0.68.

2.5.1.7 Results for PDS 7: Short-Term SBO. This PDS involves station blackout (without any dc power) scenarios where LOSP is not recoverable. Coolant injection is lost early such that core damage occurs in the short term. The ADS requires dc power. Thus, the operator cannot depressurize the vessel before core damage. Also, because offsite power is not recoverable, the functions of injection, heat removal, and those of the miscellaneous systems are not available. This PDS also includes cut sets with either one or two stuck open SRVs. If the RPV is depressurized through the stuck open SRVs, the firewater system can be used as a source of low pressure injections.

Table 2.5-7 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. Evaluation of the APET produced 1473 source term bins for this PDS. To capture 95% of the probability, 552 bins are required. The 10 most probable bins capture 21% of the probability.

In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. The containment sprays do not operate during the accident but because there are no stuck open SRV tailpipe vacuum breakers all of the in-vessel releases are still scrubbed by the suppression pool.

Because dc power is lost, ac power can not be recovered and the ADS is unavailable such that the RPV is at high pressure. There is a small probability (4%), however, that a SRV will stick open and depressurize the RPV. Once the RPV has been depressurized, the firewater system can be used to provide coolant injection to the RPV. The firewater system has its own power supply. Thus, the probability that vessel breach is averted is only 0.01. The probability that the containment fails early is 0.60.

2.5.1.8 Results for PDS 8: Long-Term SBO. This PDS involves station blackout (without any dc power) scenarios where LOSP is not recoverable. Coolant injection is lost late such that core damage occurs in the long term. The ADS requires dc power. Thus, the operator cannot depressurize the vessel before core damage. Since offsite power is not recoverable, the injection and heat removal functions and the miscellaneous systems are not available. Table 2.5-8 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early

Table 2.5-7  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 7: Short-Term SBO

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
Ten Most Probable Bins*										
1	ABABAEAEB	0.041	LoZrOx	HiPnLPI	LoDCH	SPBE0L0	CVB-Rpt	noCS	noCCI	cSRVBkr
2	AAABAEAEB	0.028	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CVB-Rpt	noCS	noCCI	cSRVBkr
3	AAABAIAEB	0.025	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CnFail	noCS	noCCI	cSRVBkr
4	AAABAFaEB	0.024	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CL-Lk	noCS	noCCI	cSRVBkr
5	ABABBEAEB	0.018	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	noCS	noCCI	cSRVBkr
6	AAABEBAEB	0.015	HiZrOx	HiPnLPI	LoDCH	SPBE2L2	CE-Rpt	noCS	noCCI	cSRVBkr
7	ABABAGAEB	0.015	LoZrOx	HiPnLPI	LoDCH	SPBE0L0	CL-Rpt	noCS	noCCI	cSRVBkr
8	ABABAFaEB	0.014	LoZrOx	HiPnLPI	LoDCH	SPBE0L0	CL-Lk	noCS	noCCI	cSRVBkr
9	AAABABAEB	0.014	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CE-Rpt	noCS	noCCI	cSRVBkr
10	AACBAFAEB	0.013	HiZrOx	HiP-LPI	LoDCH	SPBE0L0	CL-Lk	noCS	noCCI	cSRVBkr
Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*										
5	ABABBEAEB	0.018	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	noCS	noCCI	cSRVBkr
6	AAABEBAEB	0.015	HiZrOx	HiPnLPI	LoDCH	SPBE2L2	CE-Rpt	noCS	noCCI	cSRVBkr
13	AAABBEAEB	0.012	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	noCS	noCCI	cSRVBkr
16	AAABHBAEB	0.011	HiZrOx	HiPnLPI	LoDCH	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr
18	AACBHBAEB	0.010	HiZrOx	HiP-LPI	LoDCH	SPBE3L3	CE-Rpt	noCS	noCCI	cSRVBkr

\* 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 2.5-8  
 Results of the Accident Progression Analysis for Grand Gulf  
 Internal Initiators: PDS 8: Long-Term SBO

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
Ten Most Probable Bins*										
1	BAABAAEB	0.067	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Lk	noCS	noCCI	cSRVBkr
2	BBABAAEB	0.040	LoZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Lk	noCS	noCCI	cSRVBkr
3	BAABAEAE	0.030	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	noCS	noCCI	cSRVBkr
4	BACBAAEB	0.030	HiZrOx	HiP-LPI	LoDCH	SPBE0LO	CE-Lk	noCS	noCCI	cSRVBkr
5	BBABAEAE	0.027	LoZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	noCS	noCCI	cSRVBkr
6	BAABABAEB	0.027	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Rpt	noCS	noCCI	cSRVBkr
7	BAABAAEA	0.021	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Lk	noCS	noCCI	oSRVBkr
8	BAABAAACB	0.017	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Lk	noCS	FLDCCI	cSRVBkr
9	BAABBBADB	0.016	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Rpt	noCS	DlyCCI	cSRVBkr
10	BAABAGAEB	0.016	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CL-Rpt	noCS	noCCI	cSRVBkr
Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*										
9	BAABBBADB	0.016	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Rpt	noCS	DlyCCI	cSRVBkr
13	BAABBAADB	0.014	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Lk	noCS	DlyCCI	cSRVBkr
18	BAABEAAEB	0.010	HiZrOx	HiPnLPI	LoDCH	SPBE2L2	CE-Lk	noCS	noCCI	cSRVBkr
21	BBABBAADB	0.009	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Lk	noCS	DlyCCI	cSRVBkr
23	BBABBBADB	0.009	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Rpt	noCS	DlyCCI	cSRVBkr

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

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

suppression pool bypass. Evaluation of the APET produced 494 source term bins for this PDS. To capture 95% of the probability, 232 bins are required. The 10 most probable bins capture 29% of the probability.

In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. The containment sprays do not operate during the accident. There is only one bin that has a stuck-open tailpipe vacuum breaker; however for this bin the drywell does not fail. Thus, all of the in-vessel releases are scrubbed by the suppression pool. Only one of the 10 most probable bins has drywell failure.

Because dc power is lost, ac power can not be recovered and the ADS is unavailable such that the RPV is at high pressure. Because there is no early coolant injection to the RPV, the probability of vessel breach is 1.0. The probability that the containment fails early is 0.54.

2.5.1.9 Results for PDS 9: Short-Term ATWS. This PDS involves ATWS scenarios. Coolant injection is lost early such that core damage occurs in the short term with the vessel at high pressure because the operator failed to depressurize it. The low pressure injection is recoverable with reactor depressurization. Heat removal via the sprays is available and the miscellaneous systems (i.e., venting, SBT, CI and H2I) are available.

Table 2.5-9 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. Evaluation of the APET produced 1793 source term bins for this PDS. To capture 95% of the probability, 477 bins are required. The 10 most probable bins capture 33% of the probability.

In the 10 most probable bins vessel breach occurs with RPV at high pressure. In nine of the 10 most probable bins a DCH event occurs at vessel breach, and in the other bin an ex-vessel steam explosion follows vessel breach. In all but one of the 10 most probable bins the containment fails at vessel breach. Containment sprays are operating during the intermediate time period in all of these 10 bins. There are no CCI releases in all but one of these bins and in the bin that CCI does occur the releases are scrubbed by a flooded cavity.

Electrical power is always available in this PDS. The probability that the RPV will be at high pressure during core damage is 0.84. The probability that coolant injection will be restored to the RPV and vessel breach will be averted is only 0.04. This low probability of core damage arrest is driven by the failure of the operators to depressurize the RPV. The probability that the containment fails early is 0.67.

2.5.1.10 Results for PDS 10: Long-Term ATWS. This PDS involves ATWS scenarios. Coolant injection is lost late such that core damage occurs in the long term with the vessel at high pressure because the operator failed to depressurize it. Low pressure injection is recoverable with reactor depressurization. Heat removal via the sprays is available, and the miscellaneous systems (i.e., venting, SBT, CI and H2I) are available.



Table 2.5-9  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 9: Short-Term ATWS

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
Ten Most Probable Bins*										
1	EAABAECEB	0.087	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
2	EBABAECEB	0.055	LoZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
3	EACBAECEB	0.035	HiZrOx	HiP-LPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
4	EAABBECEB	0.033	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
5	EBABBECEB	0.028	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
6	EBCBAECEB	0.021	LoZrOx	HiP-LPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
7	EAABAECCB	0.021	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	FLDCCI	cSRVBkr
8	EAADAECEB	0.018	HiZrOx	HiPnLPI	LoEXSE	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
9	EAABAFCEB	0.017	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CL-Lk	LCS	noCCI	cSRVBkr
10	EACBBECEB	0.017	HiZrOx	HiP-LPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*										
4	EAABBECEB	0.033	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
5	EBABBECEB	0.028	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
10	EACBBECEB	0.017	HiZrOx	HiP-LPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
15	EACBHECEB	0.017	HiZrOx	HiP-LPI	LoDCH	SPBE3L3	CVB-Rpt	LCS	noCCI	cSRVBkr
20	EAABEECEB	0.017	HiZrOx	HiP-LPI	LoDCH	SPBE2L2	CVB-Rpt	LCS	noCCI	cSRVBkr

\* 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 2.5-10 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. Evaluation of the APET produced 1461 source term bins for this PDS. To capture 95% of the probability, 496 bins are required. The 10 most probable bins capture 21% of the probability.

In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. In all of these bins the containment fails early; however, there is coincident drywell failure in only one of these bins. The containment sprays operate before vessel breach in all of these bins and continue to operate during the entire accident in all but two of these bins.

Electrical power is always available in this PDS. The probability that the RPV will be at high pressure during core damage is 0.97. The probability that coolant injection will be restored to the RPV and vessel breach will be averted is only 0.01. This low probability of core damage arrest is driven by the failure of the operators to depressurize the RPV. The probability that the containment fails early is 1.0. The containment always fails in this PDS because the energy dumped into the suppression pool from the RPV during an ATWS transient exceeds the capacity of the RHR system which results in a large buildup of steam in the containment.

2.5.1.11 Results for PDS 11: Short-Term T2. This PDS involves transient scenarios where the PCS is lost (T2). Coolant injection is lost early such that core damage occurs in the short term with the vessel at high pressure because the operator failed to depressurize it. Both high and low pressure injection systems are recoverable since the failures involved operator failures. Heat removal via the sprays is available and the miscellaneous systems (i.e., venting, SBT, CI and H<sub>2</sub>I) are available.

Table 2.5-11 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. Evaluation of the APET produced 2136 source term bins for this PDS. To capture 95% of the probability, 705 bins are required. The 10 most probable bins capture 22% of the probability.

In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. The containment fails early in all but two of these bins. Only two of these bins have coincident early containment failure and early drywell failure. The containment sprays operate during the intermediate time period in all of these bins and there are no CCI release in all but one of these bins.

Electrical power is always available in this PDS. The probability that the RPV will be at high pressure during core damage is 0.84. The probability that coolant injection will be restored to the RPV and vessel breach will be averted is only 0.05. This low probability of core damage arrest is driven by the failure of the operators to depressurize the RPV. The probability that the containment fails early is 0.56.

Table 2.5-10  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 10: Long-Term ATWS

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Ten Most Probable Bins*</b>										
1	FAABAADEB	0.047	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Lk	ECS	noCCI	cSRVBkr
2	FACBAADEB	0.026	HiZrOx	HiP-LPI	LoDCH	SPBE0LO	CE-Lk	ECS	noCCI	cSRVBkr
3	FACBABDEB	0.025	HiZrOx	HiP-LPI	LoDCH	SPBE0LO	CE-Rpt	ECS	noCCI	cSRVBkr
4	FAABABBEb	0.024	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Rpt	ECSnoL	noCCI	cSRVBkr
5	FBABAADEB	0.016	LoZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Lk	ECS	noCCI	cSRVBkr
6	FACBBADEB	0.016	HiZrOx	HiP-LPI	LoDCH	SPBE0I3	CE-Lk	ECS	noCCI	cSRVBkr
7	FAABAADDB	0.015	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Lk	ECS	DlyCCI	cSRVBkr
8	FAABABDEB	0.015	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Rpt	ECS	noCCI	cSRVBkr
9	FBABABDEB	0.014	LoZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Rpt	ECS	noCCI	cSRVBkr
10	FAABABBDB	0.013	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CE-Rpt	ECSnoL	DlyCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*</b>										
6	FACBBADEB	0.016	HiZrOx	HiP-LPI	LoDCH	SPBE0I3	CE-Lk	ECS	noCCI	cSRVBkr
13	FBABBADEB	0.011	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Lk	ECS	noCCI	cSRVBkr
16	FAABBBBEB	0.010	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Rpt	ECSnoL	noCCI	cSRVBkr
17	FAABBADEB	0.010	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Lk	ECS	noCCI	cSRVBkr
22	FAABBBDEB	0.008	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CE-Rpt	ECS	noCCI	cSRVBkr

\* 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 2.5-11  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 11: Short-Term T2

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Ten Most Probable Bins*</b>										
1	CAABAECEB	0.060	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CVB-Rpt	LCS	noCCI	cSRVBkr
2	CBABAECEB	0.030	LoZrOx	HiPnLPI	LoDCH	SPBE0L0	CVB-Rpt	LCS	noCCI	cSRVBkr
3	CACBAECEB	0.025	HiZrOx	HiP-LPI	LoDCH	SPBE0L0	CVB-Rpt	LCS	noCCI	cSRVBkr
4	CBABBECEB	0.018	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
5	CAABBECEB	0.018	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
6	CBCBAECEB	0.016	LoZrOx	HiP-LPI	LoDCH	SPBE0L0	CVB-Rpt	LCS	noCCI	cSRVBkr
7	CAABAECEA	0.015	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CVB-Rpt	LCS	noCCI	oSRVBkr
8	CAABAECCB	0.014	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CVB-Rpt	LCS	FLDCCI	cSRVBkr
9	CAABAFCEB	0.014	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CL-Lk	LCS	noCCI	cSRVBkr
10	CAABAICEB	0.013	HiZrOx	HiPnLPI	LoDCH	SPBE0L0	CnFail	LCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*</b>										
4	CBABBECEB	0.018	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
5	CAABBECEB	0.018	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
13	CACBBECEB	0.011	HiZrOx	HiP-LPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
18	CAABBECEA	0.009	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	oSRVBkr
20	CACBHECEB	0.008	HiZrOx	HiP-LPI	LoDCH	SPBE3L3	CVB-Rpt	LCS	noCCI	cSRVBkr

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

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

2.5.1.12 Results for PDS 12: Long-Term T2. PDS 12 is the same as PDS 11 except that core damage occurs in the long-term.

Table 2.5-12 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. Evaluation of the APET produced 2136 source term bins for this PDS. To capture 95% of the probability, 705 bins are required. The 10 most probable bins capture 22% of the probability.

In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. The containment fails early in all but two of these bins. Only two of these bins have coincident early containment failure and early drywell failure. The containment sprays operate during the intermediate time period in all of these bins and there are no CCI release in all but one of these bins.

Electrical power is always available in this PDS. The probability that the RPV will be at high pressure during core damage is 0.84. The probability that coolant injection will be restored to the RPV and vessel breach will be averted is only 0.05. This low probability of core damage arrest is driven by the failure of the operators to depressurize the RPV. The probability that the containment fails early is 0.56.

2.5.1.13 Core Damage Arrest and Avoidance of Vessel Breach. Once core damage has begun, the only way vessel failure is prevented is if coolant injection is restored to the RPV. Restoration of coolant injection to the RPV, however, does not necessarily preclude vessel breach. If injection is not recovered until late in the core damage process, it is unlikely that the addition of water will prevent vessel breach. In addition, there is the possibility that the core debris that slumps into the bottom head of the vessel will trigger a steam explosion. Although steam explosions do not guarantee vessel failure, they do pose a significant challenge to the integrity of the RPV and in some cases do result in vessel failure.

Figure 2.5-1 shows the probability that core damage is arrested before the lower head of the vessel fails for the four collapsed PDS groups (super-groups). For the short-term station blackout super-group the probability of core damage arrest is driven by the likelihood that ac power is recovered early in the accident. Injection to the RPV generally follows ac power recovery. Although the mean probability of recovering ac power is high (0.62%) for short-term station blackout PDSs, there are several factors that tend to reduce the probability of core damage arrest. First, restoration of coolant injection to the RPV does not guarantee that the vessel will not fail. In some cases the core debris is not in a coolable configuration when injection is recovered and, therefore, the accident continues to vessel breach. There are other cases in which only low pressure injection systems are recovered; however, the operators have failed to depressurize the RPV. With the vessel at system pressure these low pressure systems are unable to provide coolant to the core and, therefore, the accident proceeds to vessel breach. Finally, in PDS 7, which is a significant contributor to the mean frequency of this super-group, ac power cannot be recovered. Therefore, except for the infrequent

Table 2.5-12  
Results of the Accident Progression Analysis for Grand Gulf  
Internal Initiators: PDS 12: Long-Term T2

<u>Order</u>	<u>Bin</u>	<u>Prob.**</u>	<u>ZrOxid</u>	<u>VB</u>	<u>DCH-SE</u>	<u>SPB</u>	<u>CF</u>	<u>Sprays</u>	<u>MCCI</u>	<u>SRVBkr</u>
<b>Ten Most Probable Bins*</b>										
1	DAABAECEB	0.060	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
2	DBABAECEB	0.030	LoZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
3	DACBAECEB	0.025	HiZrOx	HiP-LPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
4	DBABBECEB	0.018	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
5	DAABBECEB	0.018	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
6	DBCBAECEB	0.016	LoZrOx	HiP-LPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	cSRVBkr
7	DAABAECEA	0.015	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	noCCI	oSRVBkr
8	DAABAECCB	0.014	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CVB-Rpt	LCS	FLDCCI	cSRVBkr
9	DAABAFCEB	0.014	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CL-Lk	LCS	noCCI	cSRVBkr
10	DAABAICEB	0.013	HiZrOx	HiPnLPI	LoDCH	SPBE0LO	CnFail	LCS	noCCI	cSRVBkr
<b>Five Most Probable Bins that have Early CF and Early Suppression Pool Bypass*</b>										
4	DBABBECEB	0.018	LoZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
5	DAABBECEB	0.018	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
13	DACBBECEB	0.011	HiZrOx	HiP-LPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	cSRVBkr
18	DAABBECEA	0.009	HiZrOx	HiPnLPI	LoDCH	SPBE0I3	CVB-Rpt	LCS	noCCI	oSRVBkr
20	DACBHECEB	0.008	HiZrOx	HiP-LPI	LoDCH	SPBE3L3	CVB-Rpt	LCS	noCCI	cSRVBkr

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

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

cases which involve a stuck-open SRV that depressurizes the RPV and allows firewater to be injected into the vessel, accidents in this group progress to vessel failure.

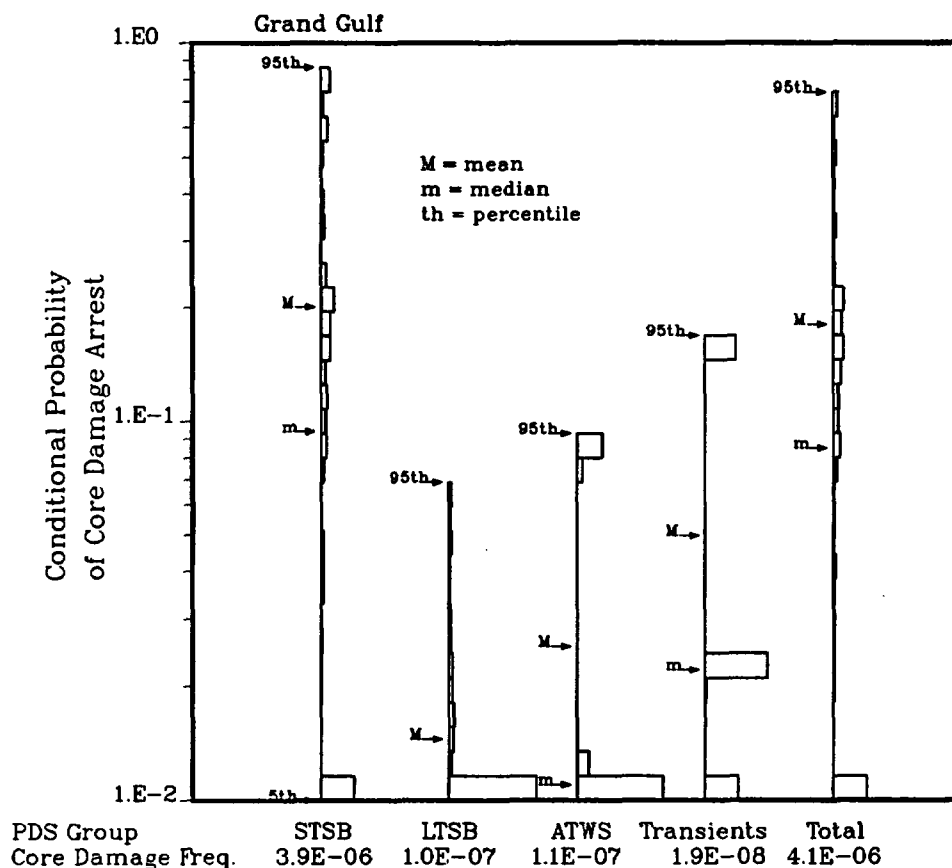


Figure 2.5-1. Probability of Core Damage Arrest.

As with the short-term station blackout super-group, the probability of core damage arrest for the long-term station blackout super-group is also driven by the likelihood that ac power is recovered. The probability of core damage arrest for the long-term station blackout super-group, however, is significantly lower than the corresponding value for the short-term station blackout super-group. Two factors are responsible for most of this difference. First, the mean probability of ac power recovery for the long-term station blackouts (given that power can be recovered) is roughly a third of the corresponding probability for a short-term station blackout. The conditional probability of recovering ac power is defined as the probability of recovering power during the core degradation process given that power was not available at the initiation of core damage. The greater

the amount of time that elapses without power recovery (i.e., the start of the time interval), the smaller the probability that power will be recovered in an ensuing time interval. For a short-term SBO accident, core damage occurs one hour after the initiating event whereas for a long-term SBO accident, core damage occurs 12 hours after the initiating event. Second, in PDS 8, which accounts for approximately half of this group's mean frequency, ac power cannot be recovered and the accident always proceeds to vessel breach.

For both the ATWS super-group and the T2 super-group, the probability of core damage arrest is driven by operator errors. In these PDSs low pressure injection systems are available; however, the operator fails to depressurize the RPV. The mean probability of core damage arrest for the ATWS super-group is slightly lower than the mean value for the T2 super-group. There are two reasons for this difference. First, the operators are more susceptible to errors during the accidents in the ATWS super-group than they are in the T2 super-group. Second, in the ATWS super-group the probability that the core debris is cooled when injection is restored is lower than the corresponding probability in the T2 super-group.

It must be remembered that core damage arrest does not necessarily mean that there will be no radionuclide releases during the accident. 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. This radionuclide release is generally small, however, because in the majority of the cases in which vessel breach is averted these releases are scrubbed as they pass through the suppression pool. Furthermore, if the vessel does not fail, there are no ex-vessel releases (e.g., CCI releases).

2.5.1.14 Early Containment Failure. The early fatality risk depends strongly on the probability of early containment failure (CF). Early containment failure includes both failures that occur before vessel breach and during the time period around vessel breach. The Grand Gulf containment is fairly weak structure when compared to the loads that can potentially occur during the course of the accident. The design pressure is only 15 psig and the assessed mean failure pressure is 55 psig. Because of its low failure pressure, the Grand Gulf containment is susceptible not only to loads from hydrogen deflagrations and detonations but can also be threatened by slow pressurization events (i.e., the accumulation of steam and hydrogen generated during the core degradation process) during accidents that do not have adequate containment heat removal capacity (e.g., long-term SBOs and long-term ATWS).

The production of hydrogen during the core damage process and later during vessel breach, should it occur, is a key factor that affects the probability of containment failure. In a BWR core there is a large inventory of zirconium. The Grand Gulf core, for example, which contains approximately 80,000 kg of zirconium, has nearly five times as much zirconium as does the Surry core (which is a PWR). Large amounts of hydrogen are produced from the oxidation of this metal during the core



damage process. If the HIS is not operating, the hydrogen will accumulate in the containment. For accidents in which the suppression pool is subcooled, the steam released from the RPV is condensed in the pool. The lack of steam in the containment atmosphere in combination with the large amount of hydrogen released during the core degradation process allows mixtures to form that have a high hydrogen concentration. Subsequent ignition of this hydrogen by either random sources or by the recovery of ac power can result in loads that cannot only threaten the containment but can also pose a significant challenge to the drywell structure.

Figure 2.5-2 shows the probability distribution for early CF at Grand Gulf. The probability distributions displayed in this figure are for accidents that proceed to vessel breach and are conditional on core damage.

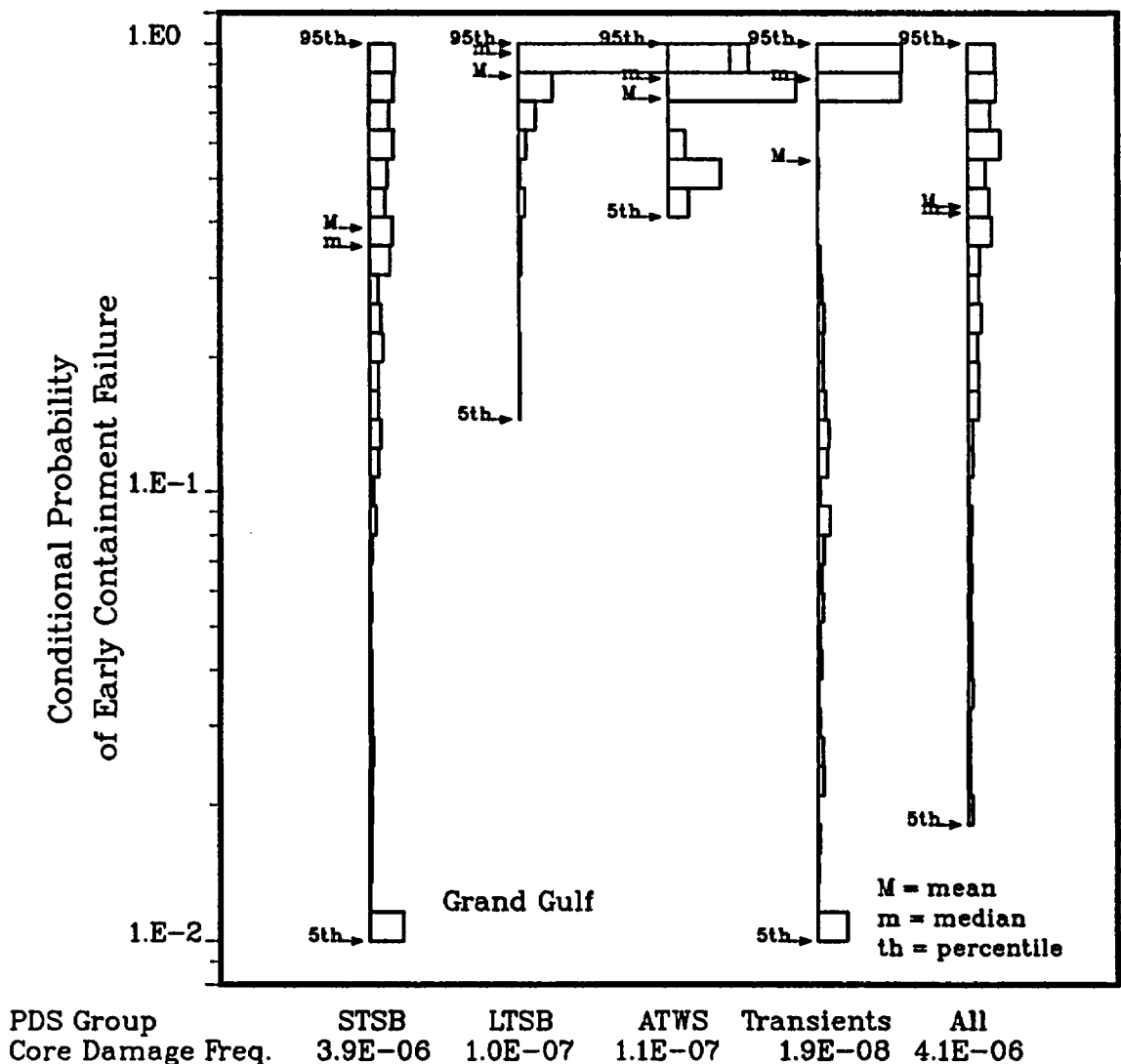


Figure 2.5-2. Probability of Early Containment Failure.

Although the mean conditional probability of early containment failure for accidents in which vessel breach is averted is approximately 0.25, the radionuclide releases are generally small because there are only in-vessel releases and these releases are typically scrubbed by the suppression pool. Thus, the early fatality risk is not strongly influenced by the cases in which vessel breach is averted and, therefore, these cases have not been included in the early containment failure probabilities.

Figure 2.5-3 shows the mean probability of containment failure before vessel breach sorted by events that can lead to containment failure. Figure 2.5-4 presents the same type of information for containment failures that occur at vessel breach. These mean values are conditional on core damage.

The weakness of the containment, relative to the loads that are imposed on it, is reflected in the relatively high containment failure probabilities. Hydrogen combustion events are the dominant events that cause early CF in the short-term station blackout and T2 super-groups. The mean probability of early containment failure for these two PDS is roughly 0.5. In both of these summary PDS groups the suppression pool is subcooled before vessel breach and, therefore, there is no significant accumulation of steam in the containment. Although this virtually eliminates the possibility of CF from slow pressurization events (e.g., accumulation of steam), it does allow mixtures to form in the containment during a short-term SBO that have a fairly high hydrogen concentration. Because the HIS is initially unavailable in during a short-term SBO, it is not uncommon for the hydrogen

SUMMARY  
ACCIDENT  
PROGRESSION  
BIN GROUP

SUMMARY PDS GROUP  
(Mean Core Damage Frequency)

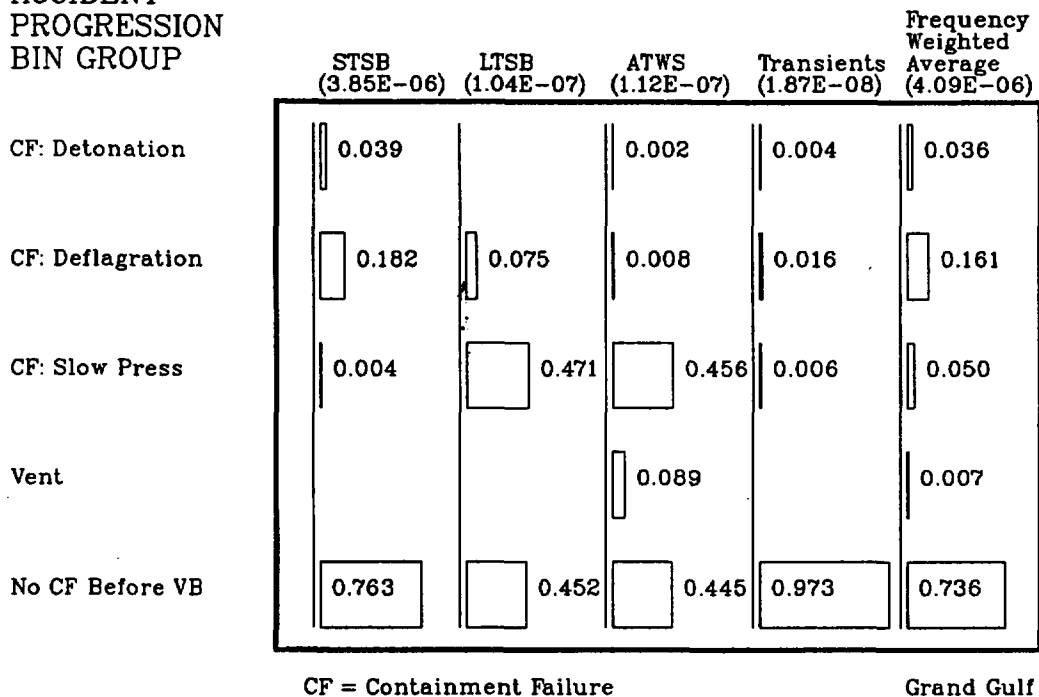


Figure 2.5-3. Mean Probability of CF Before Vessel Breach.

concentration in containment to be above 16% before vessel breach. In the short-term SBO PDS group about half of the early CF probability results from failures that occur before vessel breach and the other half results from failures shortly after vessel breach. In the T2 super-group, on the other hand, almost all of the early CFs occur at the time of vessel breach. For accidents in the T2 super-group, it is likely that the operator turned on the HIS before core damage and, therefore, the hydrogen generated before vessel breach is usually burned such that the resulting load is benign.

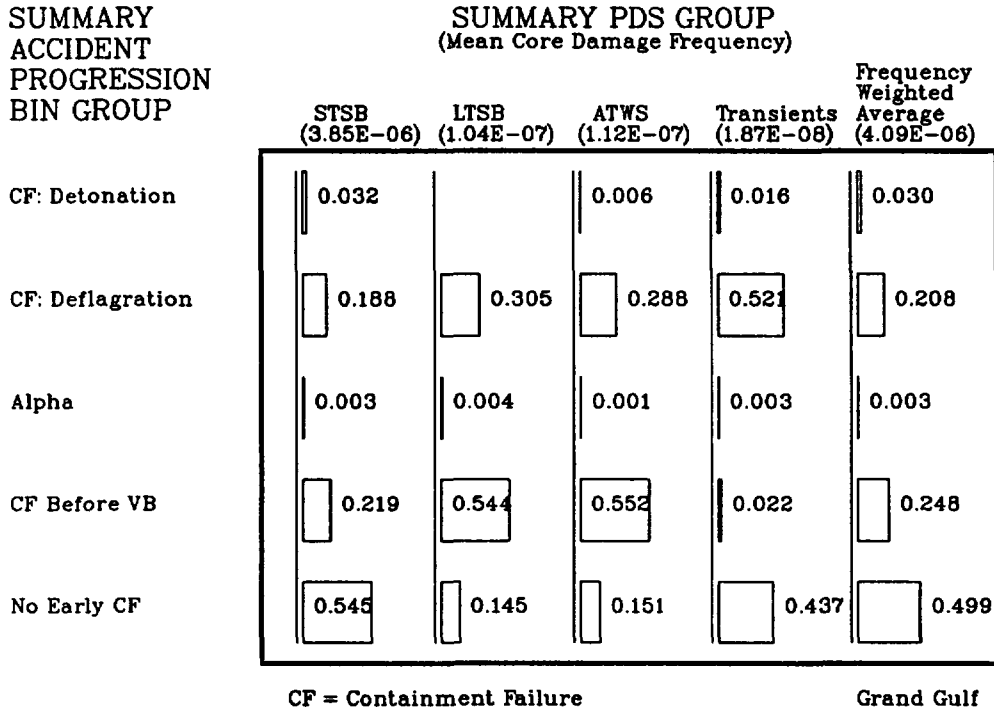


Figure 2.5-4. Mean Probability of CF at Vessel Breach.

For the long-term SBO super-group, the mean conditional probability of early CF is 0.85. Less than half of this probability comes from CFs caused by hydrogen combustion events. In this super-group the suppression pool is saturated and the containment is pressurized by the accumulation of steam that is generated by the hot pool. In most of these accidents hydrogen burns are not possible because the containment is steam inert. Thus, the preponderance of the CFs that occur before vessel breach are caused by pressurization events associated with the accumulation of steam in the containment. There are a few cases, however, in which the containment sprays are recovered before vessel breach. In these cases the sprays slowly condense the steam which allows a combustible mixture to form. Ignition of this mixture can potentially fail the containment. Roughly a third of this mean probability results from CFs that occur at vessel breach and the vast majority of these failures are caused by hydrogen combustion events.

For the ATWS super-group, the mean conditional probability of early CF is 0.76. Similar to the long-term SBO super-group, less than half of this probability comes from CFs caused by hydrogen combustion events. This super-group consists of both a long-term PDS and a short-term PDS. In the long-term PDS the suppression pool is saturated and either the operators vent the containment or the containment fails before vessel breach from the accumulation of steam in the containment. This PDS is responsible for a little more than half of this super-group's mean frequency. In the short-term PDS, on the the other hand, almost all of the early CFs occur at the time of vessel breach. The pool is subcooled in the short-term PDS. Although combustible mixtures can form in the containment before vessel breach in this PDS, the HIS is typically on during core damage and, therefore, the hydrogen generated before vessel breach is usually burned such that the resulting load is benign.

2.5.1.15 Early Drywell Failure. Early drywell failure is an important attribute of the accident progression because failure of the drywell establishes a pathway for radionuclides in the drywell to bypass the suppression pool. Radionuclides are released to the drywell atmosphere at vessel breach and during CCI. In-vessel releases can also enter the drywell if a vacuum breaker sticks open on a SRV tailpipe. Although an intact drywell guarantees that all of the releases will be scrubbed by the pool, drywell failure does not necessarily mean that the radionuclides will be released from the containment, should it fail, without being scrubbed. The in-vessel releases, except from accidents that involve a stuck-open SRV tailpipe vacuum breaker and a failed drywell, are released to the pool where they are scrubbed before entering the containment. Furthermore, if the containment sprays are operating, the ex-vessel releases will be scrubbed by this system. Similarly, if the reactor cavity contains water, which is a likely event, the pool overlaying the core debris will scrub the CCI releases.

Because accidents that result in early drywell failure coincident with early containment failure are generally the dominant risk contributors, it is appropriate to discuss the events that can lead to early drywell failure. Figure 2.5-5 shows the mean probability of drywell failure before vessel breach sorted by events that can lead to drywell failure. Figure 2.5-6 presents the same type of information for drywell failures that occur at vessel breach. These mean values are conditional on core damage; they are not conditional on either vessel breach or early containment failure. In fact, the mean probability of early drywell failure is 0.31, however, the mean probability of coincident early containment and drywell failure is 0.23. Thus, some of the accidents that have early drywell failure do not involve early containment failure. However, these figures provide useful insight into the events that are responsible for early drywell failure.

Before vessel breach the only significant event that causes drywell failure is hydrogen combustion in the wetwell. Although the containment structure is considerably weaker than the drywell wall, rapid deflagrations and detonations in the wetwell can lead to large pressure differentials across the drywell wall which can cause drywell failure. For these rapid combustion events, neither containment failure nor passage of gases through the suppression pool into the drywell occur quickly enough to instigate the

pressure rise from the burn. Slow pressurization events associated with the accumulation of steam in the containment are not a threat to the drywell structure. For the short-term SBO super-group, most of the failures are caused by deflagrations. A relatively small fraction of these failures is caused by detonations. The mean probability of drywell failure before vessel breach is considerably less for the other PDS groups. There are several reasons for the lower failure probability in these groups. In the long-term SBO PDS group the containment is frequently steam inert during this stage of the accident. In the ATWS PDS group, the containment is steam inert in some of the cases and in many of the other cases the HIS is on during core damage. In the T2 PDS group, the HIS is also generally on during the core damage process.

SUMMARY  
ACCIDENT  
PROGRESSION  
BIN GROUP

SUMMARY PDS GROUP  
(Mean Core Damage Frequency)

	STSB (3.85E-06)	LTSB (1.04E-07)	ATWS (1.12E-07)	Transients (1.87E-08)	Frequency Weighted Average (4.09E-06)
DWF: Detonation	0.021			0.002	0.019
DWF: Deflagration	0.104	0.048	0.026	0.055	0.097
No DWF Before VB	0.863	0.951	0.972	0.942	0.874

DWF = Drywell Failure Grand Gulf

Figure 2.5-5. Mean Probability of Drywell Failure Before Vessel Breach.

SUMMARY  
ACCIDENT  
PROGRESSION  
BIN GROUP

SUMMARY PDS GROUP  
(Mean Core Damage Frequency)

	STSB (3.85E-06)	LTSB (1.04E-07)	ATWS (1.12E-07)	Transients (1.87E-08)	Frequency Weighted Average (4.09E-06)
DWF: Loads Accompanying VB	0.084	0.158	0.156	0.146	0.088
DWF: Pedestal Fail. (Loads Accomp. VB)	0.051	0.090	0.109	0.096	0.058
DWF: Pedestal Fail. (Dynamic Loads)	0.019	0.018	0.007	0.008	0.016
DWF: Detonation	0.017		0.003	0.008	0.016
DWF: Deflagration	0.022	0.011			0.019
Alpha	0.005	0.006	0.002	0.002	0.005
DWF: Before VB	0.106	0.043	0.026	0.055	0.099
No Early DWF	0.686	0.672	0.696	0.683	0.688

DWF = Drywell Failure

Grand Gulf

Figure 2.5-6. Mean Probability of Drywell Failure at Vessel Breach.

For drywell failures that occur at vessel breach, loads accompanying vessel breach are responsible for the majority of these failures. These quasi-static loads, which were provided by the Containment Loads Expert Panel, include contributions from: DCH, ex-vessel steam explosions, hydrogen burns, and RPV blow down. At vessel breach these events pressurize the drywell volume before the suppression pool vents clear. Drywell failures caused by these loads are responsible for nearly 50% of the mean probability of drywell failure at vessel breach. In addition to directly pressurizing the drywell volume, these loads can also pressurize the reactor cavity and fail the pedestal. In some cases loss of reactor support can induce drywell failure. This is the second event in Figure 2.5-5 that causes drywell failure and it is responsible for almost 30% of the mean probability of drywell failure at vessel breach. As can be seen in this figure, alpha mode events are a negligible contributor to the mean probability of early drywell failure.

2.5.1.16 Summary. Figure 2.5-7 shows the mean distribution among the summary accident progression bins for the PDS super-groups. Only mean values are shown, so Figure 2.5-7 gives no indication of the range of values encountered. These mean values are conditional on core damage. The

distribution for core damage arrest is shown in Figure 2.5-1, and the distribution for early (at or before vessel breach) failure of the containment is shown in Figure 2.5-2. Nonetheless, Figure 2.5-7 gives a good idea of the relative likelihood of the possible results of the accident progression analysis. The summary bins are composed of essentially four characteristics: occurrence of vessel breach, timing of containment failure, timing of suppression pool bypass, and the availability of the containment sprays. The summary bins are listed roughly in decreasing order of the severity of the resulting source term. The last two bins are an exception to this ordering scheme. Because there are some accidents in the NO vessel breach summary bin that have early containment failure, the releases associated with this bin are higher than releases for the vessel breach, No CF summary bin. A description of these summary bins is presented in section 2.4.3.

Because roughly 90% of the total mean core damage frequency is attributed to the short-term SBO super-group, the results presented in the frequency weighted average column are heavily influenced by the short-term SBO results. If the accident proceeds to core damage, containment failure during the accident is a likely outcome. The mean conditional probability of early containment failure is approximately 0.50 and half of this mean value is associated with accidents that also involve some bypass of the suppression pool (i.e., drywell failure). If the accident proceeds to vessel breach and the containment does not fail early, there is still a fairly high probability that the containment will fail late in the accident. Events that can fail the containment late in the accident are hydrogen burns and the accumulation of noncondensibles and steam in the containment. In the SBO PDSs ac power may not be available late in the accident and, thus, the containment sprays will not be available to condense the steam. Furthermore, even if the sprays are available, the accumulation of noncondensibles generated at vessel breach and during CCI may still fail the containment. Containment venting is not a likely outcome in this analysis. There are several reasons for this result. First, the dominant PDSs are the short-term station blackouts. In these PDSs, the suppression pool remains subcooled during core damage and, therefore, the containment is not pressurized by the accumulation of steam. During core damage and after vessel breach a significant quantity of radionuclides will be released to the containment. After vessel breach it is unlikely that the operator will vent these releases to the outside environment.

The first two summary bins represent accidents in which vessel breach occurs and both the containment and the drywell fail early. The only difference between the first two bins is the availability of the containment spray system. For accidents characterized by the first bin, the majority of the ex-vessel releases will not be scrubbed by either the suppression pool or the containment sprays whereas releases associated with the second bin will be scrubbed by the sprays. For the SBO PDSs, the first bin is a more likely outcome than the second bin because in many cases ac power is not available. The opposite trend is observed for the ATWS and T2 PDSs because ac power is never lost in these PDSs.

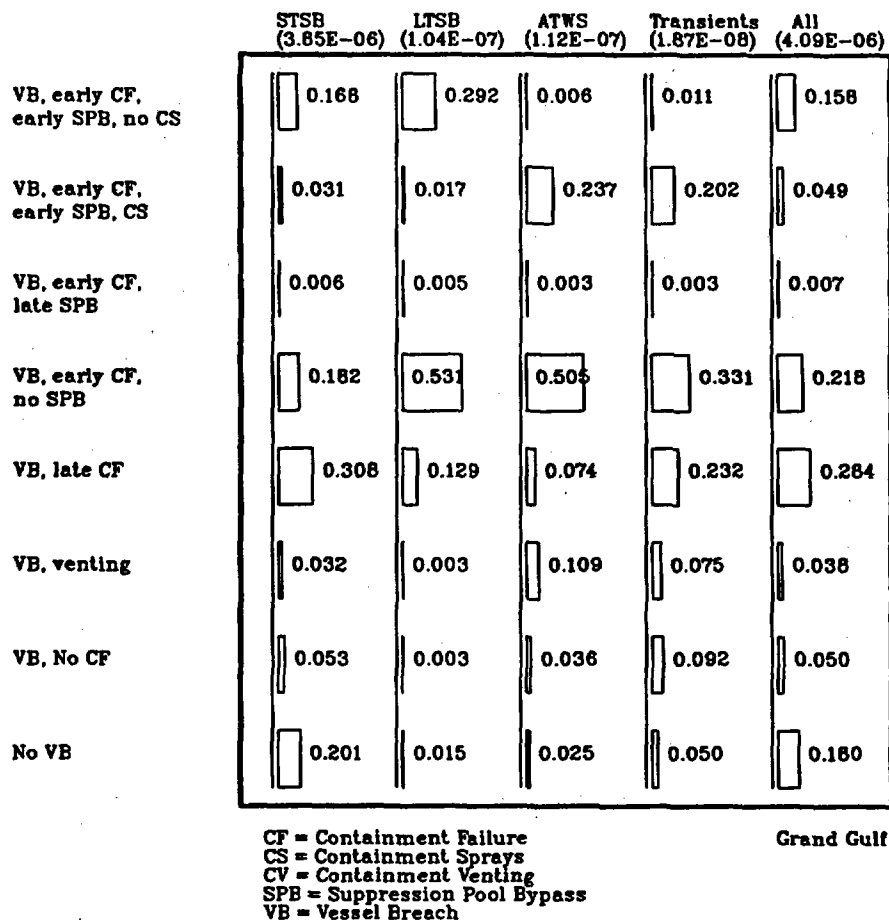


Figure 2.5-7. Mean Probability of APBs for the Summary PDSs.

The bin that involves vessel breach, early CF, and no suppression pool bypass is a likely outcome for both the long-term SBO super-group and the ATWS super-group. The reason for this result is that many of these CFs are caused by the accumulation of steam in the containment. This slow type of pressurization event can fail the containment but does not pose a threat to the drywell structure. For accidents characterized by this bin, both the in-vessel and ex-vessel releases will be scrubbed by the suppression pool.

The short-term SBO super-group is the only group that has a significant probability of core damage arrest. The mean probability that vessel breach is averted in this group is 0.20. Although a quarter of this mean value is associated with accidents in which core damage is arrested, the majority of the in-vessel releases are directed to the suppression pool where they are scrubbed before entering the containment.

For the accidents in which the core damage process is not arrested and the accident proceeds to vessel breach, there is a significant probability that core debris is cooled and there are no CCI releases. If CCI is initiated it will most likely occur in a flooded cavity. CCI releases that occur in



a flooded reactor cavity will be scrubbed by the overlaying pool of water. The mean conditional probabilities for the cases with no CCI and the cases where CCI occurs with an overlaying pool of water are:

PDS	<u>CCI with Overlaying</u>		
	<u>No CCI</u>	<u>Pool of Water</u>	<u>Dry CCI</u>
<b>Short-Term SBO PDS</b>			
1	0.64	0.35	0.01
2	0.64	0.35	0.01
3	0.61	0.38	0.01
7	0.77	0.21	0.02
<b>Long-Term SBO PDS</b>			
4	0.45	0.52	0.03
5	0.45	0.52	0.03
6	0.41	0.55	0.03
8	0.69	0.20	0.11
<b>ATWS PDS</b>			
9	0.76	0.24	<0.01
10	0.62	0.21	0.17
<b>T2 PDS</b>			
11	0.76	0.24	<0.01
12	0.76	0.24	<0.01

These mean values are conditional on core damage. Furthermore, the no CCI case includes the accident progressions in which vessel breach is arrested.

Figure 2.5-7 shows the mean frequencies for the summary PDS groups and mean conditional probabilities for the summary APBs, where the mean is taken over all 250 observations in the sample. 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 2.5-8.

## 2.6 Insights from the Accident Progression Analysis

Several insights can be drawn from the accident progression analysis. First, for the PDSs analyzed in this study, containment failure during the accident is a likely outcome. The predominant causes of these failures are hydrogen deflagrations. In the short-term SBO PDS group, which is responsible for roughly 90% of the total mean core damage frequency at Grand Gulf, ac power is not available early in the accident and, therefore, the HIS is not available at the beginning of core damage for the vast majority of the accident analyzed. Without the HIS the hydrogen that is

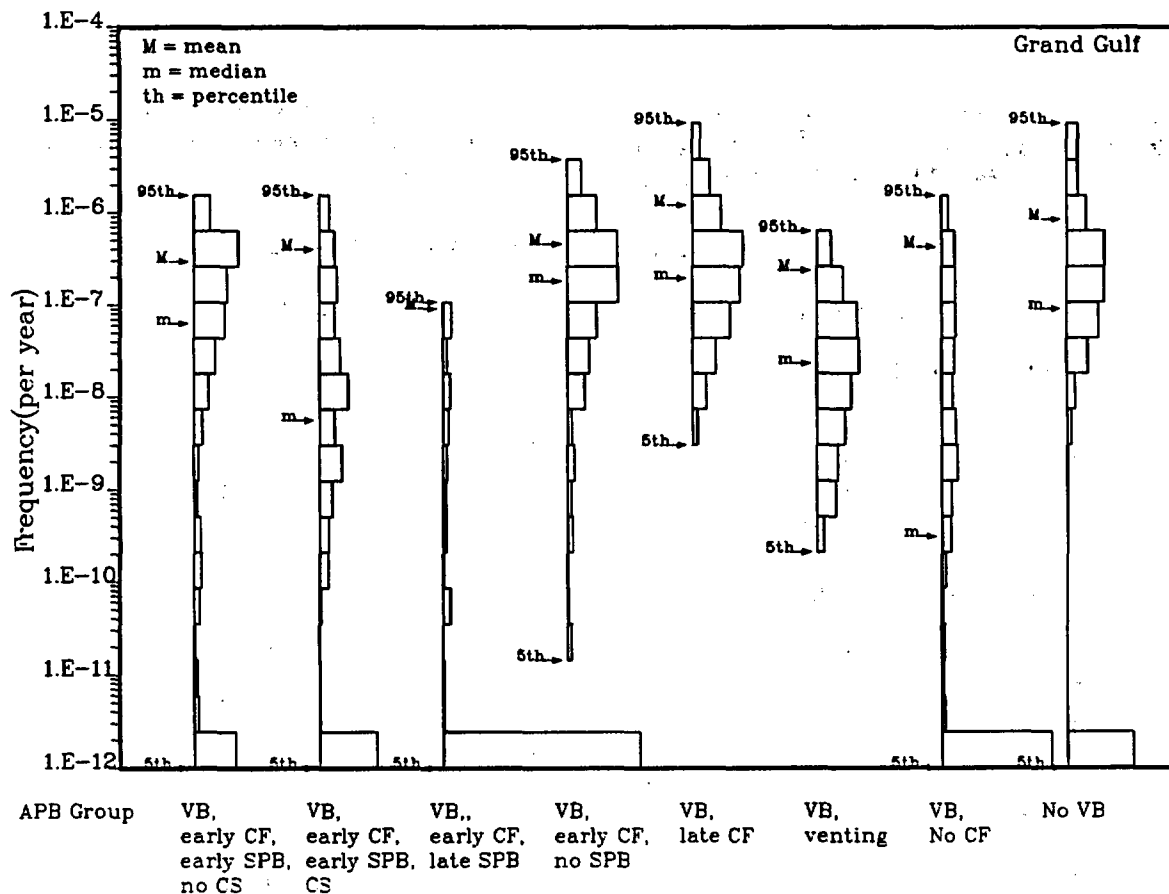


Figure 2.5-8. Distribution of Frequencies for APB Groups.

produced from the oxidation of zirconium during the core damage process can accumulate in the containment. Subsequent ignition of this hydrogen by either random sources or by the recovery of ac power can result in loads that cannot only threaten the containment but they can also pose a significant challenge to the drywell structure. For the PDSs analyzed in this study, the mean conditional probability of early containment failure is nearly 0.5. Furthermore, half of this mean value comes from accidents that also involve some bypass of the suppression pool. Increased availability of the HIS will significantly reduce the probability of both containment and drywell failure before vessel breach. However, because of the weakness of the containment and the potential for the rapid combustion of hydrogen at vessel breach, the containment will still be susceptible to loads at vessel breach. Furthermore, the integrity of the drywell will still be challenged by loads accompanying vessel breach.

The results of the analysis to determine whether there is water in the reactor cavity, as described in Appendix A.1, indicate that there is a high likelihood that the cavity will contain water at vessel breach. The presence of water in the cavity is important, and has both advantages and disadvantages. The presence of water allows for the possibility of ex-vessel steam explosions. On the other hand, this water also contributes to the high 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.

## 2.7 References

1. M. T. Drouin, J. L. LaChance, B. J. Shapiro, S. Miller, and T. A. Wheeler, "Analysis of Core Damage Frequency: Grand Gulf, Unit 1 Internal Events," NUREG/CR-4550, Volume 6, SAND86-2084, Sandia National Laboratories, September 1989.
2. D. M. Ericson, Jr., Editor, T. A. Wheeler, T. T. Sype, M. T. Drouin, W. R. Cramond, A. L. Camp, K. J. Maloney, and F. T. Harper, "Analysis of Core Damage Frequency: Internal Events Methodology," NUREG/CR-4550, Volume 1, SAND86-2084, Sandia National Laboratories, January 1990.
3. J. M. Griesmeyer and L. N. Smith, "A Reference Manual for the Event Progression Analysis Code (EVNTRE)," NUREG/CR-5174, SAND88-1607, Sandia National Laboratories, September 1989.
4. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," Second Draft for Peer Review, NUREG-1150, June 1989.

### 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. 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, and additional information about the timing of the releases, the energy associated with the releases, and the height of the releases.

Source term analysis is performed by a relatively small computer code: GGSOR. The aim 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, the purpose is to represent the results of the more detailed codes that do consider these quantities.

A more complete discussion of the source term analysis, and of GGSOR in particular, may be found in NUREG/CR-5360.\* The methods on which GGSOR 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 Grand Gulf plant that are important to the magnitude of the radionuclide release. Section 3.2 presents a brief overview of the GGSOR 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 Grand Gulf Features Important to the Source Term Analysis

Grand Gulf Unit 1 is a boiling water reactor-6 (BWR-6) that is housed in a Mark III containment. The containment is a reinforced concrete structure with a steel liner. The RPV is located inside the drywell which is in turn surrounded by the containment structure. The drywell volume communicates to the wetwell volume through the suppression pool.

The primary barrier between the radionuclides released from the core and the outside environment is the containment structure. The containment structure has a design pressure of 15 psig and an assessed mean failure pressure of 55 psig. Because of this relatively low failure pressure (relative to the loads that are imposed on it during the course of the accident), it was determined during the accident progression analysis that the containment is likely to fail during accidents that progress to core damage. In fact, the containment fails early in roughly half of the accident progressions analyzed. The drywell structure is considerably

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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).

stronger than the containment. The design pressure of the drywell from internal pressurization is 30 psid. Nevertheless, the drywell is still susceptible to the loads that occur from hydrogen combustion events and from pressurization events accompanying vessel breach. Of the accidents that result in early containment failure, half also involve early drywell failure.

Although the results of this study indicate that the containment is likely to fail, there are a number of plant characteristics that help to reduce the amount of radionuclides that can potentially be released to the environment. Because of the suppression pool's ability to effectively trap radionuclides, it provides the potential for substantial mitigation of the source term in severe accidents. In addition to the suppression pool, other features that can potentially reduce the source term are the containment sprays and the reactor cavity pool.

There are two pathways by which radionuclides enter the suppression pool. The first pathway is through the SRV tail pipes. Because the dominant contributors to the core damage frequency were transient initiated events (i.e., LOCAs were not analyzed in the accident progression analysis) the in-vessel releases exit the vessel via the steam lines, pass through the SRV tail pipes, and are then discharged into the suppression pool through the T-quenchers at the end of the tail pipes. For the in-vessel releases to bypass the suppression pool a SRV tail pipe vacuum breaker must stick open during core damage and the drywell must be failed. If the drywell is not failed, the releases will enter the drywell volume and then will be directed to the suppression pool via the horizontal vents. These horizontal vents are the second pathway for radionuclides to enter the suppression pool. If the drywell is intact, the ex-vessel releases will also enter the suppression pool via this pathway. The first pathway is more effective than the second pathway at trapping radionuclides. However, the second pathway still offers a significant mechanism for mitigating the source term.

The containment sprays can also be effective at reducing the amount of airborne radionuclides. Because the dominant PDSs are short-term SBOs, the sprays are generally not on before core damage. The unavailability of the sprays early in the accident is not particularly important because as mentioned previously, the majority of the in-vessel releases pass through the suppression pool. In the dominant short-term SBO PDS it is likely that the sprays will be on after vessel breach and, therefore, any release from CCI will be scrubbed. The decontamination factor (DF) associated with the sprays is roughly the same as the DF associated with the suppression pool when the radionuclides enter through the horizontal vents.

The Grand Gulf reactor cavity is roughly a right cylindrical volume that is located directly below the RPV. This volume is large enough to contain the core debris that is released from the RPV should vessel breach occur. (However, energetic events such as DCH and ex-vessel steam explosions can disperse core debris outside the cavity.) Thus, unlike a Mark I containment, the core debris generally remains in the reactor cavity. Because of the geometry of the Mark III containment, it is likely that the

cavity will contain water at the time of vessel breach. Water can enter the drywell when pressurization events in the wetwell depress the suppression pool sufficiently such that water is pushed up over the weir wall. The amount of water in the drywell depends on whether the upper water pool has been dumped and on the transient pressurization of the containment. During long-term PDSs leaking equipment (e.g., recirculation pumps) can also be an important source of water. Water in the drywell can enter the cavity either through the drain in the drywell floor or through a door in the pedestal wall. The presence of water in the cavity is important for three reasons. First, if there is a large amount of water in the cavity it is possible that the core debris that is released from the cavity will be cooled and, therefore, CCI will not be initiated. Second, if CCI is initiated following vessel breach and the cavity contains water, the pool above the core debris will scrub the CCI releases. Third, ex-vessel steam explosions at vessel breach are possible if the cavity contains water. An ex-vessel steam explosion will increase the amount of airborne radionuclides in the drywell. The first two effects of cavity water mitigate the source term. The last effect increases the radionuclide release. Thus, the presence of water can be both beneficial and detrimental.

### 3.2 Description of the GGSOR Code

This section describes the manner in which the source term is computed for each accident progression bin (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 GGSOR may be found in Reference 1. Section 3.2.3 presents the parameters sampled in the source term portion of this analysis.

#### 3.2.1 Overview of the Parametric Model

GGSOR is a fast-running, parametric computer code used to calculate the source terms for each APB for each observation for Grand Gulf. As there are typically a few thousand bins for each observation, and 250 observations in the sample, the need for a source calculation method that requires a minimum of computer time for one evaluation is obvious. GGSOR is not designed to calculate the behavior of the fission products from their basic chemical and physical properties and the flow and temperature conditions in the reactor and the containment. The purpose of GGSOR is to provide a framework for integrating the results of the more detailed codes that do consider these quantities. Since many of the parameters GGSOR utilizes to calculate the release fractions were determined by a panel of experts, the results of the detailed codes enter GGSOR "filtered" through 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 sixty isotopes are placed in nine radionuclide classes as shown in Table 3.2-1. It is these nine classes which are treated individually in the source term analysis.

Table 3.2-1  
Isotopes in Each Radionuclide Release Class

<u>Release Class</u>	<u>Isotopes Included</u>
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
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 GGSOR

Since the consequences will generally depend on the timing of containment failure, GGSOR considers three time regimes in which the containment can fail: before vessel breach, at or near the time of vessel breach, and late in the accident. Furthermore, GGSOR considers two releases from the containment. The first release occurs roughly at the time of containment failure (assuming the containment fails after core damage). The second release begins after the first release has finished (unless CCI initiation is delayed in which case the second release is also delayed). When the containment fails before vessel breach, the first release is due to fission products that escape from the fuel while the core is still in the RPV (i.e., in-vessel releases). For this case, the second release includes fission products that are released at the time of vessel breach and after vessel breach. Releases after vessel breach include fission products from CCI releases, material revolatilized from the RPV after vessel breach and iodine released from the suppression pool (and in some cases the RPV cavity

water). These releases will be referred to as the late releases. When the containment fails around the time of vessel breach the first release includes in-vessel releases as well as fission products that are released at the time of vessel breach. The second release is due to the late releases. For situations where the containment fails many hours after vessel breach, both releases consist of in-vessel releases, fission products released at vessel breach, and the late releases. The timing and duration of these releases depend primarily on the PDS and the time and mode of containment failure.

For radionuclide class  $i$ , the basic parametric equation for GGSOR has the following form:

$$ST_i = \text{(Eq. 3.1)}$$

$$\begin{aligned}
 &= FCOR_i * FVES_i * (REL1 + REL2 + REL3) * FCONV_i \\
 &+ VBPUF_i * (REL4 + REL5) * FCONC_i \\
 &+ (1 - FCOR_i - VBPUF_i) * FLV * FHPE * FDCH_i * (REL6 + REL7) * FCONC_i \\
 &+ (1 - FCOR_i - VBPUF_i) * FLV * FHPE * FEVSE_i * (REL6 + REL7) * FCONC_i \\
 &+ (1 - FCOR_i - VBPUF_i) * FLV * XCCI * FCCI_i * (REL8 + REL9) * FCONC_i \\
 &+ FCOR_i * (1 - FVES_i) * FREVOL_i * (REL10 + REL11) * FCONC_i \\
 &\quad i=2, 3, \text{ \& } 4 \text{ ONLY} \\
 &+ [FLT11 * POOLI + FLT12 * CAVWI * (REL12 + REL13)] * REL14
 \end{aligned}$$

where

- REL1 -  $F T L P * F P L B Y E / D F S P R V_i$
- REL2 -  $F T L P * (1 - F P L B Y E) / M A X (D F C P A_i, D F S P R V_i)$
- REL3 -  $(1 - F T L P) / M A X (D F V P A_i, D F S P R V_i)$
- REL4 -  $F P L B Y P / D F S P R C_i$
- REL5 -  $(1 - F P L B Y P) / M A X (D F C P A_i, D F S P R C_i)$
- REL6 -  $F P L B Y D / D F S P R C_i$
- REL7 -  $(1 - F P L B Y D) / M A X (D F C P A_i, D F S P R C_i)$
- REL8 -  $F P L B Y C / M A X (D F C A V_i, D F S P R C_i)$
- REL9 -  $(1 - F P L B Y C) / M A X (D F C A V_i, D F C P A_i, D F S P R C_i)$
- REL10 -  $F P L B Y C / D F S P R C_i$
- REL11 -  $(1 - F P L B Y C) / M A X (D F C P A_i, D F S P R C_i)$
- REL12 -  $F P L B Y C / D F C P A_i$
- REL13 -  $(1 - F P L B Y C) / D F C P A_i$
- REL14 -  $F C O N C_i$  if no containment failure  
           - 1.0 if containment failure
- XCCI - 1 - FHPE if DCH or ex-vessel steam explosion occurs  
        - 1.0 if neither DCH nor ex-vessel steam explosion occurs.

The first summation term on the right side of Equation 3.1 represents the in-vessel releases. The second term describes the puff release at vessel breach. The third term represents the DCH release. The fourth term represents the ex-vessel steam explosion release and is mutually exclusive with the third term (i.e., the experts said if DCH occurred, then the ex-vessel steam explosion release should not be considered separately). The fifth term represents the CCI release. The sixth term is the revolatilization release from the reactor coolant system after vessel breach and is for I, Cs, and Te classes only. The last term



represents the late iodine release from the suppression pool and reactor cavity water after the containment fails. The definitions of the various parameters in Equation 3.1 are as follows:

- CAVWI - fraction of initial iodine core inventory scrubbed by the cavity water during CCI release
- DFSPRC<sub>1</sub> - scrubbing decontamination factor for sprays acting on species i released into containment after vessel breach
- DFSPRV<sub>1</sub> - scrubbing decontamination factor for sprays acting on species i released into containment from the vessel before vessel breach
- DFCAV<sub>1</sub> - scrubbing decontamination factor for aerosol species i released into cavity water during CCI release
- DFCPA<sub>1</sub> - scrubbing decontamination factor for aerosol species i flowing from drywell to the suppression pool
- DFVPA<sub>1</sub> - scrubbing decontamination factor for aerosol species i flowing from the vessel to the suppression pool
- FCI<sub>1</sub> - fraction of material released from the melt during molten CCI
- FCNC<sub>1</sub> - fraction of species i released from containment for CCI and other releases after vessel breach, not including the effects of scrubbing by pools and sprays
- FCNV<sub>1</sub> - fraction of species i released from containment for material released into containment before vessel breach, not including the effects of scrubbing by pools and sprays
- FCOR<sub>1</sub> - fraction of initial inventory of species i released from the fuel prior to vessel failure
- FDCH<sub>1</sub> - fraction of radionuclide in the portion of the core involved in direct containment heating that is released to the drywell at vessel breach.
- FEVSE<sub>1</sub> - fraction of radionuclides in the portion of the core involved in an ex-vessel steam explosion that is released to the drywell at vessel breach
- FHPE - fraction of core material leaving the vessel that participates in either the direct containment heating or the steam explosion and therefore is not available for molten CCI release later

- FLV - fraction of the core material that leaves the vessel after the vessel breach
- FREVOL<sub>1</sub> - fraction of the core material that is deposited on the surfaces of the reactor vessel and structural materials that is revaporized and released in the drywell after vessel breach
- FPLBYC - fraction of CCI releases that bypass the suppression pool
- FPLBYD - fraction of DCH releases or ex-vessel steam releases that bypass the suppression pool
- FPLBYE - fraction of in-vessel releases that bypass the suppression pool
- FPLBYP - fraction of puff releases at vessel breach that bypass the suppression pool
- FTLP - fraction of the in-vessel releases that are released into the drywell through stuck-open SRV tailpipe vacuum breaches
- FVES<sub>1</sub> - fraction of material released from the fuel that is released from the vessel
- FLT11 - fraction of iodine in the suppression pool that is volatilized and released after vessel breach
- FLT12 - fraction of iodine in the cavity water that is volatilized and released after vessel breach
- POOLI - fraction of initial core inventory for iodine scrubbed by the pool
- ST<sub>1</sub> - fraction of the initial core inventory of species i that is ultimately released to the environment
- VBPUF<sub>1</sub> - fraction of initial core inventory of species i that is released to the drywell as puff at the time of vessel breach
- XCCI - fraction of core material that leaves the vessel that participates in CCI.

It is expected that accompanying containment failure a substantial portion of the enclosure building at Grand Gulf will fail. Thus, no credit is given for retention of radionuclides in the enclosure building. A detailed discussion of this equation is presented in NUREG/CR-5360.\* The FORTRAN listing of GGSOR is contained in Appendix B.

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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).

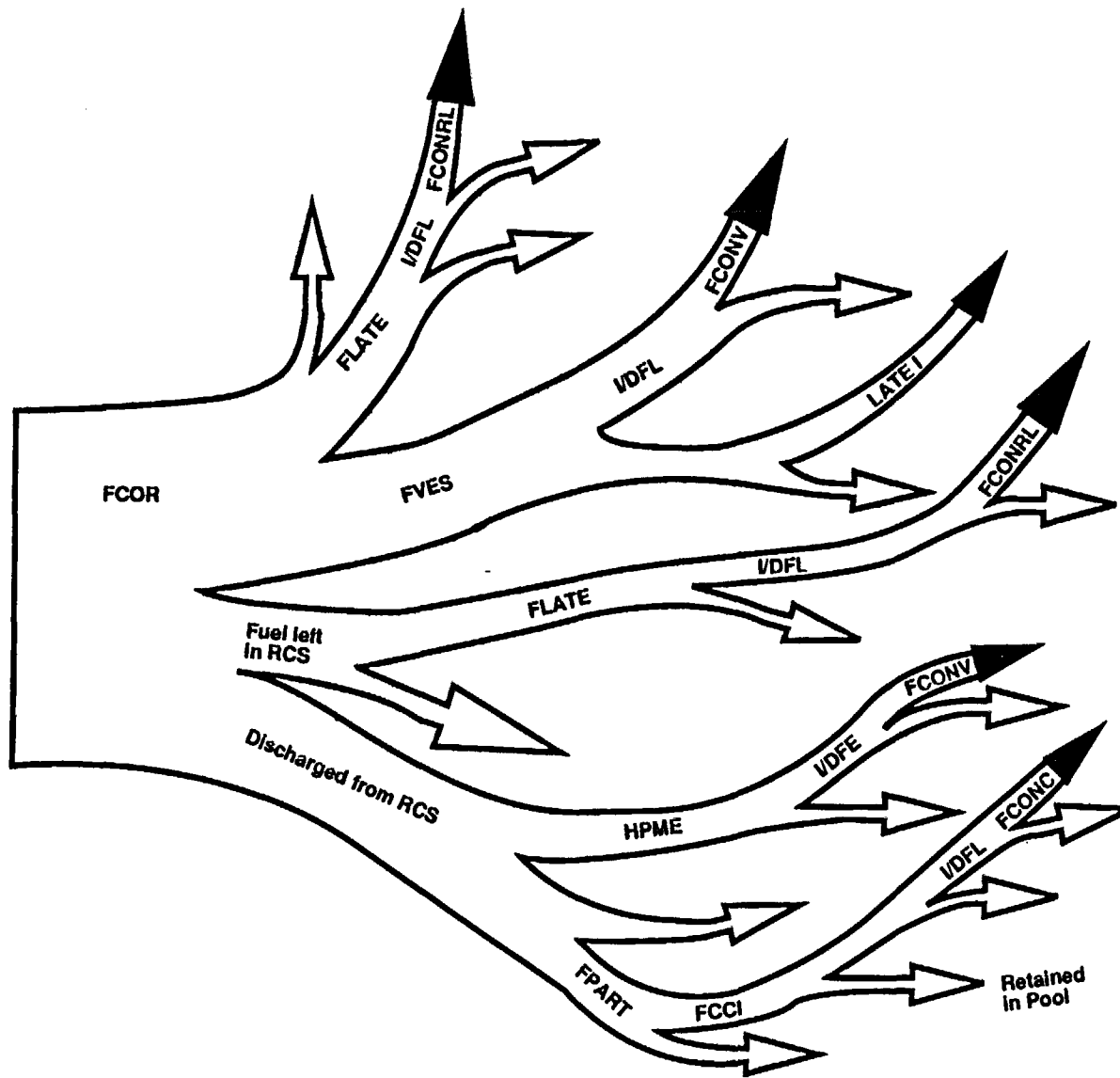


Figure 3.2-1. Blood Flow Diagram for GGSOR.

Figure 3.2-1 depicts the parametric equations schematically in terms of 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. This figure is read as follows: the first division of the radioactive material is indicated by FCOR. The top branch, indicated by FCOR, represents the fraction released from the core before vessel breach, and the lower branch, an amount  $1-FCOR$ , represents the amount still in the RCS at vessel breach. The FCOR branch is then split into that which leaves the RCS before or at vessel breach, FVES, and that which is retained in the RCS past vessel breach,  $1-FVES$ . Of the material retained in the RCS at vessel breach, a fraction FLATE is revolatilized later. Of the revolatilized fraction, a portion is removed by engineered removal mechanisms such as sprays, parameter  $1/DFL$ , and another portion is removed by natural mechanisms such as deposition, parameter FCONRL. The part of the revolatilized fraction that is not removed escapes to the environment 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, GGSOR is run in the "sampling mode". That is, most of the parameters in the release fraction equations are determined by sampling from distributions for that parameter, and the value for each parameter varies from observation to observation. Many of these distributions were provided by an expert panel.

The equation above contains 25 parameters. Nine of them were considered by the Source Term Expert Panel. An additional 12 parameters were quantified either by the expert panel for the previous draft of this report or internally. The values for four of these parameters (i.e., CAVWI, FLV, POOLI, XCCI) are determined by various combinations of previously defined parameters.

Many of these parameters in the equation above are determined directly by sampling from distributions provided by a panel of experts, see NUREG/CR-4551, Volume 2, Part 4. Other parameters are derived from such values, and still others were determined internally, see the XSOR document.\*

### 3.2.3 Variables Sampled in the Source Term Analysis

The twelve parameters sampled for the source term analysis are listed in Table 3.2-2. When GGSOR was evaluated for all the bins generated by the APET evaluation for a given observation, all the sampled parameters in GGSOR had values chosen specifically for that observation. These values were selected by the Latin Hypercube Sampling (LHS) program from distributions that were previously defined. Many of these distributions were determined by the expert panel on source terms. Eight issues were considered by the Source Term Expert Panel:

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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).

1. FCOR and FVES
2. Ice Condenser DF (not applicable to Grand Gulf)
3. Late Releases from the RPV
4. FCCI
5. FCONV and FCONC
6. Late Iodine
7. Reactor Building DF (not applicable to Grand Gulf)
8. DCH Releases

Table 3.2-2  
Variables Sampled in the Source Term Analysis

<u>Variable</u>	<u>Description</u>
FCOR	Fraction of each fission product group released from the core to the vessel before vessel breach. There are two cases: high and low zirconium oxidation. This parameter was assessed by the Source Term Expert Panel.
FVES	Fraction of each fission product group released from the core which is released from the vessel. There are three cases: short-term SBO with the RPV at system pressure, short-term SBO with the RPV at low pressure, and ATWS with the RPV at system pressure. This parameter was assessed by the Source Term Expert Panel.
FREVO	Fraction of the deposited amount of each fission product group in the RPV which revolatilized after vessel breach and released to the drywell. There are three cases: no water injection after vessel breach and a high drywell temperature, no water injection after vessel breach and low drywell temperature, and water injection to the vessel after vessel breach. This parameter was assessed by the Source Term Expert Panel.
FCONV	Fraction of each fission product group released from containment for material released into containment before vessel breach, not including the effects of scrubbing by pools and sprays. There are six cases: early containment leakage and a subcooled suppression pool, early containment leakage and a saturated suppression pool, early containment rupture and a subcooled suppression pool, early containment rupture and a saturated suppression pool, late containment leak, and late containment rupture. This parameter was assessed by the Source Term Expert Panel.
FDCH	Fraction of each fission product group in the core material that participates in a direct containment heating event (DCH) that is released to the drywell. Given the occurrence of DCH, there is only one case. This parameter was assessed by the Source Term Expert Panel.

Table 3.2-2 (continued)

Variable	Description
FEVSE	<p>Fraction of each fission product group in the core material that participates in an ex-vessel steam explosion that is released to the drywell. Given the occurrence of an ex-vessel steam explosion, there is only one case. This parameter was not assessed by the Source Term Expert Panel. It is assumed that the release fractions for the ex-vessel steam explosion phenomena are sufficiently similar to the release fractions associated with DCH that the DCH distributions are also used to quantify this parameter.</p>
FCCI	<p>Fraction of each fission product group in the the core material at the start of CCIs that is released to the drywell. There are four cases: low zirconium oxidation in the core and no overlaying water, low zirconium oxidation in the core with overlaying water, high zirconium oxidation in the core and no overlaying water, and high zirconium oxidation in the core with overlaying water. This parameter was assessed by the Source Term Expert Panel.</p>
FCONC	<p>Fraction of each fission product group released from the containment for CCI and other releases after vessel breach, not including the effects of scrubbing by pools and sprays. There are six cases: early containment leakage and a subcooled suppression pool, early containment leakage and a saturated suppression pool, early containment rupture and a subcooled suppression pool, early containment rupture and a saturated suppression pool, late containment leak, and late containment rupture. This parameter was assessed by the Source Term Expert Panel.</p>
FLTI	<p>This variable in the LHS sample is used for both FLTI1 and FLTI2 (i.e., completely correlated). These parameters were assessed by the Source Term Expert Panel.</p>
	<p>FLTI1: Fraction of iodine in the suppression pool that is volatilized and released after vessel breach. There are two cases: the suppression pool is subcooled and the suppression pool is saturated.</p>
	<p>FLTI2: Fraction of iodine in the cavity water that is volatilized and released after vessel breach. There are two cases: the reactor cavity is flooded and the reactor cavity is wet.</p>

Table 3.2-2 (continued)

<u>Variable</u>	<u>Description</u>
DFPOOL	<p>This variable in the LHS sample is used for both DFVPA and DFCPA (i.e., completely correlated). This issue was not assessed by the Source Term Expert Panel. The distributions for these parameters were obtained from the draft report of NUREG/CR-4551, Volume 7.</p> <p>DFVPA: Decontamination factor for in-vessel releases that are released into the suppression pool.</p> <p>DFCPA: Decontamination factor for aerosol releases flowing from the drywell to the suppression pool.</p>
DFCAV	<p>Decontamination factor for aerosols released into the cavity water the CCI release. This DF is applied when the core debris is not coolable and CCI proceeds under water. There are two cases: the reactor cavity is flooded and the reactor cavity is only partially filled with water. This issue was not assessed by the Source Term Expert Panel. The distributions for this parameter were obtained from the draft report of NUREG/CR-4551, Volume 7.</p>
DFSPRAY	<p>This variable in the LHS sample is used for both DFSPRV and DFSPRC (i.e., completely correlated). This issue was not assessed by the Source Term Expert Panel. The distributions for these parameters were obtained from the draft report of NUREG/CR-4551, Volume 8.</p> <p>DFSPRV: Decontamination factor for sprays acting on fission product groups released into the containment from the vessel.</p> <p>DFSPRC: Decontamination factor for sprays acting on fission product groups released into the containment after vessel breach.</p>

Two of these issues are not applicable to Grand Gulf. For each issue considered by the expert panel, the result is an aggregate distribution for the nine radionuclide release classes defined in Table 3.2-1. These distributions are not necessarily discrete. While the experts provided separate distributions for all nine classes for FCOR, for other parameters, for example, they stated that classes 5 through 9 should be considered together as an aerosol class.

The sampling process works somewhat differently for the source term analysis than it does for the accident progression analysis. In the source term analysis, LHS was used only to determine a random number between 0.0

and 1.0 for each parameter to be sampled. The actual distributions are contained in a data file (listed in Appendix B) that is read by GGSOR before execution.

The variable identifiers given in Table 3.2-2 are used in several ways in the source term analysis. Consider the first variable in Table 3.2-2: FCOR. 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 vessel breach for the 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. Furthermore, in the sampling process, FCOR is used to refer to the random number from the LHS, which is used to select the values from these distributions. That means that, as used in sampling, FCOR defines a quantile in these distributions. The release fractions associated with this quantile are used in GGSOR 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 variables selected by LHS are used to define quantiles in the parameter distributions; the values associated with these quantiles are used as parameter values in GGSOR. In use, the process works like this. Suppose LHS selects a value of 0.05 for FCOR for Observation 1. Referring to the data tables in Appendix B.2, it may be seen that, for low zirconium oxidation in-vessel, the 0.05 quantile values for FCOR are 0.084 for inert gases, 0.009 for I, 0.009 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 as there are in Table 2.3-2. 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 GGSOR are listed in Table 3.2-2; parameter values for other cases are determined internally in GGSOR, often from the values for the cases listed. For example, there is no distribution for FVES for long-term SBOs. The value of FVES for the long-term SBOs was derived from the distributions for other cases.

For each parameter that was assessed by the Source Term Expert Panel, the distribution for the parameter, 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. The distributions for the



remaining parameters are presented in Appendix B. A discussion of these parameters may be found in NUREG/CR-5360.\*

### 3.3 Results of 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 a sample of the more likely and more important APBs are discussed here. However, source terms were computed for all the APBs for each of the 250 observations in the sample. The source term is composed of release fractions for the nine radionuclide groups for a first and a second release as well as release timing, release height, and release energy. As discussed above, the source terms are computed by a fast-running parametric computer code, GGSOR.

For purposes of readability, all tables and figures related to this subsection appear at the end of it.

Section 3.3.1 presents the results for the internal initiators. The tables in this section present 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

In a manner analogous to Section 2.5.1, the results of the source term analysis for internal initiators are presented for each PDS group. The tables in this section only provide a sample of APBs and their associated mean source terms for the various PDSs.

3.3.1.1 Results for PDS 1: Short-Term SBO. As discussed in Section 2.5.1.1, this PDS involves SBO scenarios where LOSP is recoverable. Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure because depressurization did not have an effect in the prevention of core damage (the operators can depressurize the RPV during core damage). If offsite power is restored then coolant injection to the RPV, containment sprays and the HIS are all available. For this PDS the mean probability that vessel breach is averted is 0.32. The mean probability that the containment fails early (early is defined as before or around the time of vessel breach) is 0.36.

Table 2.5-1 lists the five most probable APBs for PDS 1, the five most probable APBs that have vessel breach, and the five most probable APBs that have early CF. Table 3.3-1 lists the mean source terms for these same APBs. Although the same bins are shown in both tables, and the structures of both tables are roughly analogous, there are some important differences in the nature of the material presented. In Table 2.5-1, the bin itself was well defined, i.e., the characteristics of the bin did not vary from

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

observation to observation. The only item in the table 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 250 observations in the sample. In Table 3.3-1, the bin is still well defined, but, as many of the parameters that are 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 250 observations in the sample.

For example, consider the first APB in Table 3.3-1: ABBDDGCCB. Of the 250 observations in the sample, 75 had non-zero conditional probabilities for this bin. As source terms are not computed for zero-probability bins, there are 75 source terms associated with APB ABBDDGCCB. These 75 source terms were summed and then divided by 75 to produce the mean source term given in the first two lines of Table 3.3-1.

The most probable APB, ABBDDGCCB, involves accidents that proceed to vessel breach. Once vessel breach occurs the core debris is released into the reactor cavity and CCI takes place under a pool of water. For this APB the containment is ruptured late in the accident. When the containment fails in the rupture mode late in the accident, GGSOR groups 90% of the radionuclides that are available to be released from the containment (i.e., those radionuclides that have not been trapped by the water pools or plated out in the vessel or containment) in the first release and the remaining 10% in the second release. The next four most probable APBs involve accidents that do not proceed to vessel breach (i.e., no ex-vessel releases) and the containment either fails late in the accident or does not fail. As a result the releases associated with these bins are considerably less than those associated with the most probable APB. When the containment develops a leak late in the accident (e.g., fourth most probable bin - ABEEAFCEB), GGSOR releases 50% of the radionuclides from the containment in the first release and the remaining 50% in the second release.

For APBs that involve accidents that do not proceed to vessel failure but do result in early containment failure, all of the radionuclides, except iodine, are grouped in the first release. Iodine that is released from the vessel that is not trapped in the suppression pool is contained in the first release. A fraction of the iodine that was trapped by the suppression pool is subsequently revolatilized from the pool and released into the containment. The revolatilized iodine is grouped in the second release.

The mean source terms in Table 3.3-1 can be used to compare the releases associated with 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 process.

For some of the accident progression bins the release energy assigned to the bin was wrong. The release energy affects how high the releases are

lofted in the atmosphere. For accidents in which the containment does not fail the release energy should have been set to zero but was inadvertently set to value that is used when the containment fails. The release fractions for these accidents, however, are typically very small and, therefore, the effect on risk is expected to be negligible. For accidents in which the containment is ruptured at vessel breach, the release energy was inadvertently set to zero. Because the plume is not lofted as high as it should have been, the early fatalities may be slightly overestimated for these accidents. The latent cancer fatalities are not particularly sensitive to this parameter and, thus, the effect on this consequence measure is expected to be very minor.

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. The frequency of each PDS is presented in Section 2.2. 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. When multiplied by the frequency of the PDS for that observation, a frequency for the APB is obtained. 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 vs. release fraction can be plotted for each observation. Figure 3.3-1 is a summary presentation of these curves for the 250 observations in the sample.

Instead of placing all 250 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 250 values of the exceedance frequency (one for each sample element). From these 250 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 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 ( $\text{yr}^{-1}$ ) at which a release fraction of  $10^{-5}$  is exceeded due to PDS 1 is roughly  $2.9 \times 10^{-6}$ ,  $2.4 \times 10^{-6}$ ,  $2 \times 10^{-6}$  and  $1.7 \times 10^{-6}$  for the iodine, cesium, strontium and lanthanum release classes, respectively. For a release fraction of 0.1, the corresponding mean exceedance frequencies are  $4.2 \times 10^{-7}$ ,  $8.8 \times 10^{-8}$ ,  $1.7 \times 10^{-8}$  and  $8.4 \times 10^{-11}$ , respectively. The three quantiles (i.e., the median, 95th and 5th) provide an indication of the spread between observations, which is often large. Typically, the mean curves reach a point where they 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 the occurrence of large source terms (e.g., release fractions  $\geq 0.1$ ) in conjunction with PDS 1 is very infrequent (less than  $10^{-6}$  for iodine and cesium, less than  $10^{-7}$  for cesium, strontium, and lanthanum).

**3.3.1.2 Results for PDS 2: Short-Term SBO.** PDS 2 is the same as PDS 1 except that heat removal via the sprays is not available with the recovery of offsite power. For this PDS the mean probability that coolant injection is recovered and vessel breach is averted is 0.32. The mean probability that the containment will fail early is 0.36.

Table 2.5.1-2 lists the five most probable APBs for this PDS, the five most probable APBs that have vessel breach, and the five most probable APBs that have early containment failure (CF). A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.2. Table 3.3-2 lists the mean source terms for these same APBs. Although the containment sprays are not available in these APBs, the in-vessel releases are discharged into the suppression pool where they are subjected to the pool DF. Of the APBs listed in Table 3.3-2 only one bin has a stuck open tail pipe vacuum breaker. The stuck-open vacuum breaker allows a fraction of the in-vessel releases to enter the drywell rather than being discharged directly into the suppression pool. However, in this bin the suppression pool is not bypassed until late in the accident and, therefore, the in-vessel releases that enter the drywell still pass through the pool (i.e., via the horizontal vents) before entering the containment volume. For the APBs that involve vessel failure, the core debris released into the cavity is either cooled or CCI takes place under water. Thus, any ex-vessel releases are also scrubbed. Thus, although the containment sprays are not available in this PDS the releases associated with the APBs presented in Table 3.2-2 are still mitigated by the suppression pool and the cavity water.

Figure 3.3-2 summarizes the release fraction CCDFs for PDS 2.

**3.3.1.3 Results for PDS 3: Short-Term SBO.** PDS 3 is the same as PDS 1 except that heat removal via the sprays is not available with the recovery of offsite power and the only injection system available with the recovery of offsite power is the condensate system. For this PDS the mean probability that coolant injection is recovered and vessel breach is averted is 0.21. The mean probability that the containment fails early is 0.44.

Table 2.5.1-3 lists the five most probable APBs for this PDS, the five most probable APBs that have vessel breach, and the five most probable APBs that have early containment failure (CF). A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.3. Table 3.3-3 lists the mean source terms for these same APBs. For the APBs listed in Table 3.3-3 the in-vessel releases are scrubbed by the suppression pool and any ex-vessel releases, should they occur, are scrubbed by either the suppression pool or the water in the reactor cavity. Only one of listed APBs involves early failure of both the containment and the drywell. But

for this bin there are no stuck-open tail pipe vacuum breakers (i.e., in-vessel releases directed to the suppression pool) and vessel breach is averted (i.e., no ex-vessel releases).

Figure 3.3-3 summarizes the release fraction CCDFs for PDS 3.

**3.3.1.4 Results for PDS 4: Long-Term SBO.** This PDS involves station blackout scenarios where LOSP is recoverable. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at low pressure. If offsite power is restored, then coolant injection to the RPV, containment sprays and the HIS are all available. Because this is a slow SBO (i.e., core damage occurs  $\geq 12$  h), this PDS has a much lower probability of recovering offsite power than did the fast SBO in which core damage occurs in approximately 1 h. For this PDS the mean probability that coolant injection is recovered and vessel breach is averted is only 0.05. The mean probability that the containment fails early is 0.65.

Table 2.5.1-4 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.4. Table 3.3-4 lists the mean source terms for these same APBs. In all of the 10 most probable bins vessel breach occurs, the RPV is at low pressure, and an ex-vessel steam explosion, which involves a small amount of the core, occurs at vessel breach. Containment sprays are not available in any of the 10 most probable bins. For these APBs the in-vessel releases are directed to the suppression pool and either the core debris in the cavity is cooled or CCI takes place under water. However, in three of these bins both the containment and the drywell are ruptured early in the accident and, therefore, the releases at vessel breach (i.e., releases associated with DCH) are not scrubbed by either the pool or the sprays. In all of the five most probable bins that have early containment failure and early suppression pool bypass vessel breach occurs with the RPV at low pressure followed by an ex-vessel steam explosion. There are no stuck-open tail pipe vacuum breakers in these five bins so all of the in-vessel releases pass through the suppression pool. However, because there is early drywell failure, a pathway is established which bypasses the suppression pool. Although the releases at vessel breach (i.e., releases associated with an ex-vessel steam explosion) are not scrubbed by either the suppression pool or the sprays, the core debris in the reactor cavity is either cooled or CCI takes place under water.

Figure 3.3-4 summarizes the release fraction CCDFs for PDS 4.

**3.3.1.5 Results for PDS 5: Long-Term SBO.** PDS 5 is the same as PDS 4 except that heat removal via the sprays is not available with the recovery of offsite power. However, because there is a low probability of recovering offsite power in this PDS this difference is not very important. For this PDS the mean probability that coolant injection is recovered and vessel breach is averted is only 0.05. The mean probability that the containment fails early is 0.64.

Table 2.5.1-5 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.5. Table 3.3-5 lists the mean source terms for these same APBs. In all of the 10 most probable bins vessel breach occurs, the RPV is at low pressure, and an ex-vessel steam explosion, which involves a small amount of the core, occurs at vessel breach. Containment sprays are not available in any of the 10 most probable bins. For these APBs the in-vessel releases are directed to the suppression pool and either the core debris in the cavity is cooled or CCI takes place under water. However, in three of these bins both the containment and the drywell are ruptured early in the accident and, therefore, the releases at vessel breach (i.e., releases associated with DCH) are not scrubbed by either the pool or the sprays. In all of the five most probable bins that have early containment failure and early suppression pool bypass vessel breach occurs with the RPV at low pressure followed by an ex-vessel steam explosion. There are no stuck-open tail pipe vacuum breakers in these five bins so all of the in-vessel releases pass through the suppression pool. However, because there is early drywell failure, a pathway is established which bypasses the suppression pool. Although the releases at vessel breach (i.e., releases associated with an ex-vessel steam explosion) are not scrubbed by either the suppression pool or the sprays, the core debris in the reactor cavity is either cooled or CCI takes place under water.

Figure 3.3-5 summarizes the release fraction CCDFs for PDS 5.

3.3.1.6 Results for PDS 6: Long-Term SBO. PDS 6 is the same as PDS 4 except that neither coolant injection to the RPV nor the containment sprays are available during the accident. Thus, because there is no coolant injection to the vessel, the mean probability of vessel breach is 1.0. The mean probability that the containment fails early is 0.68.

Table 2.5.1-6 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.6. Table 3.3-6 lists the mean source terms for these same APBs. In all of the 10 most probable bins vessel breach occurs with the RPV at low pressure followed by an ex-vessel steam explosion that involves a small fraction of the core. The containment sprays do not operate during the accident but because there are no stuck-open SRV tail pipe vacuum breakers all of the in-vessel releases are still scrubbed by the suppression pool. In all of the 10 most probable bin the core debris released from the vessel is cooled and there are no CCI releases.

Figure 3.3-6 summarizes the release fraction CCDFs for PDS 6.

3.3.1.7 Results for PDS 7: Short-Term SBO. This PDS involves station blackout (without any dc power) scenarios where LOSP is not recoverable. Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure and depressurization is not

possible. Since offsite power is not recoverable, neither coolant injection nor containment sprays are available during the accident. In a small fraction of these accidents (4%) a SRV will stick open and depressurize the RPV. Once the RPV has been depressurized, the firewater system can be used to provide coolant injection to the RPV. Thus, the mean probability that vessel breach is averted is only 0.01. The mean probability that the containment fails early is 0.60.

Table 2.5.1-7 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.7. Table 3.3-7 lists the mean source terms for these same APBs. In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. The containment sprays do not operate during the accident but because there are no stuck open SRV tail pipe vacuum breakers all of the in-vessel releases are still scrubbed by the suppression pool. Furthermore, the core debris that accumulates in the reactor cavity is cooled by water and, thus, there are no CCI releases. However, the drywell does fail early in two of these bins and, therefore, the releases at vessel breach (i.e., releases associated with DCH) are not scrubbed by either the pool or the sprays.

Figure 3.3-7 summarizes the release fraction CCDFs for PDS 7.

**3.3.1.8 Results for PDS 8: Long-Term SBO.** This PDS involves SBO (without any dc power) scenarios where LOSP is not recoverable. Coolant injection is lost late such that core damage occurs in the long term and with the vessel at high pressure and depressurization is not possible. Since offsite power is not recoverable, neither coolant injection nor containment sprays are available during the accident. Because there is no coolant injection to the RPV, the probability of vessel breach is 1.0. The mean probability that the containment fails early is 0.54.

Table 2.5.1-8 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.8. Table 3.3-8 lists the mean source terms for these same APBs. In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. The containment sprays do not operate during the accident. There is only one bin that has a stuck-open tail pipe vacuum breaker; however for this bin the drywell does not fail. Thus, all of the in-vessel releases are scrubbed by the suppression pool. Although the in-vessel releases for the ninth most probable bin are scrubbed by the suppression pool, the ex-vessel releases do not benefit from a pool DF. In this APB both the drywell and the containment are ruptured early in the accident. Thus, the radionuclides released at vessel breach (e.g., from DCH) and the releases from CCI bypass the suppression pool. Furthermore, the sprays are not available in this PDS and in this APB CCI does not take place under a pool of water. Thus, the ex-vessel releases are not mitigated by the sprays, the suppression pool, or the

cavity pool. Therefore, it is not surprising that the mean release fractions associated with this APB tend to be higher than the release fractions for the other nine bins.

Figure 3.3-8 summarizes the release fraction CCDFs for PDS 8.

**3.3.1.9 Results for PDS 9: Short-Term ATWS.** This PDS involves ATWS transient scenarios. Coolant injection is lost early such that core damage occurs in the short term and with the vessel at high pressure because the operator failed to depressurize. The low pressure injection is recoverable with reactor depressurization. The containment sprays are available during the accident. The mean probability that coolant injection will be restored to the RPV and vessel breach will be averted is only 0.04. The mean probability that the containment fails early is 0.67.

Table 2.5.1-9 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.9. Table 3.3-9 lists the mean source terms for these same APBs. In the 10 most probable bins, vessel breach occurs with RPV at high pressure. At vessel breach either a DCH event occurs (nine bins) or there is an ex-vessel steam explosion (one bin). In all but one of the 10 most probable bins the containment fails at vessel breach. In all of these 10 bins the in-vessel releases are directed to the suppression pool. The drywell fails early in three of these APBs and, therefore, the releases at vessel breach bypass the suppression pool. However, the containment sprays are operating around the time of vessel breach. There are no CCI releases in all but one of these bins and in the bin that CCI does occur, the releases are scrubbed by a flooded cavity.

Figure 3.3-9 summarizes the release fraction CCDFs for PDS 9.

**3.3.1.10 Results for PDS 10: Long-Term ATWS.** This PDS involves ATWS transient scenarios. Coolant injection is lost late such that core damage occurs in the long term with the vessel at high pressure because of operator error. Low pressure injection is recoverable with reactor depressurization. The containment sprays are available during the accident. The mean probability that coolant injection will be restored to the RPV and vessel breach will be averted is only 0.01. The probability that the containment fails early is 1.0. The containment always fails in this PDS because the energy dumped into the suppression pool from the RPV during an ATWS transient exceeds the capacity of the RHR system which results in a large buildup of steam in the containment.

Table 2.5.1-10 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.10. Table 3.3-10 lists the mean source terms for these same APBs. In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. In all of these bins the containment fails early; however, there is coincident drywell failure in



only one of these bins. The containment sprays operate before vessel breach in all of these bins and continue to operate during the entire accident in all but two of these bins. In these APBs both in-vessel releases and the ex-vessel releases are scrubbed by either the suppression pool or the containment sprays.

Figure 3.3-10 summarizes the release fraction CCDFs for PDS 10.

3.3.1.11 Results for PDS 11: Short-Term T2. This PDS involves transient scenarios where the PCS is lost (T2). Coolant injection is lost early such that core damage occurs in the short term with the vessel at high pressure because of operator error. The containment sprays are available during the accident. The mean probability that coolant injection will be restored to the RPV and vessel breach will be averted is only 0.05. The mean probability that the containment fails early is 0.56.

Table 2.5.1-11 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.11. Table 3.3-11 lists the mean source terms for these same APBs. In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. The containment fails early in all but two of these bins. Only two of the 10 bins have coincident early containment failure and early drywell failure. The bins that have early drywell failure do not have any stuck-open tail pipe vacuum breakers. Thus, in the 10 most probable bins the in-vessel releases are scrubbed by the suppression pool. Furthermore, the containment sprays operate around the time of vessel breach and there are no CCI releases in all but one of these bins. Thus, the ex-vessel releases are scrubbed by either the suppression pool, the sprays, or the water in the reactor cavity.

Figure 3.3-11 summarizes the release fraction CCDFs for PDS 11.

3.3.1.12 Results for PDS 12: Long-Term T2. PDS 12 is the same as PDS 11 except that core damage occurs in the long-term. The mean probability that coolant injection will be restored to the RPV and vessel breach will be averted is only 0.05. The mean probability that the containment fails early is 0.56.

Table 2.5.1-12 lists the 10 most probable APBs for this PDS and the five most probable APBs that have early containment failure and early suppression pool bypass. A discussion of the accident characteristics for these APBs is presented in Section 2.5.1.12. Table 3.3-12 lists the mean source terms for these same APBs. In all of the 10 most probable bins, vessel breach occurs with the RPV at high pressure followed by a DCH event that involves a small fraction of the core. The containment fails early in all but two of these bins. Only two of the 10 bins have coincident early containment failure and early drywell failure. The bins that have early drywell failure do not have any stuck-open tail pipe vacuum breakers. Thus, in the 10 most probable bins the in-vessel releases are scrubbed by the suppression pool. Furthermore, the containment sprays operate around

the time of vessel breach and there are no CCI releases in all but one of these bins. Thus, the ex-vessel releases are scrubbed by either the suppression pool, the sprays, or the water in the reactor cavity.

Figure 3.3-12 summarizes the release fraction CCDFs for PDS 12.

**3.3.1.13 Results for Generalized Accident Progression Bins.** The preceding twelve 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. These generalized APBs are generated by sorting all of the bins from the 12 PDS on attributes of the accident. The generalized bins are composed of essentially four characteristics: occurrence of vessel breach, timing of containment failure, timing of suppression pool bypass, and the availability of the containment sprays. These generalized APBs are listed roughly in decreasing order of the severity of the source term (i.e., release timing and release fractions). (The last two bins are an exception to this ordering scheme). A description of these reduced bins is presented in section 2.4.3.

Figure 3.3-13 shows the variation of the exceedance frequency with release fraction for the iodine, cesium, strontium, and lanthanum radionuclide classes for all the APBs in which the vessel fails and both the containment and drywell fail early in the accident. In this bin the containment sprays are not available. Although the in-vessel releases will generally be directed to the suppression pool, the releases at vessel breach and any ex-vessel releases will not be subjected to the DF associated with either the pool or the sprays. If the reactor cavity contains water, however, any CCI releases will be scrubbed by the overlaying pool.

Figure 3.3-14 shows the variation of the exceedance frequency with release fraction for all the APBs in which the vessel fails and both the containment and drywell fail early in the accident. This generalized bin is similar to generalized bin used in Figure 3.3-13 except that in these accidents the sprays are available. The release fractions associated with this bin tend to be lower than the release fractions presented in Figure 3.3-13.

Figure 3.3-15 shows the variation of the exceedance frequency with release fraction for all the APBs in which the vessel fails, the containment fails early, and the drywell fails late in the accident. Failure of the drywell late in the accident can be induced by failure of the reactor pedestal caused by concrete erosion from CCI. Thus, for this generalized APB both the in-vessel releases and the release at vessel breach are directed into the suppression pool. Initiation of CCI is relatively likely in this APB. Furthermore, because of the late failure of the drywell, the CCI release will bypass the suppression pool. This APB has a fairly low frequency of occurrence.

Figure 3.3-16 shows the variation of the exceedance frequency with release fraction for all the APBs in which the vessel fails and the containment fails early in the accident. In these APBs the drywell does not fail

during the accident. In this APB both the in-vessel and the ex-vessel releases are directed to the suppression pool.

Figure 3.3-17 shows the variation of the exceedance frequency with release fraction for all the APBs in which the vessel fails and the containment fails late in the accident. This generalized APB has a relatively high frequency of occurrence and includes a variety of different accidents (i.e., those with and without drywell failure).

Figure 3.3-18 shows the variation of the exceedance frequency with release fraction for all the APBs in which the vessel fails and the containment is vented during the accident.

Figure 3.3-19 shows the variation of the exceedance frequency with release fraction for all the APBs in which the vessel failed but the containment remained intact throughout the accident. Because in these APBs there is only nominal leakage from the containment, the release fractions tend to be quite low. It should be pointed that some of the APBs in this group involve accidents in which the containment fails even though vessel breach is averted.

Figure 3.3-20 shows the variation of the exceedance frequency with release fraction for all the APBs in which the vessel breach is averted. Although the vessel does not fail in these APBs, some of these bins involve early containment failure. Thus, the release fractions for these APBs are typically larger than the release fraction presented in the previous figure.

3.3.1.14 Summary. When all the types of accidents from internal initiators at Grand Gulf are considered together, the exceedance frequency plots shown in Figure 3.3-21 are obtained. A plot is not shown for the noble gases since 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 and cesium is on the order of  $10^{-6}$ /year and for tellurium and strontium it is on the order of  $10^{-7}$ /year. The second sheet of Figure 3.3-16 shows the release fractions for ruthenium, lanthanum, cesium, and barium, which are often treated together as aerosol species. The mean frequency of exceeding a release fraction of 0.01 for ruthenium, lanthanum, and cesium is on the order of  $10^{-7}$ /year. The releases for the barium class are slightly higher than those for the other three aerosol radionuclide classes.

Table 3.3-1  
Mean Source Terms for Grand Gulf  
Internal Initiators. PDS 1: 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	ABDDGCCB	3.6E+03	3.2E+01	3.0E+07 1.4E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.5E-01 1.7E-02	7.3E-03 8.1E-04	7.9E-03 8.7E-04	3.9E-03 4.3E-04	2.0E-04 2.3E-05	2.8E-04 3.1E-05	5.7E-04 6.3E-05	3.3E-03 3.7E-04	
2	ABEEAICEB	3.6E+03	3.2E+01	8.0E+06 0.0E+00	5.0E+04 5.8E+04	7.2E+03 2.2E+04	1.8E-03 1.8E-03	1.3E-05 1.3E-05	1.1E-08 1.1E-08	5.2E-09 5.2E-09	8.7E-10 8.7E-10	1.7E-10 1.7E-10	3.8E-11 3.8E-11	1.5E-10 1.5E-10	9.7E-10 9.7E-10	
3	ABEEAGCEB	3.6E+03	3.2E+01	3.2E+08 0.0E+00	5.0E+04 5.1E+04	1.8E+02 1.4E+04	6.1E-01 6.7E-02	6.4E-03 7.1E-04	1.8E-03 2.0E-04	8.7E-04 9.7E-05	2.1E-04 2.4E-05	3.0E-05 3.3E-06	8.6E-06 9.5E-07	3.4E-05 3.8E-06	2.2E-04 2.5E-05	
4	ABEEAFCEB	3.6E+03	3.2E+01	8.0E+06 0.0E+00	5.0E+04 5.8E+04	7.2E+03 2.2E+04	3.4E-01 3.4E-01	2.5E-03 2.5E-03	2.5E-04 2.5E-04	1.7E-04 1.7E-04	8.6E-05 8.6E-05	1.2E-05 1.2E-05	6.2E-06 6.2E-06	3.0E-05 3.0E-05	8.7E-05 8.7E-05	
5	ABEEARCEB	3.6E+03	3.2E+01	3.2E+08 0.0E+00	5.0E+04 5.1E+04	1.8E+02 1.4E+04	6.2E-01 6.8E-02	7.0E-03 7.8E-04	1.9E-03 2.2E-04	9.8E-04 1.1E-04	2.4E-04 2.7E-05	3.8E-05 4.3E-06	1.2E-05 1.4E-06	5.3E-05 5.9E-06	2.6E-04 2.9E-05	
Five Most Probable Bins That Have VB*																
1	ABDDGCCB	3.6E+03	3.2E+01	3.0E+07 1.4E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.5E-01 1.7E-02	7.3E-03 8.1E-04	7.9E-03 8.7E-04	3.9E-03 4.3E-04	2.0E-04 2.3E-05	2.8E-04 3.1E-05	5.7E-04 6.3E-05	3.3E-03 3.7E-04	
9	ABDDGCCB	3.6E+03	3.2E+01	3.0E+07 1.4E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	2.1E-01 2.4E-02	3.8E-03 4.3E-04	3.7E-03 4.2E-04	1.6E-03 1.8E-04	8.6E-05 9.5E-06	7.4E-05 8.2E-06	1.3E-04 1.4E-05	9.7E-04 1.1E-04	
12	ABDDGACB	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.9E-01 2.1E-02	1.9E-02 2.1E-03	1.9E-02 2.1E-03	1.0E-02 1.1E-03	4.4E-04 4.9E-05	7.0E-04 7.8E-05	1.4E-03 1.5E-04	8.2E-03 9.1E-04	
13	ABDDGCCA	3.6E+03	3.2E+01	3.0E+07 1.4E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.6E-01 1.7E-02	1.2E-02 1.4E-03	1.1E-02 1.3E-03	5.2E-03 5.8E-04	3.2E-04 3.6E-05	3.5E-04 3.9E-05	7.8E-04 8.7E-05	4.6E-03 5.1E-04	
14	ABDAICEB	3.6E+03	3.2E+01	7.5E+05 9.1E+04	5.0E+04 5.8E+04	7.2E+03 2.2E+04	2.0E-03 2.0E-03	2.0E-05 2.0E-05	1.7E-08 1.7E-08	8.1E-09 8.1E-09	1.5E-09 1.5E-09	6.2E-10 6.2E-10	1.5E-10 1.5E-10	3.0E-10 3.0E-10	1.7E-09 1.7E-09	
Five Most Probable Bins That Have Early CF*																
7	AAEEBAEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	8.3E-01 0.0E+00	2.7E-02 3.0E-03	2.1E-02 0.0E+00	1.5E-02 0.0E+00	6.3E-03 0.0E+00	1.2E-03 0.0E+00	4.4E-04 0.0E+00	2.3E-03 0.0E+00	6.5E-03 0.0E+00	
10	AAEEBAEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	7.7E-01 0.0E+00	1.5E-02 2.5E-03	1.4E-02 0.0E+00	1.1E-02 0.0E+00	6.9E-03 0.0E+00	1.1E-03 0.0E+00	4.4E-04 0.0E+00	2.0E-03 0.0E+00	7.0E-03 0.0E+00	
15	AAEEAACEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 2.2E+04	7.1E-01 0.0E+00	8.0E-03 3.0E-03	5.9E-03 0.0E+00	3.4E-03 0.0E+00	3.7E-04 0.0E+00	1.3E-04 0.0E+00	1.6E-05 0.0E+00	5.9E-05 0.0E+00	4.3E-04 0.0E+00	
18	AAEEHBAEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	7.2E-01 0.0E+00	2.6E-03 5.2E-03	1.5E-03 0.0E+00	5.3E-04 0.0E+00	1.7E-05 0.0E+00	1.2E-05 0.0E+00	5.5E-07 0.0E+00	1.2E-06 0.0E+00	2.8E-05 0.0E+00	
31	AABDABCB	3.6E+03	3.2E+01	1.1E+06 6.7E+06	8.3E+03 1.3E+04	4.7E+03 3.6E+03	8.6E-01 1.4E-01	2.8E-02 8.0E-02	2.0E-02 4.5E-02	1.3E-02 4.6E-02	4.1E-03 4.6E-02	8.0E-04 1.1E-03	2.3E-04 3.2E-03	1.0E-03 6.1E-03	4.2E-03 3.6E-02	

\* A listing of source terms for all bins is available on computer media

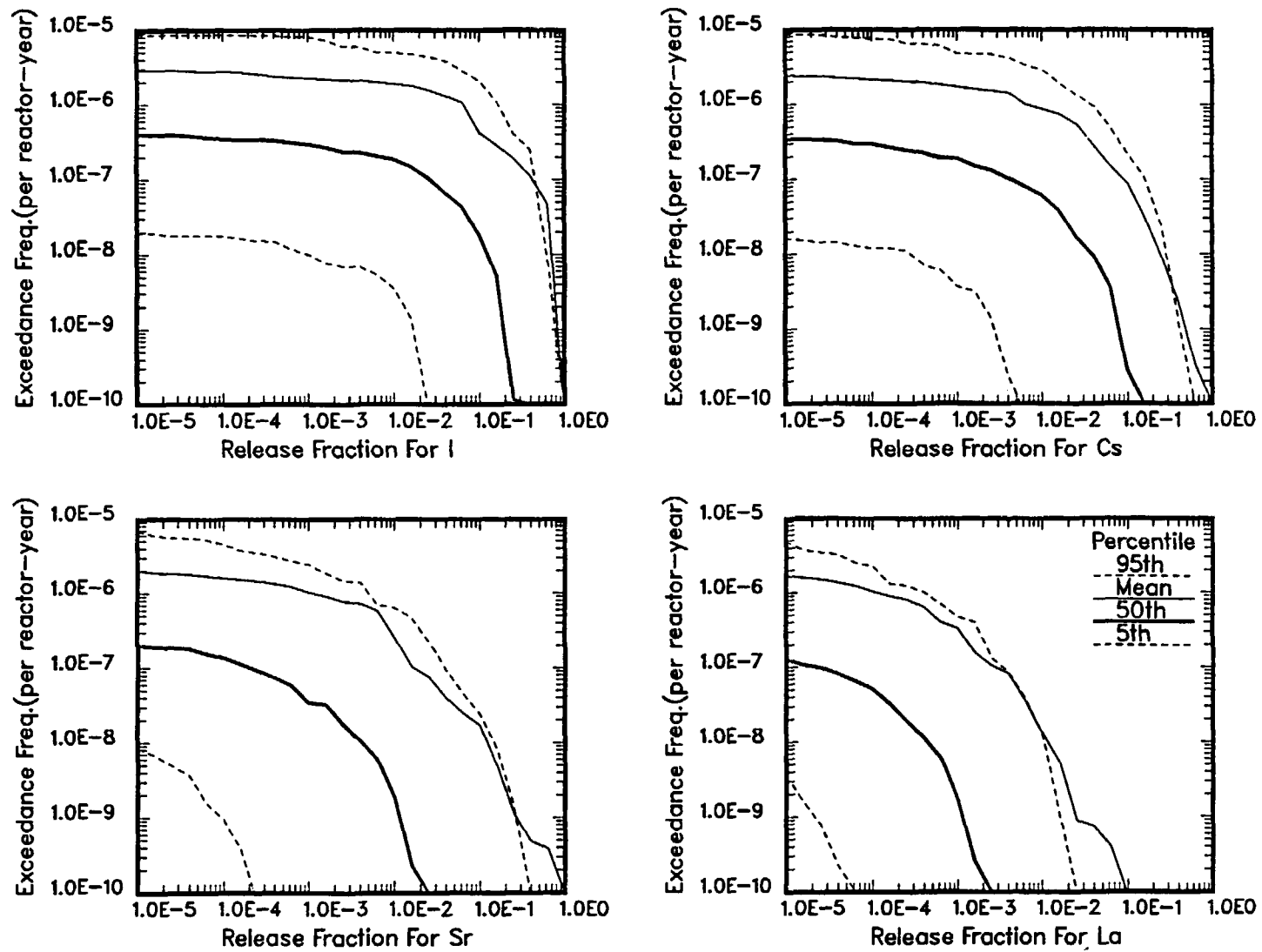


Figure 3.3-1. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 1: Fast SBO.

Table 3.3-2  
Mean Source Terms for Grand Gulf  
Internal Initiators. PDS 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	ABDDGACB	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.9E-01 2.1E-02	1.9E-02 2.1E-03	1.9E-02 2.1E-03	1.0E-02 1.1E-03	4.4E-04 4.9E-05	7.0E-04 7.8E-05	1.4E-03 1.5E-04	8.2E-03 9.1E-04
2	ABEEAIAEB	3.6E+03	3.2E+01	8.0E+06 0.0E+00	5.0E+04 5.8E+04	7.2E+03 2.2E+04	1.8E-03 1.8E-03	1.3E-05 1.3E-05	1.0E-08 1.0E-08	4.9E-09 4.9E-09	8.1E-10 8.1E-10	1.6E-10 1.6E-10	3.6E-11 3.6E-11	1.4E-10 1.4E-10	9.1E-10 9.1E-10
3	ABEEAGAEB	3.6E+03	3.2E+01	3.2E+08 0.0E+00	5.0E+04 5.1E+04	1.8E+02 1.4E+04	6.1E-01 6.8E-02	6.2E-03 6.9E-04	1.5E-03 1.7E-04	7.6E-04 8.4E-05	1.9E-04 2.1E-05	2.6E-05 2.8E-06	7.4E-06 8.3E-07	2.9E-05 3.3E-06	1.9E-04 2.2E-05
4	ABEEAFAEB	3.6E+03	3.2E+01	8.0E+06 0.0E+00	5.0E+04 5.8E+04	7.2E+03 2.2E+04	3.5E-01 3.5E-01	2.3E-03 2.3E-03	2.1E-04 2.1E-04	1.4E-04 1.4E-04	7.1E-05 7.1E-05	9.7E-06 9.7E-06	5.1E-06 5.1E-06	2.4E-05 2.4E-05	7.2E-05 7.2E-05
5	ABEEAFAEB	3.6E+03	3.2E+01	3.2E+08 0.0E+00	5.0E+04 5.1E+04	1.8E+02 1.4E+04	6.1E-01 6.8E-02	7.0E-03 7.8E-04	1.9E-03 2.1E-04	9.3E-04 1.0E-04	2.3E-04 2.6E-05	3.7E-05 4.1E-06	1.2E-05 1.3E-06	5.0E-05 5.6E-06	2.4E-04 2.7E-05
Five Most Probable Bins That Have VB*															
1	ABDDGACB	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.9E-01 2.1E-02	1.9E-02 2.1E-03	1.9E-02 2.1E-03	1.0E-02 1.1E-03	4.4E-04 4.9E-05	7.0E-04 7.8E-05	1.4E-03 1.5E-04	8.2E-03 9.1E-04
7	ABDDGACB	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	2.1E-01 2.3E-02	2.0E-02 2.2E-03	2.0E-02 2.2E-03	8.4E-03 9.4E-04	2.0E-04 2.2E-05	3.9E-04 4.4E-05	6.6E-04 7.4E-05	5.0E-03 5.6E-04
10	ABDDGACA	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.9E-01 2.1E-02	2.3E-02 2.5E-03	2.3E-02 2.5E-03	1.2E-02 1.3E-03	5.2E-04 5.7E-05	7.9E-04 8.8E-05	1.6E-03 1.8E-04	9.7E-03 1.1E-03
13	ABDDAIAEB	3.6E+03	3.2E+01	7.5E+05 4.7E+05	5.0E+04 5.8E+04	7.2E+03 2.2E+04	1.9E-03 1.9E-03	1.6E-05 1.6E-05	2.2E-08 2.2E-08	1.1E-08 1.1E-08	2.1E-09 2.1E-09	1.4E-09 1.4E-09	3.1E-10 3.1E-10	4.5E-10 4.5E-10	2.4E-09 2.4E-09
14	ABDDAFAEB	3.6E+03	3.2E+01	7.5E+05 4.7E+05	5.0E+04 5.8E+04	7.2E+03 2.2E+04	3.1E-01 3.1E-01	4.3E-03 4.3E-03	1.2E-03 1.2E-03	4.1E-04 4.1E-04	9.4E-05 9.4E-05	1.1E-04 1.1E-04	2.6E-05 2.6E-05	3.2E-05 3.2E-05	1.1E-04 1.1E-04
Five Most Probable Bins That Have Early CF*															
8	AAEEBAEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	8.3E-01 0.0E+00	2.7E-02 3.0E-03	2.1E-02 0.0E+00	1.5E-02 0.0E+00	6.3E-03 0.0E+00	1.2E-03 0.0E+00	4.4E-04 0.0E+00	2.3E-03 0.0E+00	6.5E-03 0.0E+00
12	AAEEBAEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	7.7E-01 0.0E+00	1.5E-02 2.5E-03	1.4E-02 0.0E+00	1.1E-02 0.0E+00	6.9E-03 0.0E+00	1.1E-03 0.0E+00	4.4E-04 0.0E+00	2.0E-03 0.0E+00	7.0E-03 0.0E+00
17	AAEEBAEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	7.2E-01 0.0E+00	2.6E-03 5.2E-03	1.5E-03 0.0E+00	5.3E-04 0.0E+00	1.7E-05 0.0E+00	1.2E-05 0.0E+00	5.5E-07 0.0E+00	1.2E-06 0.0E+00	2.8E-05 0.0E+00
18	AAEEAAAEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 2.2E+04	7.1E-01 0.0E+00	8.0E-03 3.0E-03	5.9E-03 0.0E+00	3.4E-03 0.0E+00	3.7E-04 0.0E+00	1.3E-04 0.0E+00	1.6E-05 0.0E+00	5.9E-05 0.0E+00	4.3E-04 0.0E+00
30	ABEEAAAEB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 2.2E+04	6.8E-01 0.0E+00	7.0E-03 4.8E-03	5.0E-03 0.0E+00	2.6E-03 0.0E+00	6.3E-04 0.0E+00	1.0E-04 0.0E+00	3.2E-05 0.0E+00	1.4E-04 0.0E+00	6.7E-04 0.0E+00

\* A listing of source terms for all bins is available on computer media

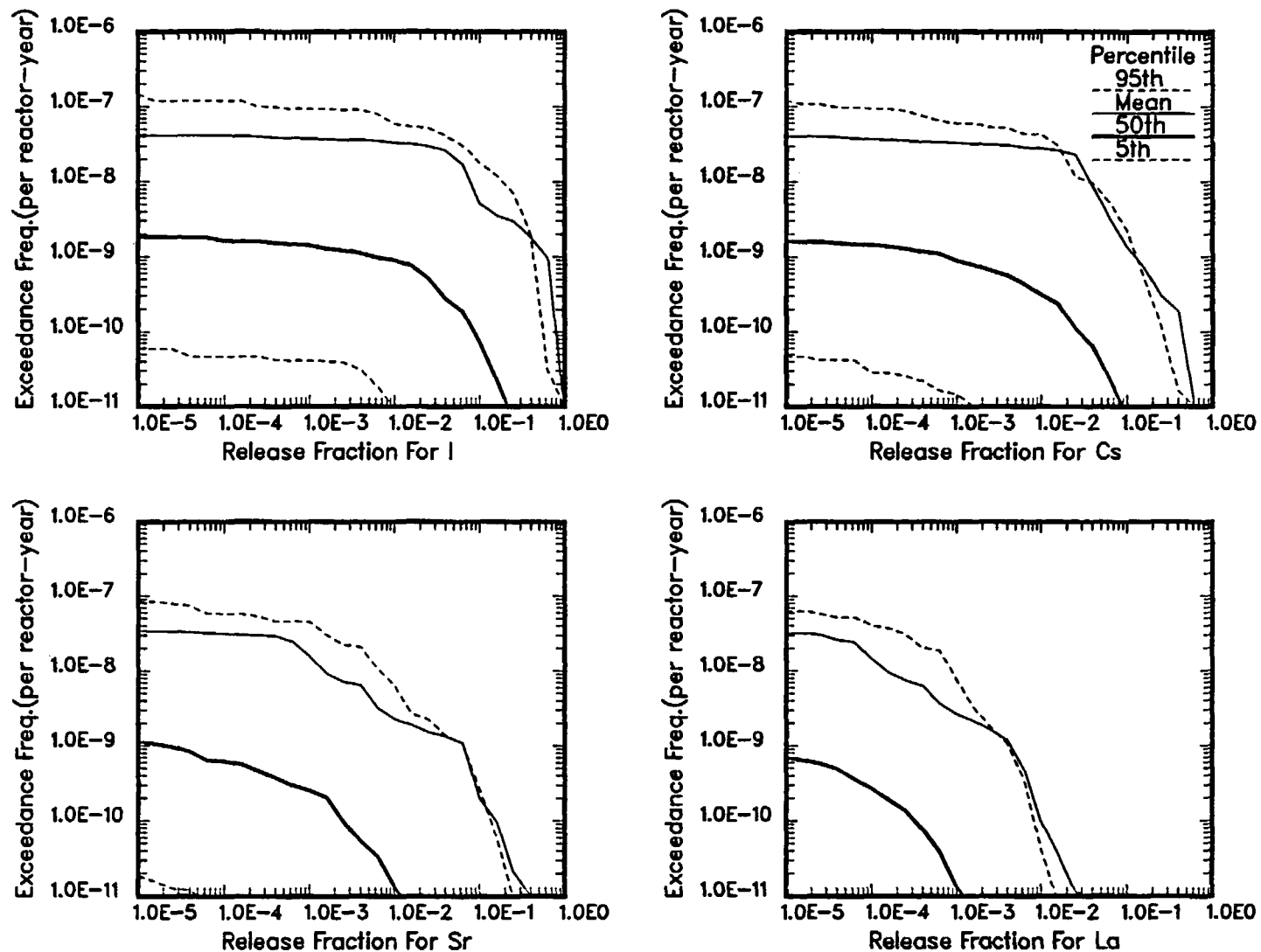


Figure 3.3-2. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 2: Fast SBO.

Table 3.3-3  
 Mean Source Terms for Grand Gulf  
 Internal Initiators. PDS 3: 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	ABDDGACB	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.9E-01 2.1E-02	1.9E-02 2.1E-03	1.9E-02 2.1E-03	1.0E-02 1.1E-03	4.4E-04 4.9E-05	7.0E-04 7.8E-05	1.4E-03 1.5E-04	8.2E-03 9.1E-04	
2	ABEEAGAEB	3.6E+03	3.2E+01	3.2E+08 0.0E+00	5.0E+04 5.1E+04	1.8E+02 1.4E+04	6.1E-01 6.8E-02	6.2E-03 6.9E-04	1.5E-03 1.7E-04	7.6E-04 8.4E-05	1.9E-04 2.1E-05	2.6E-05 2.8E-06	7.4E-06 8.3E-07	2.9E-05 3.3E-06	1.9E-04 2.2E-05	
3	ABEEAIAEB	3.6E+03	3.2E+01	8.0E+06 0.0E+00	5.0E+04 5.8E+04	7.2E+03 2.2E+04	1.8E-03 1.8E-03	1.3E-05 1.3E-05	1.0E-08 1.0E-08	4.9E-09 4.9E-09	8.1E-10 8.1E-10	1.6E-10 1.6E-10	3.6E-11 3.6E-11	1.4E-10 1.4E-10	9.1E-10 9.1E-10	
4	ABEEFAEAB	3.6E+03	3.2E+01	8.0E+06 0.0E+00	5.0E+04 5.8E+04	7.2E+03 2.2E+04	3.5E-01 3.5E-01	2.3E-03 2.3E-03	2.1E-04 2.1E-04	1.4E-04 1.4E-04	7.1E-05 7.1E-05	9.7E-06 9.7E-06	5.1E-06 5.1E-06	2.4E-05 2.4E-05	7.2E-05 7.2E-05	
5	ABDDGACB	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	2.1E-01 2.3E-02	2.0E-02 2.2E-03	2.0E-02 2.2E-03	8.4E-03 9.4E-04	2.0E-04 2.2E-05	3.9E-04 4.4E-05	6.6E-04 7.4E-05	5.0E-03 5.6E-04	
Five Most Probable Bins That Have VB*																
1	ABDDGACB	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.9E-01 2.1E-02	1.9E-02 2.1E-03	1.9E-02 2.1E-03	1.0E-02 1.1E-03	4.4E-04 4.9E-05	7.0E-04 7.8E-05	1.4E-03 1.5E-04	8.2E-03 9.1E-04	
5	ABDDGACB	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	2.1E-01 2.3E-02	2.0E-02 2.2E-03	2.0E-02 2.2E-03	8.4E-03 9.4E-04	2.0E-04 2.2E-05	3.9E-04 4.4E-05	6.6E-04 7.4E-05	5.0E-03 5.6E-04	
6	ABABAEAEB	3.6E+03	3.2E+01	0.0E+00 0.0E+00	1.3E+04 1.3E+04	1.8E+02 1.4E+04	7.1E-01 0.0E+00	1.5E-02 2.3E-02	1.4E-02 9.4E-03	6.5E-03 3.5E-03	2.0E-03 0.0E+00	1.3E-03 0.0E+00	3.8E-04 0.0E+00	5.4E-04 0.0E+00	2.3E-03 0.0E+00	
8	ABDDGACA	3.6E+03	3.2E+01	3.0E+07 7.0E+05	5.0E+04 5.1E+04	1.8E+02 1.4E+04	9.0E-01 1.0E-01	1.9E-01 2.1E-02	2.3E-02 2.5E-03	2.3E-02 2.5E-03	1.2E-02 1.3E-03	5.2E-04 5.7E-05	7.9E-04 8.8E-05	1.6E-03 1.8E-04	9.7E-03 1.1E-03	
10	ABDAIAEB	3.6E+03	3.2E+01	7.5E+05 4.7E+05	5.0E+04 5.8E+04	7.2E+03 2.2E+04	1.9E-03 1.8E-05	1.6E-05 2.2E-08	2.2E-08 1.1E-08	1.1E-08 2.1E-09	1.4E-09 1.4E-09	3.1E-10 3.1E-10	4.5E-10 4.5E-10	2.4E-09 2.4E-09		
Five Most Probable Bins That Have Early CF*																
6	ABABAEAEB	3.6E+03	3.2E+01	0.0E+00 0.0E+00	1.3E+04 1.3E+04	1.8E+02 1.4E+04	7.1E-01 0.0E+00	1.5E-02 2.3E-02	1.4E-02 9.4E-03	6.5E-03 3.5E-03	2.0E-03 0.0E+00	1.3E-03 0.0E+00	3.8E-04 0.0E+00	5.4E-04 0.0E+00	2.3E-03 0.0E+00	
16	AAEEBAEAB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	8.3E-01 0.0E+00	2.7E-02 3.0E-03	2.1E-02 0.0E+00	1.5E-02 0.0E+00	6.3E-03 0.0E+00	1.2E-03 0.0E+00	4.4E-04 0.0E+00	2.3E-03 0.0E+00	6.5E-03 0.0E+00	
18	AAEEBAEAB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	7.7E-01 0.0E+00	1.5E-02 2.5E-03	1.4E-02 0.0E+00	1.1E-02 0.0E+00	6.9E-03 0.0E+00	1.1E-03 0.0E+00	4.4E-04 0.0E+00	2.0E-03 0.0E+00	7.0E-03 0.0E+00	
20	ABABEAEB	3.6E+03	3.2E+01	0.0E+00 0.0E+00	1.3E+04 1.3E+04	1.8E+02 1.4E+04	6.7E-01 0.0E+00	3.6E-02 4.2E-02	3.9E-02 1.8E-02	1.4E-02 6.3E-03	5.3E-03 0.0E+00	4.6E-03 0.0E+00	1.3E-03 0.0E+00	1.5E-03 0.0E+00	6.1E-03 0.0E+00	
22	AAEEBAEAB	3.6E+03	3.2E+01	1.2E+07 0.0E+00	8.3E+03 1.3E+04	4.7E+03 1.4E+04	7.2E-01 0.0E+00	2.6E-03 5.2E-03	1.5E-03 0.0E+00	5.3E-04 0.0E+00	1.7E-05 0.0E+00	1.2E-05 0.0E+00	5.5E-07 0.0E+00	1.2E-06 0.0E+00	2.8E-05 0.0E+00	

\* A listing of source terms for all bins is available on computer media

3.29



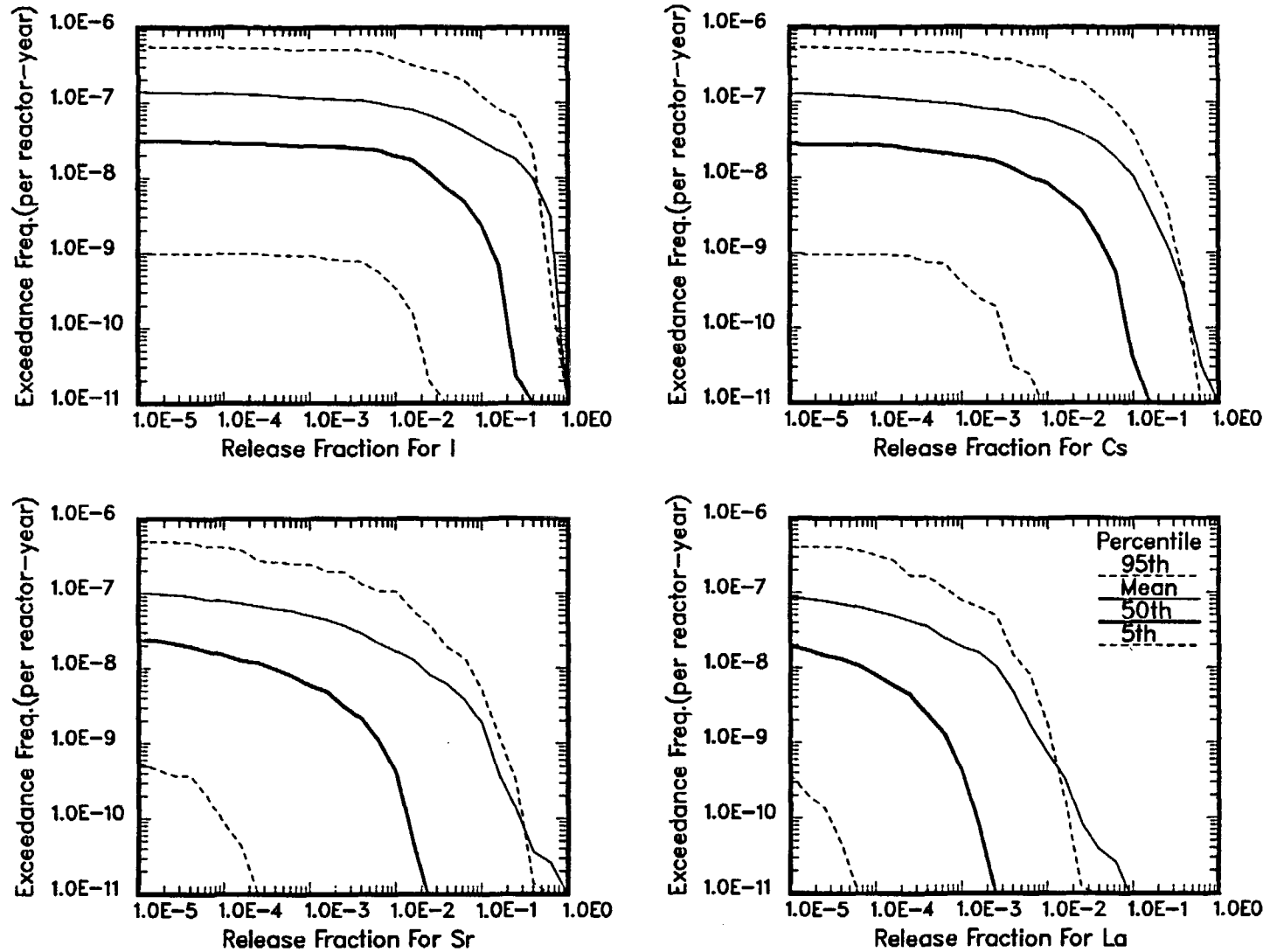


Figure 3.3-3. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 3: Fast SBO.

Table 3.3-4  
Mean Source Terms for Grand Gulf  
Internal Initiators. PDS 4: 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
Ten Most Probable Bins*															
1	BABDAGACB	4.3E+04	3.2E+01	2.3E+07 8.3E+05	7.2E+04 7.3E+04	9.0E+02 1.4E+04	9.0E-01 1.0E-01	1.1E-01 1.2E-02	8.3E-03 9.3E-04	9.9E-03 1.1E-03	6.1E-03 6.8E-04	2.5E-04 2.7E-05	3.7E-04 4.1E-05	6.8E-04 7.5E-05	4.6E-03 5.1E-04
2	BABDAEACB	4.3E+04	3.2E+01	0.0E+00 0.0E+00	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.8E-01 2.2E-01	2.1E-02 1.5E-01	1.7E-02 4.3E-02	1.0E-02 2.9E-02	3.4E-03 2.2E-02	1.3E-03 1.5E-05	3.4E-04 1.1E-03	8.0E-04 2.1E-03	3.6E-03 1.5E-02
3	BABDHBACB	4.3E+04	3.2E+01	4.4E+06 7.9E+06	4.9E+04 5.3E+04	4.7E+03 3.6E+03	7.2E-01 2.8E-01	9.7E-03 2.9E-01	7.3E-03 1.1E-01	4.1E-03 6.1E-02	1.0E-03 3.0E-02	2.0E-04 3.1E-03	4.2E-05 2.3E-03	1.6E-04 3.5E-03	1.1E-03 2.2E-02
4	BBBDAGACB	4.3E+04	3.2E+01	2.3E+07 8.3E+05	7.2E+04 7.3E+04	9.0E+02 1.4E+04	9.0E-01 1.0E-01	1.1E-01 1.2E-02	1.1E-02 1.2E-03	1.1E-02 1.2E-03	5.0E-03 5.5E-04	4.1E-04 4.5E-05	3.6E-04 4.0E-05	7.1E-04 7.8E-05	4.2E-03 4.7E-04
5	BABDAEAE	4.3E+04	3.2E+01	0.0E+00 0.0E+00	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.8E-01 0.0E+00	2.2E-02 9.2E-02	1.8E-02 9.5E-03	1.1E-02 3.6E-03	3.4E-03 0.0E+00	1.6E-03 0.0E+00	3.6E-04 0.0E+00	7.7E-04 0.0E+00	3.8E-03 0.0E+00
6	BABDHEACB	4.3E+04	3.2E+01	0.0E+00 0.0E+00	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.6E-01 2.4E-01	2.5E-02 2.6E-01	2.4E-02 9.4E-02	1.2E-02 4.7E-02	4.3E-03 2.4E-02	4.0E-03 4.7E-05	8.6E-04 1.3E-03	1.1E-03 2.2E-03	5.0E-03 1.5E-02
7	BABDAGAE	4.3E+04	3.2E+01	2.3E+07 8.3E+05	7.2E+04 7.3E+04	9.0E+02 1.4E+04	7.0E-01 7.8E-02	6.2E-02 6.9E-03	3.8E-03 4.2E-04	1.9E-03 2.2E-04	2.6E-04 2.9E-05	2.4E-04 2.7E-05	5.6E-05 6.2E-06	7.0E-05 7.7E-06	3.1E-04 3.4E-05
8	BABDHEAE	4.3E+04	3.2E+01	4.4E+06 7.9E+06	4.9E+04 5.3E+04	4.7E+03 3.6E+03	7.2E-01 1.4E-02	9.7E-03 1.0E-01	7.3E-03 4.0E-02	4.1E-03 1.5E-02	1.0E-03 2.5E-03	2.0E-04 3.0E-03	4.2E-05 7.3E-04	1.6E-04 7.2E-04	1.1E-03 3.1E-03
9	BBBDAEACB	4.3E+04	3.2E+01	0.0E+00 0.0E+00	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.1E-01 2.9E-01	2.3E-02 1.3E-01	1.8E-02 4.1E-02	8.8E-03 3.3E-02	2.6E-03 2.8E-02	1.4E-03 1.4E-05	2.9E-04 1.6E-03	4.9E-04 3.0E-03	2.9E-03 2.0E-02
10	BBBDAGAE	4.3E+04	3.2E+01	2.3E+07 8.3E+05	7.2E+04 7.3E+04	9.0E+02 1.4E+04	6.0E-01 6.6E-02	4.1E-02 4.5E-03	5.0E-03 5.5E-04	3.8E-03 4.2E-04	7.2E-04 8.0E-05	2.7E-04 3.1E-05	6.4E-05 7.1E-06	1.3E-04 1.5E-05	7.7E-04 8.6E-05
Five Most Probable Bins That Have Early CF and Early Suppression Pool Bypass*															
3	BABDHBACB	4.3E+04	3.2E+01	4.4E+06 7.9E+06	4.9E+04 5.3E+04	4.7E+03 3.6E+03	7.2E-01 2.8E-01	9.7E-03 2.9E-01	7.3E-03 1.1E-01	4.1E-03 6.1E-02	1.0E-03 3.0E-02	2.0E-04 3.1E-03	4.2E-05 2.3E-03	1.6E-04 3.5E-03	1.1E-03 2.2E-02
6	BABDHEACB	4.3E+04	3.2E+01	0.0E+00 0.0E+00	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.6E-01 2.4E-01	2.5E-02 2.6E-01	2.4E-02 9.4E-02	1.2E-02 4.7E-02	4.3E-03 2.4E-02	4.0E-03 4.7E-05	8.6E-04 1.3E-03	1.1E-03 2.2E-03	5.0E-03 1.5E-02
8	BABDHEAE	4.3E+04	3.2E+01	4.4E+06 7.9E+06	4.9E+04 5.3E+04	4.7E+03 3.6E+03	7.2E-01 1.4E-02	9.7E-03 1.0E-01	7.3E-03 4.0E-02	4.1E-03 1.5E-02	1.0E-03 2.5E-03	2.0E-04 3.0E-03	4.2E-05 7.3E-04	1.6E-04 7.2E-04	1.1E-03 3.1E-03
11	BABDHEAE	4.3E+04	3.2E+01	0.0E+00 0.0E+00	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.6E-01 0.0E+00	2.6E-02 9.3E-02	2.4E-02 2.3E-02	1.2E-02 7.3E-03	4.3E-03 0.0E+00	4.0E-03 0.0E+00	8.7E-04 0.0E+00	1.1E-03 0.0E+00	5.1E-03 0.0E+00
12	BBBDHEACB	4.3E+04	3.2E+01	0.0E+00 0.0E+00	5.4E+04 5.5E+04	9.0E+02 1.4E+04	6.3E-01 3.7E-01	3.5E-02 3.5E-01	3.4E-02 1.3E-01	1.5E-02 6.8E-02	7.1E-03 3.8E-02	5.9E-03 5.2E-05	1.2E-03 2.6E-03	1.8E-03 4.6E-03	8.1E-03 2.6E-02

\* A listing of source terms for all bins is available on computer media

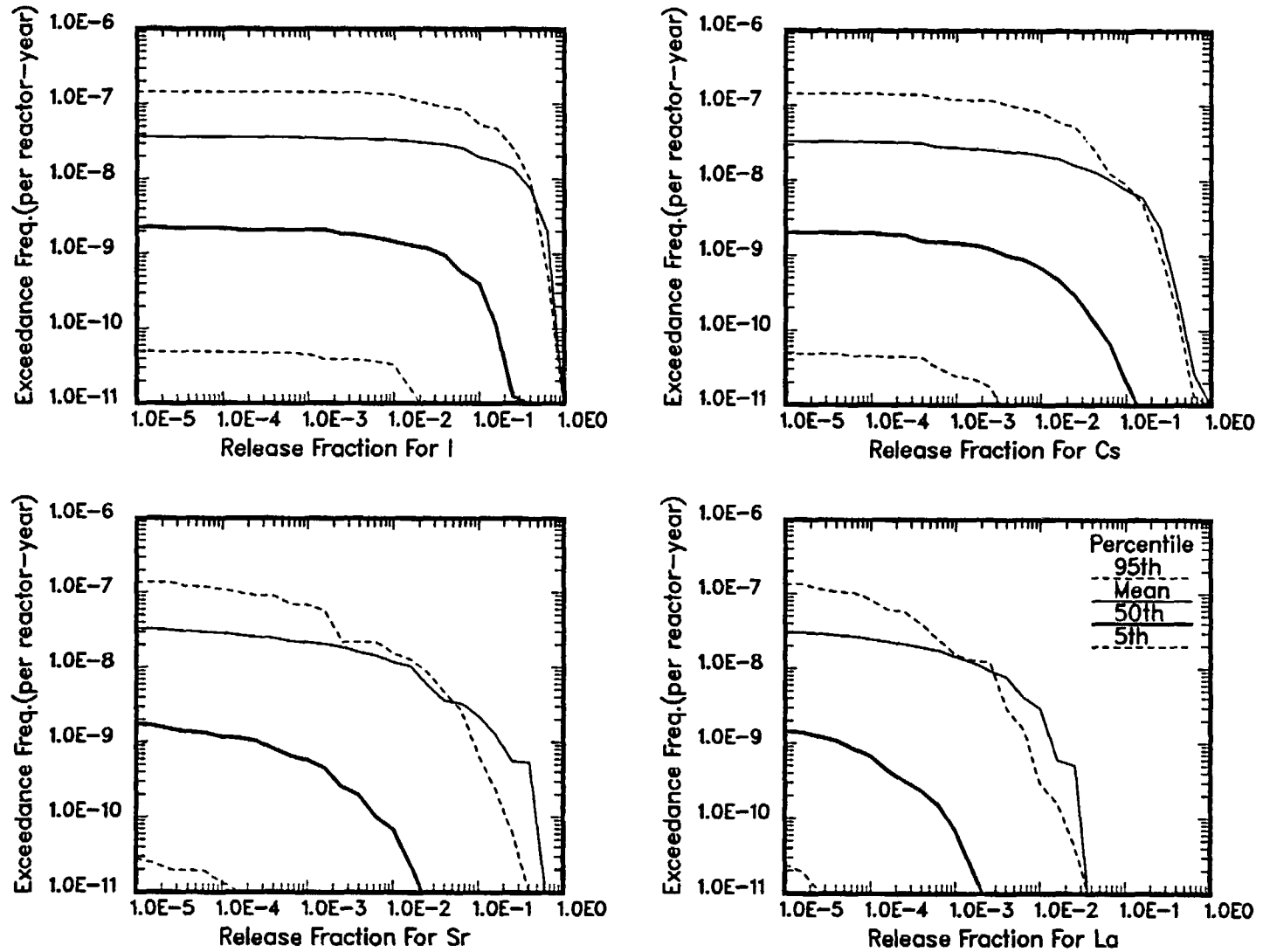


Figure 3.3-4. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 4: Slow SBO.

Table 3.3-5  
 Mean Source Terms for Grand Gulf  
 Internal Initiators. PDS 5: Slow SBO

Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (W)	Release Start (s)	Release Duration (s)	Release Fractions								
							MG	I	Cs	Te	Sr	Ru	La	Ce	Ba
Ten Most Probable Bins*															
1	BABDAGACB	4.3E+04	3.2E+01	2.3E+07 8.3E+05	7.2E+04 7.3E+04	9.0E+02 1.4E+04	9.0E-01 1.0E-01	1.1E-01 1.2E-02	8.3E-03 9.3E-04	9.9E-03 1.1E-03	6.1E-03 6.8E-04	2.5E-04 2.7E-05	3.7E-04 4.1E-05	6.8E-04 7.5E-05	4.6E-03 5.1E-04
2	BABDAEACB	4.3E+04	3.2E+01	0.0E+00 5.5E+04	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.8E-01 2.2E-01	2.1E-02 1.5E-01	1.7E-02 4.3E-02	1.0E-02 2.9E-02	3.4E-03 2.2E-02	1.3E-03 1.5E-05	3.4E-04 1.1E-03	8.0E-04 2.1E-03	3.6E-03 1.5E-02
3	BABDHBACB	4.3E+04	3.2E+01	4.4E+06 7.9E+06	4.9E+04 5.3E+04	4.7E+03 3.6E+03	7.2E-01 2.8E-01	9.7E-03 2.9E-01	7.3E-03 1.1E-01	4.1E-03 6.1E-02	1.0E-03 3.0E-02	2.0E-04 3.1E-03	4.2E-05 2.3E-03	1.6E-04 3.5E-03	1.1E-03 2.2E-02
4	BBBDAGACB	4.3E+04	3.2E+01	2.3E+07 8.3E+05	7.2E+04 7.3E+04	9.0E+02 1.4E+04	9.0E-01 1.0E-01	1.1E-01 1.2E-02	1.1E-02 1.2E-03	1.1E-02 1.2E-03	5.0E-03 5.5E-04	4.1E-04 4.5E-05	3.6E-04 4.0E-05	7.1E-04 7.8E-05	4.2E-03 4.7E-04
5	BBBDAEACB	4.3E+04	3.2E+01	0.0E+00 5.5E+04	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.1E-01 2.3E-02	2.9E-01 1.3E-01	1.8E-02 4.1E-02	8.8E-03 3.3E-02	2.6E-03 2.8E-02	1.4E-03 1.4E-05	2.9E-04 1.6E-03	4.9E-04 3.0E-03	2.9E-03 2.0E-02
6	BABDAEAEB	4.3E+04	3.2E+01	0.0E+00 5.5E+04	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.8E-01 2.2E-02	2.9E-01 2.2E-02	1.8E-02 1.8E-02	1.1E-02 3.4E-03	5.0E-03 1.6E-03	4.1E-04 3.6E-04	3.6E-04 7.7E-04	7.1E-04 3.8E-03	4.2E-03 3.8E-03
7	BABDAGAEB	4.3E+04	3.2E+01	2.3E+07 8.3E+05	7.2E+04 7.3E+04	9.0E+02 1.4E+04	7.0E-01 7.8E-02	6.2E-02 6.9E-03	3.8E-03 4.2E-04	1.9E-03 2.2E-04	2.6E-04 2.9E-05	2.4E-04 2.7E-05	5.6E-05 6.2E-06	7.0E-05 7.7E-06	3.1E-04 3.4E-05
8	BABDBEACB	4.3E+04	3.2E+01	0.0E+00 5.5E+04	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.6E-01 2.4E-01	2.5E-02 2.6E-01	2.4E-02 9.4E-02	1.2E-02 4.7E-02	4.3E-03 2.4E-02	4.0E-03 4.7E-05	8.6E-04 1.3E-03	1.1E-03 2.2E-03	5.0E-03 1.5E-02
9	BABDHAEB	4.3E+04	3.2E+01	4.4E+06 7.9E+06	4.9E+04 5.3E+04	4.7E+03 3.6E+03	7.2E-01 1.4E-02	9.7E-03 1.0E-01	7.3E-03 4.0E-02	4.1E-03 1.5E-02	1.0E-03 2.5E-03	2.0E-04 3.0E-03	4.2E-05 7.3E-04	1.6E-04 7.2E-04	1.1E-03 3.1E-03
10	BBBDAGAEB	4.3E+04	3.2E+01	2.3E+07 8.3E+05	7.2E+04 7.3E+04	9.0E+02 1.4E+04	6.0E-01 6.6E-02	4.1E-02 4.5E-03	5.0E-03 5.5E-04	3.8E-03 4.2E-04	7.2E-04 8.0E-05	2.7E-04 3.1E-05	6.4E-05 7.1E-06	1.3E-04 1.5E-05	7.7E-04 8.6E-05
Five Most Probable Bins That Have Early CF and Early Suppression Pool Bypass*															
3	BABDHBACB	4.3E+04	3.2E+01	4.4E+06 7.9E+06	4.9E+04 5.3E+04	4.7E+03 3.6E+03	7.2E-01 2.8E-01	9.7E-03 2.9E-01	7.3E-03 1.1E-01	4.1E-03 6.1E-02	1.0E-03 3.0E-02	2.0E-04 3.1E-03	4.2E-05 2.3E-03	1.6E-04 3.5E-03	1.1E-03 2.2E-02
8	BABDBEACB	4.3E+04	3.2E+01	0.0E+00 5.5E+04	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.6E-01 2.4E-01	2.5E-02 2.6E-01	2.4E-02 9.4E-02	1.2E-02 4.7E-02	4.3E-03 2.4E-02	4.0E-03 4.7E-05	8.6E-04 1.3E-03	1.1E-03 2.2E-03	5.0E-03 1.5E-02
9	BABDHAEB	4.3E+04	3.2E+01	4.4E+06 7.9E+06	4.9E+04 5.3E+04	4.7E+03 3.6E+03	7.2E-01 1.4E-02	9.7E-03 1.0E-01	7.3E-03 4.0E-02	4.1E-03 1.5E-02	1.0E-03 2.5E-03	2.0E-04 3.0E-03	4.2E-05 7.3E-04	1.6E-04 7.2E-04	1.1E-03 3.1E-03
12	BBBDBEACB	4.3E+04	3.2E+01	0.0E+00 5.5E+04	5.4E+04 5.5E+04	9.0E+02 1.4E+04	6.3E-01 3.7E-01	3.5E-02 3.5E-01	3.4E-02 1.3E-01	1.5E-02 6.8E-02	7.1E-03 3.8E-02	5.9E-03 5.2E-05	1.2E-03 2.6E-03	1.9E-03 4.6E-03	8.1E-03 2.6E-02
13	BABDBEAEB	4.3E+04	3.2E+01	0.0E+00 5.5E+04	5.4E+04 5.5E+04	9.0E+02 1.4E+04	7.6E-01 0.0E+00	2.6E-02 9.3E-02	2.4E-02 2.3E-02	1.2E-02 7.3E-03	4.3E-03 0.0E+00	4.0E-03 0.0E+00	8.7E-04 0.0E+00	1.1E-03 0.0E+00	5.1E-03 0.0E+00

\* A listing of source terms for all bins is available on computer media

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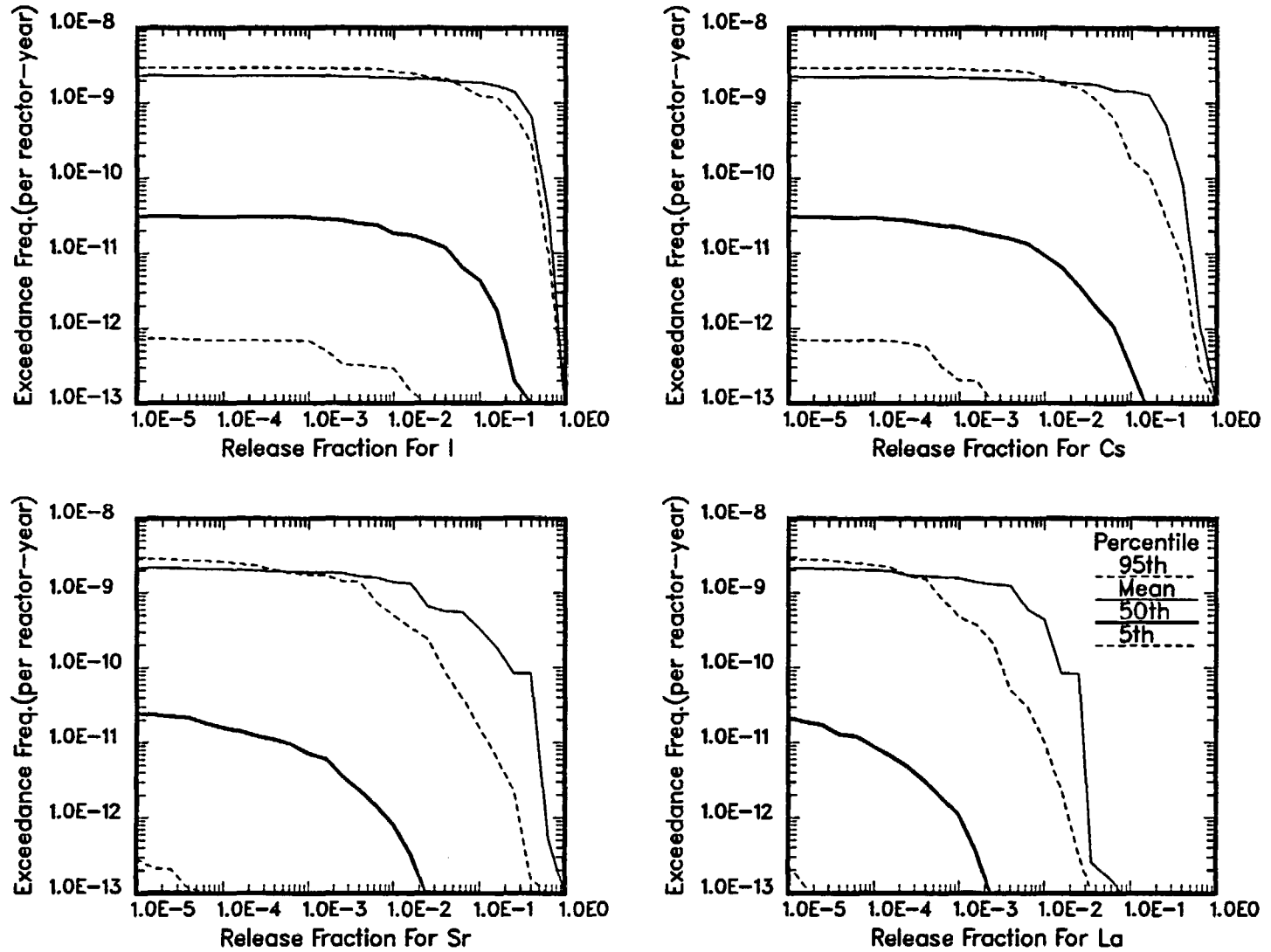


Figure 3.3-5. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 5: Slow SBO.

Table 3.3-6  
Mean Source Terms for Grand Gulf  
Internal Initiators. PDS 6: 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	
Ten Most Probable Bins*																
1	BABDHACB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	7.2E-01	9.7E-03	7.3E-03	4.1E-03	1.0E-03	2.0E-04	4.2E-05	1.6E-04	1.1E-03	
				7.9E+06	5.3E+04	3.6E+03	2.8E-01	2.9E-01	1.1E-01	6.1E-02	3.0E-02	3.1E-03	2.3E-03	3.5E-03	2.2E-02	
2	BABDAGCB	4.3E+04	3.2E+01	2.3E+07	7.2E+04	9.0E+02	9.0E-01	1.1E-01	8.3E-03	9.9E-03	6.1E-03	2.5E-04	3.7E-04	6.8E-04	4.6E-03	
				8.3E+05	7.3E+04	1.4E+04	1.0E-01	1.2E-02	9.3E-04	1.1E-03	6.8E-04	2.7E-05	4.1E-05	7.5E-05	5.1E-04	
3	BABDAEACB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	7.8E-01	2.1E-02	1.7E-02	1.0E-02	3.4E-03	1.3E-03	3.4E-04	8.0E-04	3.6E-03	
				0.0E+00	5.5E+04	1.4E+04	2.2E-01	1.5E-01	4.3E-02	2.9E-02	2.2E-02	1.5E-05	1.1E-03	2.1E-03	1.5E-02	
4	BBBDAGACB	4.3E+04	3.2E+01	2.3E+07	7.2E+04	9.0E+02	9.0E-01	1.1E-01	1.1E-02	1.1E-02	5.0E-03	4.1E-04	3.6E-04	7.1E-04	4.2E-03	
				8.3E+05	7.3E+04	1.4E+04	1.0E-01	1.2E-02	1.2E-03	1.2E-03	5.5E-04	4.5E-05	4.0E-05	7.8E-05	4.7E-04	
5	BABDHBAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	7.2E-01	9.7E-03	7.3E-03	4.1E-03	1.0E-03	2.0E-04	4.2E-05	1.6E-04	1.1E-03	
				7.9E+06	5.3E+04	3.6E+03	1.4E-02	1.0E-01	4.0E-02	1.5E-02	2.5E-03	3.0E-03	7.3E-04	7.2E-04	3.1E-03	
6	BABDAEAEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	7.8E-01	2.2E-02	1.8E-02	1.1E-02	3.4E-03	1.6E-03	3.6E-04	7.7E-04	3.8E-03	
				0.0E+00	5.5E+04	1.4E+04	0.0E+00	9.2E-02	9.5E-03	3.6E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
7	BABDAGAEB	4.3E+04	3.2E+01	2.3E+07	7.2E+04	9.0E+02	7.0E-01	6.2E-02	3.8E-03	1.9E-03	2.6E-04	2.4E-04	5.6E-05	7.0E-05	3.1E-04	
				8.3E+05	7.3E+04	1.4E+04	7.8E-02	6.9E-03	4.2E-04	2.2E-04	2.9E-05	2.7E-05	6.2E-06	7.7E-06	3.4E-05	
8	BBBDAAECB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	7.1E-01	2.3E-02	1.8E-02	8.8E-03	2.6E-03	1.4E-03	2.9E-04	4.9E-04	2.9E-03	
				0.0E+00	5.5E+04	1.4E+04	2.9E-01	1.3E-01	4.1E-02	3.3E-02	2.8E-02	1.4E-05	1.6E-03	3.0E-03	2.0E-02	
9	BABDBEACB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	7.6E-01	2.5E-02	2.4E-02	1.2E-02	4.3E-03	4.0E-03	8.6E-04	1.1E-03	5.0E-03	
				0.0E+00	5.5E+04	1.4E+04	2.4E-01	2.6E-01	9.4E-02	4.7E-02	2.4E-02	4.7E-05	1.3E-03	2.2E-03	1.5E-02	
10	BBBDAGAEB	4.3E+04	3.2E+01	2.3E+07	7.2E+04	9.0E+02	6.0E-01	4.1E-02	5.0E-03	3.8E-03	7.2E-04	2.7E-04	6.4E-05	1.3E-04	7.7E-04	
				8.3E+05	7.3E+04	1.4E+04	6.6E-02	4.5E-03	5.5E-04	4.2E-04	8.0E-05	3.1E-05	7.1E-06	1.5E-05	8.6E-05	
Five Most Probable Bins That Have Early CF and Early Suppression Pool Bypass*																
1	BABDHACB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	7.2E-01	9.7E-03	7.3E-03	4.1E-03	1.0E-03	2.0E-04	4.2E-05	1.6E-04	1.1E-03	
				7.9E+06	5.3E+04	3.6E+03	2.8E-01	2.9E-01	1.1E-01	6.1E-02	3.0E-02	3.1E-03	2.3E-03	3.5E-03	2.2E-02	
5	BABDHBAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	7.2E-01	9.7E-03	7.3E-03	4.1E-03	1.0E-03	2.0E-04	4.2E-05	1.6E-04	1.1E-03	
				7.9E+06	5.3E+04	3.6E+03	1.4E-02	1.0E-01	4.0E-02	1.5E-02	2.5E-03	3.0E-03	7.3E-04	7.2E-04	3.1E-03	
9	BABDBEACB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	7.6E-01	2.5E-02	2.4E-02	1.2E-02	4.3E-03	4.0E-03	8.6E-04	1.1E-03	5.0E-03	
				0.0E+00	5.5E+04	1.4E+04	2.4E-01	2.6E-01	9.4E-02	4.7E-02	2.4E-02	4.7E-05	1.3E-03	2.2E-03	1.5E-02	
12	BBBDBEACB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	6.3E-01	3.5E-02	3.9E-02	1.5E-02	7.1E-03	5.9E-03	1.2E-03	1.9E-03	8.1E-03	
				0.0E+00	5.5E+04	1.4E+04	3.7E-01	3.5E-01	1.3E-01	6.8E-02	3.8E-02	5.2E-05	2.6E-03	4.6E-03	2.6E-02	
13	BABDBEAEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	7.6E-01	2.6E-02	2.4E-02	1.2E-02	4.3E-03	4.0E-03	8.7E-04	1.1E-03	5.1E-03	
				0.0E+00	5.5E+04	1.4E+04	0.0E+00	9.3E-02	2.3E-02	7.3E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	

\* A listing of source terms for all bins is available on computer media

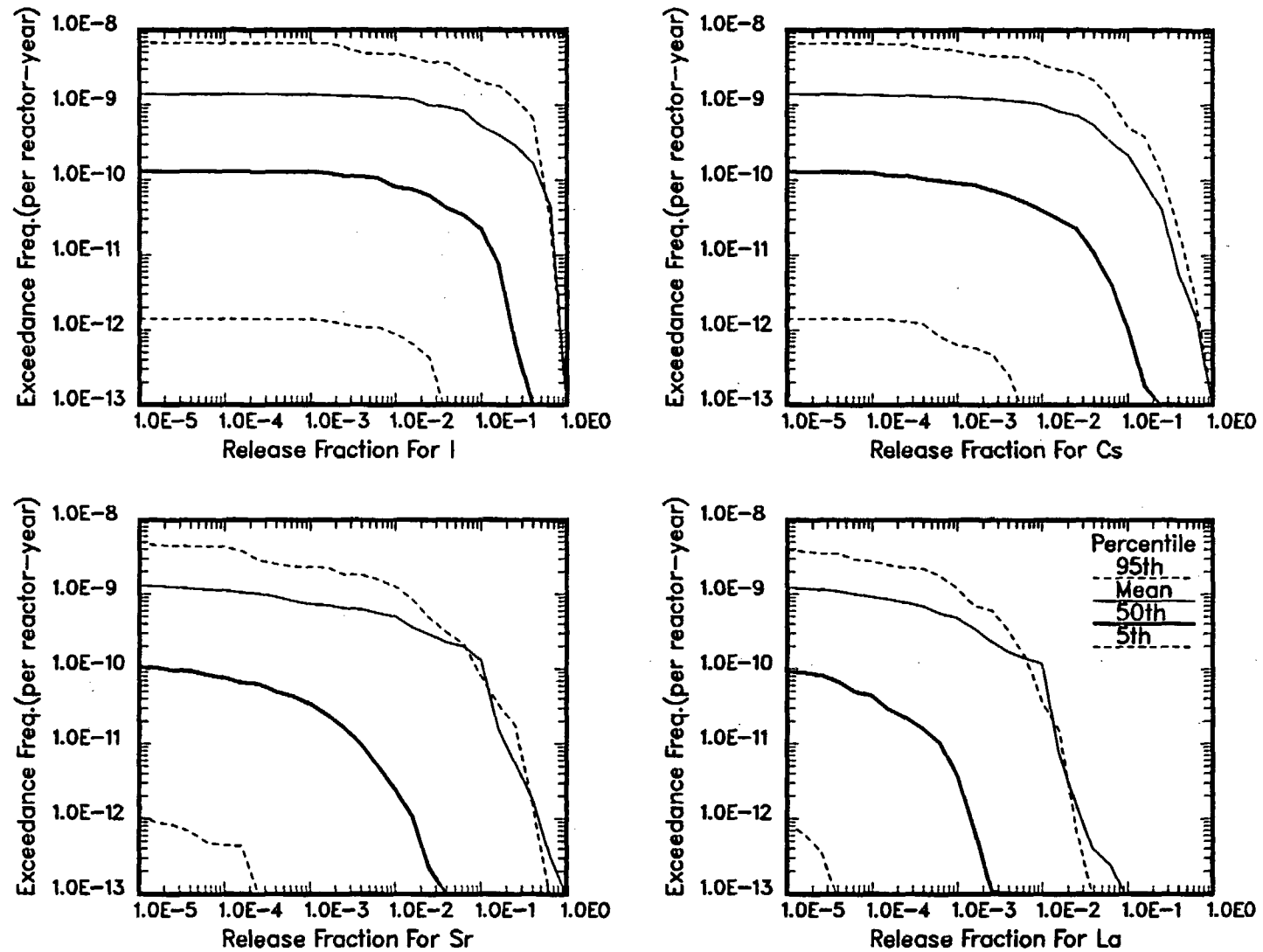


Figure 3.3-6. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 6: Slow SBO.

Table 3.3-7  
Mean Source Terms for Grand Gulf  
Internal Initiators. PDS 7: 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
Ten Most Probable Bins*															
1	ABABAEAEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.1E-01	1.5E-02	1.4E-02	6.5E-03	2.0E-03	1.3E-03	3.8E-04	5.4E-04	2.3E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	2.3E-02	9.4E-03	3.5E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
2	AAABAEAEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.8E-01	1.0E-02	1.1E-02	7.1E-03	3.6E-03	2.8E-03	6.7E-04	1.1E-03	4.0E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	2.3E-02	7.3E-03	1.1E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
3	AAABAEAEB	3.6E+03	3.2E+01	7.5E+05	5.0E+04	7.2E+03	1.9E-03	8.3E-06	1.4E-08	5.6E-09	1.7E-09	1.9E-09	4.4E-10	4.6E-10	2.1E-09
				4.7E+05	5.8E+04	2.2E+04	1.9E-03	8.3E-06	1.4E-08	5.6E-09	1.7E-09	1.9E-09	4.4E-10	4.6E-10	2.1E-09
4	AAABAEAEB	3.6E+03	3.2E+01	7.5E+05	5.0E+04	7.2E+03	3.7E-01	4.2E-03	1.9E-03	1.2E-03	4.5E-04	3.4E-04	7.1E-05	1.3E-04	5.1E-04
				4.7E+05	5.8E+04	2.2E+04	3.7E-01	4.2E-03	1.9E-03	1.2E-03	4.5E-04	3.4E-04	7.1E-05	1.3E-04	5.1E-04
5	ABABAEAEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	6.7E-01	3.6E-02	3.9E-02	1.4E-02	5.3E-03	4.6E-03	1.3E-03	1.5E-03	6.1E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	4.2E-02	1.8E-02	6.3E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
6	AAABAEAEB	3.6E+03	3.2E+01	1.1E+06	8.3E+03	4.7E+03	6.9E-01	9.2E-03	8.2E-03	6.4E-03	2.6E-03	4.4E-04	1.6E-04	7.1E-04	2.6E-03
				6.7E+06	1.3E+04	3.6E+03	3.1E-02	9.5E-02	4.9E-02	1.8E-02	6.6E-03	6.7E-03	1.5E-03	1.7E-03	7.7E-03
7	ABABAEAEB	3.6E+03	3.2E+01	3.0E+07	5.0E+04	1.8E+02	6.2E-01	1.1E-02	6.1E-03	3.5E-03	9.8E-04	3.6E-04	9.1E-05	1.9E-04	1.1E-03
				7.0E+05	5.1E+04	1.4E+04	6.9E-02	1.3E-03	6.8E-04	3.9E-04	1.1E-04	4.0E-05	1.0E-05	2.1E-05	1.2E-04
8	ABABAEAEB	3.6E+03	3.2E+01	7.5E+05	5.0E+04	7.2E+03	3.4E-01	3.9E-03	1.8E-03	7.4E-04	2.2E-04	2.2E-04	4.0E-05	5.3E-05	2.7E-04
				4.7E+05	5.8E+04	2.2E+04	3.4E-01	3.9E-03	1.8E-03	7.4E-04	2.2E-04	2.2E-04	4.0E-05	5.3E-05	2.7E-04
9	AAABAEAEB	3.6E+03	3.2E+01	1.1E+06	8.3E+03	4.7E+03	8.5E-01	6.5E-03	5.6E-03	4.5E-03	2.4E-03	4.0E-04	1.7E-04	9.0E-04	2.4E-03
				6.7E+06	1.3E+04	3.6E+03	1.5E-02	3.1E-02	1.4E-02	4.0E-03	1.7E-03	2.2E-03	5.5E-04	6.3E-04	1.9E-03
10	AACBAFAEB	3.6E+03	3.2E+01	7.5E+05	5.0E+04	7.2E+03	3.9E-01	4.0E-03	1.9E-03	9.9E-04	2.1E-04	8.7E-05	3.5E-05	1.5E-04	2.2E-04
				4.7E+05	5.8E+04	2.2E+04	3.9E-01	4.0E-03	1.9E-03	9.9E-04	2.1E-04	8.7E-05	3.5E-05	1.5E-04	2.2E-04
Five Most Probable Bins That Have Early CF and Early Suppression Pool Bypass*															
5	ABABAEAEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	6.7E-01	3.6E-02	3.9E-02	1.4E-02	5.3E-03	4.6E-03	1.3E-03	1.5E-03	6.1E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	4.2E-02	1.8E-02	6.3E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
6	AAABAEAEB	3.6E+03	3.2E+01	1.1E+06	8.3E+03	4.7E+03	6.9E-01	9.2E-03	8.2E-03	6.4E-03	2.6E-03	4.4E-04	1.6E-04	7.1E-04	2.6E-03
				6.7E+06	1.3E+04	3.6E+03	3.1E-02	9.5E-02	4.9E-02	1.8E-02	6.6E-03	6.7E-03	1.5E-03	1.7E-03	7.7E-03
13	AAABAEAEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.6E-01	3.0E-02	3.4E-02	2.0E-02	8.2E-03	6.9E-03	2.0E-03	2.3E-03	9.8E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	6.7E-02	2.7E-02	5.7E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
16	AAABAEAEB	3.6E+03	3.2E+01	1.1E+06	8.3E+03	4.7E+03	7.1E-01	2.0E-03	1.5E-03	8.5E-04	4.7E-04	1.0E-04	4.9E-05	3.5E-04	4.8E-04
				6.7E+06	1.3E+04	3.6E+03	2.9E-02	8.7E-02	5.9E-02	2.5E-02	7.7E-03	6.4E-03	1.9E-03	1.9E-03	9.4E-03
18	AACBHAEB	3.6E+03	3.2E+01	1.1E+06	8.3E+03	4.7E+03	7.2E-01	2.2E-03	1.8E-03	1.1E-03	6.7E-04	1.4E-04	6.6E-05	4.7E-04	6.8E-04
				6.7E+06	1.3E+04	3.6E+03	2.8E-02	4.6E-02	3.8E-02	1.5E-02	6.2E-03	8.3E-03	2.1E-03	2.3E-03	7.2E-03

\* A listing of source terms for all bins is available on computer media

3.37



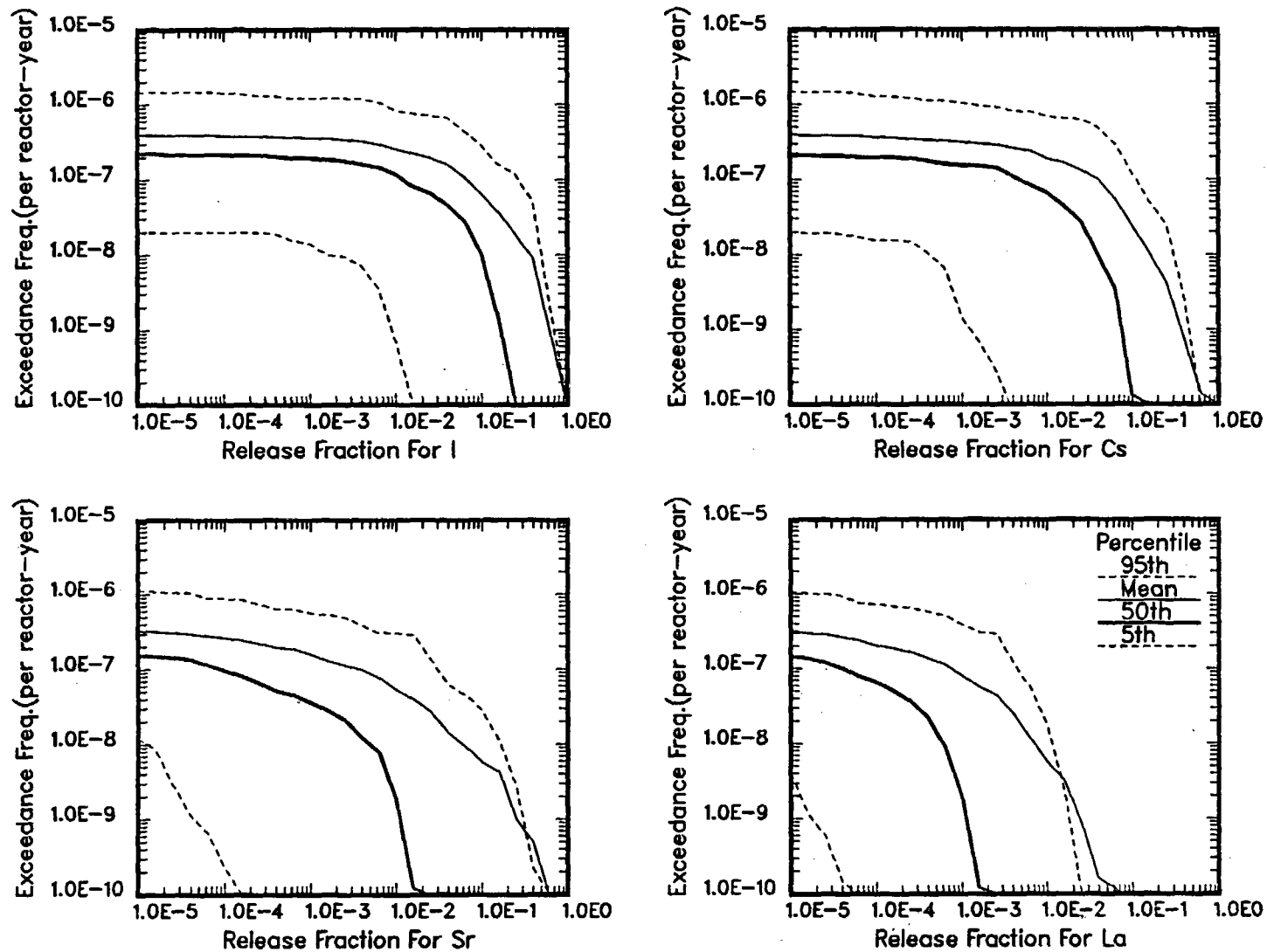


Figure 3.3-7. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 7: Fast SBO.

Table 3.3-8  
Mean Source Terms for Grand Gulf  
Internal Initiators. PDS 8: 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		
Ten Most Probable Bins*				1	BAABAAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	7.6E-01	3.4E-03	2.8E-03	1.9E-03	8.1E-04	1.4E-04	5.1E-05
	2.7E-04	8.2E-04		1.3E+06	5.3E+04	2.2E+04	2.4E-02	3.8E-02	6.5E-03	2.1E-03	7.9E-04	9.0E-04	2.3E-04	2.5E-04	9.2E-04		
2	BBABAAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	6.3E-01	3.3E-03	2.6E-03	1.3E-03	3.3E-04	4.6E-05	1.3E-05	5.2E-05	3.4E-04		
				1.3E+06	5.3E+04	2.2E+04	3.7E-02	4.1E-02	9.5E-03	3.9E-03	4.4E-04	5.0E-04	1.4E-04	1.4E-04	5.4E-04		
3	BAABAAEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	7.7E-01	1.4E-02	1.3E-02	7.6E-03	3.1E-03	2.2E-03	5.7E-04	9.4E-04	3.5E-03		
				0.0E+00	5.5E+04	1.4E+04	0.0E+00	5.6E-02	6.4E-03	1.2E-03	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00		
4	BACBAAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	8.0E-01	2.0E-03	1.4E-03	8.2E-04	5.0E-04	1.0E-04	5.2E-05	3.8E-04	5.1E-04		
				1.3E+06	5.3E+04	2.2E+04	2.0E-02	3.0E-02	8.0E-03	3.4E-03	2.1E-04	3.4E-04	1.6E-04	1.6E-04	3.2E-04		
5	BBABAAEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	9.0E+02	6.6E-01	1.6E-02	1.5E-02	6.8E-03	2.0E-03	1.4E-03	3.6E-04	4.6E-04	2.3E-03		
				0.0E+00	5.5E+04	1.4E+04	0.0E+00	2.9E-02	5.3E-03	8.2E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00		
6	BAABAAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	8.5E-01	5.0E-03	4.0E-03	2.8E-03	6.4E-04	1.6E-04	5.6E-05	3.5E-04	8.6E-04		
				7.9E+06	5.3E+04	3.6E+03	1.5E-02	8.9E-02	1.5E-02	5.3E-03	1.1E-03	1.3E-03	3.9E-04	4.0E-04	1.4E-03		
7	BAABAAEA	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	7.3E-01	8.6E-03	7.4E-03	5.0E-03	1.8E-03	3.2E-04	1.0E-04	4.6E-04	1.9E-03		
				1.3E+06	5.3E+04	2.2E+04	2.8E-02	3.1E-02	6.1E-03	2.2E-03	1.2E-03	1.3E-03	2.9E-04	3.2E-04	1.3E-03		
8	BAABAAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	7.3E-01	4.4E-03	3.7E-03	2.5E-03	9.4E-04	1.6E-04	5.3E-05	2.4E-04	9.6E-04		
				1.3E+06	5.3E+04	2.2E+04	2.7E-01	7.8E-02	2.1E-02	1.1E-02	6.9E-03	1.2E-03	6.2E-04	9.9E-04	5.6E-03		
9	BAABBAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	6.9E-01	1.5E-03	1.3E-03	1.0E-03	4.6E-04	7.8E-05	2.1E-05	8.9E-05	4.8E-04		
				7.9E+06	6.4E+04	3.6E+03	3.1E-01	4.1E-01	2.8E-01	1.5E-01	4.3E-02	6.1E-03	2.8E-03	3.8E-03	2.8E-02		
10	BAABGAEB	4.3E+04	3.2E+01	2.3E+07	7.2E+04	9.0E+02	7.3E-01	3.4E-02	3.1E-03	1.5E-03	3.3E-04	4.8E-04	1.4E-04	1.5E-04	4.1E-04		
				8.3E+05	7.3E+04	1.4E+04	8.1E-02	3.8E-03	3.5E-04	1.7E-04	3.7E-05	5.4E-05	1.5E-05	1.7E-05	4.6E-05		
Five Most Probable Bins That Have Early CF and Early Suppression Pool Bypass*																	
9	BAABBAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	6.9E-01	1.5E-03	1.3E-03	1.0E-03	4.8E-04	7.8E-05	2.1E-05	8.9E-05	4.8E-04		
				7.9E+06	6.4E+04	3.6E+03	3.1E-01	4.1E-01	2.8E-01	1.5E-01	4.3E-02	6.1E-03	2.8E-03	3.8E-03	2.8E-02		
13	BAABBAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	9.2E-01	4.5E-03	3.7E-03	2.1E-03	6.9E-05	7.0E-05	4.0E-06	9.9E-06	1.1E-04		
				1.3E+06	6.4E+04	2.2E+04	8.1E-02	1.3E-01	7.3E-02	6.7E-02	4.9E-02	1.2E-03	1.9E-03	2.5E-03	3.1E-02		
18	BAABAAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	8.1E-01	7.5E-03	7.2E-03	6.2E-03	3.5E-03	5.3E-04	2.2E-04	1.0E-03	3.5E-03		
				1.3E+06	5.3E+04	2.2E+04	1.9E-02	5.3E-02	2.2E-02	8.4E-03	2.7E-03	2.7E-03	7.5E-04	9.5E-04	2.9E-03		
21	BBABBAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	6.1E-01	1.2E-03	1.1E-03	8.9E-04	4.4E-04	5.1E-05	1.8E-05	7.2E-05	4.4E-04		
				1.3E+06	6.4E+04	2.2E+04	3.9E-01	1.4E-01	1.5E-01	1.8E-01	2.0E-01	1.8E-03	1.9E-02	3.1E-02	1.6E-01		
23	BBABBAEB	4.3E+04	3.2E+01	4.4E+06	4.9E+04	4.7E+03	9.2E-01	5.1E-02	4.7E-02	3.4E-02	1.3E-02	1.7E-03	5.2E-04	2.1E-03	1.3E-02		
				7.9E+06	6.4E+04	3.6E+03	8.1E-02	4.3E-01	2.4E-01	2.4E-01	1.5E-01	2.6E-03	1.7E-02	2.6E-02	1.4E-01		

\* A listing of source terms for all bins is available on computer media

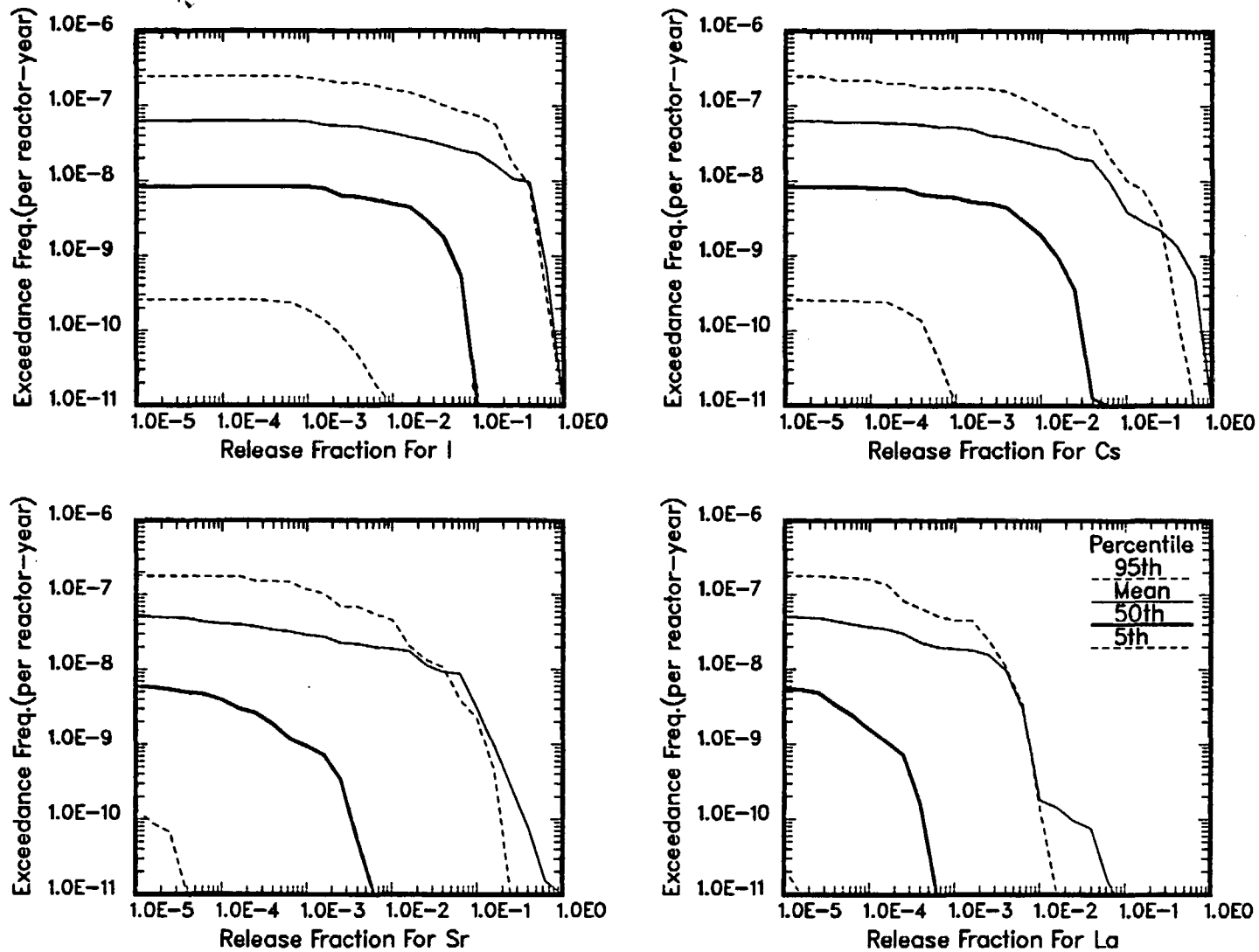


Figure 3.3-8. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 8: Slow SBO.

Table 3.3-9  
 Mean Source Terms for Grand Gulf  
 Internal Initiators. PDS 9: Fast ATWS

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	EAABAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.5E-01	1.6E-02	1.3E-02	6.7E-03	2.7E-03	1.3E-03	3.2E-04	8.0E-04	3.0E-03
2	EBABAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.0E-01	1.7E-02	1.3E-02	3.7E-03	8.5E-04	7.0E-04	1.2E-04	1.4E-04	1.0E-03
3	EACBAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.4E+04	0.0E+00	5.0E-03	4.7E-04	8.8E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
4	EAABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.1E-02	2.4E-04	4.4E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
5	EBABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.5E-01	2.8E-02	2.5E-02	1.4E-02	5.7E-03	1.3E-03	4.7E-04	1.3E-03	5.9E-03
6	EBCBAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	5.2E-01	1.2E-02	1.1E-02	6.8E-03	3.3E-03	1.2E-03	3.1E-04	6.4E-04	3.5E-03
7	EAABAECCB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.5E-01	1.6E-02	1.4E-02	6.9E-03	2.8E-03	1.4E-03	3.3E-04	8.2E-04	3.1E-03
8	EAADAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	8.5E-01	2.4E-02	1.4E-02	5.0E-03	1.2E-03	5.1E-04	1.0E-04	2.1E-04	1.3E-03
9	EAABAFCEB	3.6E+03	3.2E+01	8.0E+06	5.0E+04	7.2E+03	3.9E-01	3.5E-03	1.3E-03	4.9E-04	9.9E-05	4.7E-05	8.8E-06	1.9E-05	1.1E-04
10	EACBBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.6E-01	6.8E-03	6.9E-03	2.2E-03	6.7E-04	3.8E-04	1.4E-04	2.1E-04	7.3E-04
Five Most Probable Bins That Have Early CP and Early Suppression Pool Bypass*															
4	EAABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.7E-01	2.2E-02	1.9E-02	9.2E-03	1.8E-03	1.8E-03	3.0E-04	3.2E-04	2.2E-03
5	EBABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.5E-01	2.8E-02	2.5E-02	1.4E-02	5.7E-03	1.3E-03	4.7E-04	1.3E-03	5.9E-03
10	EACBBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.6E-01	6.8E-03	6.9E-03	2.2E-03	6.7E-04	3.8E-04	1.4E-04	2.1E-04	7.3E-04
15	EACBBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.6E-01	1.3E-02	9.0E-03	3.6E-03	1.0E-03	1.1E-03	3.1E-04	3.9E-04	1.2E-03
20	EAABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	6.1E-01	2.0E-02	1.8E-02	1.0E-02	5.2E-03	1.1E-03	4.3E-04	1.6E-03	5.2E-03

\* A listing of source terms for all bins is available on computer media

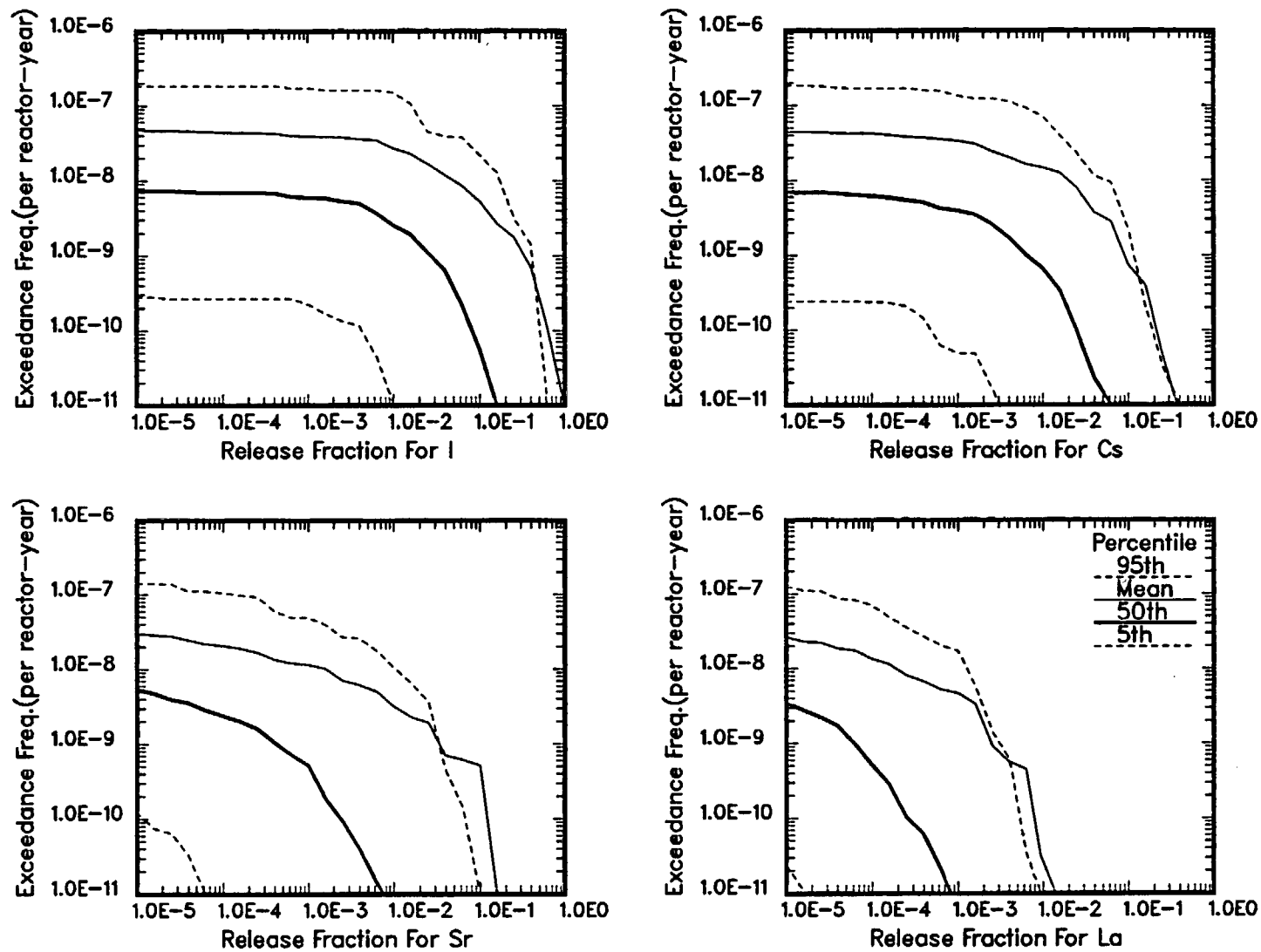


Figure 3.3-9. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 9: Fast ATWS.

Table 3.3-10  
Mean Source Terms for Grand Gulf  
Internal Initiators. PDS 10: Slow ATWS

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	FAABAADEB	2.9E+04	3.2E+01	2.4E+06 8.5E+05	3.2E+04 3.7E+04	4.7E+03 2.2E+04	6.8E-01 3.2E-02	7.9E-03 2.5E-02	6.4E-03 2.3E-03	3.5E-03 1.3E-03	1.9E-03 5.7E-04	3.1E-04 5.7E-04	1.4E-04 1.0E-04	7.4E-04 1.0E-04	1.9E-03 6.7E-04
2	FACBAADEB	2.9E+04	3.2E+01	2.4E+06 8.5E+05	3.2E+04 3.7E+04	4.7E+03 2.2E+04	7.4E-01 2.6E-02	2.4E-03 1.8E-02	1.9E-03 1.6E-03	7.7E-04 6.2E-04	1.0E-04 1.2E-04	3.1E-05 1.6E-04	4.6E-06 5.9E-05	1.7E-05 5.9E-05	1.2E-04 1.6E-04
3	FACBABDEB	2.9E+04	3.2E+01	2.4E+06 5.1E+06	3.2E+04 3.7E+04	4.7E+03 3.6E+03	7.9E-01 2.1E-02	2.7E-03 1.2E-01	2.0E-03 1.4E-03	8.6E-04 3.0E-04	3.1E-04 2.9E-04	5.4E-05 3.9E-04	1.3E-05 8.9E-05	5.1E-05 1.1E-04	3.2E-04 3.1E-04
4	FAABABBE	2.9E+04	3.2E+01	2.4E+06 2.6E+07	3.2E+04 3.7E+04	4.7E+03 3.6E+03	7.9E-01 2.1E-02	4.1E-03 9.4E-02	2.6E-03 9.5E-03	7.0E-04 4.9E-03	1.8E-05 1.8E-03	1.7E-05 1.9E-03	8.6E-07 4.4E-04	2.1E-06 4.4E-04	2.8E-05 2.2E-03
5	FBABAADEB	2.9E+04	3.2E+01	2.4E+06 8.5E+05	3.2E+04 3.7E+04	4.7E+03 2.2E+04	6.4E-01 3.6E-02	7.5E-04 3.5E-02	6.6E-04 5.1E-04	4.3E-04 4.2E-05	2.2E-04 3.9E-06	2.6E-05 9.2E-06	2.2E-04 4.8E-06	3.6E-05 4.7E-06	2.2E-04 5.1E-06
6	FACBBADEB	2.9E+04	3.2E+01	2.4E+06 8.5E+05	3.2E+04 3.7E+04	4.7E+03 2.2E+04	7.6E-01 2.4E-02	2.6E-04 2.8E-02	1.2E-04 1.2E-03	2.0E-05 3.7E-04	5.6E-07 2.0E-05	4.8E-07 3.9E-05	2.9E-08 2.3E-05	7.5E-08 2.3E-05	9.3E-07 3.4E-05
7	FAABAADDB	2.9E+04	3.2E+01	2.4E+06 8.5E+05	3.2E+04 4.8E+04	4.7E+03 2.2E+04	7.1E-01 2.9E-01	4.2E-04 1.2E-01	2.3E-04 4.7E-02	4.5E-05 3.6E-02	1.1E-06 1.8E-02	4.3E-07 2.4E-04	2.6E-08 1.5E-03	5.4E-08 5.0E-04	2.3E-06 1.6E-02
8	FAABABDEB	2.9E+04	3.2E+01	2.4E+06 5.1E+06	3.2E+04 3.7E+04	4.7E+03 3.6E+03	6.9E-01 3.1E-02	5.1E-04 3.2E-02	2.7E-04 1.5E-03	5.3E-05 4.0E-04	1.6E-06 3.2E-05	1.1E-06 6.5E-05	4.3E-08 3.9E-05	9.1E-08 3.9E-05	2.5E-06 6.0E-05
9	FBABABDEB	2.9E+04	3.2E+01	2.4E+06 5.1E+06	3.2E+04 3.7E+04	4.7E+03 3.6E+03	6.2E-01 3.8E-02	9.3E-03 9.8E-02	7.0E-03 4.4E-03	3.3E-03 2.8E-03	1.3E-03 2.6E-05	1.5E-04 8.1E-05	5.3E-05 4.7E-05	2.1E-04 4.5E-05	1.3E-03 4.6E-05
10	FAABABDB	2.9E+04	3.2E+01	2.4E+06 2.6E+07	3.2E+04 4.8E+04	4.7E+03 3.6E+03	8.8E-01 1.2E-01	2.6E-04 2.8E-02	1.2E-04 3.3E-02	2.3E-05 3.7E-02	1.4E-06 3.6E-02	4.8E-07 8.9E-04	5.6E-08 1.9E-03	1.9E-07 1.9E-03	1.9E-06 2.8E-02
Five Most Probable Bins That Have Early CF and Early Suppression Pool Bypass*															
6	FACBBADEB	2.9E+04	3.2E+01	2.4E+06 8.5E+05	3.2E+04 3.7E+04	4.7E+03 2.2E+04	7.6E-01 2.4E-02	2.6E-04 2.8E-02	1.2E-04 1.2E-03	2.0E-05 3.7E-04	5.6E-07 2.0E-05	4.8E-07 3.9E-05	2.9E-08 2.3E-05	7.5E-08 2.3E-05	9.3E-07 3.4E-05
13	FBABBADEB	2.9E+04	3.2E+01	2.4E+06 8.5E+05	3.2E+04 3.7E+04	4.7E+03 2.2E+04	6.7E-01 3.3E-02	2.5E-03 1.3E-02	2.1E-03 1.6E-03	1.2E-03 7.3E-04	5.9E-04 2.2E-04	7.2E-05 2.5E-04	3.3E-05 7.4E-05	1.5E-04 7.3E-05	6.0E-04 2.8E-04
16	FAABBBBE	2.9E+04	3.2E+01	2.4E+06 2.6E+07	3.2E+04 3.7E+04	4.7E+03 3.6E+03	7.5E-01 2.5E-02	1.9E-03 2.3E-01	1.1E-03 2.9E-02	5.4E-04 7.9E-03	3.0E-04 3.3E-03	4.8E-05 3.5E-03	1.5E-05 7.2E-04	6.8E-05 7.0E-04	3.0E-04 3.8E-03
17	FAABBADEB	2.9E+04	3.2E+01	2.4E+06 8.5E+05	3.2E+04 3.7E+04	4.7E+03 2.2E+04	9.4E-01 5.8E-03	4.9E-03 7.6E-02	3.5E-03 2.0E-03	1.3E-03 1.6E-03	4.4E-05 6.7E-04	3.8E-05 6.8E-04	2.5E-06 1.3E-04	6.9E-06 1.3E-04	6.6E-05 8.1E-04
22	FAABBBDEB	2.9E+04	3.2E+01	2.4E+06 5.1E+06	3.2E+04 3.7E+04	4.7E+03 3.6E+03	6.0E-01 4.0E-02	4.3E-03 1.4E-01	3.5E-03 1.0E-02	1.5E-03 5.2E-03	6.4E-05 2.4E-03	5.1E-05 2.3E-03	3.4E-06 3.6E-04	9.5E-06 3.6E-04	9.4E-05 2.7E-03

\* A listing of source terms for all bins is available on computer media

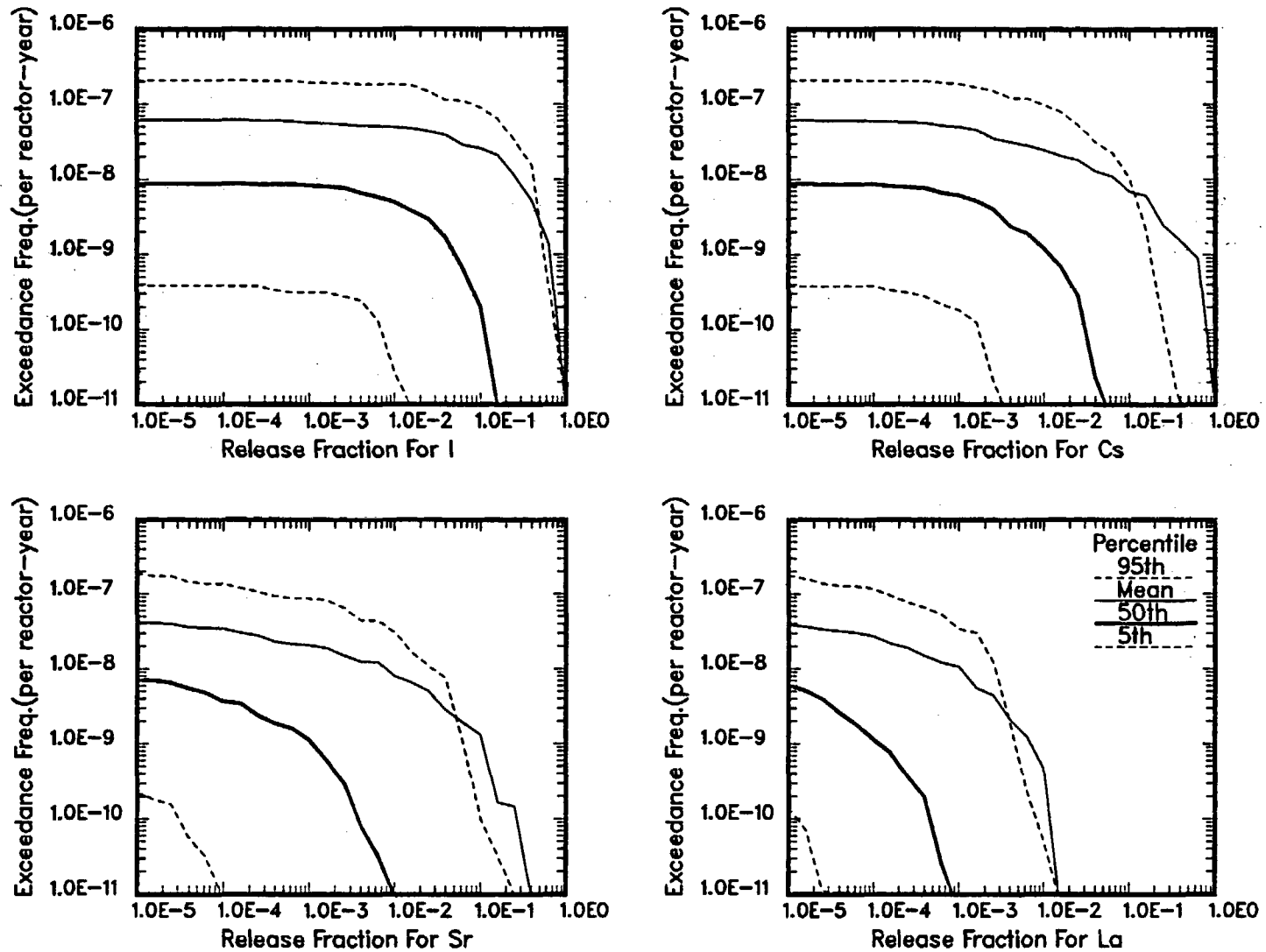


Figure 3.3-10. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 10: Slow ATWS.

Table 3.3-11  
 Mean Source Terms for Grand Gulf  
 Internal Initiators. PDS 11: Fast T2

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	CAABAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.6E-01	2.0E-02	1.6E-02	8.4E-03	3.4E-03	1.7E-03	3.9E-04	9.9E-04	3.7E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.1E-02	3.4E-03	7.7E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
2	CBABAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	6.9E-01	1.1E-02	8.2E-03	2.8E-03	9.1E-04	4.2E-04	8.6E-05	1.5E-04	9.9E-04
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	7.9E-03	1.3E-03	7.2E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
3	CACBAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	8.1E-01	5.6E-03	4.4E-03	1.5E-03	5.8E-04	3.8E-04	8.3E-05	1.3E-04	6.3E-04
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.1E-02	2.5E-04	4.7E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
4	CBABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.3E-01	2.4E-02	2.1E-02	1.1E-02	4.3E-03	9.8E-04	3.6E-04	9.7E-04	4.5E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.1E-02	2.4E-03	6.6E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
5	CAABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.3E-01	2.7E-02	2.4E-02	1.2E-02	2.3E-03	2.3E-03	3.8E-04	4.0E-04	2.8E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.7E-02	9.6E-04	3.4E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
6	CBABAECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	5.9E-01	2.5E-02	2.0E-02	9.8E-03	3.1E-03	1.4E-03	3.8E-04	6.7E-04	3.4E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.8E-03	1.4E-04	4.0E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
7	CAABAECEA	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.6E-01	3.1E-02	2.4E-02	1.2E-02	5.2E-03	2.0E-03	5.4E-04	1.9E-03	5.5E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.1E-02	3.3E-03	7.4E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
8	CAABAECCB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.5E-01	2.0E-02	1.7E-02	8.7E-03	3.6E-03	1.7E-03	4.1E-04	1.0E-03	3.9E-03
				0.0E+00	1.3E+04	1.4E+04	2.5E-01	6.3E-02	2.5E-02	1.3E-02	1.0E-02	7.9E-05	7.9E-04	1.4E-03	7.9E-03
9	CAABAFCEB	3.6E+03	3.2E+01	7.5E+05	5.0E+04	7.2E+03	4.0E-01	3.9E-03	1.2E-03	4.5E-04	8.9E-05	4.3E-05	7.9E-06	1.7E-05	1.0E-04
				9.1E+04	5.8E+04	2.2E+04	4.0E-01	3.9E-03	1.2E-03	4.5E-04	8.9E-05	4.3E-05	7.9E-06	1.7E-05	1.0E-04
10	CAABAICEB	3.6E+03	3.2E+01	7.5E+05	5.0E+04	7.2E+03	2.2E-03	8.5E-06	5.0E-09	2.9E-09	1.6E-09	1.5E-09	2.4E-10	2.5E-10	1.9E-09
				9.1E+04	5.8E+04	2.2E+04	2.2E-03	8.5E-06	5.0E-09	2.9E-09	1.6E-09	1.5E-09	2.4E-10	2.5E-10	1.9E-09
Five Most Probable Bins That Have Early CF and Early Suppression Pool Bypass*															
4	CBABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.3E-01	2.4E-02	2.1E-02	1.1E-02	4.3E-03	9.8E-04	3.6E-04	9.7E-04	4.5E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.1E-02	2.4E-03	6.6E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
5	CAABBECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.3E-01	2.7E-02	2.4E-02	1.2E-02	2.3E-03	2.3E-03	3.8E-04	4.0E-04	2.8E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.7E-02	9.6E-04	3.4E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
13	CACBHECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	8.1E-01	7.8E-03	7.9E-03	2.6E-03	8.2E-04	4.6E-04	1.7E-04	2.5E-04	8.9E-04
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.2E-02	8.7E-04	3.2E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
18	CAABBECEA	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.3E-01	4.0E-02	3.5E-02	1.7E-02	2.4E-03	2.4E-03	3.9E-04	4.2E-04	3.1E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	1.6E-02	9.6E-04	3.4E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
20	CACBHECEB	3.6E+03	3.2E+01	0.0E+00	1.3E+04	1.8E+02	7.6E-01	1.3E-02	9.0E-03	3.6E-03	1.0E-03	1.1E-03	3.1E-04	3.9E-04	1.2E-03
				0.0E+00	1.3E+04	1.4E+04	0.0E+00	9.2E-03	4.2E-04	1.3E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

\* A listing of source terms for all bins is available on computer media



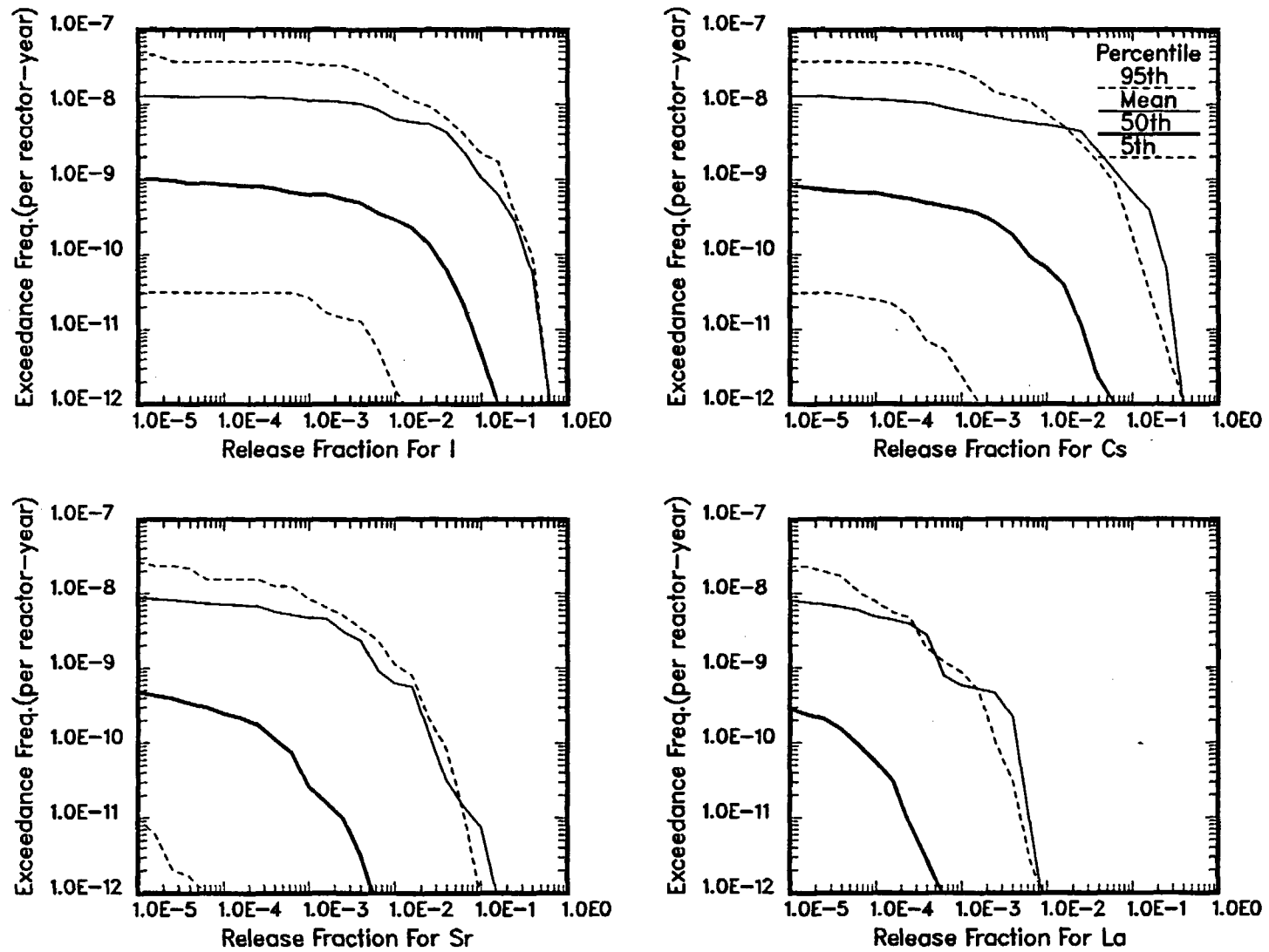


Figure 3.3-11. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 11: Fast T2.

Table 3.3-12  
 Mean Source Terms for Grand Gulf  
 Internal Initiators. PDS 12: Slow T2

Order	Bin	Warning Time (s)	Elevation (m)	Release Energy (H)	Release Start (s)	Release Duration (s)	Release Fractions									
							NG	I	Cs	Te	Sr	Ru	La	Ce	Ba	
Ten Most Probable Bins*																
1	DAABAECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.6E-01	2.0E-02	1.6E-02	8.4E-03	3.4E-03	1.7E-03	3.9E-04	9.9E-04	3.7E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.1E-02	3.4E-03	7.7E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
2	DBABAECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	8.9E-01	1.1E-02	8.2E-03	2.8E-03	9.1E-04	4.2E-04	8.6E-05	1.5E-04	9.8E-04	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	7.9E-03	1.3E-03	7.2E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
3	DACBAECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	8.1E-01	5.6E-03	4.4E-03	1.5E-03	5.6E-04	3.8E-04	8.3E-05	1.3E-04	6.3E-04	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.1E-02	2.5E-04	4.7E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
4	DBABBECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.3E-01	2.4E-02	2.1E-02	1.1E-02	4.3E-03	8.8E-04	3.6E-04	9.7E-04	4.5E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.1E-02	2.4E-03	6.6E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
5	DAABBECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.3E-01	2.7E-02	2.4E-02	1.2E-02	2.3E-03	2.3E-03	3.8E-04	4.0E-04	2.8E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.7E-02	9.6E-04	3.4E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
6	DBCBAECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	5.9E-01	2.5E-02	2.0E-02	9.8E-03	3.1E-03	1.4E-03	3.8E-04	6.7E-04	3.4E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.8E-03	1.4E-04	4.0E-05	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
7	DAABAECEA	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.6E-01	3.1E-02	2.4E-02	1.2E-02	5.2E-03	2.0E-03	5.4E-04	1.9E-03	5.5E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.1E-02	3.3E-03	7.4E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
8	DAABAECCB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.5E-01	2.0E-02	1.7E-02	8.7E-03	3.6E-03	1.7E-03	4.1E-04	1.0E-03	3.9E-03	
				0.0E+00	5.4E+04	1.4E+04	2.5E-01	6.3E-02	2.5E-02	1.3E-02	1.0E-02	7.9E-05	7.9E-04	1.4E-03	7.9E-03	
9	DAABAFCEB	4.3E+04	3.2E+01	2.9E+06	7.2E+04	7.2E+03	4.0E-01	3.9E-03	1.2E-03	4.5E-04	8.9E-05	4.3E-05	7.9E-06	1.7E-05	1.0E-04	
				1.1E+05	7.9E+04	2.2E+04	4.0E-01	3.9E-03	1.2E-03	4.5E-04	8.9E-05	4.3E-05	7.9E-06	1.7E-05	1.0E-04	
10	DAABAICED	4.3E+04	3.2E+01	2.9E+06	7.2E+04	7.2E+03	2.2E-03	8.5E-06	5.0E-09	2.6E-09	1.6E-09	1.5E-09	2.4E-10	2.5E-10	1.9E-09	
				1.1E+05	7.9E+04	2.2E+04	2.2E-03	8.5E-06	5.0E-09	2.6E-09	1.6E-09	1.5E-09	2.4E-10	2.5E-10	1.9E-09	
Five Most Probable Bins That Have Early CF and Early Suppression Pool Bypass*																
4	DBABBECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.3E-01	2.4E-02	2.1E-02	1.1E-02	4.3E-03	9.8E-04	3.6E-04	9.7E-04	4.5E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.1E-02	2.4E-03	6.6E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
5	DAABBECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.3E-01	2.7E-02	2.4E-02	1.2E-02	2.3E-03	2.3E-03	3.8E-04	4.0E-04	2.8E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.7E-02	9.6E-04	3.4E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
13	DACBBECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	8.1E-01	7.8E-03	7.9E-03	2.8E-03	8.2E-04	4.6E-04	1.7E-04	2.5E-04	8.9E-04	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.2E-02	8.7E-04	3.2E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
18	DAABBECEA	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.3E-01	4.0E-02	3.5E-02	1.7E-02	2.4E-03	2.4E-03	3.9E-04	4.2E-04	3.1E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	1.6E-02	9.6E-04	3.4E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
20	DACBBECEB	4.3E+04	3.2E+01	0.0E+00	5.4E+04	1.8E+02	7.6E-01	1.3E-02	9.0E-03	3.6E-03	1.0E-03	1.1E-03	3.1E-04	3.9E-04	1.2E-03	
				0.0E+00	5.4E+04	1.4E+04	0.0E+00	9.2E-03	4.2E-04	1.3E-04	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	

\* A listing of source terms for all bins is available on computer media

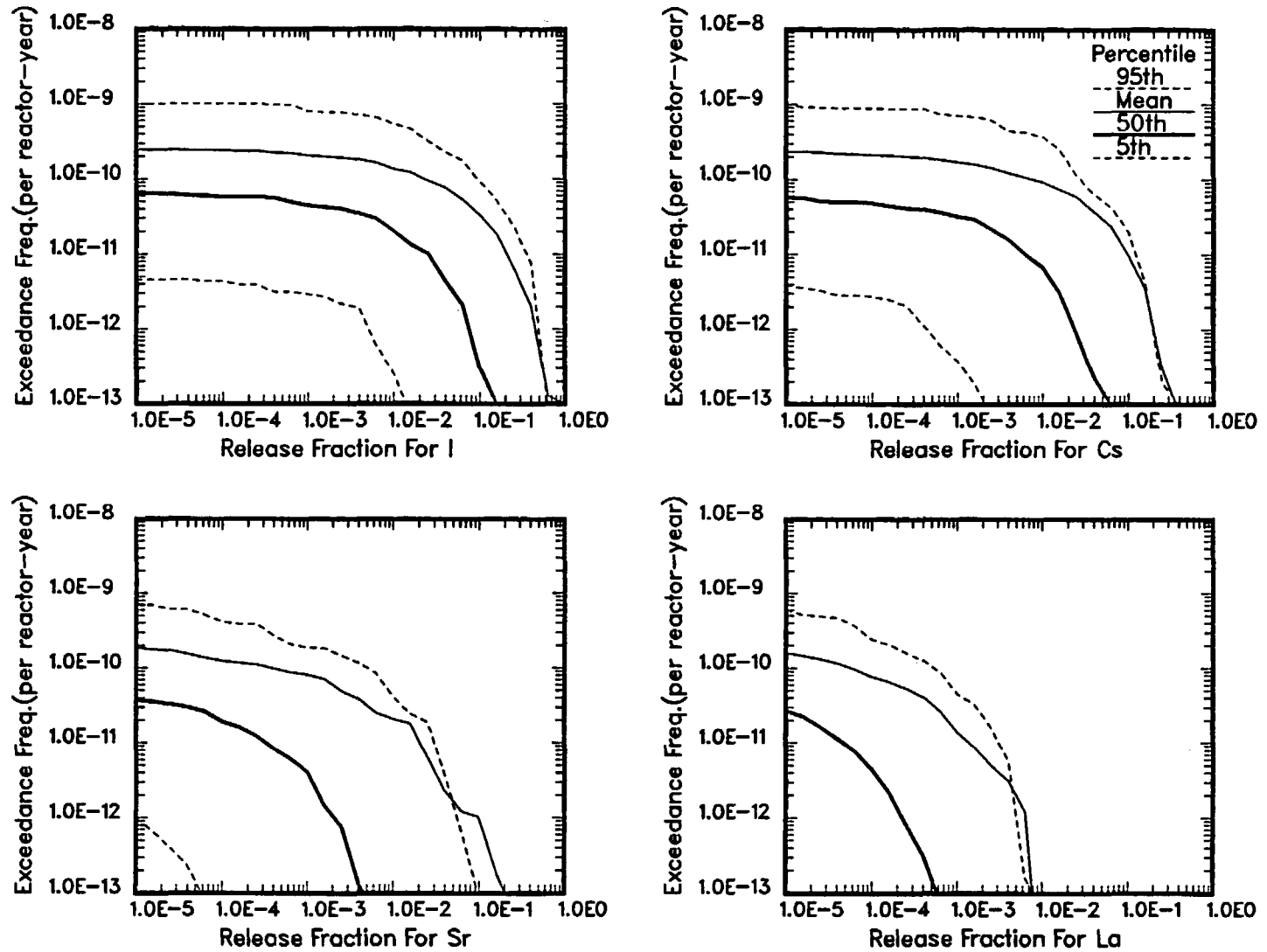


Figure 3.3-12. Exceedance Frequencies for Release Fractions for Grand Gulf Internal Initiators, PDS 12: Slow T2.

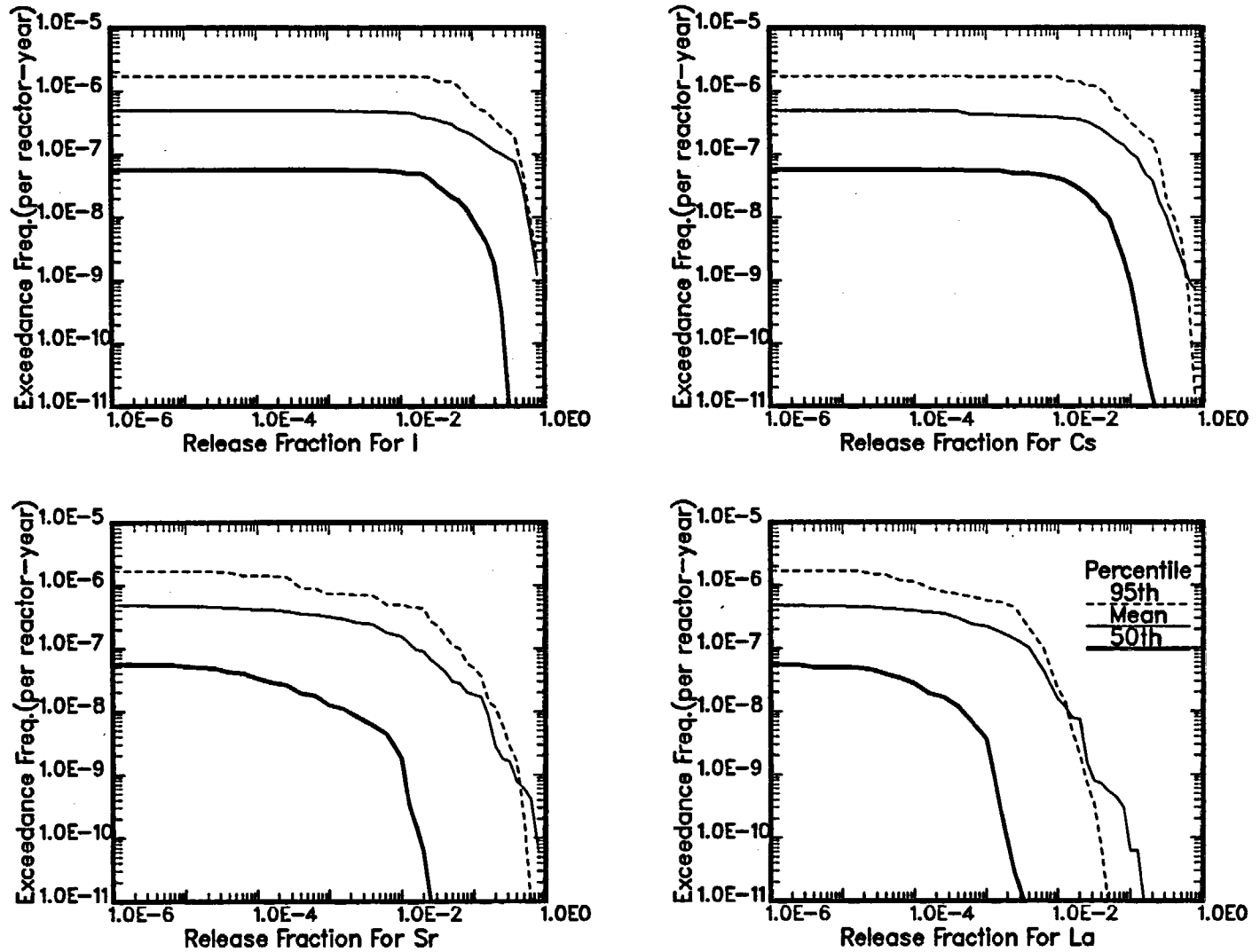


Figure 3.3-13. Exceedance Frequencies for Release Fractions for Grand Gulf. Summary APB 1: Vessel Breach, Early CF, Early SP Bypass, and No CS.

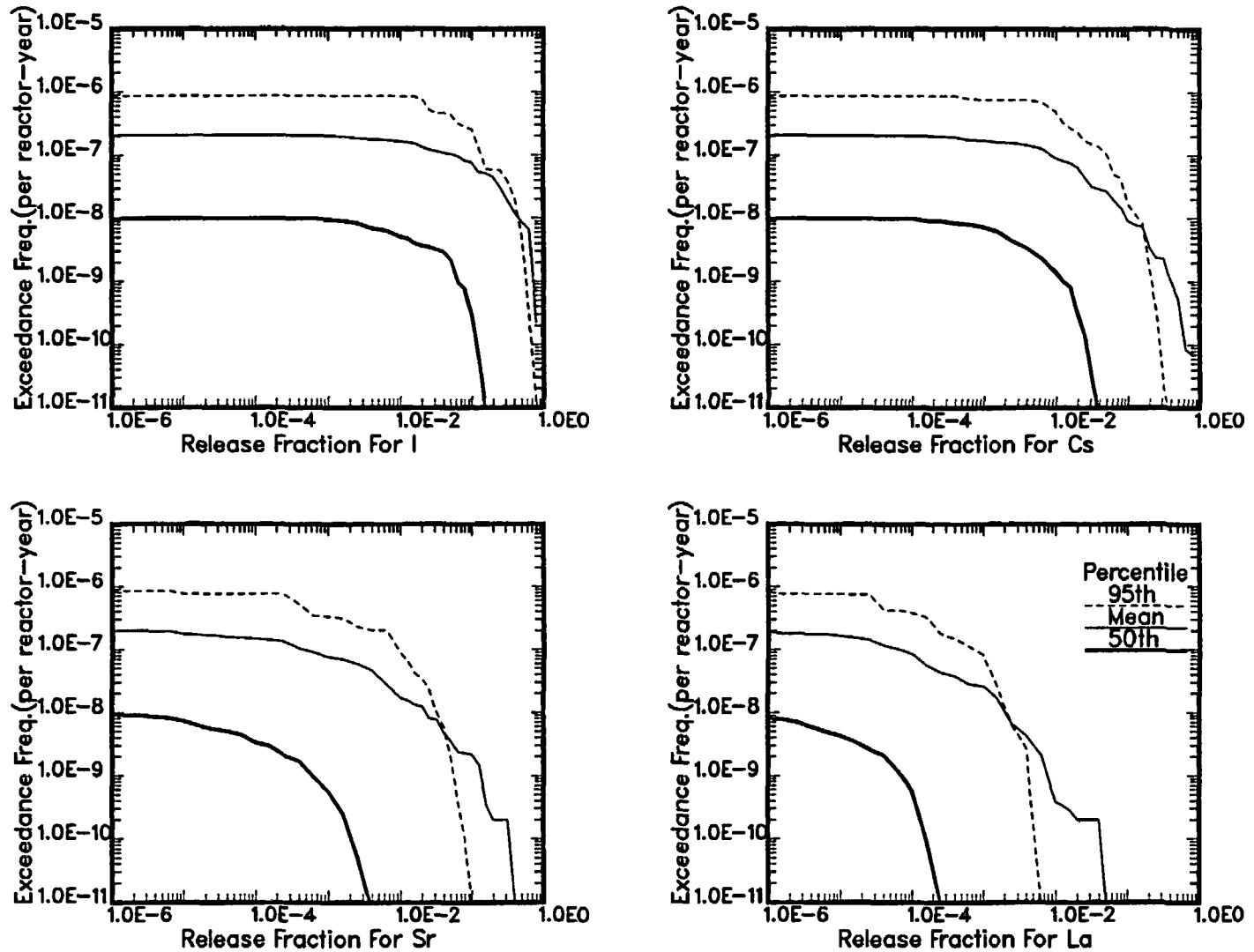


Figure 3.3-14. Exceedance Frequencies for Release Fractions for Grand Gulf. Summary APB 2: Vessel Breach, Early CF, Early SP Bypass, and CS Available.

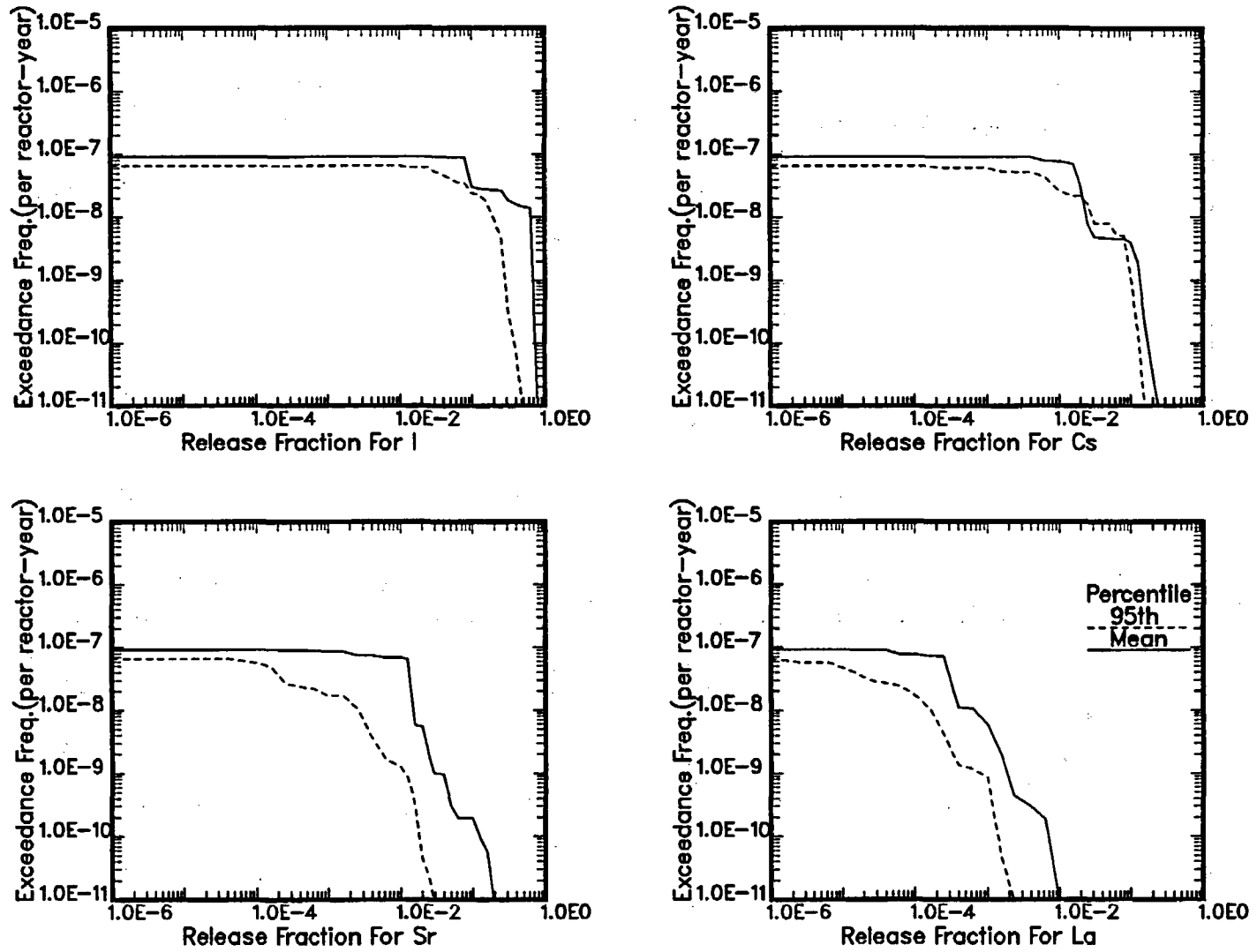


Figure 3.3-15. Exceedance Frequencies for Release Fractions for Grand Gulf. Summary APB 3: Vessel Breach, Early CF, Late SP Bypass.

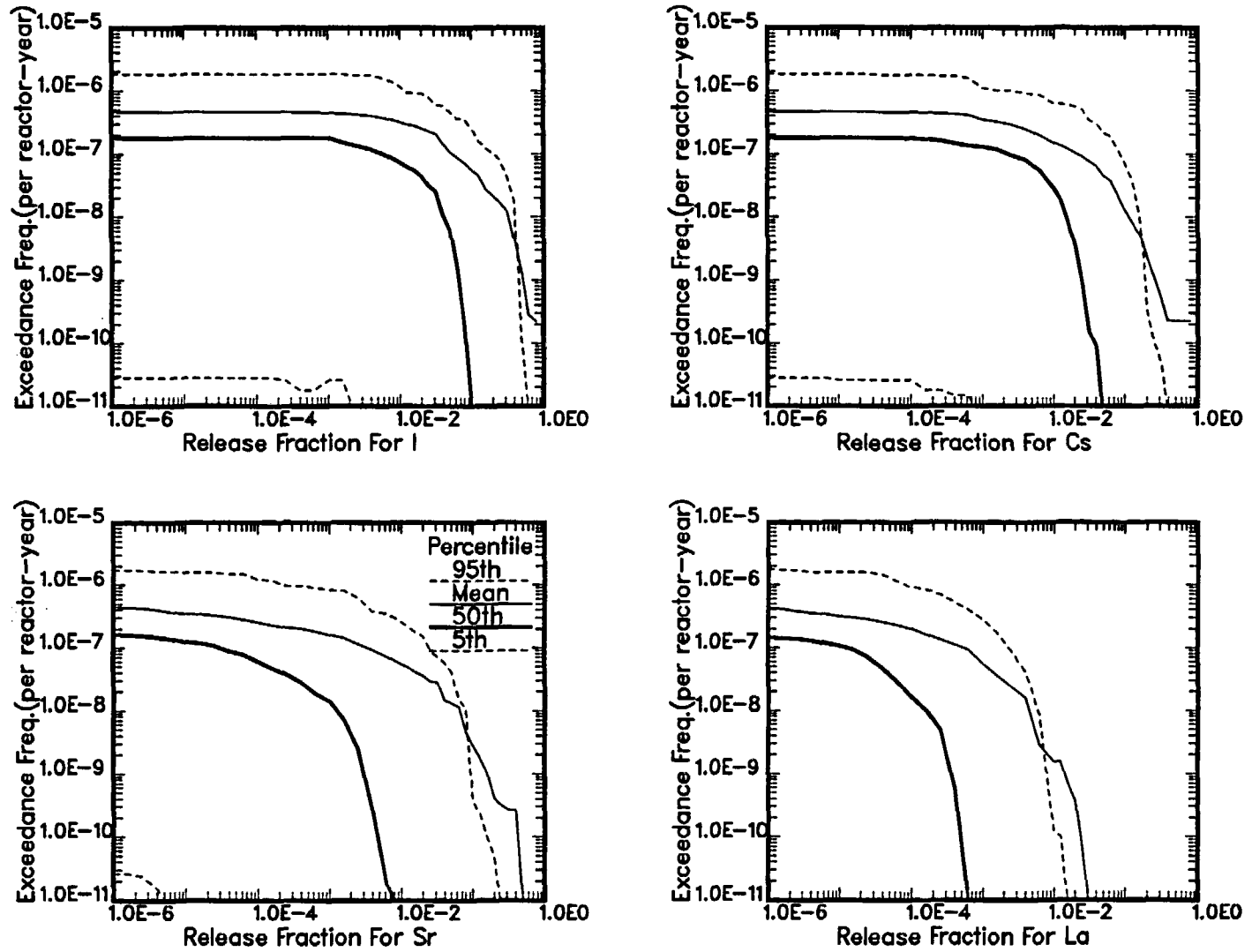


Figure 3.3-16. Exceedance Frequencies for Release Fractions for Grand Gulf. Summary APB 4: Vessel Breach, Early CF, No SP Bypass.

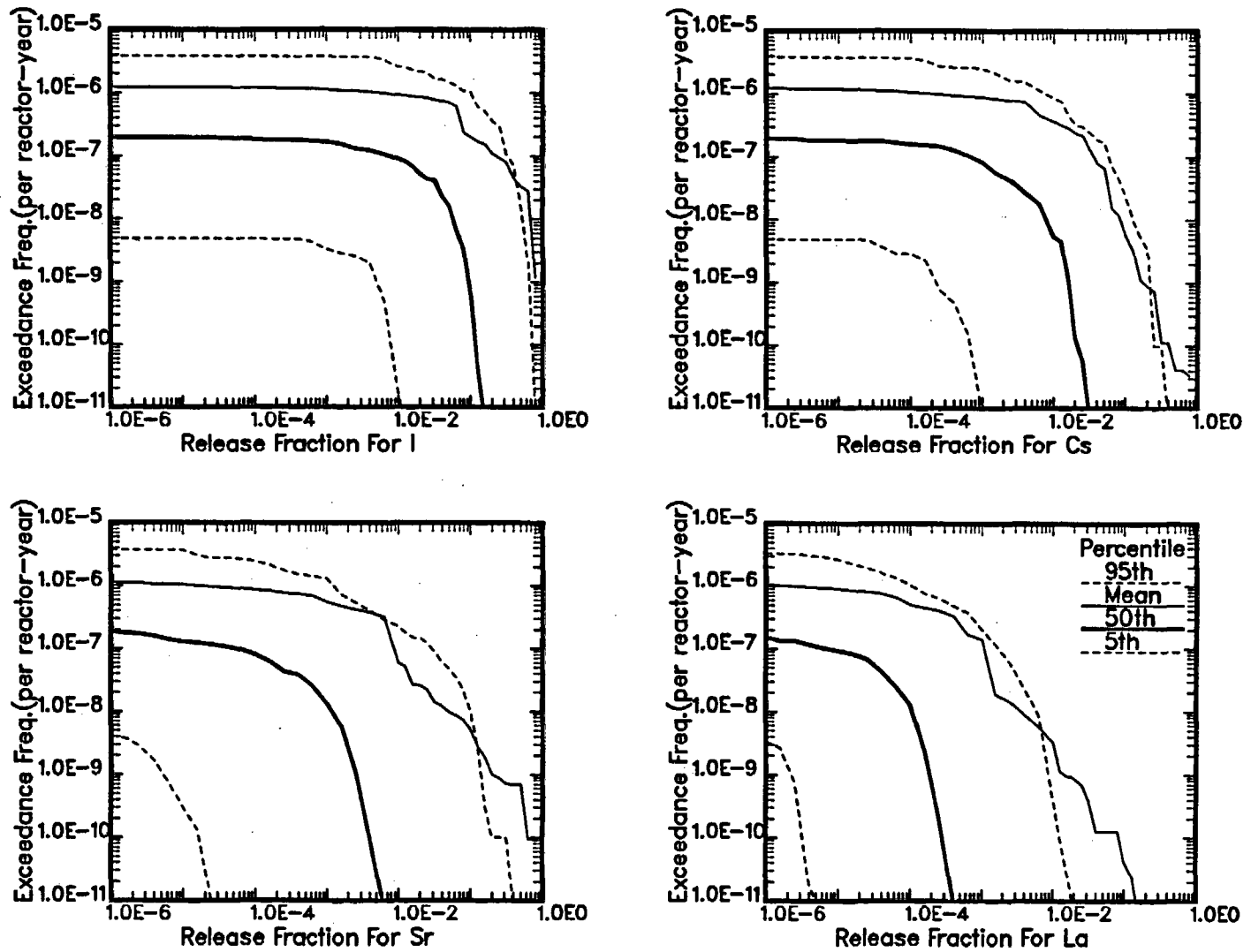


Figure 3.3-17. Exceedance Frequencies for Release Fractions for Grand Gulf. Summary APB 5: Vessel Breach, Late CF.



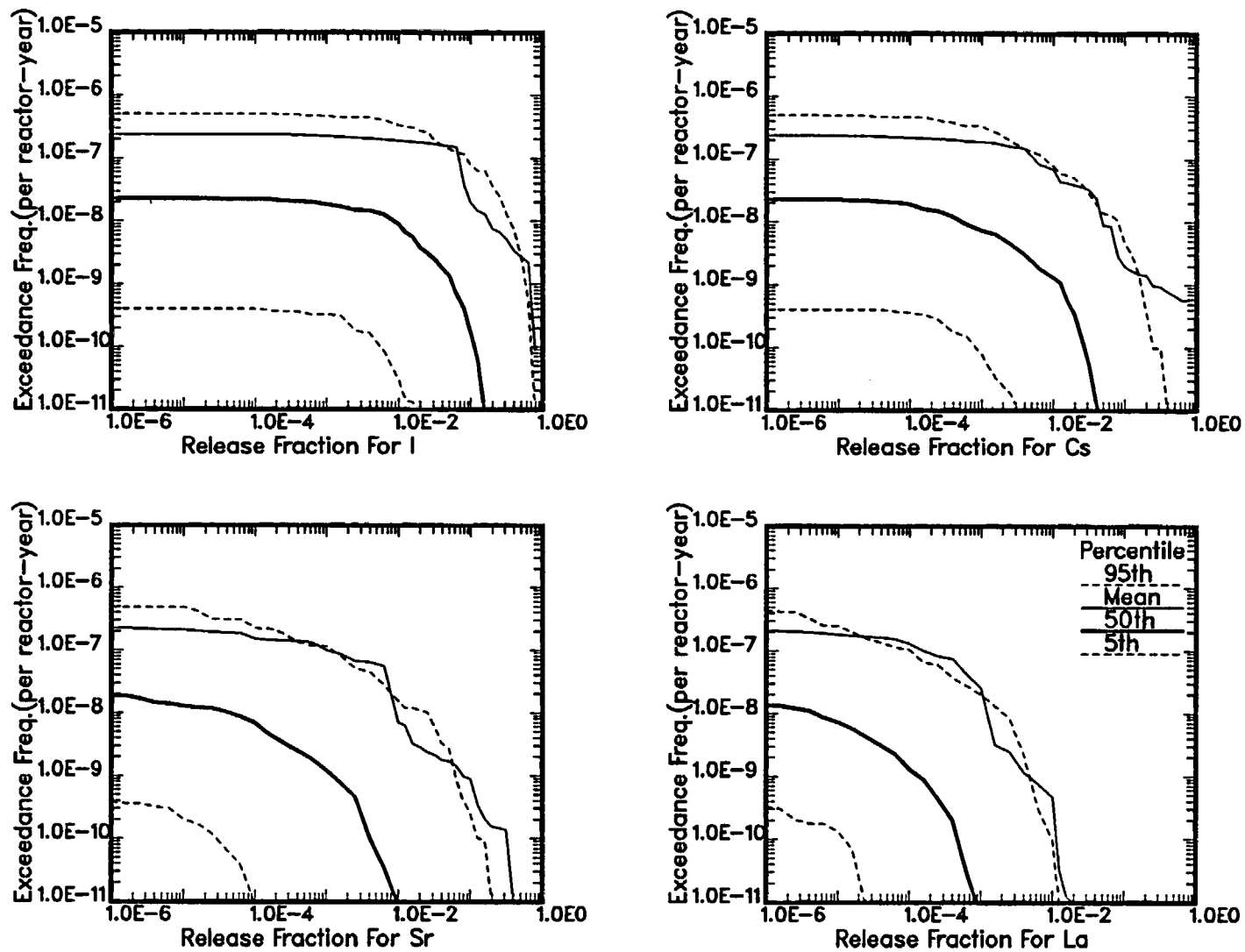


Figure 3.3-18. Exceedance Frequencies for Release Fractions for Grand Gulf. Summary APB 6: Vessel Breach, Containment Vented.

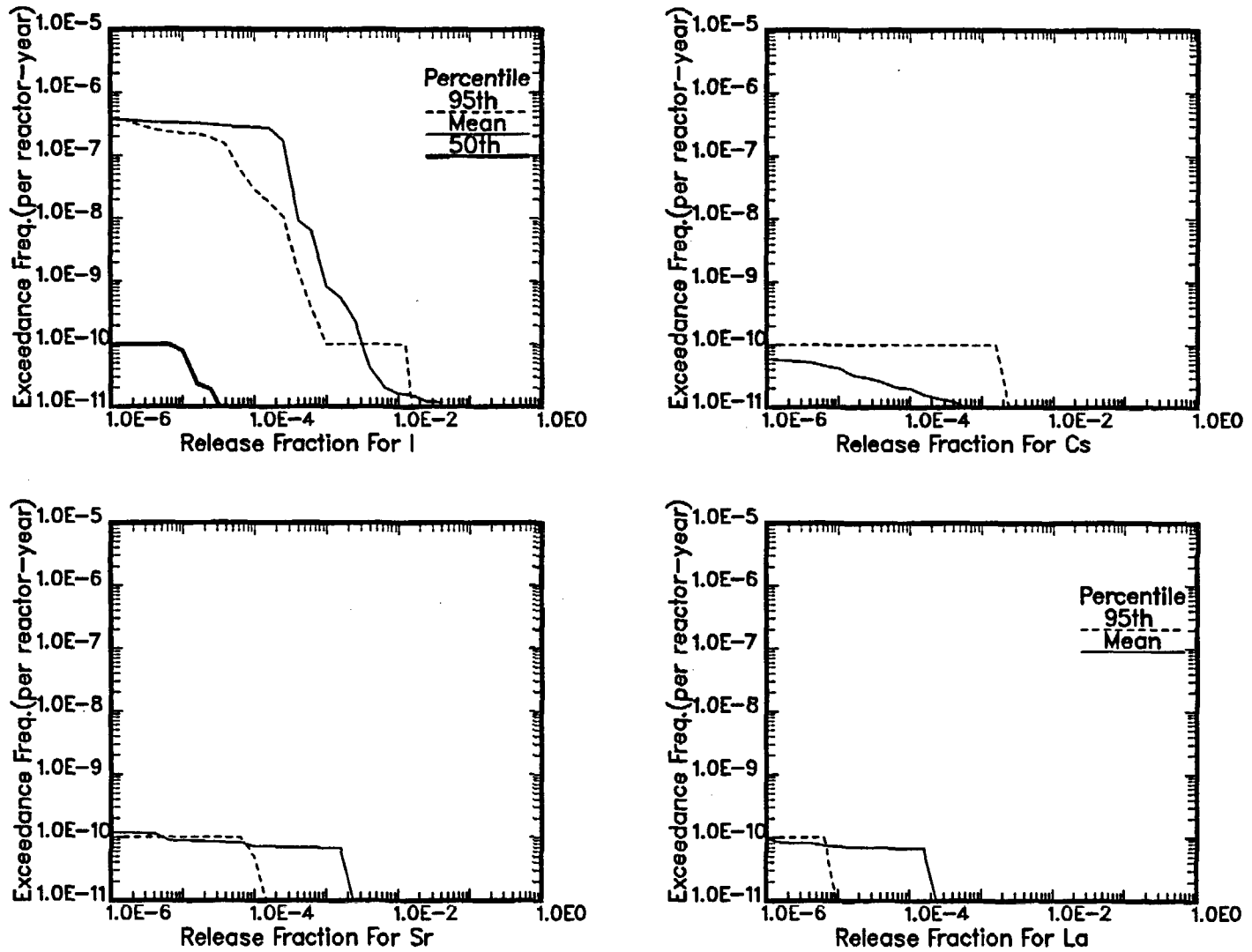


Figure 3.3-19. Exceedance Frequencies for Release Fractions for Grand Gulf. Summary APB 7: Vessel Breach, No CF.

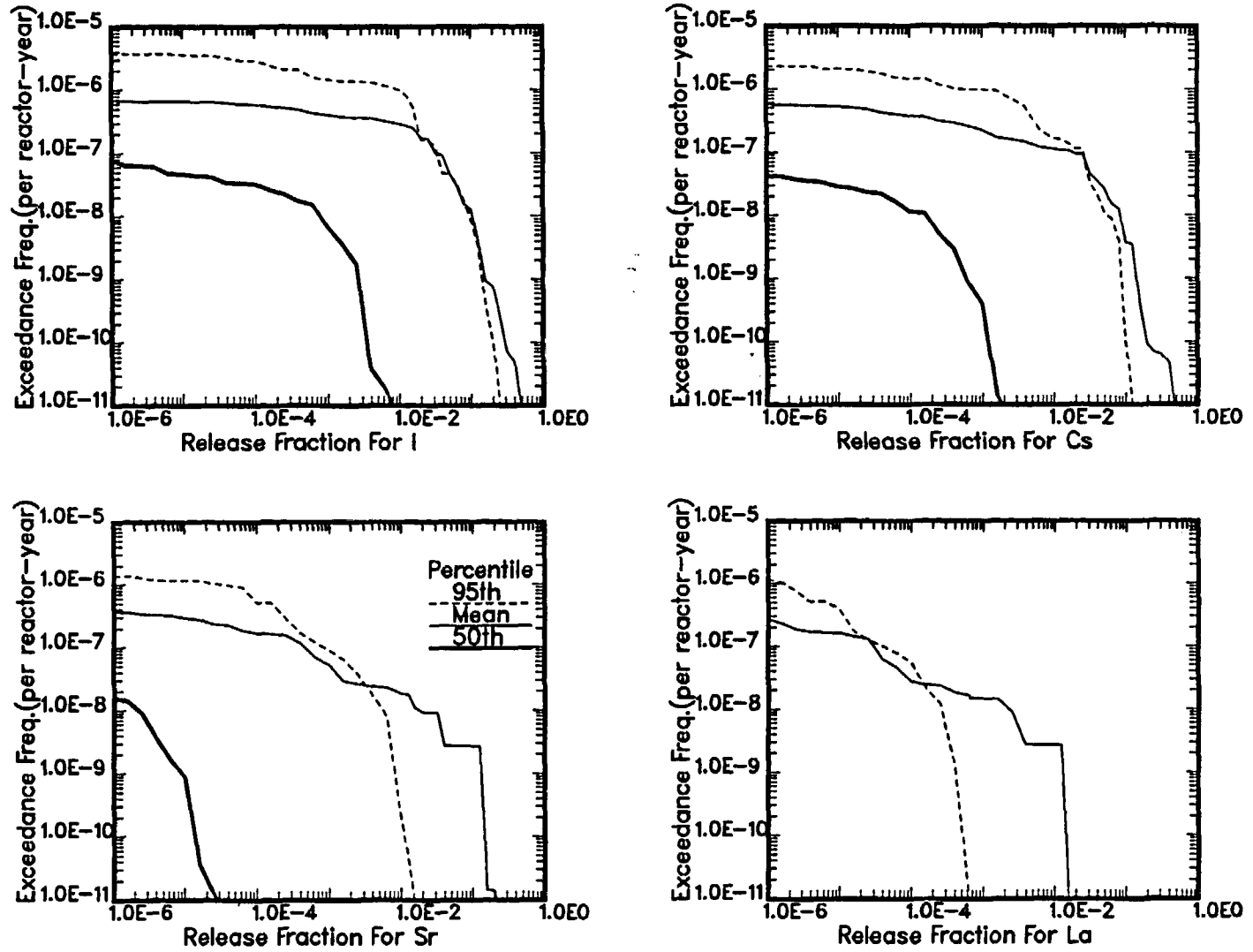


Figure 3.3-20. Exceedance Frequencies for Release Fractions for Grand Gulf. Summary APB 8: No Vessel Breach.

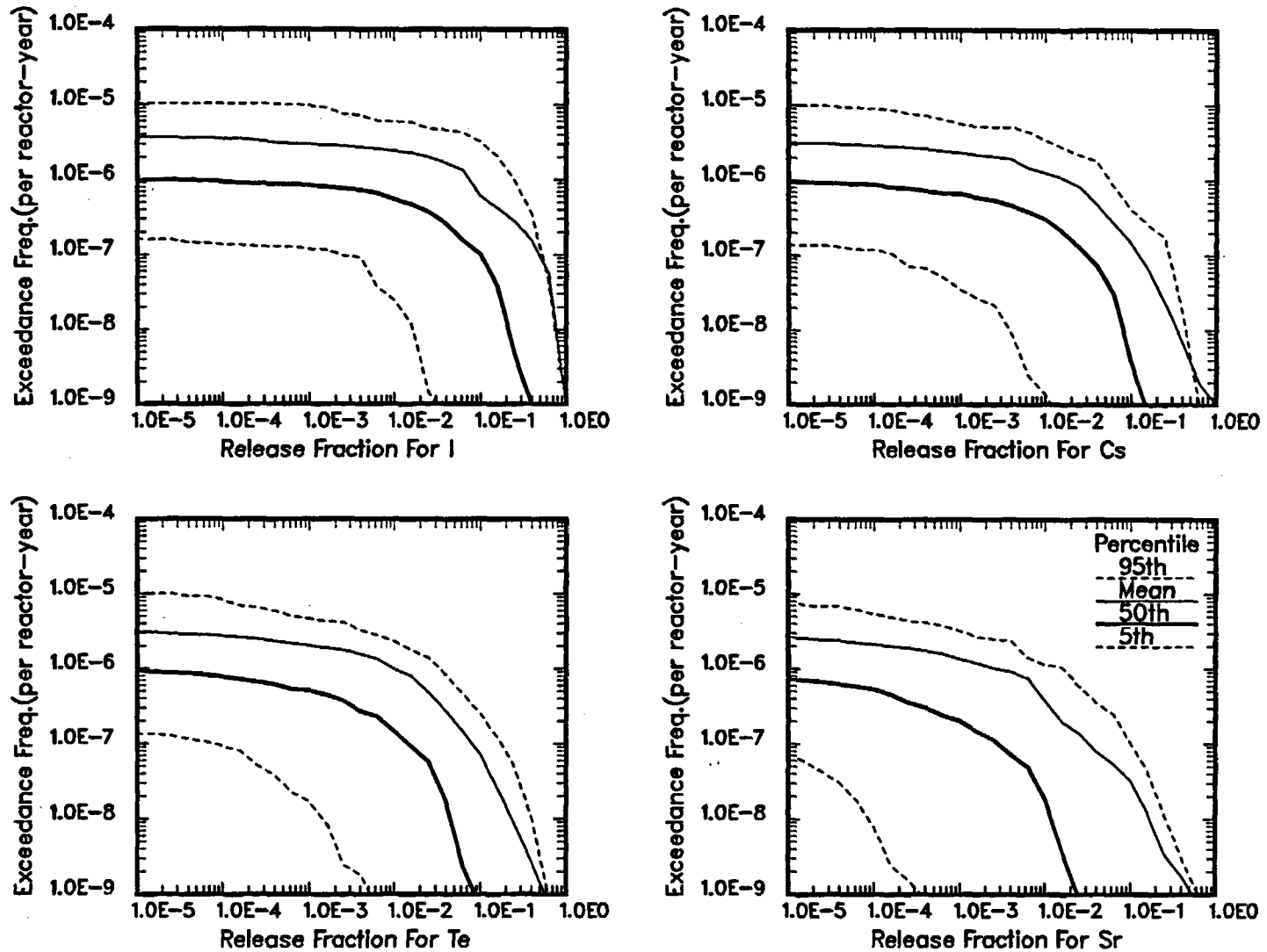


Figure 3.3-21. Exceedance Frequencies for Release Fractions for Grand Gulf. All Internal Initiators.

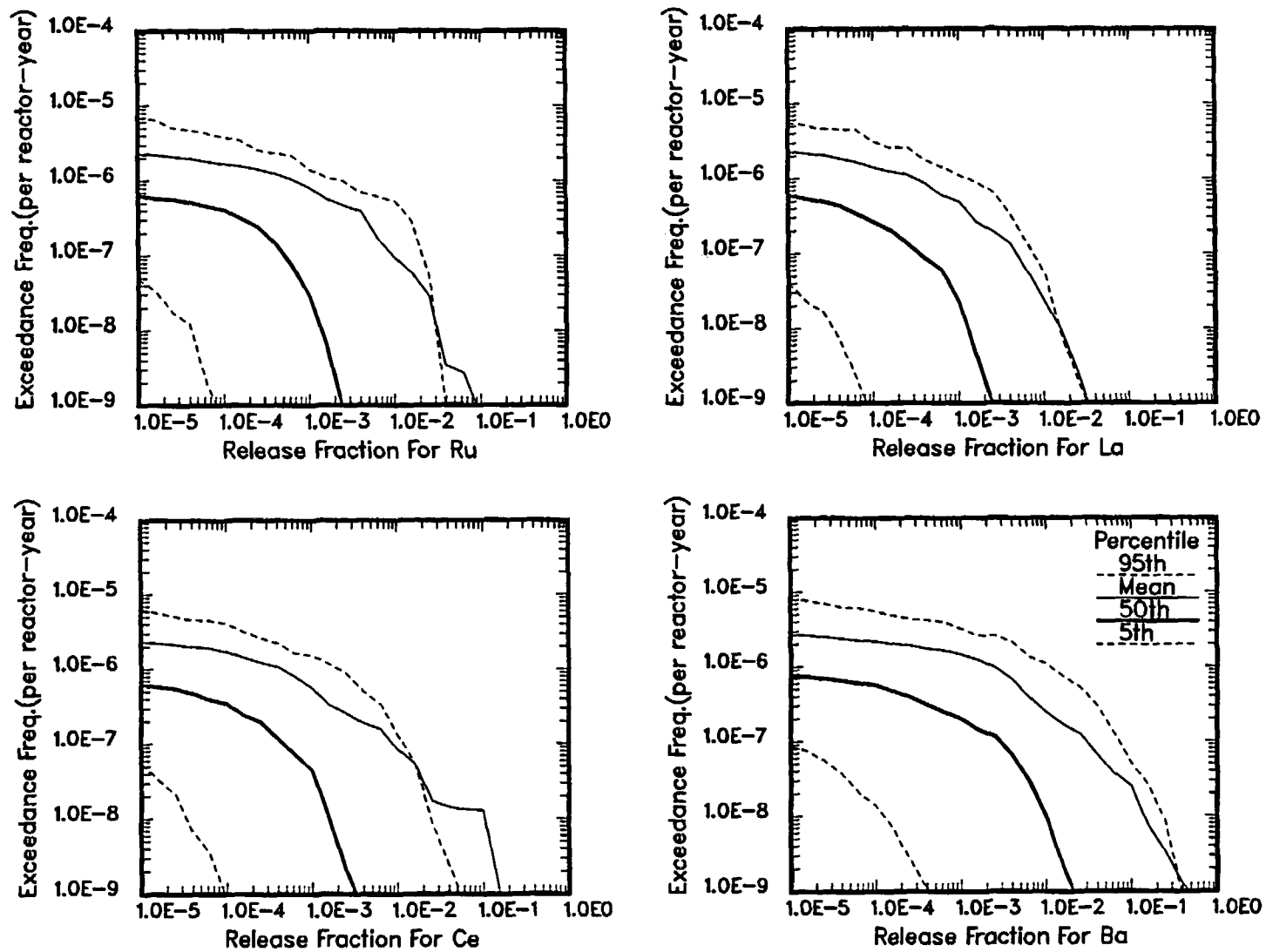


Figure 3.3-21. (continued).

### 3.4 Partitioning of the Source Terms for the Consequence Analysis

The first subsection discusses the partitioning process in some detail in the course of presenting 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 74,762 source terms for internal initiators. It is not computationally possible to perform a calculation with the MACCS consequence model<sup>1</sup> for each of these source terms. Therefore, the interface between the source term analysis and the consequence analysis is formed by grouping this large number of 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 frequency-weighted mean source term is determined for each group. Then, a single MACCS calculation is performed for each mean source term. This grouping of the source terms is performed with the PARTITION program,<sup>2</sup> 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.<sup>2</sup> This section describes the details of the partitioning process for source terms generated in the source term analysis for internal initiators.

The early health effect weight 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 early health effect weight 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 chronic health effect weight 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. Furthermore, 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 -0.5990 to 5.2741 and log EH ranges from -0.7824 to 1.9782. Figure 3.4-1 shows a plot of the pairs (CH, EH) for the 45752 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 3.9, which resulted in a loguniform division of the range of CH into ten 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.

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

	<u>Number of Source Terms</u>	<u>Percent of Total Frequency</u>
EH>0 and CH>0	45752	59.93
EH=0 and CH>0	29010	40.07
EH=0 and CH=0	0	0.00
<b>Total</b>	<b>74762</b>	<b>100.00</b>

For EH>0 and CH>0, Range LOG10(CH) = -0.5990 to 5.2741  
Range LOG10(EH) = -0.7824 to 1.9782

For EH=0 and CH>0, Range LOG10(CH) = -3.8723 to 3.7615

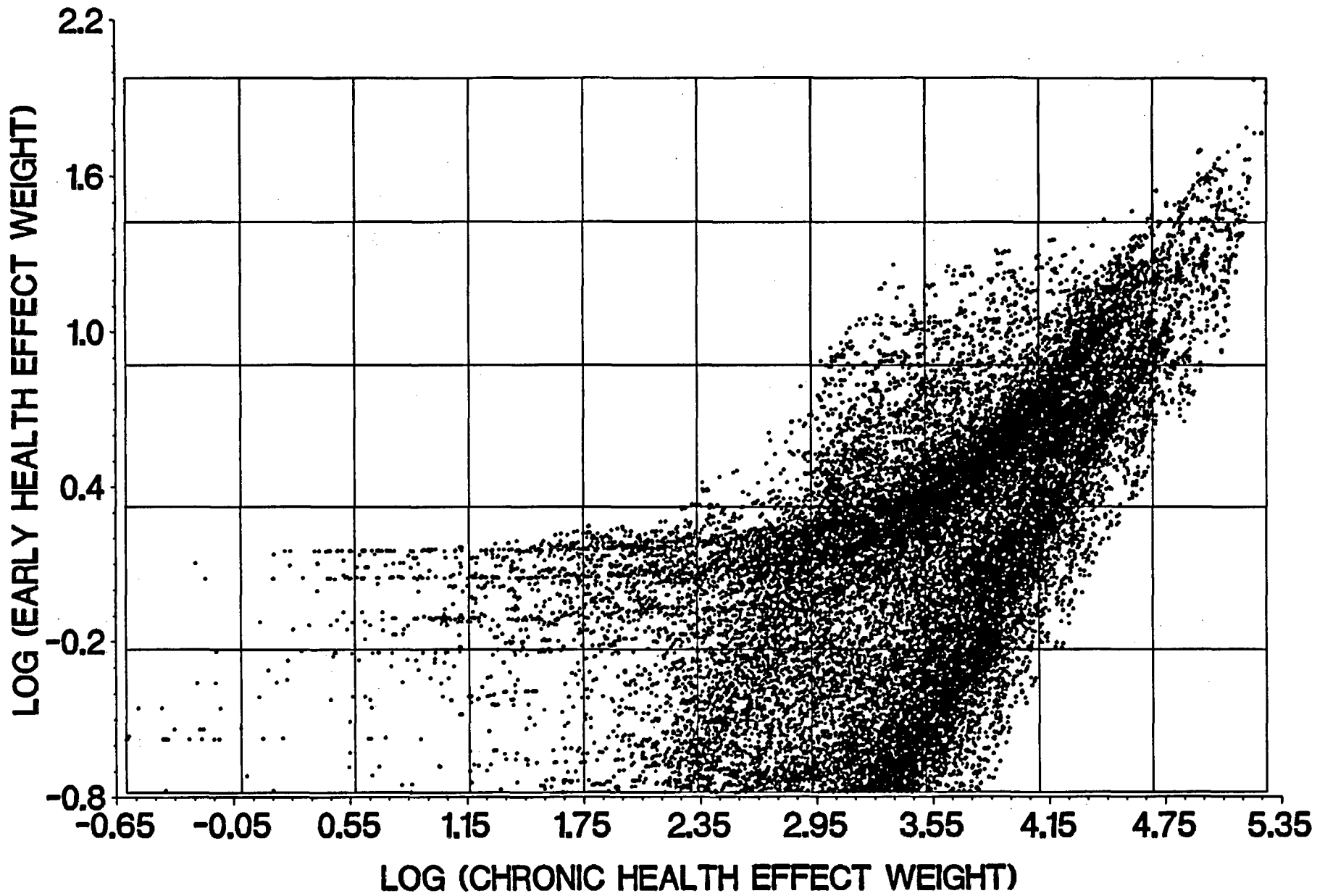


Figure 3.4-1. Distribution of Nonzero Early and Chronic Health Effect Weights for Internal Initiators



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 page 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 page 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 page 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 -3.8723 to 3.7615. The cells shown in Table 3.4-3 are based on a log uniform division of the range of CH into six intervals.

At this point, the result of partitioning is 19 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 minutes before the release begins;
- Subgroup 2: Evacuation starts between 30 minutes 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 Grand Gulf is 1.25 h. Additional discussion of evacuation delay time is given in Volume 2, Part 7 of this report.

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, GG-I and GG-I-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, GG-I-J in Table 3.4-4, that are actually used for the risk calculations.

Although not part 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 (250 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.

The highest release fractions are associated with group 13, as would be expected from Figure 3.4-1 and Table 3.4-2. The dominant accidents in this group are short-term station blackouts that have early containment failures. The frequency for this group, however, is fairly low; relatively few source terms fall in the grid represented by group GG-13, and they are not exceptionally frequent. The most likely source term groups are GG-18, GG-07, GG-16, and GG-08. Of these four groups, only GG-07 and GG-08 have the potential to cause early fatalities.

Although more source terms fall into GG-08 than GG-07, the total frequency of the source terms in group GG-08 is less than the frequency of the source terms associated with GG-07. It should be noted that when comparing the percent frequencies in Table 3.4-2 with the percentages in Table 3.4-3, they must be "weighted" by the percentages in Table 3.4-1. For example, the percent frequency associated with GG-18, relative to all of the source terms, is  $(32.8)(0.401) = 13.1\%$ , whereas the percent frequency associated with GG-07 is  $(20.5)(0.599) = 12.3\%$ . Thus, GG-18 is slightly more frequent than GG-07.

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 45752:

	1	2	3	4	5	6	7	8	9	10
1									7	217
2							219	484	2550	797
3					2	131	2672	5976	4609	114
4	5	66	172	510	1344	3475	5014	5502	745	
5	26	42	64	174	780	2626	4718	2711		

PERCENT OF FREQUENCY CONTAINED IN EACH CELL:

	1	2	3	4	5	6	7	8	9	10
1									0.04	0.09
2							0.78	0.72	4.24	0.37
3					0.02	0.11	6.98	14.55	3.62	0.00
4	0.00	0.35	0.86	0.98	6.45	7.30	9.35	8.10	0.24	
5	0.02	0.13	0.05	0.33	1.65	3.31	20.50	8.85		

AFTER PARTITIONING:

CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 45752:

	1	2	3	4	5	6	7	8	9	10
1										217
2									2712	882
3							2904	6261	5094	
4			740		1825	3524	5014	5890		
5						3260	4718	2711		

Table 3.4-2 (continued)

PERCENT OF FREQUENCY CONTAINED IN EACH CELL:

	1	2	3	4	5	6	7	8	9	10
1										0.09
2									4.50	0.41
3							7.57	15.14	3.91	
4			2.45		6.93	7.33	9.35	8.16		
5						4.83	20.50	8.85		

LABELING AFTER PARTITIONING:

	1	2	3	4	5	6	7	8	9	10
1										GG-13
2									GG-11	GG-14
3							GG-05	GG-08	GG-12	
4			GG-01		GG-02	GG-03	GG-06	GG-09		
5						GG-04	GG-07	GG-10		

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

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BEFORE PARTITIONING:

CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 29010:

	1	2	3	4	5	6
	+-----+-----+-----+-----+-----+-----+					
1	567	1722	1695	3457	12231	9338
	+-----+-----+-----+-----+-----+-----+					

PERCENT OF FREQUENCY CONTAINED IN EACH CELL:

	1	2	3	4	5	6
	+-----+-----+-----+-----+-----+-----+					
1	8.67	11.94	23.90	11.50	32.80	11.19
	+-----+-----+-----+-----+-----+-----+					

AFTER PARTITIONING:

CELL COUNTS WITHIN THE GRID FOR A TOTAL COUNT OF 29010:

	1	2	3	4	5	6
	+-----+-----+-----+-----+-----+-----+					
1		2289	1695	3457	12231	9338
	+-----+-----+-----+-----+-----+-----+					

PERCENT OF FREQUENCY CONTAINED IN EACH CELL:

	1	2	3	4	5	6
	+-----+-----+-----+-----+-----+-----+					
1		20.61	23.90	11.50	32.80	11.19
	+-----+-----+-----+-----+-----+-----+					

LABELING AFTER PARTITIONING:

	1	2	3	4	5	6
	+-----+-----+-----+-----+-----+-----+					
1		GG-15	GG-16	GG-17	GG-18	GG-19
	+-----+-----+-----+-----+-----+-----+					

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Table 3.4-4  
 Mean Source Terms Resulting from Partitioning for Internal Initiators - Grand Gulf

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
GG-01	5.8E-08		3.6E+03	1.4E+03	32	8.1E+06	9.5E+03	4.0E+03	7.4E-01	1.2E-04	6.0E-05	1.8E-05	2.9E-06	2.6E-06	5.5E-07	7.4E-07	3.4E-06
						3.4E+05	1.4E+04	1.7E+04	9.6E-03	1.7E-03	1.0E-05	9.0E-07	1.1E-07	4.7E-07	1.4E-07	1.3E-07	2.0E-07
GG-01-1	0.258		3.6E+03	4.9E+03	32	1.2E+06	1.3E+04	2.1E+03	7.6E-01	8.8E-05	6.1E-05	2.0E-05	8.4E-06	8.9E-06	2.0E-06	2.6E-06	9.3E-06
						8.8E+04	1.5E+04	1.6E+04	3.3E-02	1.5E-03	1.2E-05	4.8E-07	1.1E-08	3.5E-15	5.5E-10	7.4E-10	5.5E-09
GG-01-2	0.742		3.6E+03	1.8E+02	32	1.1E+07	8.3E+03	4.7E+03	7.3E-01	1.2E-04	6.0E-05	1.8E-05	9.3E-07	3.4E-07	2.7E-06	9.3E-08	1.3E-06
						4.3E+05	1.3E+04	1.8E+04	1.7E-03	1.8E-03	9.6E-06	1.0E-06	1.4E-07	6.3E-07	1.9E-07	1.7E-07	2.7E-07
GG-01-3	0.000																
GG-02	1.7E-07		3.7E+03	2.5E+03	32	4.3E+06	1.1E+04	4.0E+03	8.9E-01	5.7E-04	4.5E-04	2.4E-04	1.3E-04	1.3E-04	3.2E-05	5.8E-05	1.4E-04
						4.4E+05	1.5E+04	1.6E+04	6.7E-03	1.6E-02	2.4E-04	1.1E-04	5.1E-05	7.3E-05	1.4E-05	1.2E-05	5.4E-05
GG-02-1	0.499		3.6E+03	4.9E+03	32	3.8E+05	1.3E+04	3.3E+03	9.5E-01	4.2E-04	3.6E-04	2.2E-04	1.4E-04	2.4E-04	5.4E-05	6.5E-05	1.6E-04
						1.9E+05	1.6E+04	1.8E+04	7.2E-04	1.8E-02	3.2E-04	1.0E-04	1.1E-05	7.8E-10	1.0E-06	8.4E-07	4.1E-06
GG-02-2	0.501		3.8E+03	1.7E+02	32	8.3E+06	8.5E+03	4.7E+03	8.3E-01	7.2E-04	5.3E-04	2.7E-04	1.2E-04	1.9E-05	1.0E-05	5.0E-05	1.2E-04
						6.9E+05	1.3E+04	1.5E+04	1.3E-02	1.3E-02	1.5E-04	1.2E-04	9.1E-05	1.5E-04	2.8E-05	2.2E-05	1.0E-04
GG-02-3	0.000																
GG-03	1.8E-07		4.6E+03	1.1E+04	32	7.6E+06	2.1E+04	2.5E+03	8.7E-01	2.9E-02	1.4E-03	1.0E-03	3.8E-04	2.5E-04	6.4E-05	9.2E-05	3.0E-04
						1.3E+06	2.3E+04	1.4E+04	4.6E-02	2.0E-02	7.9E-04	5.4E-04	6.0E-04	8.5E-04	2.6E-04	2.7E-04	5.0E-04
GG-03-1	0.673		3.8E+03	1.7E+04	32	9.8E+06	2.5E+04	1.4E+03	8.8E-01	4.3E-02	1.6E-03	1.3E-03	5.0E-04	3.5E-04	9.0E-05	1.1E-04	3.7E-04
						1.7E+05	2.7E+04	1.6E+04	6.2E-02	2.0E-02	5.8E-04	5.6E-04	5.3E-04	5.8E-06	1.4E-04	2.0E-04	3.7E-04
GG-03-2	0.327		6.4E+03	9.1E+01	32	3.2E+06	1.1E+04	4.7E+03	8.5E-01	1.6E-03	9.4E-04	4.5E-04	1.5E-04	2.5E-05	1.1E-05	5.5E-05	1.5E-04
						3.6E+06	1.6E+04	1.1E+04	1.3E-02	2.2E-02	1.2E-03	5.2E-04	7.2E-04	2.6E-03	4.9E-04	4.2E-04	7.4E-04
GG-03-3	0.000																
GG-04	1.2E-07		8.3E+03	2.0E+04	32	1.3E+07	3.2E+04	2.8E+03	6.4E-01	2.6E-02	1.3E-03	5.4E-04	2.4E-04	2.3E-04	4.4E-05	5.1E-05	2.3E-04
						1.1E+06	3.5E+04	1.5E+04	8.0E-02	1.5E-02	4.7E-04	1.7E-04	7.7E-05	5.9E-05	2.8E-05	3.1E-05	7.2E-05
GG-04-1	0.794		5.2E+03	2.5E+04	32	1.6E+07	3.4E+04	2.3E+03	6.8E-01	3.3E-02	1.5E-03	6.3E-04	3.0E-04	2.8E-04	5.5E-05	6.2E-05	2.9E-04
						2.6E+05	3.7E+04	1.7E+04	8.3E-02	1.0E-02	1.6E-04	9.5E-05	5.7E-05	2.5E-05	1.7E-05	2.3E-05	5.1E-05
GG-04-2	0.206		2.0E+04	-3.0E+02	32	3.0E+06	2.4E+04	4.7E+03	4.7E-01	7.4E-04	3.9E-04	1.7E-04	2.5E-05	3.8E-06	1.7E-06	8.4E-06	2.9E-05
						4.3E+06	2.9E+04	9.3E+03	6.9E-02	3.2E-02	1.6E-03	4.6E-04	1.5E-04	1.9E-04	7.1E-05	6.3E-05	1.5E-04
GG-04-3	0.000																
GG-05	1.8E-07		4.1E+03	1.8E+04	32	1.2E+07	2.6E+04	2.7E+03	8.5E-01	1.6E-01	2.8E-03	1.9E-03	1.0E-03	5.1E-04	1.3E-04	3.1E-04	9.5E-04
						1.6E+06	2.9E+04	1.4E+04	1.4E-01	1.1E-01	2.7E-03	2.1E-03	2.9E-03	1.2E-03	1.1E-03	1.5E-03	2.6E-03
GG-05-1	0.593		3.8E+03	2.8E+04	32	1.9E+07	3.8E+04	1.3E+03	8.4E-01	2.7E-01	3.7E-03	2.4E-03	1.4E-03	8.0E-04	2.0E-04	4.2E-04	1.2E-03
						2.6E+05	3.9E+04	1.6E+04	1.6E-01	1.3E-01	1.9E-03	1.3E-03	9.4E-04	8.0E-06	1.9E-04	2.7E-04	7.5E-04
GG-05-2	0.407		4.6E+03	1.5E+02	32	1.2E+06	9.3E+03	4.7E+03	8.7E-01	2.2E-03	1.6E-03	1.2E-03	5.5E-04	9.1E-05	3.1E-05	1.4E-04	5.6E-04
						3.6E+06	1.4E+04	1.2E+04	1.2E-01	9.1E-02	3.9E-03	3.2E-03	5.8E-03	3.1E-03	2.5E-03	3.2E-03	5.2E-03
GG-05-3	0.000																

3.67

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
GG-06	2.2E-07		4.8E+03	1.3E+04	32	8.0E+06	2.2E+04	2.9E+03	8.1E-01	3.9E-02	6.2E-03	4.3E-03	2.3E-03	4.6E-04	1.4E-04	2.3E-04	1.6E-03
						1.2E+06	2.5E+04	1.5E+04	8.7E-02	2.7E-02	4.0E-03	1.6E-03	8.5E-04	2.0E-04	1.1E-04	1.5E-04	6.2E-04
GG-06-1		0.718	3.9E+03	1.8E+04	32	9.6E+06	2.6E+04	2.2E+03	8.3E-01	5.1E-02	6.7E-03	5.2E-03	3.1E-03	6.2E-04	1.9E-04	3.0E-04	2.1E-03
						1.9E+05	2.9E+04	1.6E+04	1.1E-01	2.7E-02	3.0E-03	1.5E-03	8.8E-04	1.5E-05	5.6E-05	8.9E-05	5.4E-04
GG-06-2		0.282	6.9E+03	1.6E+02	32	3.8E+06	1.2E+04	4.7E+03	7.7E-01	8.7E-03	5.1E-03	1.9E-03	2.2E-04	5.7E-05	9.1E-06	3.5E-05	2.5E-04
						3.8E+06	1.6E+04	1.2E+04	4.0E-02	2.7E-02	6.6E-03	1.8E-03	7.5E-04	6.7E-04	2.6E-04	2.9E-04	8.2E-04
GG-06-3		0.000															
GG-07	4.9E-07		4.2E+03	4.0E+04	32	2.9E+07	4.9E+04	9.0E+02	8.5E-01	5.7E-02	5.9E-03	5.9E-03	5.1E-03	2.1E-03	5.8E-04	6.7E-04	2.7E-03
						4.4E+05	5.0E+04	1.5E+04	1.3E-01	8.4E-03	1.4E-03	1.2E-03	7.0E-04	3.5E-04	8.3E-05	9.8E-05	4.5E-04
GG-07-1		0.978	3.8E+03	4.1E+04	32	2.9E+07	4.9E+04	8.1E+02	8.5E-01	5.8E-02	6.0E-03	6.0E-03	5.1E-03	2.1E-03	5.9E-04	6.8E-04	2.8E-03
						2.8E+05	5.0E+04	1.5E+04	1.3E-01	8.1E-03	1.2E-03	1.1E-03	6.9E-04	3.5E-04	8.0E-05	9.6E-05	4.4E-04
GG-07-2		0.022	2.3E+04	-2.0E+02	32	2.8E+06	2.8E+04	4.7E+03	6.2E-01	5.0E-03	3.7E-03	1.8E-03	6.0E-04	1.3E-04	5.1E-05	3.4E-04	6.2E-04
						7.5E+06	3.3E+04	1.1E+04	7.8E-02	2.2E-02	8.7E-03	2.6E-03	1.2E-03	3.8E-04	2.4E-04	2.0E-04	9.0E-04
GG-07-3		0.000															
GG-08	3.6E-07		5.4E+03	5.2E+03	32	2.8E+06	1.5E+04	3.5E+03	8.4E-01	4.1E-02	1.5E-02	1.1E-02	5.7E-03	1.7E-03	6.3E-04	2.4E-03	5.8E-03
						2.1E+06	1.9E+04	1.3E+04	1.2E-01	1.2E-01	2.2E-02	1.5E-02	8.6E-03	1.4E-03	9.0E-04	1.0E-03	5.1E-03
GG-08-1		0.613	4.1E+03	8.3E+03	32	3.0E+06	1.7E+04	2.7E+03	8.4E-01	5.4E-02	1.4E-02	9.8E-03	4.4E-03	1.9E-03	5.8E-04	1.3E-03	4.5E-03
						1.9E+05	2.0E+04	1.7E+04	1.4E-01	1.3E-01	1.9E-02	1.4E-02	7.6E-03	2.2E-05	3.3E-04	5.2E-04	3.7E-03
GG-08-2		0.387	7.4E+03	2.0E+02	32	2.5E+06	1.2E+04	4.7E+03	8.4E-01	1.9E-02	1.6E-02	1.2E-02	7.8E-03	1.3E-03	7.0E-04	4.0E-03	7.8E-03
						5.1E+06	1.7E+04	7.3E+03	9.5E-02	1.1E-01	2.8E-02	1.6E-02	1.0E-02	3.6E-03	1.8E-03	1.8E-03	7.5E-03
GG-08-3		0.000															
GG-09	1.9E-07		5.3E+03	2.6E+04	32	1.8E+07	3.5E+04	2.9E+03	6.5E-01	4.7E-02	2.8E-02	1.7E-02	2.7E-03	8.1E-04	3.0E-04	4.6E-04	2.4E-03
						1.4E+06	3.8E+04	1.5E+04	1.9E-01	2.9E-02	1.5E-02	7.4E-03	1.2E-03	5.6E-04	1.6E-04	2.0E-04	1.1E-03
GG-09-1		0.814	4.2E+03	3.1E+04	32	2.1E+07	4.0E+04	2.5E+03	6.7E-01	5.7E-02	3.4E-02	2.0E-02	3.3E-03	9.7E-04	3.6E-04	5.1E-04	2.8E-03
						2.6E+05	4.3E+04	1.7E+04	2.1E-01	2.4E-02	1.2E-02	7.4E-03	7.9E-04	2.5E-04	6.1E-05	1.1E-04	7.4E-04
GG-09-2		0.186	9.9E+03	1.8E+02	32	1.9E+06	1.5E+04	4.7E+03	5.7E-01	4.2E-03	2.8E-03	1.1E-03	3.8E-04	8.5E-05	3.6E-05	2.6E-04	4.0E-04
						6.6E+06	1.9E+04	5.8E+03	7.4E-02	5.2E-02	3.1E-02	7.2E-03	2.8E-03	1.9E-03	5.9E-04	6.1E-04	2.8E-03
GG-09-3		0.000															
GG-10	2.1E-07		4.4E+03	4.1E+04	32	2.7E+07	5.0E+04	5.9E+03	5.5E-01	3.2E-02	1.8E-02	1.1E-02	7.1E-04	5.7E-04	6.5E-05	1.2E-04	9.1E-04
						3.2E+05	5.6E+04	2.0E+04	4.2E-01	2.6E-02	1.4E-02	8.6E-03	4.6E-04	4.3E-04	4.6E-05	7.5E-05	6.1E-04
GG-10-1		0.981	3.8E+03	4.2E+04	32	2.7E+07	5.0E+04	5.9E+03	5.5E-01	3.3E-02	1.9E-02	1.1E-02	7.2E-04	5.7E-04	6.6E-05	1.2E-04	9.2E-04
						2.0E+05	5.6E+04	2.0E+04	4.2E-01	2.5E-02	1.3E-02	8.6E-03	4.0E-04	4.1E-04	4.0E-05	6.9E-05	5.7E-04
GG-10-2		0.019	3.3E+04	2.4E+02	32	3.3E+06	3.7E+04	4.7E+03	5.8E-01	6.0E-03	4.7E-03	2.6E-03	4.1E-04	1.1E-04	2.4E-05	1.2E-04	4.5E-04
						6.8E+06	4.3E+04	1.0E+04	1.9E-01	4.0E-02	2.8E-02	6.8E-03	3.6E-03	1.4E-03	3.7E-04	4.1E-04	2.6E-03
GG-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
									GG-11	1.1E-07		4.5E+03	2.0E+03	32	1.5E+06	1.1E+04	3.2E+03
						4.2E+06	1.4E+04	8.3E+03	2.7E-01	3.1E-01	1.1E-01	6.3E-02	1.7E-02	2.6E-03	1.7E-03	2.2E-03	1.3E-02
GG-11-1		0.321	4.3E+03	5.8E+03	32	1.0E+06	1.5E+04	2.1E+02	7.3E-01	6.2E-02	5.3E-02	3.0E-02	1.2E-02	4.6E-03	1.6E-03	5.0E-03	1.3E-02
						2.0E+04	1.5E+04	1.4E+04	2.6E-01	3.2E-01	8.9E-02	6.1E-02	2.4E-02	8.7E-05	1.5E-03	2.7E-03	1.8E-02
GG-11-2		0.679	4.6E+03	1.7E+02	32	1.7E+06	9.3E+03	4.7E+03	7.2E-01	5.4E-02	4.7E-02	3.2E-02	2.2E-02	4.6E-03	2.2E-03	1.6E-02	2.2E-02
						6.2E+06	1.4E+04	5.4E+03	2.8E-01	3.0E-01	1.2E-01	6.4E-02	1.4E-02	3.8E-03	1.7E-03	1.9E-03	1.1E-02
GG-11-3		0.000															
GG-12	9.3E-08		1.3E+04	7.1E+03	32	5.5E+06	2.5E+04	3.3E+03	7.8E-01	7.8E-02	5.7E-02	3.4E-02	9.8E-03	1.7E-03	8.1E-04	1.6E-03	8.7E-03
						3.8E+06	2.8E+04	1.1E+04	1.4E-01	1.1E-01	5.2E-02	2.8E-02	1.6E-02	3.7E-03	1.4E-03	2.0E-03	1.3E-02
GG-12-1		0.387	1.0E+04	1.8E+04	32	8.4E+06	3.3E+04	1.1E+03	8.1E-01	9.0E-02	5.9E-02	4.8E-02	2.2E-02	3.1E-03	1.9E-03	3.4E-03	1.8E-02
						2.6E+05	3.4E+04	1.5E+04	1.7E-01	1.1E-01	4.3E-02	2.7E-02	1.6E-02	9.0E-05	8.7E-04	1.6E-03	1.1E-02
GG-12-2		0.613	1.5E+04	1.7E+02	32	3.6E+06	2.0E+04	4.7E+03	7.6E-01	7.0E-02	5.6E-02	2.6E-02	2.1E-03	7.7E-04	1.1E-04	4.3E-04	2.5E-03
						6.1E+06	2.4E+04	8.8E+03	1.2E-01	1.1E-01	5.7E-02	2.9E-02	1.5E-02	6.0E-03	1.7E-03	2.2E-03	1.4E-02
GG-12-3		0.000															
GG-13	2.2E-09		5.3E+03	5.3E+03	32	2.3E+06	1.5E+04	1.8E+03	8.2E-01	2.5E-01	2.3E-01	2.2E-01	1.9E-01	3.6E-02	2.0E-02	1.2E-01	1.9E-01
						2.3E+06	2.0E+04	1.1E+04	1.8E-01	2.4E-01	2.6E-01	2.6E-01	2.4E-01	7.5E-03	1.9E-02	2.8E-02	1.9E-01
GG-13-1		0.635	4.3E+03	8.3E+03	32	2.7E+06	1.7E+04	1.9E+02	7.4E-01	1.5E-01	1.6E-01	1.6E-01	1.7E-01	3.1E-02	2.0E-02	9.3E-02	1.7E-01
						6.4E+04	2.2E+04	1.4E+04	2.6E-01	3.1E-01	3.3E-01	3.2E-01	2.9E-01	1.1E-02	2.2E-02	3.3E-02	2.4E-01
GG-13-2		0.365	7.1E+03	2.4E+02	32	1.6E+06	1.2E+04	4.7E+03	9.5E-01	4.1E-01	3.7E-01	3.1E-01	2.2E-01	4.6E-02	2.2E-02	1.5E-01	2.2E-01
						6.3E+06	1.8E+04	4.8E+03	4.8E-02	1.3E-01	1.2E-01	1.4E-01	1.4E-01	2.2E-03	1.3E-02	2.0E-02	1.2E-01
GG-13-3		0.000															
GG-14	9.8E-09		2.0E+04	3.4E+03	32	2.9E+06	2.8E+04	3.1E+03	6.4E-01	1.2E-01	1.1E-01	9.5E-02	4.6E-02	6.7E-03	4.2E-03	1.4E-02	4.1E-02
						6.8E+06	3.6E+04	8.4E+03	3.6E-01	3.4E-01	3.3E-01	1.8E-01	6.6E-02	2.5E-03	3.4E-03	4.3E-03	4.2E-02
GG-14-1		0.387	2.0E+04	8.8E+03	32	3.9E+06	3.4E+04	4.8E+02	7.1E-01	1.1E-01	1.1E-01	1.2E-01	7.3E-02	5.9E-03	6.4E-03	4.6E-03	6.1E-02
						1.3E+05	4.1E+04	1.4E+04	2.8E-01	3.2E-01	3.1E-01	1.6E-01	6.4E-02	5.7E-05	2.7E-03	4.0E-03	4.5E-02
GG-14-2		0.613	2.0E+04	4.3E+01	32	2.4E+06	2.4E+04	4.7E+03	6.0E-01	1.4E-01	1.2E-01	8.1E-02	2.8E-02	7.1E-03	2.8E-03	2.0E-02	2.9E-02
						1.1E+07	3.3E+04	4.7E+03	4.0E-01	3.6E-01	3.4E-01	1.9E-01	6.7E-02	4.0E-03	3.8E-03	4.6E-03	4.0E-02
GG-14-3		0.000															
GG-15	3.3E-07		3.7E+03	4.2E+04	32	6.0E+06	5.0E+04	7.2E+03	2.3E-03	8.5E-07	1.0E-08	4.9E-09	9.6E-10	3.4E-10	9.2E-11	1.8E-10	1.0E-09
						8.8E+04	5.8E+04	2.2E+04	2.1E-03	8.5E-07	1.0E-08	4.9E-09	9.5E-10	3.4E-10	9.2E-11	1.8E-10	1.0E-09
GG-15-1		0.999	3.7E+03	4.2E+04	32	6.0E+06	5.0E+04	7.2E+03	2.2E-03	8.5E-07	1.0E-08	4.9E-09	9.6E-10	3.4E-10	9.2E-11	1.8E-10	1.0E-09
						8.8E+04	5.8E+04	2.2E+04	2.1E-03	8.5E-07	1.0E-08	4.9E-09	9.6E-10	3.4E-10	9.2E-11	1.8E-10	1.0E-09
GG-15-2		0.001	3.6E+03	1.8E+02	32	1.2E+07	8.3E+03	4.7E+03	7.2E-02	1.3E-08	7.1E-09	5.6E-09	1.1E-10	0.0E+00	0.0E+00	0.0E+00	1.7E-10
						0.0E+00	1.3E+04	2.2E+04	0.0E+00	2.4E-06	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
GG-15-3		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
GG-16	3.8E-07		3.6E+03	4.2E+04	32	2.1E+07 1.5E+05	5.0E+04 5.7E+04	6.7E+03 2.1E+04	2.5E-02 6.2E-03	1.2E-04 1.1E-04	8.5E-08 3.9E-08	4.6E-08 2.2E-08	8.8E-09 8.2E-09	3.8E-09 3.3E-09	9.6E-10 8.1E-10	2.0E-09 1.8E-09	9.5E-09 8.3E-09
GG-16-1		0.999	3.6E+03	4.2E+04	32	2.1E+07 1.5E+05	5.0E+04 5.7E+04	6.7E+03 2.1E+04	2.5E-02 6.2E-03	1.2E-04 1.1E-04	8.3E-08 3.9E-08	4.4E-08 2.2E-08	8.8E-09 8.2E-09	3.8E-09 3.3E-09	9.6E-10 8.1E-10	2.0E-09 1.8E-09	9.5E-09 8.3E-09
GG-16-2		0.001	4.6E+03	1.6E+02	32	1.2E+07 1.0E+04	9.2E+03 1.4E+04	4.7E+03 2.1E+04	1.4E-01 8.8E-04	3.8E-06 7.8E-06	2.1E-06 4.0E-08	1.8E-06 8.1E-09	2.9E-08 6.0E-10	1.4E-09 1.1E-09	5.5E-10 6.4E-10	5.5E-10 6.4E-10	5.9E-08 1.1E-09
GG-16-3		0.000															
GG-17	1.8E-07		4.2E+03	4.1E+04	32	9.1E+07 1.6E+05	5.0E+04 5.4E+04	3.7E+03 1.8E+04	4.1E-01 1.7E-01	3.9E-04 1.8E-04	1.9E-05 9.4E-06	1.0E-05 4.8E-06	2.1E-06 4.2E-07	4.3E-06 7.0E-07	4.0E-07 1.0E-07	4.4E-07 1.1E-07	2.5E-06 5.4E-07
GG-17-1		0.980	3.8E+03	4.2E+04	32	9.3E+07 1.5E+05	5.0E+04 5.4E+04	3.7E+03 1.8E+04	4.1E-01 1.7E-01	4.0E-04 1.8E-04	1.9E-05 9.2E-06	1.0E-05 4.6E-06	2.1E-06 3.8E-07	4.4E-06 6.3E-07	4.1E-07 7.5E-08	4.4E-07 8.4E-08	2.5E-06 4.7E-07
GG-17-2		0.020	2.1E+04	-2.5E+02	32	5.6E+06 7.2E+05	2.5E+04 3.0E+04	4.7E+03 1.8E+04	4.5E-01 1.6E-02	4.9E-05 3.4E-04	2.2E-05 1.9E-05	7.7E-06 1.2E-05	1.9E-07 2.8E-06	3.2E-08 4.0E-06	5.8E-09 1.4E-06	1.9E-08 1.3E-06	3.9E-07 4.0E-06
GG-17-3		0.000															
GG-18	5.2E-07		5.4E+03	3.9E+04	32	6.4E+07 3.2E+05	4.9E+04 5.3E+04	4.0E+03 1.8E+04	5.7E-01 2.4E-01	6.3E-03 3.9E-03	3.3E-04 1.6E-04	2.1E-04 8.7E-05	8.0E-05 2.7E-05	3.4E-05 1.4E-05	1.8E-05 4.9E-06	1.6E-05 5.7E-06	7.7E-05 2.6E-05
GG-18-1		0.953	4.2E+03	4.1E+04	32	6.7E+07 2.1E+05	5.0E+04 5.4E+04	3.9E+03 1.8E+04	5.7E-01 2.6E-01	6.6E-03 3.8E-03	3.4E-04 1.4E-04	2.2E-04 8.6E-05	8.4E-05 2.2E-05	3.6E-05 8.9E-06	1.9E-05 3.9E-06	1.7E-05 4.5E-06	8.0E-05 2.1E-05
GG-18-2		0.047	3.0E+04	1.6E+02	32	4.1E+06 2.6E+06	3.5E+04 4.0E+04	4.7E+03 1.5E+04	5.7E-01 3.1E-02	2.0E-04 6.7E-03	1.0E-04 4.3E-04	4.9E-05 1.2E-04	5.6E-06 1.2E-04	9.4E-07 1.2E-04	3.6E-07 2.6E-05	1.7E-06 3.1E-05	6.5E-06 1.2E-04
GG-18-3		0.000															
GG-19	1.8E-07		8.9E+03	3.6E+04	32	6.2E+07 5.9E+05	4.9E+04 5.2E+04	3.2E+03 1.7E+04	6.3E-01 2.0E-01	9.5E-03 4.5E-03	4.2E-03 2.0E-03	2.1E-03 1.1E-03	4.8E-04 3.1E-04	1.9E-04 1.4E-04	6.4E-05 3.8E-05	1.1E-04 6.2E-05	4.8E-04 3.1E-04
GG-19-1		0.904	5.9E+03	4.0E+04	32	6.8E+07 3.0E+05	5.0E+04 5.3E+04	3.0E+03 1.7E+04	6.3E-01 2.2E-01	1.0E-02 4.0E-03	4.5E-03 1.7E-03	2.3E-03 1.0E-03	5.2E-04 2.6E-04	2.1E-04 7.8E-05	7.0E-05 2.3E-05	1.2E-04 4.9E-05	5.2E-04 2.5E-04
GG-19-2		0.096	3.7E+04	4.7E+02	32	4.0E+06 3.4E+06	4.2E+04 4.7E+04	4.7E+03 1.7E+04	6.1E-01 4.7E-02	1.3E-03 9.4E-03	7.3E-04 5.2E-03	3.1E-04 1.9E-03	6.0E-05 7.7E-04	1.1E-05 6.9E-04	2.4E-06 1.8E-04	9.5E-06 1.8E-04	6.6E-05 8.5E-04
GG-19-3		0.000															

### 3.5 Insights from the Source Term Analysis

The range in the release fractions for similar accidents is large; typically several orders of magnitude. Although the containment is predicted to fail in most of the accidents analyzed, there are several features of the Grand Gulf plant that tend to mitigate the release. First, the in-vessel releases are generally directed to the suppression pool where they are subjected to the pool DF. Although not as effective as the suppression pool, the containment sprays and the reactor cavity pool also offer a mechanism for reducing the release of radionuclides from the containment. The largest releases tend to occur when the suppression pool is bypassed and the containment sprays are not operating. As mentioned in Section 2.5 coincident failures of the containment and the drywell is a distinct possibility at Grand Gulf. Furthermore, because the dominant accidents are SBOs, it is not uncommon for the containment sprays to be unavailable at the time of vessel breach. In these accidents, releases that occur at vessel breach (e.g., release associated with DCH or an ex-vessel steam explosion) and after vessel breach (e.g., CCI releases) bypass the suppression pool and are not subjected to either a pool DF or a spray DF.

### 3.6 References

1. 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.
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 Interface in the NUREG-1150 Probabilistic Risk Assessments," NUREG/CR-5253, SAND88-2940, Sandia National Laboratories, May 1990.
3. C. N. Amos et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Grand Gulf, Unit 1," NUREG/CR-4551, SAND86-1309, Volume 4, Sandia National Laboratories, (Draft for Comment) 1987.
4. A. S. Benjamin et al., "Evaluation of Severe Accident Risks and the Potential for Risk Reduction: Surry Power Station, Unit 1," NUREG/CR-4551, SAND86-1309, Volume 1, Sandia National Laboratories, (Draft for Comment) 1987.

#### 4. CONSEQUENCE ANALYSIS

Offsite consequences were calculated with MACCS<sup>1,2,3</sup> for each of the source term groups defined in the partitioning process. This code has been in use for some time and will not be described in any 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 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. Both 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. 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 crops 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 out of the emergency planning 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 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.

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% of the population evacuates, and 0.5% 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.

Table 4.1-1  
Definition of Consequence Analysis Results

Variable	Definition
Early fatalities	Number of fatalities occurring within 1 year 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 one year for an individual within one mile of the exclusion boundary (i.e., $\sum (ef/pop)p$ , 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).

Table 4.1-1 (continued)

<u>Variable</u>	<u>Definition</u>
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., $\Sigma (cf/pop)p$ , 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 $t = \infty$ ).

#### 4.2 MACCS Input for Grand Gulf

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 do 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 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 have strong site dependencies and presents the values of these parameters used in the MACCS calculations for the Grand Gulf site. The evacuation delay period begins when general emergency conditions occur and ends when the general public starts to evacuate. Non-farm wealth includes personal, 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 Grand Gulf MACCS calculations used one year of meteorological data from the Grand Gulf 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. Table 4.2-2 lists the shielding parameters used in this analysis.

<u>Distance From Plant</u>		<u>Population</u>
<u>(km)</u>	<u>(miles)</u>	
1.6	1.0	34
4.8	3.0	879
16.1	10.0	10,255
48.3	30.0	97,395
160.9	100.0	1,614,883
563.3	350.0	22,259,422
1609.3	1000.0	142,024,448

There is considerable variation in the sector populations (out to 1000 miles) as well. The NNE sector has a population of about 28 million and the NE and ENE sectors each have populations of about 25 million, while the SSW sector has a population of about one-half a million.

Table 4.2-1  
Site Specific Input Data for Grand Gulf MACCS Calculations

<u>Parameter</u>	
Reactor Power Level (MWt)	3833
Containment Height (m)	32
Containment Width (m)	32
Exclusion Zone Distance (km)	0.696
Evacuation Delay (h)	1.25
Evacuation Speed (m/s)	3.7
<b>Farmland Fractions by Crop Categories</b>	
Pasture	0.7
Stored Forage	0.05
Grains	0.18
Green Leafy Vegetables	0.0005
Legumes and Seeds	0.13
Roots and Tubers	0.0008
Other Food Crops	0.004
Non-Farm Wealth (\$/person)	53,000
<b>Farm Wealth</b>	
Value (\$/hectare)	1824
Fraction in Improvements	0.30

Table 4.2-2  
Shielding Factors used for Grand Gulf MACCS Calculations

<u>Radiation Pathway</u>	<u>Population Response</u>		
	<u>Evacuate</u>	<u>Normal Activity</u>	<u>Take Shelter</u>
<b>Internal Initiators</b>			
Cloudshine	1.0	0.75	0.70
Groundshine	0.5	0.33	0.25
Inhalation	1.0	0.41	0.33
Skin	1.0	0.41	0.33

### 4.3 Results of MACCS Consequence Calculations

The results given 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 a 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 Grand Gulf, nSTG = 58 and nC = 6. The resultant 58 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<sup>4</sup> and RISQUE in the calculation of the mean risk results for internal initiators at Grand Gulf. 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 among individuals who evacuate, continue normal activities, and actively take shelter; information on the division of results between early and chronic exposure is also given. In addition to the six consequence measures which are reported in the text of this report, Table C.1-1 contains results for early injuries (prodromal vomiting), economic cost, and individual early fatality risk at 1 mile. (Note that individual early fatality risk at one mile is distinct from individual early fatality risk within one mile. The risk at one mile [listed in Appendix C only] 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 within 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 that exist 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 (GG-I-J) in Table 3.4-4 which have a non-zero frequency. The CCDFs were generated using the estimate 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 and in NUREG/CR-5382.<sup>4</sup> The CCDFs for risk are presented in the next chapter; they relate consequence values with the frequency at which these values are exceeded.

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

Source Term	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 Fatality 0-10 mi.
GG-01-1	2.94E-06	6.45E+00	1.24E+02	4.15E+02	3.62E-08	1.64E-05
GG-01-2	0.00E+00	7.40E+00	1.40E+02	4.74E+02	0.00E+00	1.29E-05
GG-01-3	-----	-----	-----	-----	-----	-----
GG-02-1	8.50E-05	3.19E+01	4.18E+02	2.09E+03	1.02E-06	5.85E-05
GG-02-2	1.36E-06	4.28E+01	4.18E+02	2.72E+03	1.72E-08	5.16E-05
GG-02-3	-----	-----	-----	-----	-----	-----
GG-03-1	1.01E-05	9.38E+01	6.99E+02	6.23E+03	9.50E-08	6.70E-05
GG-03-2	5.15E-06	9.85E+01	8.00E+02	6.21E+03	6.35E-08	7.63E-05
GG-03-3	-----	-----	-----	-----	-----	-----
GG-04-1	1.79E-07	8.05E+01	5.69E+02	5.22E+03	1.72E-09	5.87E-05
GG-04-2	5.90E-07	8.58E+01	8.07E+02	5.30E+03	7.45E-09	8.21E-05
GG-04-3	-----	-----	-----	-----	-----	-----
GG-05-1	2.19E-03	2.03E+02	1.82E+03	1.44E+04	2.02E-05	9.55E-05
GG-05-2	4.18E-04	1.83E+02	1.61E+03	1.26E+04	5.00E-06	1.06E-04
GG-05-3	-----	-----	-----	-----	-----	-----
GG-06-1	3.66E-05	2.06E+02	1.46E+03	1.29E+04	3.68E-07	1.36E-04
GG-06-2	6.55E-06	2.98E+02	1.63E+03	1.73E+04	8.30E-08	1.54E-04
GG-06-3	-----	-----	-----	-----	-----	-----
GG-07-1	1.71E-05	1.83E+02	1.28E+03	1.20E+04	1.14E-07	9.39E-05
GG-07-2	1.74E-07	2.98E+02	1.60E+03	1.75E+04	2.20E-09	1.47E-04
GG-07-3	-----	-----	-----	-----	-----	-----
GG-08-1	3.52E-03	3.97E+02	2.63E+03	2.47E+04	3.81E-05	1.67E-04
GG-08-2	9.25E-04	5.73E+02	3.18E+03	3.58E+04	1.08E-05	1.40E-04
GG-08-3	-----	-----	-----	-----	-----	-----
GG-09-1	6.45E-05	6.11E+02	2.56E+03	3.59E+04	5.05E-07	1.73E-04
GG-09-2	5.60E-05	4.67E+02	2.19E+03	2.75E+04	7.00E-07	1.28E-04
GG-09-3	-----	-----	-----	-----	-----	-----

Table 4.3-1 (continued)

Source Term	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 Fatality 0-10 mi.
GG-10-1	4.91E-06	6.27E+02	2.58E+03	3.59E+04	6.20E-08	1.97E-04
GG-10-2	7.05E-06	5.02E+02	2.30E+03	2.93E+04	8.90E-08	1.50E-04
GG-10-3	-----	-----	-----	-----	-----	-----
GG-11-1	4.02E-02	9.41E+02	4.73E+03	5.74E+04	2.05E-04	1.41E-04
GG-11-2	1.89E-02	1.28E+03	5.85E+03	7.84E+04	1.06E-04	1.38E-04
GG-11-3	-----	-----	-----	-----	-----	-----
GG-12-1	4.99E-03	8.43E+02	4.03E+03	5.17E+04	5.10E-05	1.54E-04
GG-12-2	1.29E-03	1.07E+03	4.18E+03	6.36E+04	1.42E-05	1.36E-04
GG-12-3	-----	-----	-----	-----	-----	-----
GG-13-1	1.63E+00	2.38E+03	1.88E+04	1.42E+05	4.02E-04	2.13E-04
GG-13-2	1.48E+00	2.73E+03	2.01E+04	1.60E+05	3.07E-04	2.31E-04
GG-13-3	-----	-----	-----	-----	-----	-----
GG-14-1	5.25E-02	1.67E+03	8.92E+03	1.01E+05	2.11E-04	1.72E-04
GG-14-2	2.95E-02	2.18E+03	9.35E+03	1.32E+05	1.20E-04	1.49E-04
GG-14-3	-----	-----	-----	-----	-----	-----
GG-15-1	0.00E+00	1.02E-02	3.47E-01	7.87E-01	0.00E+00	6.38E-09
GG-15-2	0.00E+00	1.07E-01	1.17E+00	6.45E+00	0.00E+00	1.68E-08
GG-15-3	-----	-----	-----	-----	-----	-----
GG-16-1	0.00E+00	3.13E-01	1.29E+01	3.47E+01	0.00E+00	1.48E-07
GG-16-2	0.00E+00	4.54E-01	9.40E+00	2.65E+01	0.00E+00	3.69E-07
GG-16-3	-----	-----	-----	-----	-----	-----
GG-17-1	0.00E+00	3.94E+00	7.29E+01	2.61E+02	0.00E+00	4.51E-06
GG-17-2	0.00E+00	4.05E+00	8.70E+01	2.47E+02	0.00E+00	9.81E-06
GG-17-3	-----	-----	-----	-----	-----	-----
GG-18-1	0.00E+00	3.65E+01	3.45E+02	2.36E+03	0.00E+00	4.32E-05
GG-18-2	0.00E+00	3.12E+01	3.71E+02	1.96E+03	0.00E+00	4.99E-05
GG-18-3	-----	-----	-----	-----	-----	-----
GG-19-1	0.00E+00	2.07E+02	1.12E+03	1.20E+04	0.00E+00	1.04E-04
GG-19-2	0.00E+00	1.77E+02	1.11E+03	1.02E+04	0.00E+00	1.30E-04
GG-19-3	-----	-----	-----	-----	-----	-----
GG-20	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

# GRAND GULF - INTERNAL EVENTS

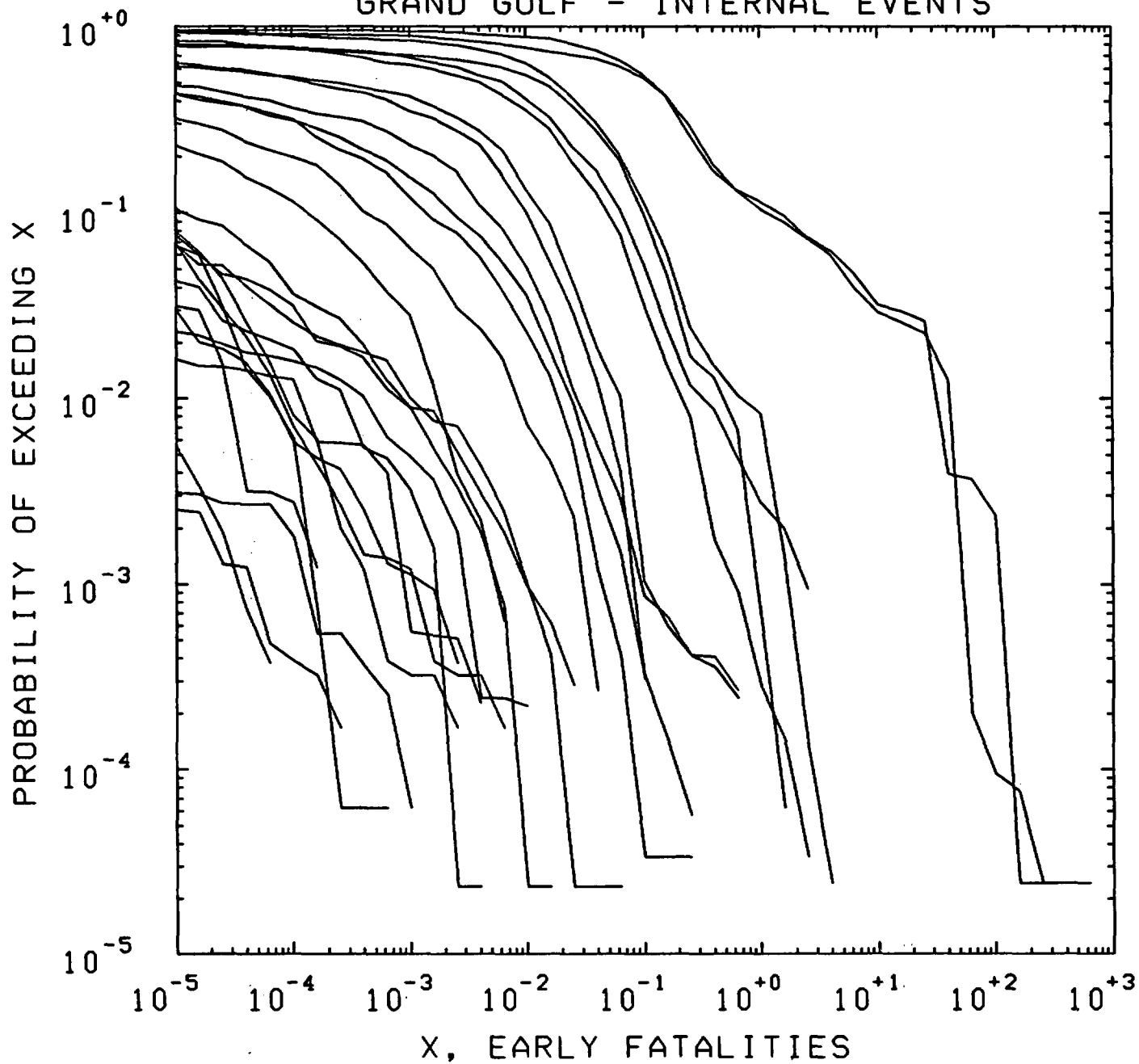


Figure 4.2-1. Consequences Conditional on Source Terms.  
Grand Gulf Internal Initiators.

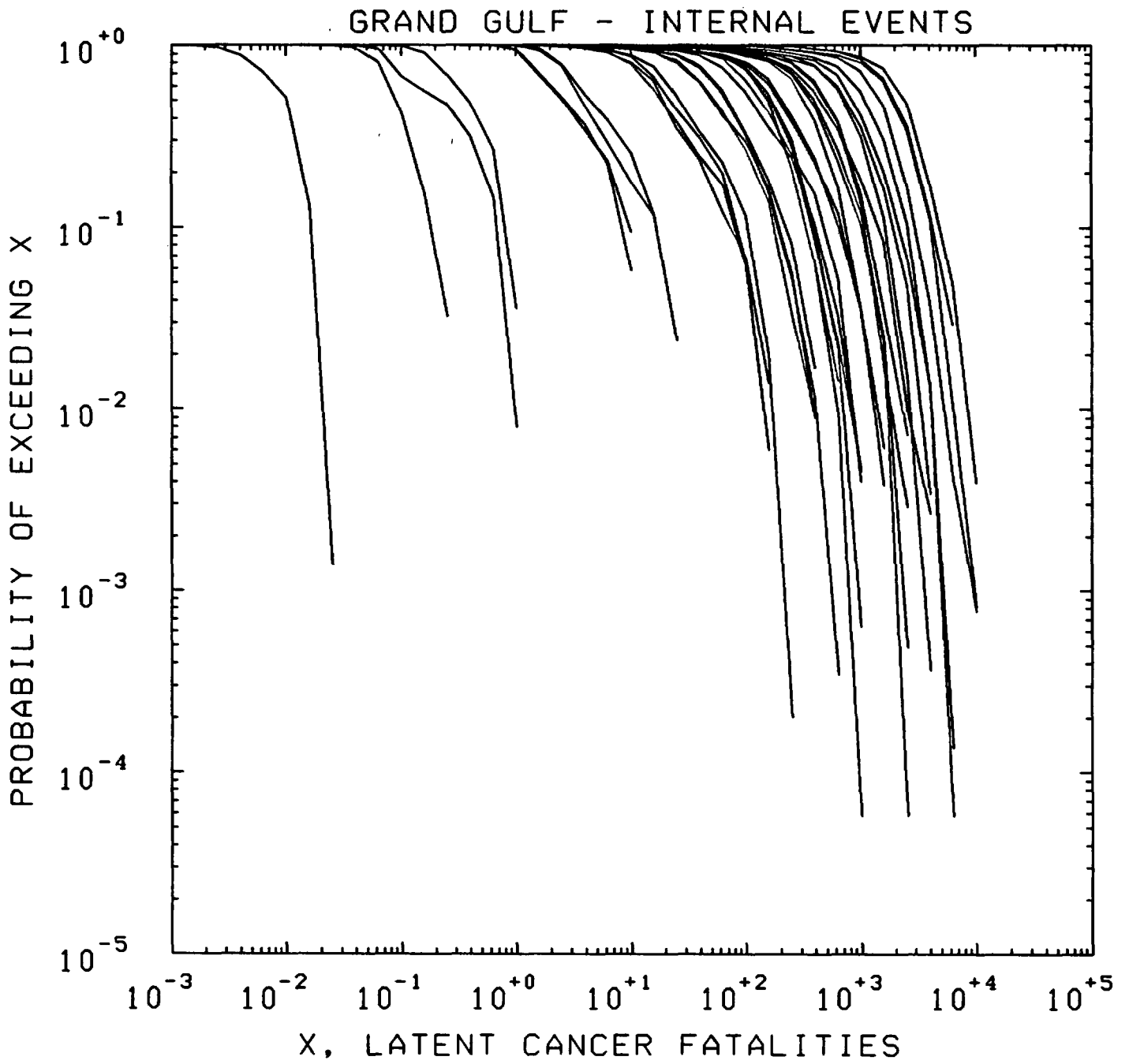


Figure 4.2-1. (continued)

GRAND GULF - INTERNAL EVENTS

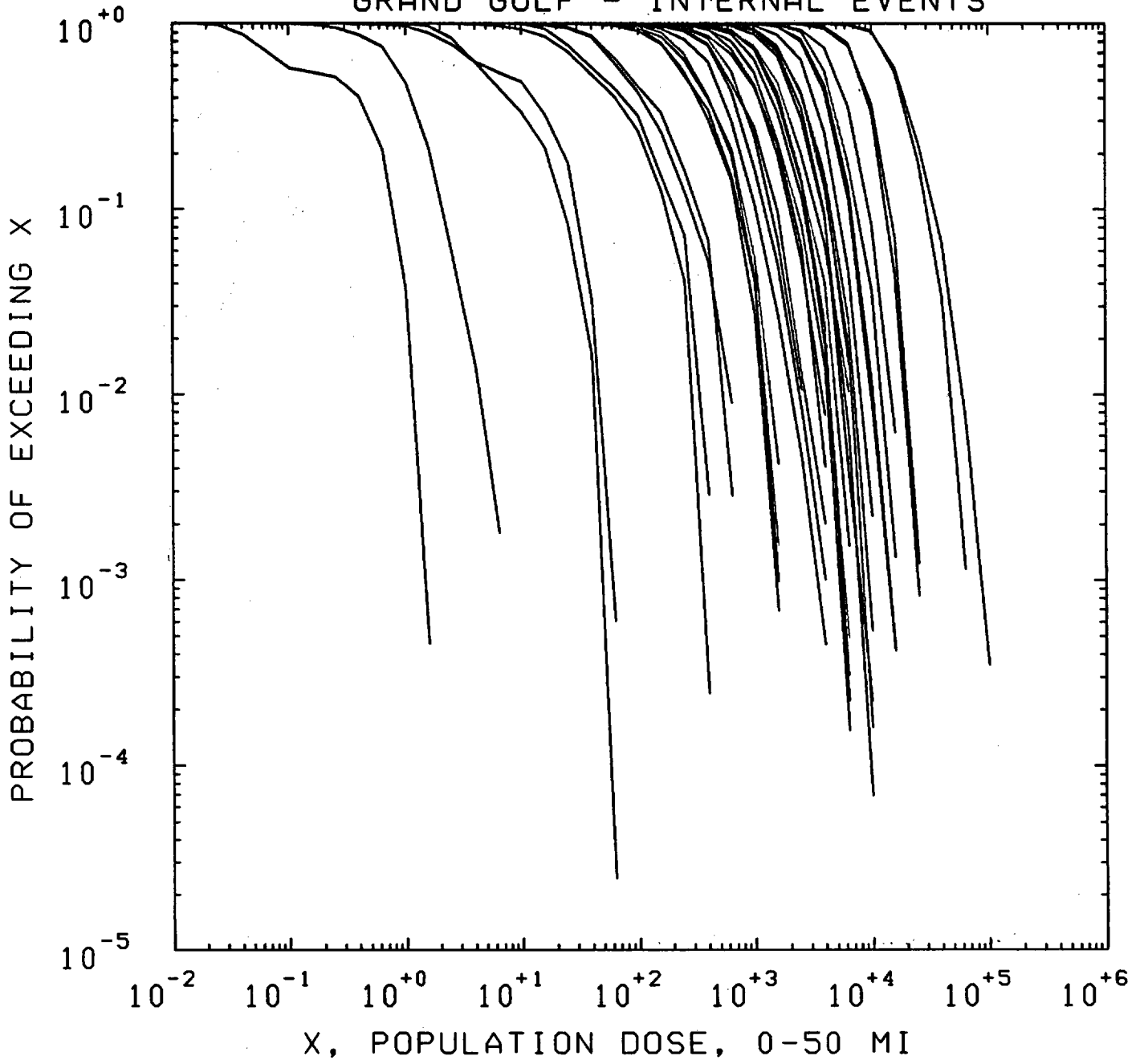


Figure 4.2-1. (continued)

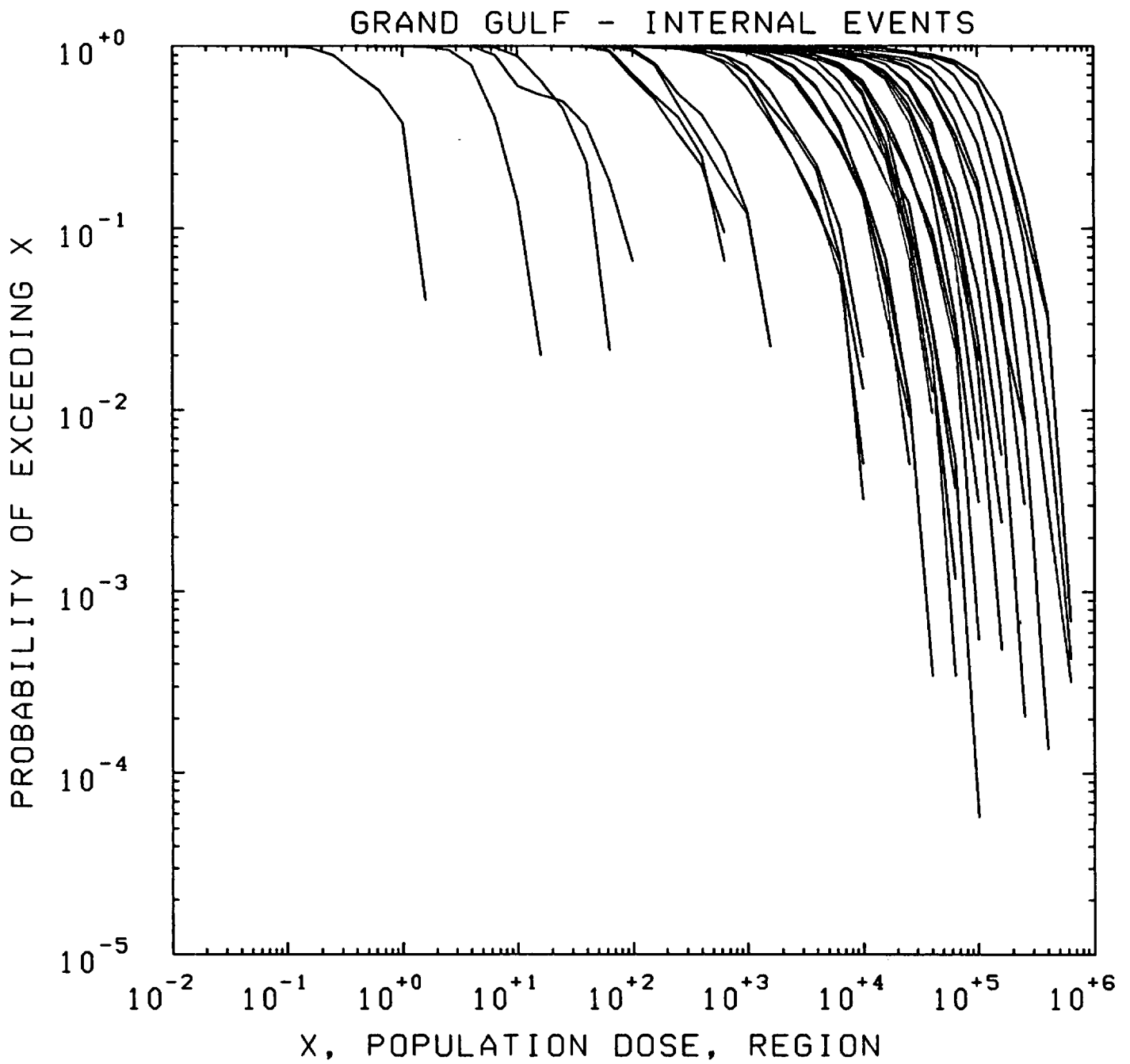


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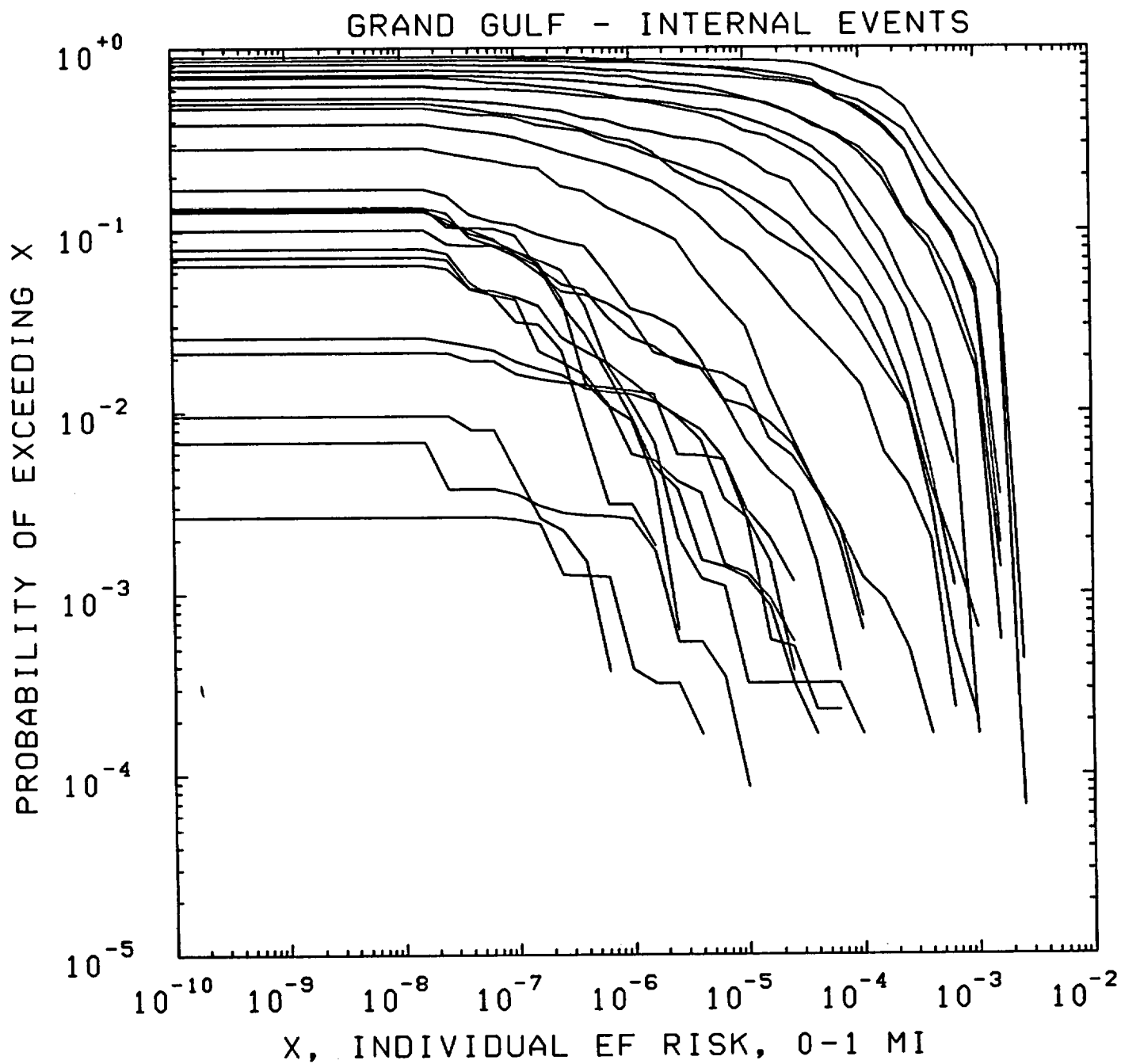


Figure 4.2-1. (continued)



GRAND GULF - INTERNAL EVENTS

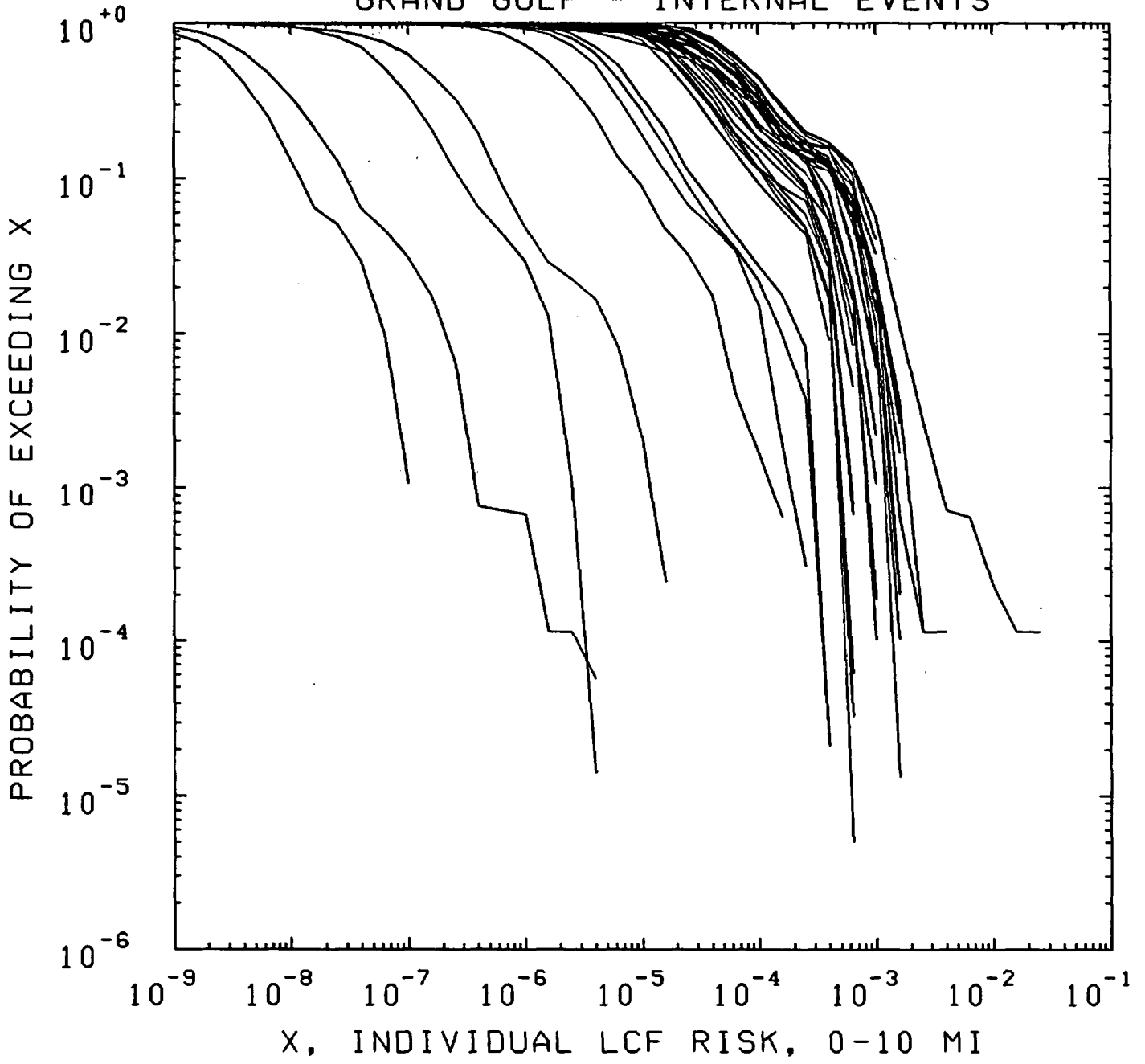
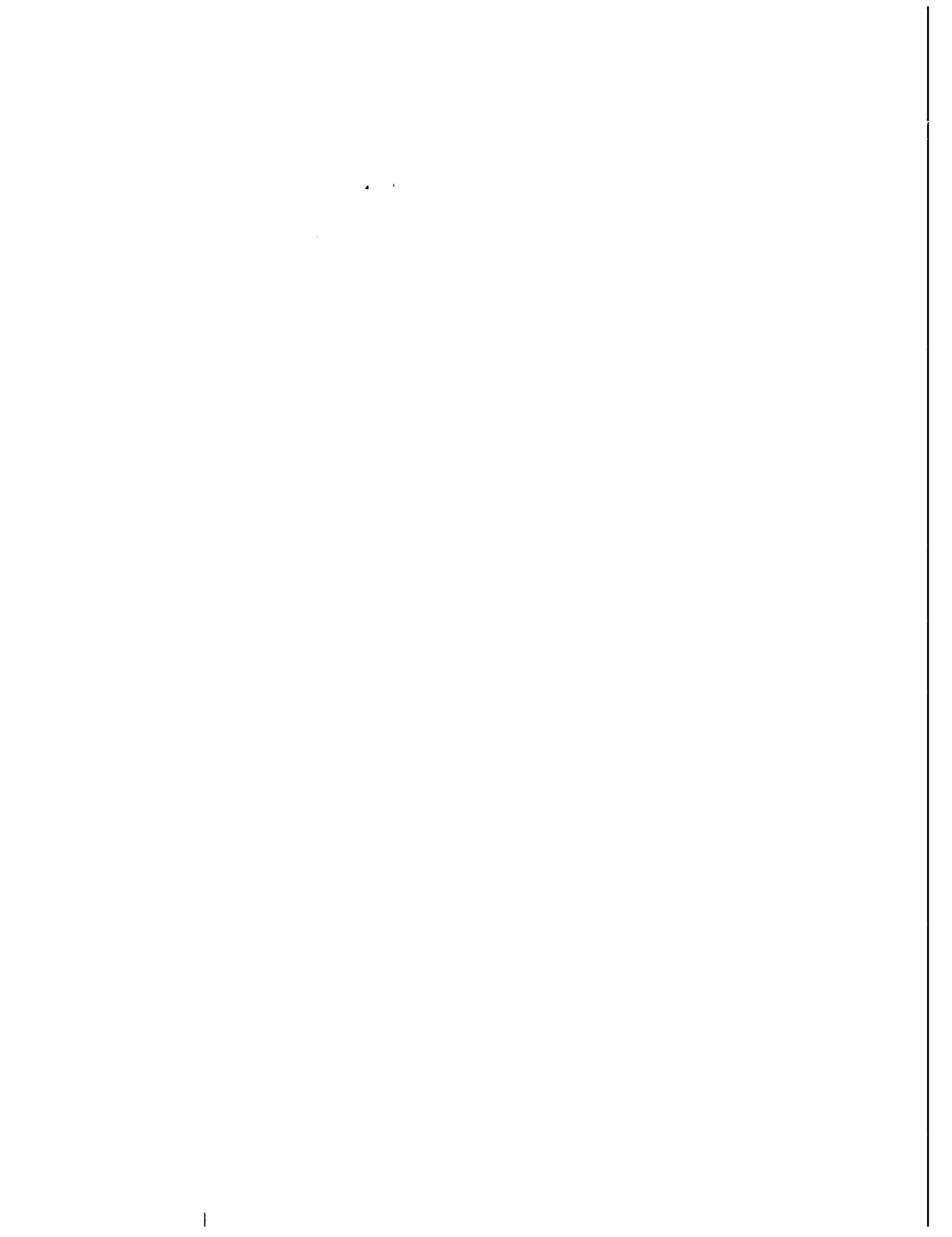


Figure 4.2-1. (continued)

#### 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, "PRAMIS: Probabilistic Risk Assessment Model Integration System User's Guide," NUREG/CR-5262, SAND88-3093, Sandia National Laboratories, May 1989.



## 5. RISK RESULTS FOR GRAND GULF

This section gives the results of the integrated risk analysis for the Grand Gulf 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 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.

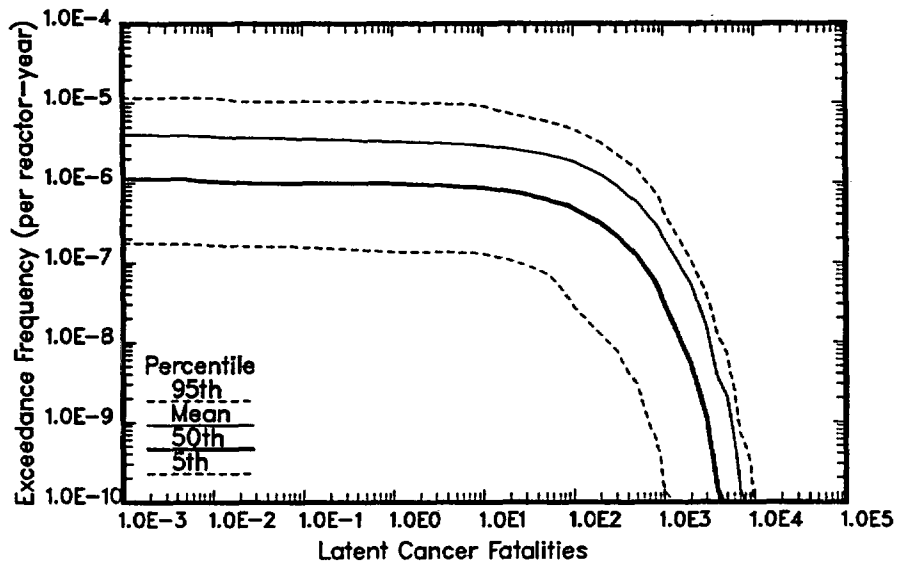
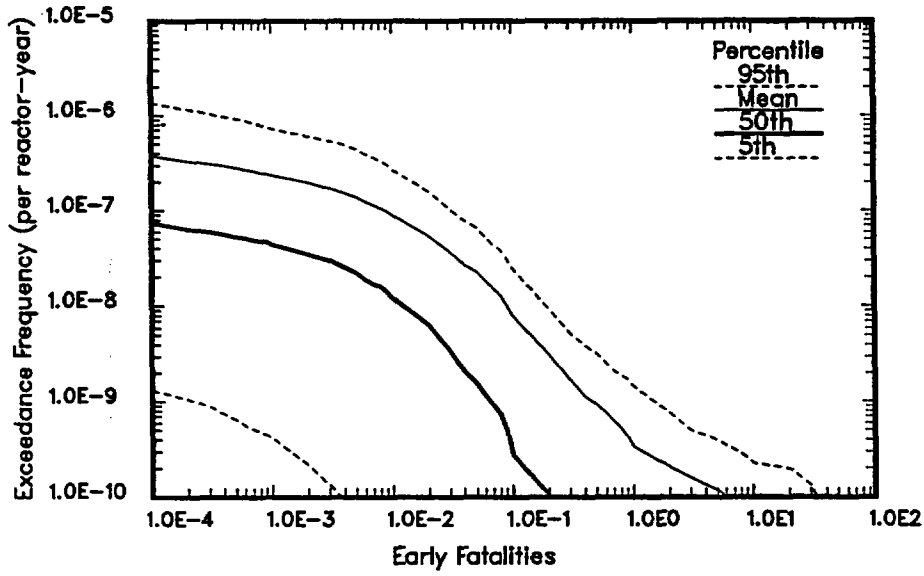
The figures in this section present only a very small portion of the total risk output available. Detailed listings of results are available on computer media by request.

### 5.1 Results for Internal Initiators

This section describes the results of the integrated risk analysis for internal initiators at the Grand Gulf plant. Section 5.1.1 is a discussion of basic risk results for internal initiators. Section 5.1.2 addresses the types of accidents and plant features, which are important in determining the risk from internal initiators at Grand Gulf. Finally, Section 5.1.3 constitutes the results of the regression analysis performed to determine the important contributors to the uncertainty in risk.

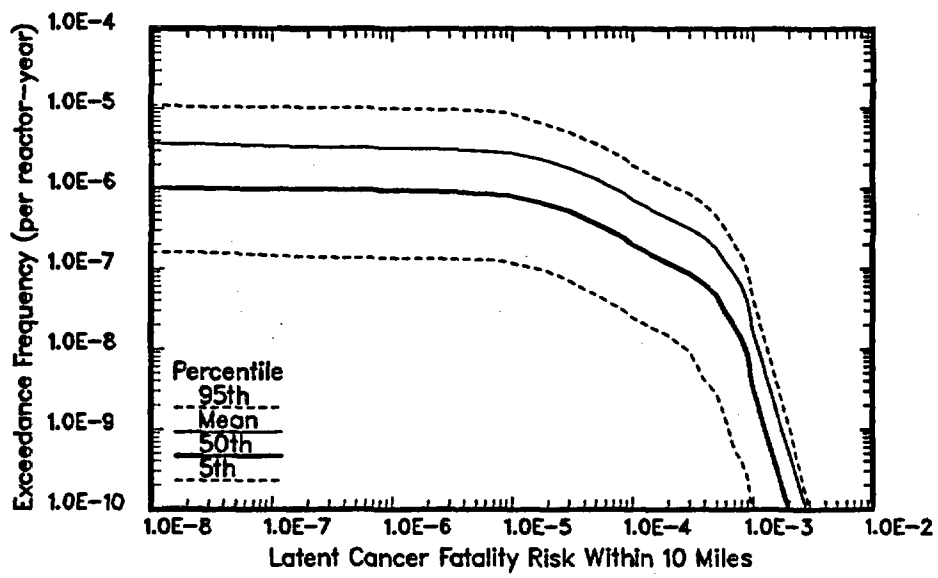
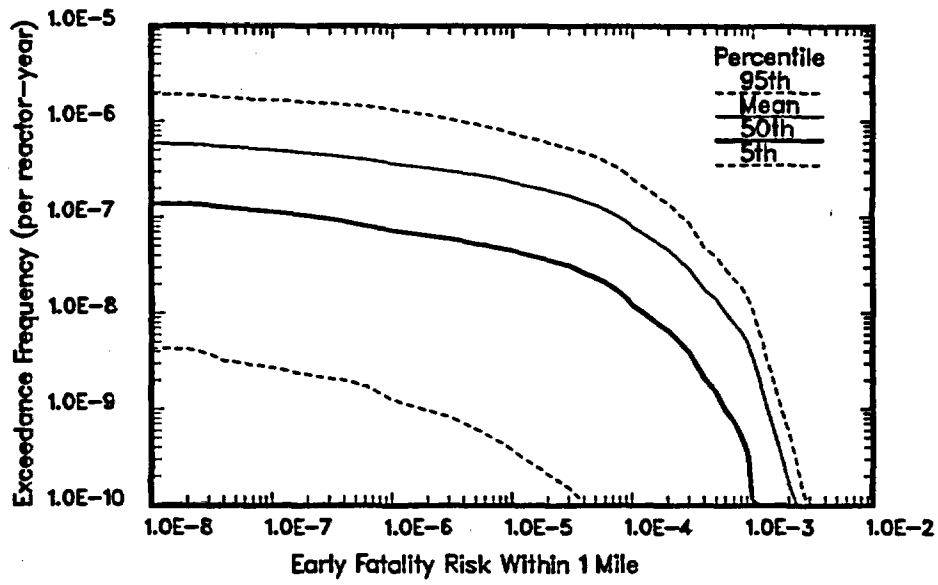
#### 5.1.1 Risk Results

Figure 5.1-1 shows the basic results of the integrated risk analysis for internal initiators at Grand Gulf. 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 250 observations in the sample for Grand Gulf, the complete set of risk results, at the most basic level, consists of 250 CCDFs for each consequence measure. Plots showing these 250 curves are contained in Appendix D; only four statistical measures of the 250 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 250 values of the exceedance frequency (one for each observation or sample element) and from these 250 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 relationship. The percentile and mean curves in Figure 5.1-1 and similar figures are only valid when read from the abscissa.



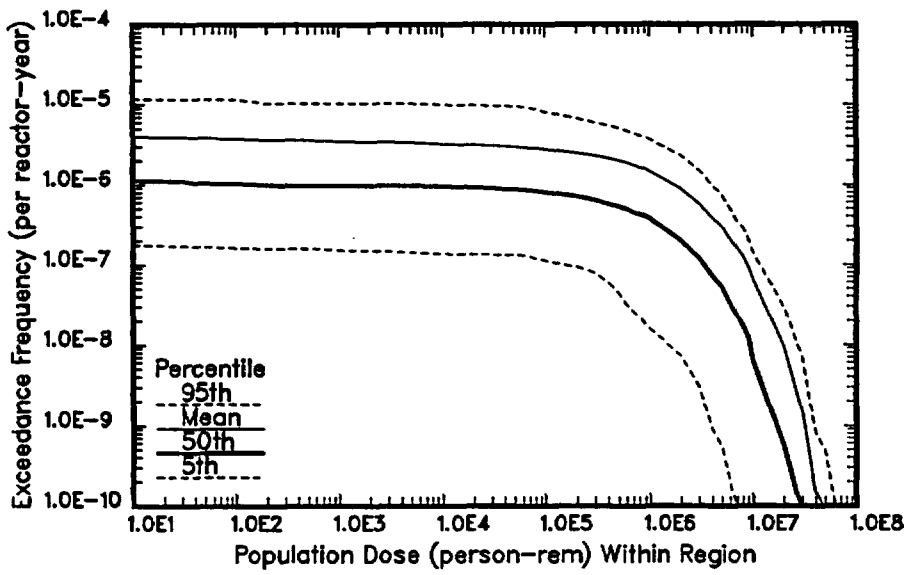
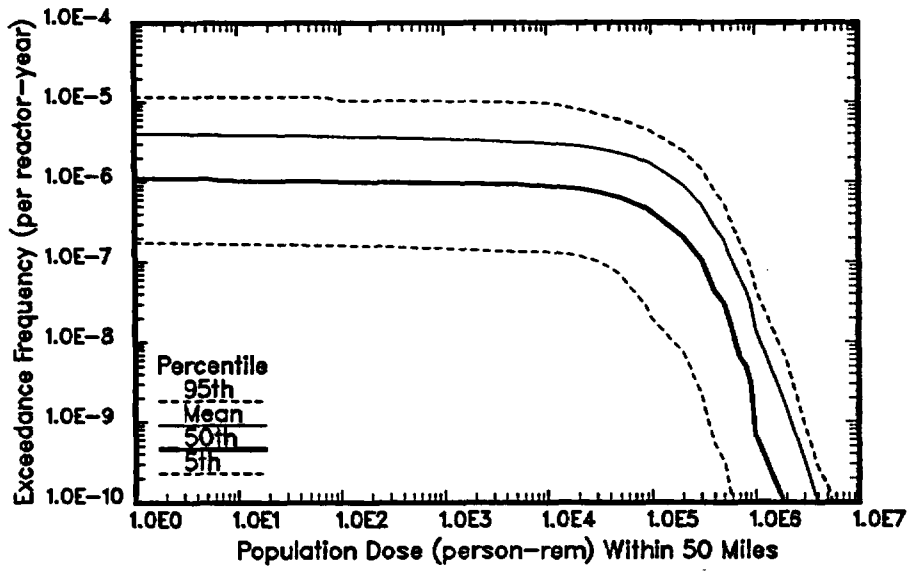
GRAND GULF BaseCase

Figure 5.1-1. Results of the Integrated Risk Analysis for Internal Initiators at Grand Gulf: Statistical Measures of the 250 Exceedance Frequency Curves for Six Consequence Measures



GRAND GULF BaseCase

Figure 5.1-1. (continued)



GRAND GULF BaseCase

Figure 5.1-1. (continued)

Although the abscissa in the third and fourth 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 given that 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 third and fourth 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 0.001 to 70 fatalities. This means that latent cancer fatalities in this range are very unlikely. Any type of containment failure is likely to lead to more than 70 delayed fatalities; it is extremely unlikely, however, that an accident will result in more than 10,000 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 0.001 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 non-stochastic. 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 (EF) and latent cancer fatalities (LCF):

<u>Consequence</u>	<u>Exceedance Frequency (1/R-yr)</u>	
	<u>Mean</u>	<u>Median</u>
1 EF	3E-10	< 1E-12
100 EF	3E-12	< 1E-12
100 LCF	2E-6	5E-7
5000 LCF	2E-9	8E-11

Although the latent cancer fatality values mentioned above may appear large, they must be considered in perspective; the calculated latent cancer fatalities 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 22 million and the population within 1000 miles of the plant is about 142 million. When spread over two or three decades, even tens of thousands of additional latent cancer fatalities 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 that are 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 250 observations, there are 250 values of annual risk for each consequence measure. These 250 values may be ordered and plotted as histograms, which is done in Figure 5.1-2. The four statistical measures utilized 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 early fatality consequence measures than for the latent cancer fatality or population dose consequence measures due to the threshold effect for early fatalities. 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 individual fatality risks, which are really an individual's probability of becoming a casualty of a reactor accident in a given year. The individual early fatality 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 250 values is plotted in the last two frames of Figure 5.1-2. Although the values are really

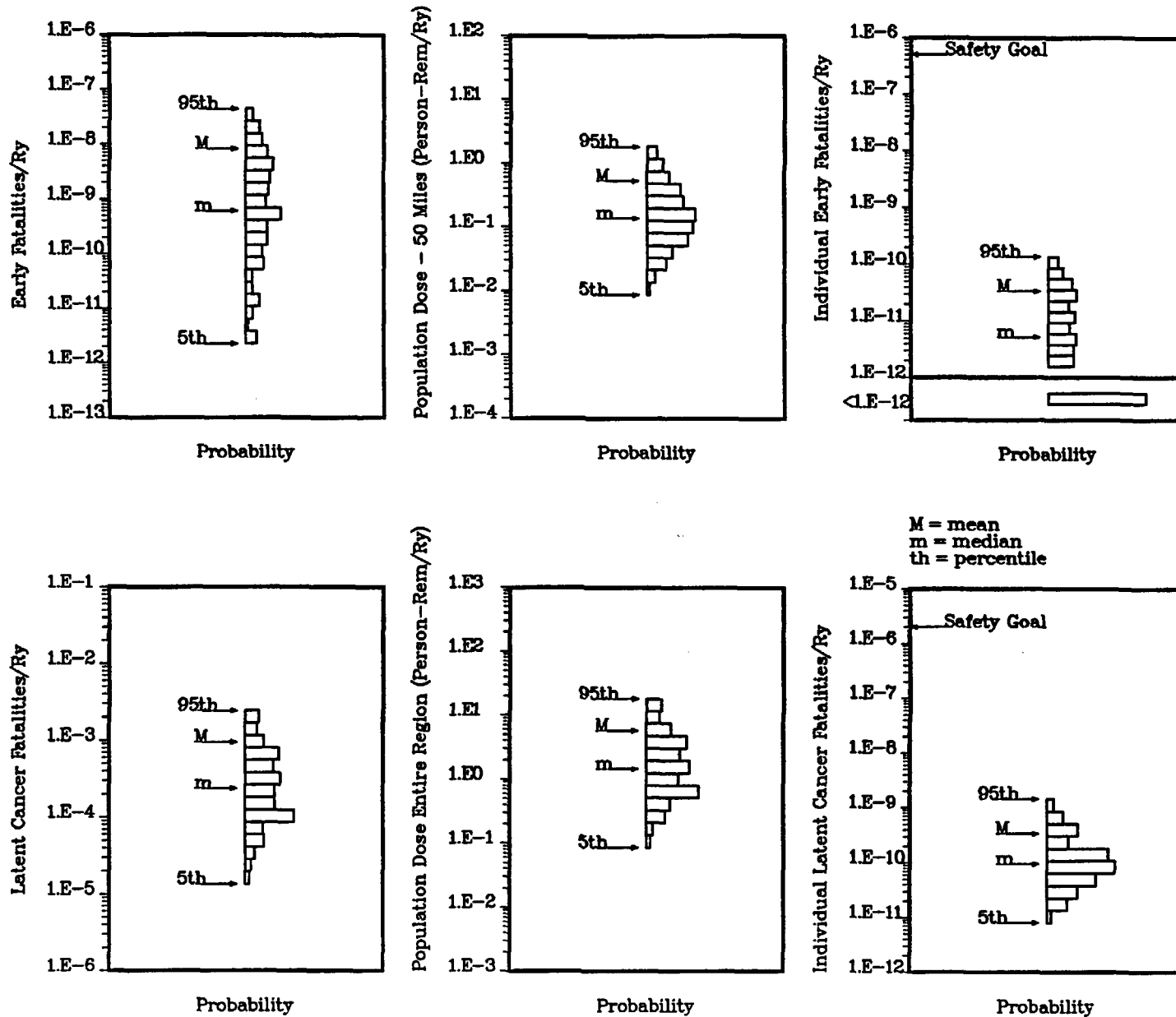


Figure 5.1-2. Annual Risks. Grand Gulf: All Internal Initiators.

Table 5.1-1  
Distributions for Annual Risk at Grand Gulf due to Internal Initiators  
(All values per reactor-year)  
(Population doses in person-rem)

<u>Risk Measure</u>	<u>5th%tile</u>	<u>Median</u>	<u>Mean</u>	<u>95th%tile</u>
Core Damage	1.8E-7	1.1E-6	4.1E-6	1.4E-5
Early Fatalities	2.5E-12	6.1E-10	8.2E-09	2.6E-08
Latent Cancer Fat.	1.4E-5	2.4E-04	9.5E-4	2.3E-3
Population Dose 50 mi.	1.2E-2	1.3E-1	5.2E-01	1.4E+0
Population Dose Entire Region	9.0E-2	1.4E+0	5.8E+0	1.5E+1
Ind. Early Fat. Risk 1 mile	2.3E-14	5.2E-12	3.3E-11	1.0E-10
Ind. L. C. Fatalities Risk--10 miles	1.3E-11	9.4E-11	3.4E-10	9.7E-10

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. The plots for individual risk in Figure 5.1-2 show that both risk distributions for Grand Gulf 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 subsection 5.1.2.

The offsite risk at Grand Gulf is relatively low, both with respect to the safety goals and to the other plants analyzed in NUREG-1150. There are several factors that lead to these low values for risk. First, the core damage frequency for Grand Gulf is very low. The mean core damage frequency is 4.0E-06. Although it is likely that the containment will fail given that core damage occurs, there are several features of the Grand Gulf plant and surrounding area that tend to reduce the consequences. The early fatality risk depends on both the magnitude of the release and on the timing of containment failure. If the containment fails early in the accident 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. The low early fatality risk can in part be attributed to the fast evacuation of the population around the plant. The population

in the vicinity of the plant is fairly sparse. This in part leads to a short evacuation delay and a fast evacuation speed. Thus, in many of the accidents analyzed, most of the population was evacuated so that they were not exposed to the plume from the accident. Furthermore, there is a threshold effect associated with early fatalities. That is, to cause an early fatality the release must be of a certain magnitude (i.e., above a certain threshold). There are several features of the Grand Gulf plant that reduce the magnitude of the source term. First, in the majority of the accidents analyzed, the in-vessel releases are scrubbed by the suppression pool. Second, because the dominant PDS group is the short-term SBO, there is a significant probability that ac power will be recovered and coolant injection will be restored to the core such that the core damage process is arrested before the vessel fails. Third, given that the vessel does fail, it is likely that either the core debris released from the vessel will be cooled or if CCI is initiated it will occur under a pool of water.

The latent cancer fatalities are generally associated with the population that is located beyond the emergency evacuation zone. Thus, this risk measure is not particularly sensitive to the timing of containment failure, but rather whether the containment fails or not. Furthermore, because there is no threshold effect for latent cancer fatalities, this consequence measure is not as sensitive to the magnitude of the release as is the early fatality risk. Thus, latent cancer fatality risk is primarily dependent on frequency of containment failure. Unlike early fatality risk, late containment failures as well as early failures of the containment are important to the latent cancers. Because the conditional probability of containment failure is high, the low values for latent cancer fatalities can be attributed to the low core damage frequency.

### 5.1.2 Contributors to Risk

There exist two distinct ways to calculate contribution to risk. To facilitate their definition, the following quantities are introduced:

$rC_j$  = risk (units: consequences/reactor-year) for consequence measure  $j$ ,

$rC_{i,j}$  = value for  $rC_j$  obtained for observation  $i$ ,

$rC_{jk}$  = risk (units: consequences/reactor-year) for consequence measure  $j$  due to PDS group  $k$ ,

$rC_{i,jk}$  = value for  $rC_{jk}$  obtained for observation  $i$ , and

$nLHS$  = number of observations in the Latin Hypercube Sample.

The notation used here is similar to that used in Section 1.4. The value of  $nLHS$  is 250 for Grand Gulf. The risk  $rC_{i,j}$  is the  $j$ th element of the vector  $rC_i$  in Equation (1.9) of Section 1.4. The risk  $rC_{i,jk}$  is the  $j$ th element of the vector  $rC_i$  when the frequencies of all the PDS groups except group  $k$  in the vector  $fPDS_i$  are set to zero. The vector  $fPDS_i$  is equal to the product  $fIE_i P_i(IE \rightarrow PDS)$ .

The result of the first method for computing contribution to risk is denoted the fractional contribution to mean risk and abbreviated 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

$$FCMR_{jk} = [ \sum rC_{ijk} / nLHS ] / [ \sum rC_{ij} / nLHS ] \\ - \sum rC_{ijk} / \sum rC_{ij},$$

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 and abbreviated 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

$$MFCR_{jk} = \sum ( rC_{ijk} / rC_{ij} ) / nLHS,$$

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 four summary PDS groups used for reporting results in NUREG-1150. 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 the short-term SBO groups is the major contributor to mean early fatality risk at Grand Gulf. It is not valid to state that the short-term SBO group contributes 93.2% of the early fatality risk at Grand Gulf. Although the exact values are different for each method, the basic conclusions that can be drawn from these results are the same. That is, both the mean early fatality risk and the mean latent cancer fatality risk are dominated by the short-term SBO group. The long-term SBO group and the ATWS group contribute considerably less to these risk measures and the T2 group is a very minor contributor.

Table 5.1-2  
 Fractional PDS Contributions (in percent) to Annual  
 Risk at Grand Gulf Due to Internal Initiators

<u>Summary PDS Group</u>	<u>Method</u>	<u>Core Damage</u>	<u>Early Fatalities</u>	<u>Latent Cancer Fatalities</u>	<u>Population Dose 50 miles</u>	<u>Population Dose Region</u>	<u>Ind. E. F. Risk-1 mile</u>	<u>Ind. L.C.F. Risk-10 mile</u>
Short-Term SBO	FCMR	94.2	93.2	91.3	91.9	91.4	92.2	92.8
	MFCR	87.7	84.1	85.3	85.7	85.5	84.2	85.5
Long-Term SBO	FCMR	2.5	4.7	4.8	4.3	4.7	4.4	3.6
	MFCR	2.5	6.5	5.0	4.4	4.8	5.9	4.1
ATWS	FCMR	2.7	2.0	3.5	3.4	3.5	3.1	3.2
	MFCR	7.8	7.9	8.2	8.3	8.2	8.3	8.6
T2	FCMR	0.5	0.2	0.4	0.4	0.4	0.3	0.4
	MFCR	2.0	1.5	1.5	1.6	1.5	1.6	1.8

Pie charts for both methods of computing the contribution to risk are shown in Figure 5.1-3 for early fatalities and latent cancer fatalities for the four summary PDS groups. The differences are readily apparent when this method of displaying the results is utilized, 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 Surry. The second sample is just as valid as the first sample, and differs from the first sample only in the fact that a different random seed was used in the LHS program. Therefore, the differences in the results between the two samples are an indication of the robustness of the analysis methods. In addition, a comparison of the two samples provides an indication of which method of calculating the contribution to risk tends to be more stable. The results from the 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 early fatality and latent cancer fatality 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 a relatively few observations. Next, 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 the annual risk for each observation is typically dominated by a few APBs which have both high frequency and high source terms and the mean risk is dominated by a few observations which 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 observations that determine 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 probabilistic risk analyses (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  $r_{C_{jk}}/r_{C_j}$  (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.

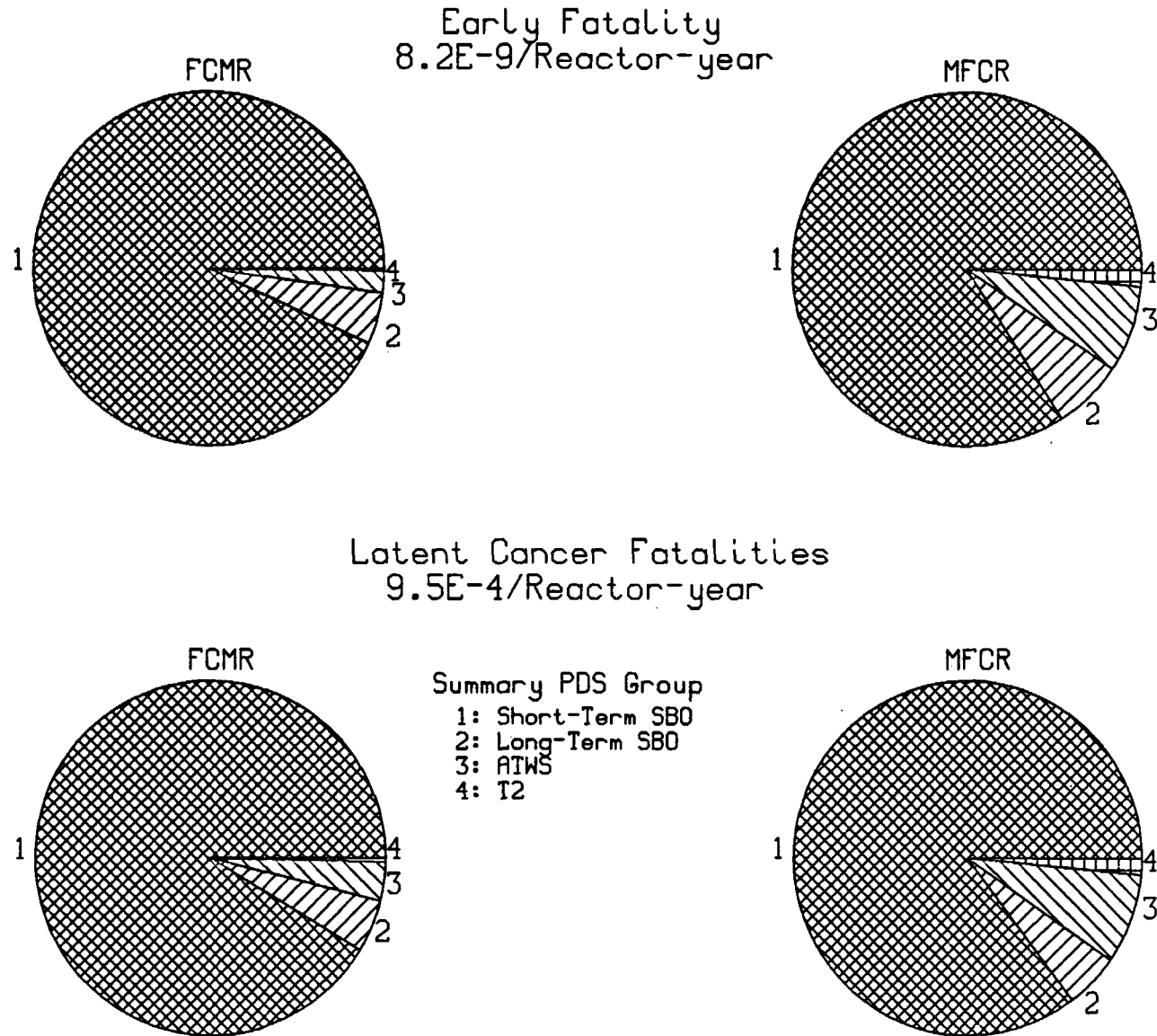


Figure 5.1-3. Fractional PDS Contributions to Annual Risk; Grand Gulf: Internal Initiators.  
MFCR= Mean Fractional Contribution to Risk; FCMR=Fractional Contribution to Risk.

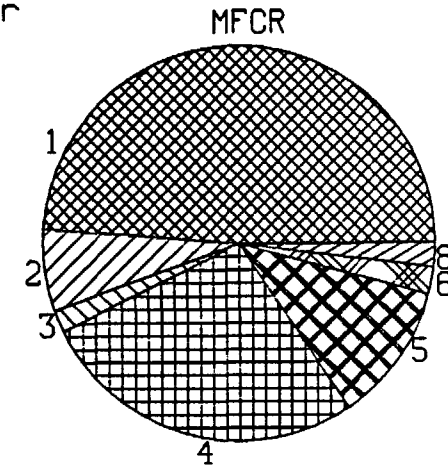
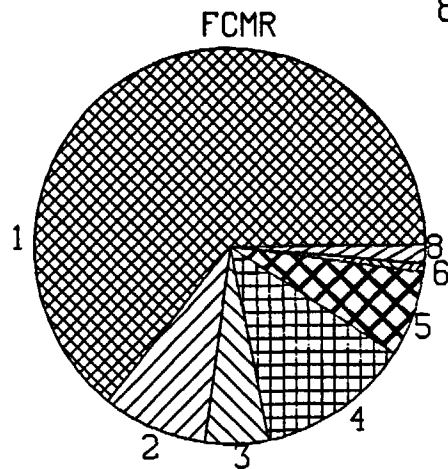


Table 5.1-3  
 Fractional APB Contributions (in percent) to Annual  
 Risk at Grand Gulf Due to Internal Initiators

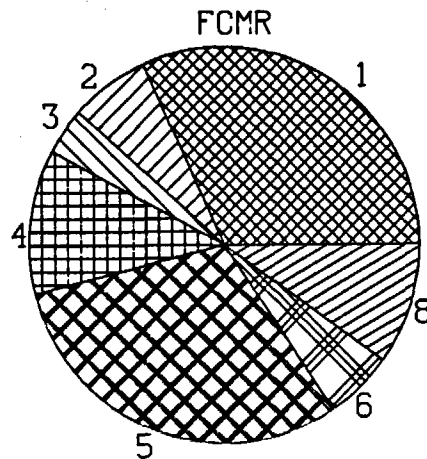
<u>Summary Accident Progression</u>	<u>Method</u>	<u>Early Fatalities</u>	<u>Latent Cancer Fatalities</u>	<u>Population Dose Dose 50 miles</u>	<u>Population Dose Region</u>	<u>Ind. E. F. Risk-1 mile</u>	<u>Ind. L.C.F. Risk-10 mile</u>
VB, Early CF, Early SP Bypass, No CS	FCMR	64.2	31.7	29.3	31.7	52.9	19.7
	MFCR	48.7	28.7	26.8	28.5	44.0	21.9
VB, Early CF, Early SP Bypass, CS Avail.	FCMR	8.7	6.5	7.2	6.6	9.4	7.1
	MFCR	6.7	5.4	5.6	5.5	7.3	5.8
VB, Early CF Late SP Bypass	FCMR	5.2	3.9	4.7	4.0	11.4	4.3
	MFCR	1.8	1.1	1.1	1.1	1.9	1.0
VB, Early CF No SP Bypass	FCMR	12.8	11.8	12.9	11.9	13.1	14.2
	MFCR	27.1	26.3	26.7	26.3	29.6	27.8
VB, Late CF	FCMR	7.0	30.4	30.8	30.4	9.9	36.3
	MFCR	11.4	27.5	28.1	27.6	12.9	30.2
VB, Vent	FCMR	0.7	5.7	5.7	5.7	1.0	6.7
	MFCR	2.2	4.0	3.9	4.0	2.1	4.1
VB, No CF	FCMR	0.0	0.01	0.08	0.00	0.0	0.02
	MFCR	0.0	0.0	0.03	0.00	0.0	0.01
No VB	FCMR	1.5	9.9	9.4	9.6	2.2	11.6
	MFCR	2.0	7.0	7.7	6.9	2.1	9.2

5.14

Early Fatality  
8.2E-9/Reactor-year



Latent Cancer Fatalities  
9.5E-4/Reactor-year



Summary Accident Progression

- 1: VB, Early CF, Early SP Bypass, No CS
- 2: VB, Early CF, Early SP Bypass, CS
- 3: VB, Early CF, Late SP Bypass
- 4: VB, Early CF, No SP Bypass
- 5: VB, Late CF
- 6: VB, Venting
- 7: VB, No CF
- 8: No VB

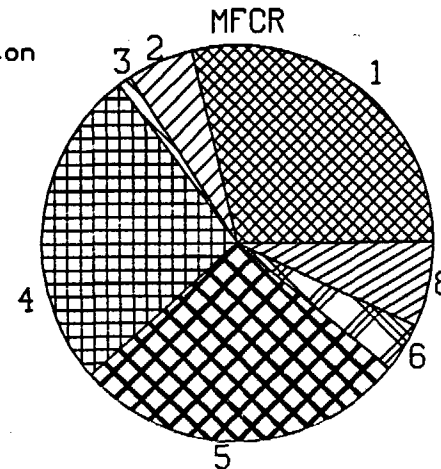


Figure 5.1-4. Fractional APB Contributions to Annual Risk. Grand Gulf: Internal Initiators. MFCR= Mean Fractional Contribution to Risk; FCMR=Fractional Contribution to Risk.

MFCR results from averaging over the 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 Grand Gulf are clear. For all of the consequence measures, the risk is dominated by the short-term SBO PDS group. This group is the dominant contributor to the core damage frequency and because ac power is not initially available in these PDSs, there is a significant probability that these accidents will involve early containment failure.

For the two consequence measures that depend on a large early release, early fatalities and individual risk of early fatality within one mile, the risk is dominated by accidents that progress to vessel breach and that involve early containment failures. Accidents in which the containment fails late are much less significant. In Table 5.1-3 the first bin (VB, Early CF, Early SP Bypass, No CS) is the dominant contributor to these risk measures because the containment fails early and the releases at vessel breach and after vessel breach are not scrubbed by either the pool or the containment sprays. Although the fourth bin in Table 5.1-3 (VB, Early CF, No SP Bypass) does not involve drywell failure, its contribution to early fatality risk is higher than the second bin (VB, Early CF, Early SP Bypass, CS Avail.) in which the drywell fails early in the accident. The reason for this is that the mean probability of the fourth bin is roughly four times the mean probability of the second bin. Thus, although the fourth bin does not involve drywell failure, the probability of this bin coupled with the fact that the containment fails early is sufficient to make this bin a significant contributor to early fatality risk.

Latent cancer fatalities 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. Furthermore, if the suppression pool is bypassed there is a greater likelihood that the release will be large. Thus, accidents in which some of the releases are not scrubbed by either the pool or the sprays tend to contribute more to latent cancer fatality risk than accidents in which the drywell remains intact. It is for this reason that the first bin in Table 5.1-3 (VB, Early CF, Early SP Bypass, No CS) is the dominant contributor to the latent cancer fatality risk. The following three risk measures also depend on the total amount of radioactivity released: population dose

within 50 miles, population dose within the entire region, and individual risk of latent cancer fatality within 10 miles.

The bin that involves accidents in which the vessel does not fail makes a minor contribution to the early fatality risk; however, it makes a noticeable contribution to the latent cancer fatality risk. It must be remembered that although the vessel does not fail in these accidents, the containment can still fail early in these accidents from the combustion of hydrogen in the wetwell. Early failure of the containment will allow a portion of the in-vessel releases to escape into the environment. The combination of the threshold effect associated with early fatalities with the fact that the releases associated with this bin are fairly small results in few early fatalities. For latent cancers, on the other hand, there is no threshold effect. Thus, any releases that are not trapped by the suppression pool or removed by the containment sprays can contribute to the latent cancer risk.

### 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 Grand Gulf due to internal initiators. The 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 Grand Gulf. 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, 2.3-3, 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 2.3-3 the variables H2INVES1 through

H2INVES6 are all correlated and, therefore, in the sensitivity analysis they are treated as a single variable (i.e., H2INVES).

Sensitivity analysis results for the six consequence measures used to express risk are presented in Table 5.1-4. This table contains the results of performing a stepwise regression on the risk as expressed by: early fatalities, latent cancer fatalities, population dose within 50 miles, population dose within the entire region, individual risk of early fatality within 1 mile, and individual risk of latent cancer fatality within 10 miles. The statistical package SAS<sup>1</sup> was used to perform the regression.

For each consequence measure, 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<sup>2</sup> values that result with the entry of successive variables into the model. The tendency of a dependent variable to increase and decrease with an increase in the 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 analyses for early fatalities and individual risk of early fatality within 1 mile only account for about 45% 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 analyses for the other four consequence measures are somewhat more successful as they are able to account for about 60% of the variability. The independent variables that account for this variability are predominantly those variables that determine the frequencies of the accident.

**Table 5.1-4  
Summary of Regression Analyses for  
Annual Risk at Grand Gulf for Internal Initiators**

Step	Early Fatalities			Latent Cancer Fatalities			Population Dose--50 miles		
	VAR <sup>a</sup>	RC <sup>b</sup>	R <sup>2c</sup>	VAR	RC	R <sup>2</sup>	VAR	RC	R <sup>2</sup>
1	H2INVES	Neg	0.06	IE-LOSP	Pos	0.13	IE-LOSP	Pos	0.15
2	FCONG	Pos	0.11	DGN-FSTR	Pos	0.22	DGN-FSTR	Pos	0.25
3	IE-LOSP	Pos	0.16	BAT-LP	Pos	0.29	DGN-FRUN	Pos	0.32
4	BAT-LP	Pos	0.19	DGN-FRUN	Pos	0.35	BAT-LP	Pos	0.38
5	DGN-FSTR	Pos	0.23	TDP-FRUN	Pos	0.39	TDP-FRUN	Pos	0.44
6	DFPOOL	Neg	0.27	FCONG	Pos	0.44	F-RPS	Pos	0.47
7	DW-Ped-F	Pos	0.31	DFPOOL	Neg	0.47	FCONG	Pos	0.50
8	DWPVB1	Pos	0.34	F-RPS	Pos	0.50	MOV-FOP	Pos	0.52
9	H2AVB	Neg	0.36	MOV-FOP	Pos	0.52	DFPOOL	Neg	0.54
10	FVES	Pos	0.39	AC	Neg	0.54	AC	Neg	0.57
11	AC	Neg	0.42	DFSPRAY	Neg	0.56	BETA-BAT	Pos	0.58
12	FCOR	Neg	0.44	BETA-BAT	Pos	0.57	MDP-FSTR	Pos	0.60
13				DW-Ped-F	Pos	0.59	DFSPRAY	Neg	0.61
14				DWPVB1	Pos	0.60	DWPVB1	Pos	0.62

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<sup>2</sup> values with the entry of successive variables into the regression model.

Table 5.1-4 (continued)

Step	Population Dose Entire Region			Individual Early Fat. Risk 0-1 mile			Individual Latent Can. Fat. Risk 0-10 mi.		
	VAR <sup>a</sup>	RC <sup>b</sup>	R <sup>2c</sup>	VAR	RC	R <sup>2</sup>	VAR	RC	R <sup>2</sup>
1	IE-LOSP	Pos	0.13	H2INVES	Neg	0.06	IE-LOSP	Pos	0.17
2	DGN-FSTR	Pos	0.22	IE-LOSP	Pos	0.11	DGN-FSTR	Pos	0.28
3	BAT-LP	Pos	0.29	DW-Ped-F	Pos	0.16	DGN-FRUN	Pos	0.36
4	DGN-FRUN	Pos	0.35	DGN-FSTR	Pos	0.20	BAT-LP	Pos	0.43
5	TDP-FRUN	Pos	0.40	H2AVB	Neg	0.24	TDP-FRUN	Pos	0.48
6	FCONG	Pos	0.44	FCONG	Pos	0.28	MOV-FOP	Pos	0.52
7	DFPOOL	Neg	0.47	DFPOOL	Neg	0.31	F-RPS	Pos	0.55
8	F-RPS	Pos	0.50	BAT-LP	Pos	0.34	AC	Neg	0.57
9	MOV-FOP	Pos	0.52	FVES	Pos	0.37	BETA-BAT	Pos	0.59
10	AC	Neg	0.54	AC	Neg	0.40	MDP-FSTR	Pos	0.60
11	DW-Ped-F	Pos	0.56	DWPVB1	Pos	0.42	FCONG	Pos	0.61
12	BETA-BAT	Pos	0.57	DGN-FRUN	Pos	0.44	DFSPRAY	Neg	0.62
13	DFSPRAY	Neg	0.59	BETA-BAT	Pos	0.46			
14	DWPVB1	Pos	0.60	EffBrnP	Pos	0.47			

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<sup>2</sup> values with the entry of successive variables into the regression model.

## 5.2 References

1. SAS Institute Inc., SAS<sup>®</sup> User's Guide: Statistics, Version 5 Edition, Cary, NC, SAS Institute Inc., 1985.





## 6. INSIGHTS AND CONCLUSIONS

Core Damage Arrest. For the dominant summary PDS group, short-term SBO, there is a significant probability that the core damage process will be arrested and vessel failure will be averted. For the accidents in which the vessel does not fail, there are no ex-vessel fission product releases (e.g., DCH or CCI). Furthermore, loads accompanying vessel breach, which pose a significant challenge to both the drywell and the containment, are avoided. The conditional probability of core damage arrest in the short-term SBO PDS group is driven by the ac power recovery probability. In the other summary PDS groups (i.e., long-term SBO, ATWS, and T2) it is unlikely that core damage process will be arrested. The core damage arrest probability for the long-term SBO group is low because the probability of recovering ac power early in the accident is fairly low for this PDS group. In the ATWS and T2 PDS groups the low values for core damage arrest are attributed to the fairly high likelihood that the operators fail to depressurize the RPV to allow coolant injection to be restored to the core.

Containment Failure. Given that core damage occurs, it is likely that the containment will fail during the course of the accident. Furthermore, for the dominant PDS summary group, short-term SBO, there is a substantial probability that the containment will fail early in the accident. Hydrogen combustion events are the dominant events that cause early CF in the short-term SBO and T2 PDS groups. The combination of a relatively weak containment, the copious production of hydrogen during core damage, and the unavailability of the HIS during a SBO leads to a high conditional probability of containment failure. The mean conditional probability of early containment failure for these two groups is approximately 0.5. In the short-term SBO group about half of the early CF probability results from failures that occur before vessel breach and the other half results from failures shortly after vessel breach. In the T2 PDS group the vast majority of the early containment failures occur around the time of vessel breach. For both the long-term SBO PDS group and the ATWS PDS group, hydrogen combustion events and pressurization of the containment from the accumulation of steam contribute to their high conditional probabilities of early containment failure.

Drywell Failure. Early drywell failure is an important attribute of the accident progression because failure of the drywell establishes a pathway for radionuclides in the drywell to bypass the suppression pool. The suppression pool offers an important mechanism for reducing the source term. Accidents that result in early drywell failure coincident with early containment failure are generally the dominant contributors to risk. Roughly 50% of the mean condition probability of early containment failure is attributed to accidents that also involve early drywell failure. Early drywell failures include failures that occur before vessel breach and failures that occur at vessel breach. Only the short-term SBO PDS group has significant probability of drywell failure before vessel breach. The vast majority of these drywell failures are caused by hydrogen combustion events. All of the PDS groups have a significant probability of drywell failure at the time of vessel breach. The majority of these failures are

caused by loads accompanying vessel breach. These quasi-static loads include contributions from DCH, ex-vessel steam explosions, hydrogen burns and RPV blow down.

Fission Product Releases. There is considerable uncertainty in the release fractions for all types of accidents. There are several features of the Grand Gulf plant that tend to mitigate the release. First, the in-vessel releases are generally directed to the suppression pool where they are subjected to the pool decontamination factor. Provided the drywell has not failed, the radionuclides released into the drywell will also pass through the pool. Although generally not as effective as the suppression pool, the containment sprays and the reactor cavity pool also offer mechanisms for reducing the release of radionuclides from the containment when the suppression pool has been bypassed. The largest releases tend to occur when the suppression pool is bypassed and the containment sprays are not operating.

Risk. The offsite risk from internal initiating events was found to be quite low, both with respect to the safety goals and to the other plants analyzed in NUREG-1150. The offsite risk is dominated by short-term station blackout PDSs. The long-term station blackout group and the ATWS group contribute considerably less to these risk measures and the T2 group is a very minor contributor. The low values for risk can be attributed to the low core damage frequency, the good emergency response, and plant features that reduce the potential source term.

Uncertainty in Risk. Considerable uncertainty is associated with the risk estimates produced in this analysis. The largest contributors to this uncertainty are the uncertainties in the parameters that determine the frequency of core damage and the uncertainty in some of the parameters that determine the magnitude of the fission product release to the environment. Propagation of the uncertainties in the accident frequency, accident progression, and source term analyses through to risk allows the uncertainty to be calculated and displayed.

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 95<sup>th</sup> percentile value for annual risk falls nearly three orders of magnitude below the safety goals. Furthermore, for both of these risk measures, the maximum of the 250 values that make up the annual risk distributions also falls well below the safety goal.

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11. ABSTRACT *(200 words or less)*

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. report in NUREG-1150, the Severe Accident Risk Reduction Program (SARRP) has completed a revised calculation of the risk to the general public from severe accidents at the Grand Gulf Nuclear Station, Unit 1. This power plant, located in Port Gibson, Mississippi, is operated by the System Energy Resources, Inc. (SERI).

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 risk initiated by events internal to the power plant was assessed.

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