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Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

Summary of Results

Edited by
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Operated by
Sandia Corporation

Prepared for
U.S. Nuclear Regulatory Commission

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Summary of Results

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Abstract

During 1989 the Nuclear Regulatory Commission (NRC) initiated an extensive program to examine the potential risks during low power and shutdown operations. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied by Brookhaven National Laboratory (Surry) and Sandia National Laboratories (Grand Gulf).

The program objectives included assessing the risks of severe accidents initiated during plant operational states other than full power operation and comparing the estimated core damage frequencies, risks, important accident sequences, and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program included a Level 3 probabilistic risk assessment (PRA) for traditional internal events and a Level 1 PRA on fire, flooding, and seismically induced core damage sequences. This report documents the work performed during the analysis of the Grand Gulf plant.

A phased approach was used for the overall study. In Phase 1, the objectives were to identify potential vulnerable plant configurations, to characterize (on a high, medium, or low basis) the potential core damage accident scenario frequencies and risks, and to provide a foundation for a detailed Phase 2 analysis. It was in Phase 1 that the concept of plant operational states (POSSs) was developed to allow the analysts to better represent the plant as it transitions from power operation to nonpower operation than was possible with the traditional technical specification divisions of modes of operation. This phase consisted of a coarse screening analysis performed for all POSSs, including seismic and internal fire and flood for some POSSs.

In Phase 2, POS 5 (approximately cold shutdown as defined by Grand Gulf Technical Specifications) during a refueling outage was selected as the plant configuration to be analyzed based on the results of the Phase 1 study. The scope of the Level 1 study includes plant damage state analysis and uncertainty analysis and is documented in a multi-volume NUREG/CR report (i.e., NUREG/CR-6143). The internal events analysis is documented in Volume 2. Internal fire and internal flood analyses are documented in Volumes 3 and 4, respectively. A separate study on seismic analysis, documented in Volume 5, was performed for the NRC by Future Resources Associates, Inc. The Level 2/3 study of the traditional internal events is documented in Volume 6, and a summary of the results for all analyses is documented in Volume 1.

In the Phase 2 study, system models were developed for POS 5 on the way down to refueling and POS 5 on the way back up from refueling, and supporting thermal hydraulic analyses were performed. Initiating events that may occur during POS 5 were identified and accident sequence event trees were developed and quantified using the IRRAS PRA code. Surviving sequences were examined for recovery potential, appropriate human recovery actions were incorporated into the sequence cut sets, and the sequences were then requantified. Those sequences surviving this preliminary recovery analysis were then reexamined during a "time window" analysis, which allows for a more realistic incorporation of the effects of the decrease in decay heat and a more time-specific incorporation of equipment unavailabilities as the plant transitions from the beginning to the end of POS 5.

Core damage frequency estimates on a per calendar year basis for the Grand Gulf plant are as follows:

	<u>Mean</u>	<u>5th percentile</u>	<u>95th percentile</u>
Internal events (excluding fire and flood)	2.0E-06	4.1E-07	5.4E-06
Internal fire events	<1E-8	-	-
Internal flood events	2.3 E-8	8.2E-11	8.6E-6
Seismic events			
for the LLNL (1993) hazard curves	7.1E-8	2.1E-11	2.2E-7
for the EPRI hazard curves	2.5E-9	2.5E-12	1.1E-8

This is comparable with the total internal event (excluding fire and flood) mean core damage frequency of 4.0E-06 per year estimated in the NUREG-1150 study of full power operations.

The risk associated with Grand Gulf as it operates in POS 5 during a refueling outage was shown to be comparable with the risk associated with full power operation. In NUREG-1150 the risk from full power operation of Grand Gulf was shown to be quite low. While the risk associated with POS 5 is low, there are very few features of the plant that are available to attenuate a release should one occur. The most likely accidents in POS 5 have an open containment, the suppression pool is bypassed, the containment sprays are not available, and the vessel fails, releasing the core debris into the containment. The low values for risk given the high conditional releases are, in part, due to the extremely low core damage frequency and the sparse population around the plant.

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Foreword

(NUREG/CR-6143 and 6144)

Low Power and Shutdown Probabilistic Risk Assessment Program

Traditionally, probabilistic risk assessments (PRA) of severe accidents in nuclear power plants have considered initiating events potentially occurring only during full power operation. Some previous screening analyses that were performed for other modes of operation suggested that risks during those modes were small relative to full power operation. However, more recent studies and operational experience have implied that accidents during low power and shutdown could be significant contributors to risk.

During 1989, the Nuclear Regulatory Commission (NRC) initiated an extensive program to carefully examine the potential risks during low power and shutdown operations. The program includes two parallel projects performed by Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL), with the seismic analysis performed by Future Resources Associates. Two plants, Surry (pressurized water reactor) and Grand Gulf (boiling water reactor), were selected as the plants to be studied.

The objectives of the program are to assess the risks of severe accidents due to internal events, internal fires, internal floods, and seismic events initiated during plant operational states other than full power operation and to compare the estimated core damage frequencies, important accident sequences and other qualitative and quantitative results with those accidents initiated during full power operation as assessed in NUREG-1150. The scope of the program includes that of a level-3 PRA.

The results of the program are documented in two reports, NUREG/CR-6143 and 6144. The reports are organized as follows:

For Grand Gulf:

NUREG/CR-6143 - Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1

- Volume 1: Summary of Results
- Volume 2: Analysis of Core Damage Frequency from Internal Events for Plant Operational State 5 During a Refueling Outage
 - Part 1: Main Report
 - Part 1A: Sections 1 - 9
 - Part 1B: Section 10
 - Part 1C: Sections 11 - 14
 - Part 2: Internal Events Appendices A to H
 - Part 3: Internal Events Appendices I and J
 - Part 4: Internal Events Appendices K to M
- Volume 3: Analysis of Core Damage Frequency from Internal Fire Events for Plant Operational State 5 During a Refueling Outage
- Volume 4: Analysis of Core Damage Frequency from Internal Flooding Events for Plant Operational State 5 During a Refueling Outage
- Volume 5: Analysis of Core Damage Frequency from Seismic Events for Plant Operational State 5 During a Refueling Outage
- Volume 6: Evaluation of Severe Accident Risks for Plant Operational State 5 During a Refueling Outage
 - Part 1: Main Report
 - Part 2: Supporting MELCOR Calculations

For Surry:

NUREG/CR-6144 - Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry Unit-1

Volume 1: Summary of Results

Volume 2: Analysis of Core Damage Frequency from Internal Events During Mid-loop Operations

Part 1: Main Report

Part 1A: Chapters 1 - 6

Part 1B: Chapters 7 - 12

Part 2: Internal Events Appendices A to D

Part 3: Internal Events Appendix E

Part 3A: Sections E.1 - E.8

Part 3B: Sections E.9 - E.16

Part 4: Internal Events Appendices F to H

Part 5: Internal Events Appendix I

Volume 3: Analysis of Core Damage Frequency from Internal Fires During Mid-loop Operations

Part 1: Main Report

Part 2: Appendices

Volume 4: Analysis of Core Damage Frequency from Internal Floods During Mid-loop Operations

Volume 5: Analysis of Core Damage Frequency from Seismic Events During Mid-loop Operations

Volume 6: Evaluation of Severe Accident Risks During Mid-loop Operations

Part 1: Main Report

Part 2: Appendices

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Acronyms

ADHR	Alternate decay heat removal
ADHRS	Alternate decay heat removal system
APET	Accident progression event tree
ATWS	Anticipated transient without scram
BWR	Boiling water reactor
BWROG	Boiling water reactor owners' group
CCI	Core-concrete interaction
CD	Core damage
CDF	Core damage frequency
COMPBRN	Computer code for compartment fire propagation analysis
DG	Diesel generator
ECCS	Emergency core cooling system
EPRI	Electric Power Research Institute
HPCS	High pressure core spray
IPE	Individual plant examination
IRRAS	Integrated reliability and risk analysis system (computer code)
LHS	Latin hypercube sample
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss-of-coolant accident
LOSP	Loss of offsite power
LP&S	Low power and shutdown
MACCS	MELCOR accident consequence code system (computer code)
MSIV	Main steam isolation valve
NRC	Nuclear Regulatory Commission
PDS	Plant damage state
POS	Plant operational state
PRA	Probabilistic risk assessment
RES	Research (Office of NRC)
RHR	Residual heat removal
SBO	Station blackout
SDC	Shutdown cooling
SPMU	Suppression pool make-up
SRV	Safety relief valve
SSW	Standby service water
TW	Time window

1. Background

Traditionally, probabilistic risk assessments (PRAs) of severe accidents in nuclear power plants have considered initiating events that could occur only during full power operation. Some previous screening analyses that have been performed for other than full-power modes of operation suggested that risks during those modes of operation were small relative to those occurring during full power operation. However, recent studies and operational experiences indicate that the risks of accidents during low power and shutdown (LP&S) may be significant. Although the power of the reactor core is much less in off power conditions than at full power, the technical specifications allow for more equipment to be inoperable in off power conditions. In certain conditions the containment can be open.

In response to the concerns over risk during low power and shutdown conditions, the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (NRC RES) has undertaken a two phase project to analyze the frequencies, consequences, and risk of accidents occurring during modes of operation other than full power.

Phase 1 of the project was completed in September of 1991 [Whitehead et al., 1991]. This phase involved a coarse screening of potential accidents that could occur at a boiling water reactor (BWR) while the reactor was operating at other than full power. The coarse screening approach was adopted as a means of obtaining, in a relatively short time, some estimate of the potential for accidents during low power and shutdown conditions and some idea of the magnitude of the work necessary for a more detailed analysis of these operating states. The BWR examined was the Grand Gulf Nuclear Power Station, a single-unit 1250 MWe (net) BWR 6 power plant with a Mark III containment, located near Port Gibson, Mississippi.

Results from the coarse screening analysis of seven plant operational states (POSS) indicated that to accurately evaluate accidents in low power or shutdown conditions, detailed modeling would be required because the risk during these conditions could not be shown to be insignificant by a screening analysis. (NOTE: Plant operational states are artificial subdivisions of the time plants spend in LP&S conditions. This concept was developed during Phase 1 of the LP&S Project to allow the analysts to better represent the plant as it transitions from power operation to nonpower operation. See Table 1 for a brief description of each POS.) Thus, NRC RES decided to have detailed follow-on analyses performed.

Since a very large effort would be required to accurately address each of the conditions identified in the Phase 1 study in detail, the NRC decided to perform a detailed analysis on one of the off-power conditions.

Table 1 POS Descriptions

POS	Description
1	Vessel pressure from rated conditions to 500 psig and thermal power not greater than 15%; core coolant at any temperature
2	Vessel pressure from rated conditions to 500 psig
3	Vessel pressure from 500 psig to above 100 psig
4	Vessel pressure less than 100 psig and the shutdown cooling system operating
5	Until vessel head is detensioned (all of cold shutdown and the initial part of Operating Condition 5—Refueling)
6	Head off and coolant level raised to the steam lines
7	Head off, upper pool filled, and refueling transfer tube open

An examination of the results of the screening study yielded Figures 1 and 2 [Whitehead et al., 1994a]. As can be seen from Figure 1, approximately 60 percent of the total core damage frequency occurs in POS 5 (consisting mainly of the cold shutdown operating condition). Therefore, from a frequency point of view, POS 5 was the most logical choice for detailed analysis.

However, core damage frequency is not always the most important discriminator for risk. In an attempt to identify the more important sequences from a risk perspective, Figure 2 was constructed. This figure provided a Venn diagram of the sequences classified as having a potentially high frequency with regard to an open containment and early core damage -- important characteristics from the limited plant damage state analysis performed during the screening study. From Figure 2 it can be seen that out of a total of 303 potentially high core damage sequences, 186 had an open containment and core damage was predicted to occur early in the accident. Of the 186 sequences that can potentially have high risk, 178 are from POS 5. This information lent additional support to the choice of POS 5 for detailed analysis.

In addition to the numerical results, engineering insights supported the selection of POS 5 for detailed study for the following reason:

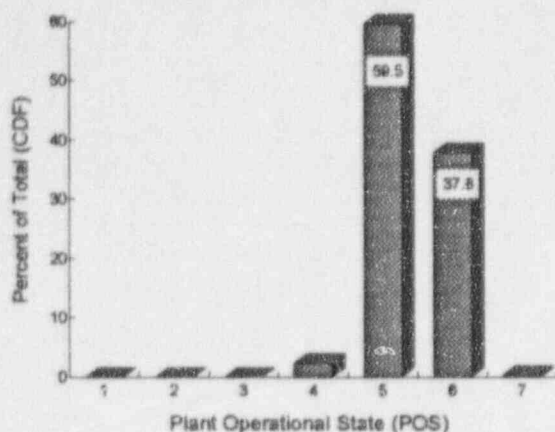


Figure 1 POS vs Percent CDF

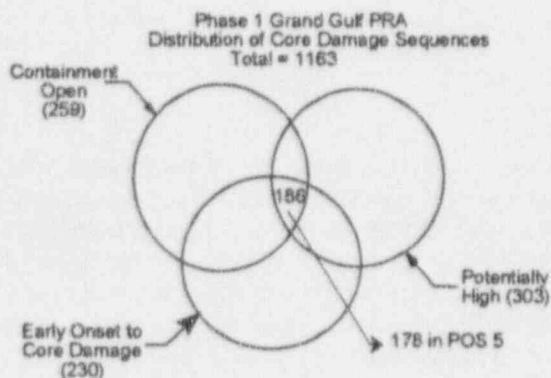


Figure 2 Open Containment and Early Core Damage Sequences in POS 5 with a Potentially High Frequency

- In POSs 1, 2, and 3, the state of the plant is essentially the same as for full power except that the power is lower and pressure/temperature can be lower. Therefore, the initiating events and configuration of mitigating systems are essentially the same as for full power. Since the plant is in these POSs less often than it is at full power, the risk in these POSs is less than at full power, by a factor approximately equal to the fraction of time in these POSs divided by the fraction of time at full power. Based on this rationale, neither POSs 1, 2, nor 3 would be selected for detailed analysis.

- In POSs 6 and 7, the vessel head is off, thus alleviating concerns about the overpressurization of components of decay heat removal systems. Also, in POSs 6 and 7, the water level is raised, thereby providing more time for mitigation of accident-initiating events than in POSs 4 or 5.
- POS 4 and POS 5 both are shutdown states. The plant is in the hot shutdown mode during POS 4, and it is in the cold shutdown mode during POS 5 (except for that part of POS 5 associated with removing the vessel head, for which the plant is in the refueling mode.) The vessel head is on in POS 4, and it is assumed to be on in POS 5. The core is cooled with the shutdown cooling (SDC) system in both POS 4 and POS 5 and with the alternate decay heat removal system in portions of POS 5. These systems are not designed for high-pressure service. If an uncontrolled pressurization transient occurs, failure of the components of low pressure shutdown cooling systems is possible in these POSs, if the systems are not isolated. Such a scenario would lead to an interfacing systems loss of coolant accident (LOCA) outside containment which cannot be mitigated with emergency core cooling systems (ECCS), in the long term, since the suppression pool inventory will be lost through the break. In POS 4, shutdown cooling is carried out by the residual heat removal (RHR) system, which has a pressure rating of 220 psig. In POS 5, shutdown cooling can be provided with either RHR, or with the alternate decay heat removal (ADHR) system (ADHRS) (after 24 hours), which has a pressure rating of 80 psig. The maximum expected decay heat in POSs 4 and 5 is almost identical: 1.0% of full power for POS 4 and 0.9% for POS 5.

When the POS was selected for detailed analysis in August 1991, it was understood that in POS 5 at Grand Gulf, auto-isolation of SDC on high pressure (135 psig) is inactive, while in POS 4 it is active. This understanding was based on information that was received during a plant visit in January 1991.

Isolation of the components for low pressure shutdown cooling during pressurization transients is less likely if the auto-isolation function on high pressure is inactive since operator recognition and intervention would be required to isolate the low pressure components. (Isolation on low level is active in POS 5, as well as in POS 4, thus providing the ability to isolate an interfacing systems LOCA in the shutdown cooling system(s) after the break occurs.) Because of the inoperability of auto-isolation on high pressure in POS 5, and because of the possible use of ADHR during POS 5, POS 5 was chosen for detailed analysis.

This volume of the report summarizes information contained in the documentation of the detailed analyses (see Volumes 2 through 6 of NUREG/CR-6143) performed for the Grand Gulf facility in POS 5 during a refueling outage.

Brookhaven National Laboratory conducted a companion project for the Surry Pressurized Water Reactor during midloop, documented in NUREG/CR-6144.

2. Objectives

The primary objective of this study was to perform a detailed analysis of potential accidents that could occur at Grand Gulf while the plant is in POS 5 during a refueling outage. The initiating events to be examined included: (1) internal initiators - including fire and flood, and (2) seismic initiators.

Specific Level 1 objectives included:

- (1) Compare the results of this study with the results of the full power analysis for Grand Gulf [USNRC, 1989] [Drouin et al., 1989].
- (2) Develop a methodology for performing PRAs for nuclear power plants in conditions other than at full power.
- (3) Provide an analytical tool with which the NRC can evaluate the potential benefits of proposed changes in regulations affecting the required operability of equipment when a plant is in a condition other than full power.

Specific Level 2/3 objectives included:

- (1) Perform a characterization of the accident progressions following core damage resulting from traditional internal events (excludes fire, flood, and seismically induced sequences), and estimate the consequences that result from these accidents.
- (2) Quantitatively determine the risk and estimate the uncertainty for the risk-significant mode of operation in POS 5.
- (3) Compare the risk associated with POS 5 during a refueling outage with the risk associated with full power operation.
- (4) Assess the potential for a radioactive release to cause onsite consequences.

3. Approach and Limitations

3.1 Level 1

The approach used was a modification of a standard Level 1 PRA approach. Event trees were constructed, top events were modeled using fault trees of various complexities, and the top events were quantified using point estimates to produce the sequence frequencies. These sequences were then examined for recovery potential and validity. For those sequences where recovery was applicable, appropriate recovery actions were incorporated. The sequences that survived the recovery analysis were then reexamined with a "time window" analysis approach. In this analysis, the surviving sequence cut sets were requantified based on their contribution to three distinct time regimes for POS 5 (i.e., entry into POS 5 to 24 hours, 24 hours to entry into POS 6, and POS 5 after core alterations).

The fault trees from the NUREG-1150 full power PRA for Grand Gulf [Drouin et al., 1989] were utilized wherever possible. The IRRAS computer code [Russell et al., 1992] was used in the construction of the event trees, modification of existing fault trees and construction of new trees, and the quantification of the accident sequences. In addition, information contained in the NUREG-1150 analysis of the Grand Gulf plant was used wherever applicable.

The IRRAS code was used to quantify frequencies for accident sequences leading to core damage. Sequences having point estimate frequencies below the Phase 2 truncation limit were considered to be noncontributors to the overall core damage frequency and were discarded. An uncertainty analysis was performed for all sequences surviving the time window screening analysis.

3.1.1 Traditional Internal Events

In comparison with the full power PRA, the event trees for POS 5 are more complex and lengthy:

- (1) Event trees for 34 initiating events were developed,
- (2) More than 165 transfer event trees were developed,
- (3) More than 110 operator action/decision points were included in the event trees,
- (4) Each transfer tree generally contained from 10 to 100 outcomes (i.e., transfer, core damage, or no core damage).

This event tree complexity is due to the relatively low decay heat in cold shutdown, resulting in a large number of ways by which cooling can be provided to the core if the coolant is

initially in a subcooled state. Also, the availability and configuration of plant systems in POS 5, compared with full power, are more complex to specify owing to the less stringent requirements on operability imposed by the technical specifications.

The methodology used in this study is the small event tree/large fault tree technique. In practical applications, this technique assumes a fixed initial plant state prior to an accident-initiating event. Through the use of seven analysis "rules" (or a set of assumptions) we were able to consider numerous different conditions that can, and in fact do, exist at shutdown before an accident-initiating event occurs.

Test and maintenance-induced loss of coolant accidents were not addressed in this study. Development of a detailed methodology for analyzing human actions during shutdown conditions is under way, and analysis of such events is deferred until this improved methodology is available.

3.1.2 Internal Fire Events

This assessment has made full use of insights gained during the past 15 years in assessing fire risk. The methodology utilized previously completed traditional internal event fault and event tree models. Thus, the level of detail of the fire analysis is consistent with the level of detail of the traditional internal events analysis. A three-step overview of the methodology used during this project is given below.

Step 1: Initial Plant Visit

The general location of safety-related components of the systems of interest was known from initial location analyses. The plant visit allowed the analyst to verify the physical arrangements in each of these areas. The analyst completed a fire zone checklist which aided in the screening analysis and in the quantification of risk.

The second purpose of the initial plant visit was to confirm with plant personnel that the documentation being used is in fact the best available information, and to get clarification about any questions that might have arisen in a review of the documentation.

Also, fire-fighting procedures were thoroughly reviewed to determine the probability of manual suppression in any given time for all critical plant areas.

Step 2: Screening

It was necessary to select those fire locations within the power plant that have the greatest potential for producing risk-dominant accident sequences. The objectives of location selection are somewhat competing and should be balanced for a meaningful risk assessment. The first objective is to

maximize the possibility that all important locations are analyzed, and this leads to the consideration of a potentially large number of candidate locations. The second objective is to minimize the effort spent quantifying event trees and fault trees for fire locations that turn out to be unimportant. A proper balance of these objectives is one that results in an ideal allocation of analytical resources and efficient assessment.

The screening analysis consisted of the following steps:

1. Potentially important fire areas were identified. Areas which had either safety-related equipment or power and control cables for that equipment were identified as requiring further analysis.
2. Fire areas were screened for probable fire-induced initiating events.
3. Fire areas were screened both on order and frequency of cut sets.
4. Each remaining fire area was numerically evaluated and truncated on frequency.

Step 3: Final Quantification

After the screening analysis had eliminated all but the probabilistically significant fire areas, dominant cut sets were quantified as follows:

1. The temperature response in each fire area for each postulated fire was determined.
2. Fire fragilities were computed. The latest version of the COMPBRN fire growth code [V. Ho et al., 1991] was used to calculate fire propagation and equipment damage.
3. The probabilities of barrier failure for all remaining combinations of adjacent fire areas were assessed. A barrier failure analysis was conducted for those combinations of two adjacent fire areas which, with or without additional random failures, remained after the screening analysis.
4. An initial recovery analysis was performed. In a fashion similar to the traditional internal events analysis, recovery of nonfire-related random failures was addressed. Appropriate modifications to recovery probabilities were made as necessary to account for fire conditions.
5. A time window analysis and an uncertainty analysis were performed on any sequence surviving the initial recovery analysis.

3.1.3 Internal Flooding Events

The analysis was performed in three steps:

1. The purpose of this step was to identify potential flood zones, flood sources within each zone, and equipment in each zone. Equipment whose failure could have safety implications for the plant, and the susceptibility of this equipment to failure caused by a flood in its location were also determined. The potential water inventory released from each source was also quantified.
2. The purpose of this step was to develop flood scenarios, determine which of these might lead to core damage, and characterize the frequency of flood-initiating events. The number of scenarios was then reduced to a size more amenable to analysis by considering which scenarios would threaten safety-related equipment and, where appropriate, combining some scenarios into one.
3. The purpose of this step was to develop and/or adapt appropriate fault and event tree models, quantify operator actions, and perform sequence frequency quantifications and uncertainty analyses.

Quantification of accident sequences followed the same approach used in the traditional internal events analysis. Several of the event trees from that work were applicable to the flooding analysis. In these cases, the frequencies of initiating event were modified to correspond to flood frequencies; system and component failures based on flood volume and location were included; and operator actions were modified as appropriate. For cases in which no analogous event trees from previous work were available, new trees were developed.

3.1.4 Seismic Events

The seismic analysis was limited to work analogous to a Level 1 seismic PRA, in which estimates have been developed for core-damage frequency from seismic events during POS 5 for a refueling outage. The methodology is almost identical to that used for full-power seismic PRAs, as widely practiced in the nuclear industry. However, seismically-induced relay chatter was beyond the scope of this analysis. Seismic hazard curves from both the Electric Power Research Institute (EPRI) [EPRI, 1989] and the Lawrence Livermore National Laboratory (LLNL) [Sobel, 1993] were used.

The modeling assumptions for the systems, the non-seismic failure rates for components, the human error rates, and the same quantification techniques that were used in the

traditional internal events analysis were used, so that the results of the two analyses would be as comparable as possible.

3.2 Level 2/3

The risk associated with POS 5 was determined in the Level 2 and 3 portions of the PRA using a simplified form of the NUREG-1150 methodology [USNRC, 1990]. The Level 2/3 portion of the PRA is concerned with the progression of postulated accidents following the onset of severe core damage and the estimation of the consequences that result from the release of any radioactive material. As such, it consists of the following constituent analyses: plant damage state (PDS) analysis, accident progression analysis, source term analysis, consequence analysis, and risk analysis. A brief summary of the approach used in each of the constituent analyses is provided below.

Plant Damage State Analysis: PDSs were developed to define the interface between the accident frequency analysis (Level 1) and the accident progression analysis (Level 2). Core damage accidents that have similar plant and system configurations at the onset of core damage are grouped together; each group is called a plant damage state.

Accident Progression Analysis: Based on the configuration of the plant defined by the PDSs, event tree techniques were used to delineate the accident progressions following the onset of core damage. The accident progressions define the status of the containment and other features of the plant that are used to mitigate the accident during the various phases of the accident; they also identify phenomena that may impact the release of radioactive material. The accident progression event tree (APET) developed in this study is similar in concept to the APETs developed in NUREG-1150; however, it is not as detailed. Compared to the NUREG-1150 APETs, the POS 5 APET included fewer questions (i.e., top events), addressed issues in less detail, and did not use formal expert judgment procedures to quantify the APET.

Source Term Analysis: Source terms, which characterize the type and amount of radioactive material releases from the plant, were estimated for accident progression groups using the parametric approach developed in NUREG-1150 [Jow et al., 1993]. The parametric expression was quantified, to the extent possible, using information from the NUREG-1150 full power analysis of Grand Gulf [Harper et al., 1992]. The source terms were then combined into a manageable number of source term groups using a partitioning algorithm first developed in the NUREG-1150 study [Iman et al., 1990] and then modified in the full power study of the LaSalle plant [Brown et al., 1992].

Consequence Analysis: Offsite consequences were estimated for each source term group using the MACCS code. The emergency response assumption used in this study are the same as those used in the NUREG-1150 Grand Gulf plant analysis [USNRC, 1990; Brown et al., 1990]. In addition to offsite consequences, this study also included a scoping analysis of onsite consequences.

Risk Analysis: The risk results reported in this study are estimates of aggregate risk, which is the sum over all accident scenarios of the product of the accident frequency with its consequence. The aggregate risk results calculated in this study account for the amount of time, on average, that the plant is in POS 5 during a typical calendar year (i.e., the plant is in POS 5 for only a small fraction of the year—approximately 3%). The risk calculated in this study is not the risk attributable to one year of operation in POS 5. All risk results presented in this report are on a per calendar year basis.

A limited uncertainty analysis (which included variables from the PDS, accident progression, and source term analyses) was also performed. In contrast to NUREG-1150, formal expert opinion techniques were not used in this study to quantify the accident progression and source term models. Where appropriate, however, distributions developed in NUREG-1150 were used in this study. For events that could not be quantified using existing distributions, new distributions were developed by the project staff.

To analyze the potential accidents that can occur during POS 5, it was necessary to divide POS 5 into three distinct time regimes. These regimes [now called *time windows* (TWs)] are: (1) from entry into POS 5 to 24 hours after shutdown, (2) from 24 hours after shutdown to entry into POS 6 (POS 6 begins approximately 94 hours after shutdown and roughly corresponds to the refueling mode of operation), and (3) POS 5 again after core alterations (this last time regime starts approximately 40 days after shutdown and lasts for approximately 10.4 days). For each time window the appropriate core power and radionuclide inventory was used to estimate the timing of the accident and its potential consequences.

4. Results

4.1 Level 1 Results

4.1.1 Quantitative Results from Traditional Internal Events Analyses

4.1.1.1 Results from Sequence Quantification for Traditional Internal Events Analyses

The total core damage frequency (CDF) presented here results from combining the mean CDFs from all 38 sequence

cut sets for the 28 sequences that survived the sequence analysis through the time window analysis. For POS 5 during a refueling outage at Grand Gulf, the sum of the mean CDFs from the surviving sequences is $2.1\text{E-}6$ per calendar year for internally initiated events (excluding internal fires and floods).

Two classes of initiating events dominate the results from this study. As can be seen below, LOCA/Diversion and loss of offsite power (LOSP)/Blackout constitute approximately 95% of the total mean core damage frequency.

IE Class	Mean CDF	% Contribution To Mean CDF
LOCA/Diversion	1.3E-06	62
LOSP/Blackout	7.0E-07	33
Other	9.9E-08	5
Total	2.1E-06	100

Figure 3 shows the contributions of the various initiating events to core damage frequency. Two types of accident sequences are among the dominant sequences in the important initiating events. They are:

- Blackout - Initiated by a LOSP, a subsequent loss of all onsite ac power either by loss of the diesel generators (DGs) directly or indirectly-- by the loss of some DG support system, and the failure to restore either offsite or onsite ac power before core damage occurs; and
- Flooding Containment - Initiated by an event requiring the injection of water into the vessel, out the safety relief valves (SRVs) to the suppression pool, and finally out the open lower containment personnel lock due to the failure of the operators to either close the lower personnel lock or to control the injection of the water into the vessel. The resulting flood is assumed to fail equipment necessary for the prevention of core damage.

From a core damage frequency vs time window aspect, time window 2 is the most important. Figure 4 indicates that time window 2 contributes 58 percent of the total core damage frequency.

Another way to present the core damage frequency information is to plot the fractional contribution of each initiator group by time window. This results in Figure 5. From this figure it can be seen that for:

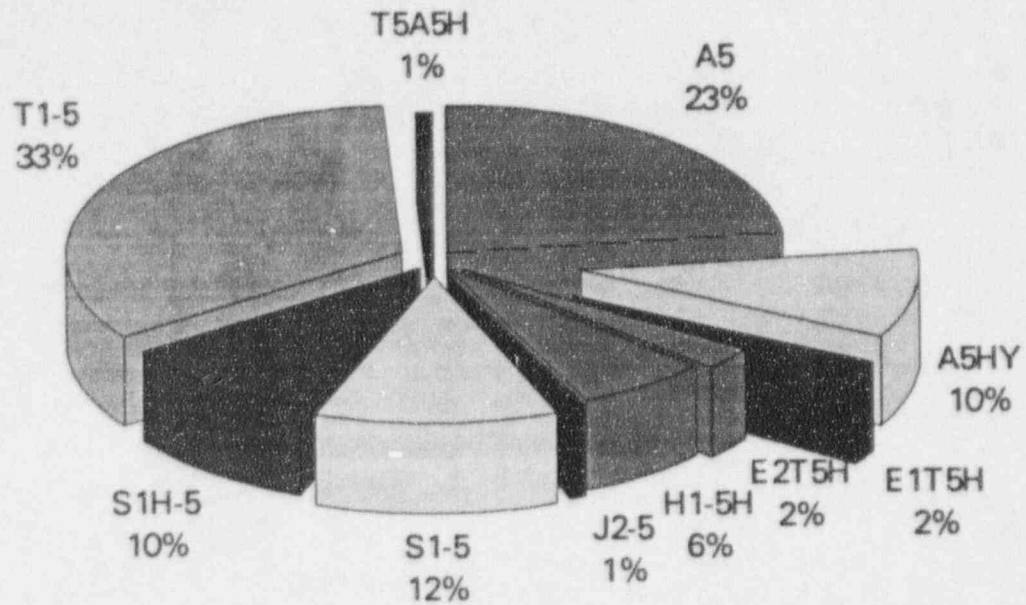


Figure 3 Contribution to CDF by Initiating Event

A5	-	Large LOCA during nonhydro conditions
A5HY	-	Large LOCA during hydro conditions
E1T5H	-	Isolation of shutdown cooling common suction line
E2T5H	-	Loss of shutdown cooling common suction line
H1-5H	-	Diversion to the suppression pool via the residual heat removal system
J2-5	-	LOCA in the residual heat removal system
S1-5	-	Intermediate LOCA during nonhydro conditions
S1H-5	-	Intermediate LOCA during hydro conditions
T1-5	-	Loss of offsite power
T5A5H	-	Loss of standby service water system

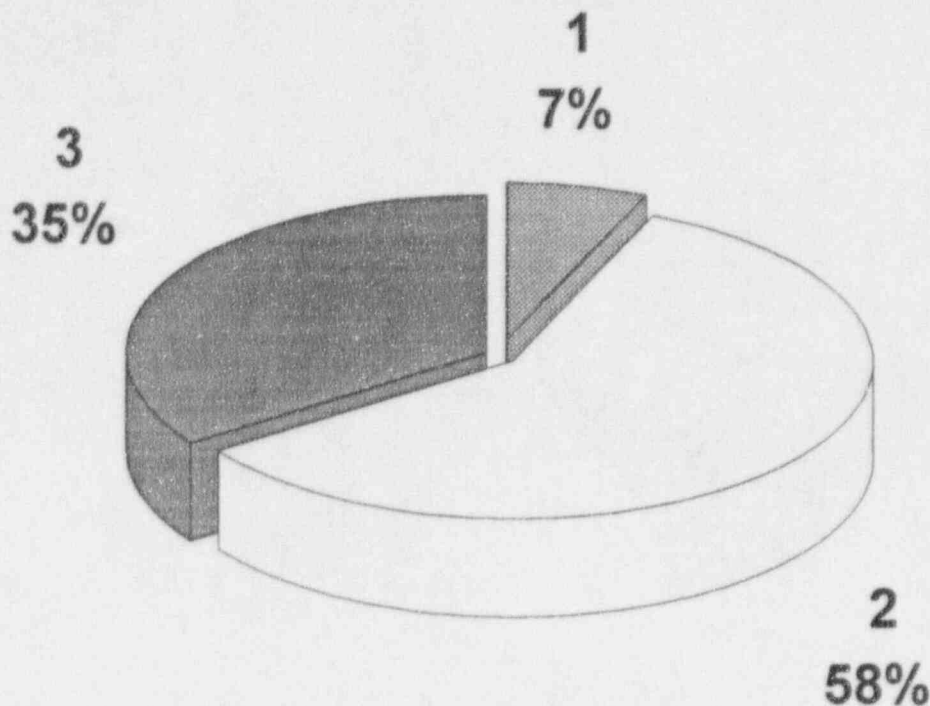


Figure 4 Percent of CDF vs Time Window

Time Window 1

The core damage frequency is split between the LOCA/Diversion and the LOSP/Blackout groups (42% and 58% respectively).

Time Window 2

The core damage frequency is split among the three groups (41% - LOCA/Diversion, 50% - LOSP/Blackout, and 9% - Other)

Time Window 3

All core damage frequency results from the LOCA/Diversion group.

One final way to present the core damage frequency information is to plot the percent contribution to the total core damage frequency and the percent of time spent in each time window vs the three time windows on the same graph. From Figure 6 it can be seen that even though the plant spends only 21 percent of the time in time window 2, this window contributes 58 percent to the total core damage frequency.

Figure 6 also indicates that time window 3 contributes 35 percent of the total core damage frequency, yet 76 percent of the time is spent in this window.

Thus, from Figures 5 and 6 we see that time window 2 is the most important time regime for POS 5 during a refueling outage.

4.1.1.2 Total Plant Model Results for Traditional Internal Events Analyses

The CDF results from the uncertainty analysis of the traditional internal events total plant model (i.e., an uncertainty analysis of all of the sequence cut sets at the same time) using 1000 samples are as follows (per calendar year):

Mean Value	2.0E-6
5th Percentile Value	4.1E-7
Median Value	1.3E-6
95th Percentile Value	5.4E-6

Comparing the results of this study with those obtained in the Grand Gulf Individual Plant Examination (IPE), we find that the mean CDF from the total plant model obtained in this study is almost an order of magnitude less than the IPE result

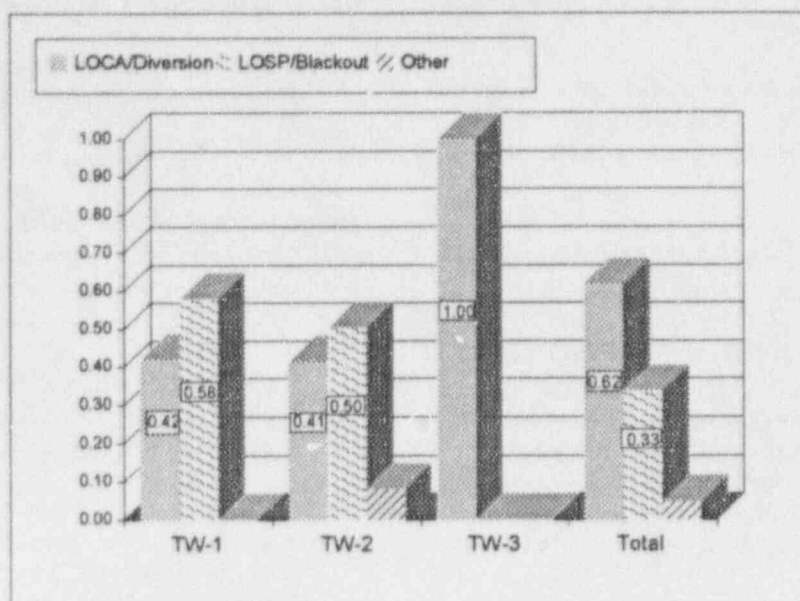


Figure 5 Fractional Contribution to CDF by IE Group vs Time Window

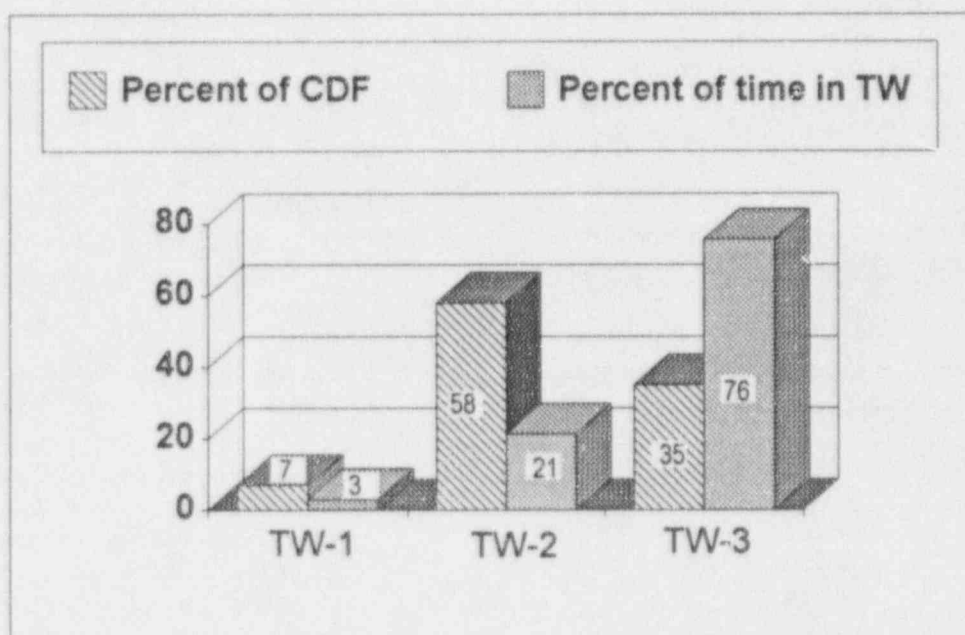


Figure 6 Percent of CDF and Percent of Time in Time Window vs Time Window

of $1.7\text{E-}5$ per reactor year. See Section 5 for a comparison with the NUREG/CR-4550 study.

In addition, the results from this study indicate that, unlike the NUREG/CR-4550 results, sequences other than those initiated by LOSP (e.g., LOCAs) contribute significantly to the core damage frequency.

4.1.2 Quantitative Results from Internal Fire Events Analyses

A detailed screening analysis was performed which showed most plant areas had a negligible contribution to the frequency of fire-induced core damage. A detailed fire propagation analysis was performed for four fire zones. There were no plant areas which were found to have a contribution to core damage frequency greater than the truncation limit of $1\text{E-}8$; thus, no fire sequences had a CDF greater than $1\text{E-}8$.

4.1.3 Quantitative Results from Internal Flooding Events Analyses

A single sequence survived through the time window analysis. This sequence is initiated by a break in a fire water system pipe. The resulting flood from this initiator disables Divisions 1, 2, and 3 Class 1E ac and dc power. Given the severity of this postulated accident sequence, no operator recovery was postulated. The mean core damage frequency for this sequence is $2.3\text{E-}8$ per calendar year. The 5th and 95th percentiles are $8.2\text{E-}11$ and $8.6\text{E-}6$ per calendar year, respectively.

4.1.4 Quantitative Results from Seismic Events Analyses

The CDF results of the seismic analyses for earthquake-initiated accidents during POS 5 for a refueling outage are as follows (per calendar year):

For the LLNL (1993) Hazard Curves

5th percentile	$2.1\text{E-}11$
Median	$2.4\text{E-}9$
Mean	$7.1\text{E-}8$
95th percentile	$2.2\text{E-}7$

For the EPRI Hazard Curves

5th percentile	$2.5\text{E-}12$
Median	$2.0\text{E-}10$
Mean	$2.5\text{E-}9$
95th percentile	$1.1\text{E-}8$

4.1.5 Qualitative Results

4.1.5.1 Plant Systems and Operations Insights from Traditional Internal Events Analyses

4.1.5.1.1 Systems Insights

Characteristics of the plant design are a major factor affecting the likelihood of core damage while in cold shutdown. For Grand Gulf, the following plant characteristics are most important:

1. Shutdown cooling system components are not rated for full pressure, but automatic isolation occurs on either high pressure or on low level;
2. Use of the residual heat removal system for shutdown cooling requires recirculation, either forced or natural, to prevent pressurization transients;
3. Due to density and pump head effects, recirculation is sensitive to actual level in the core region. The water level in the core region is related to but not equal to measured level in the downcomer;
4. At decay heat levels of concern, flooding induced dryout of the core (i.e., a steam flow sufficient to prevent water from cooling the core) at atmospheric pressure will not occur, and the core can be cooled by steaming with a maximum of 250 gpm makeup;
5. To steam at low pressure, opening of one safety relief valve in relief mode is sufficient to maintain pressure low enough that the low head pumps in the emergency cooling system can provide sufficient makeup;
6. Opening of one safety relief valve in relief requires operator action, dc power, and air;
7. In using the emergency core cooling system in a water solid mode, opening of two safety relief valves in the relief mode prevents overpressurizing the shutdown cooling system components, both in the residual heat removal system and in the ADHRS, regardless of the pump(s) used;
8. In using the emergency core cooling system in a water solid mode, opening of one safety relief valve in the relief mode prevents overpressurizing the components in the residual heat removal system used in shutdown cooling, but components in the auxiliary decay heat removal system may overpressurize;

9. Isolation of the shutdown cooling system allows the core to be cooled at full pressure by steaming on one safety relief valve at its safety setpoint, and no operator action or support systems are required to operate the valve in the safety mode;
10. Use of emergency core cooling systems in a water solid mode does not require suppression pool makeup, in the short term, to compensate for vessel fill;
11. Water can be injected into the vessel at low pressure from both service water and diesel-driven firewater pumps.

4.1.5.1.2 Operations Insights

In POS 5 (i.e., cold shutdown), the requirements of the technical specifications for the operability of systems and components are much less stringent than for power operation. The actual availability of systems depends on plant-specific practices, and on the reason for transitioning the plant to cold shutdown, in this case—a refueling outage.

For Grand Gulf, the following practices have an important impact on the ability to cool the core in POS 5:

1. At least two safety relief valves are maintained operable for both relief and safety operation;
2. Automatic isolation of the low-pressure shutdown cooling system is not bypassed, but is maintained on both high pressure and low level;
3. Some subsystems of the emergency core cooling system are available most of the time.

4.1.5.2 Insights from Internal Fire Events Analyses

The fire-induced core damage frequency is lower than full-power fire risk assessments for a number of reasons. First, the plant is in this POS only 3 percent of the time in any given year, so even if all other factors remained the same with full power, one would expect the fire-induced CDF for POS 5 to be lower. Second, the shutdown fire frequencies are somewhat lower than those at power. Third, even if active electromechanical safety-related equipment is damaged by fire, an initiating event may not necessarily occur. For instance, for the loss of TBCW (turbine building cooling water) initiator to result from fire-related damage, multiple operational pumps must fail. These pumps and their associated cabling have sufficient separation to make it highly unlikely that a single fire could lead to failure of all pumps. Thus, many initiating events at shutdown were eliminated because of the physical separation criteria of the screening process. Even for the unscreened initiating events,

very few fire zones were found to be applicable because of physical separation criteria. Also, relative to other plants, Grand Gulf utilizes more automatic fire protection systems in critical safety-related areas, which in turn reduces the probability of damage from a fire. Therefore, after taking into account the physical separation of safety-related functions, automatic fire protection systems, lower frequencies of fire initiated events, and manual fire suppression, most initiating events at shutdown and many fire zones were eliminated from further analysis.

A detailed fire propagation analysis was performed for the remaining initiators and respective fire zones. It was found that only in very limited areas could fire damage result in both the initiating event and other fire-related failures that were necessary for core damage. Even in these situations, other random failures (nonfire-related) were also necessary before core damage occurred. Therefore, when taking into account the reduction in fire frequency due to the limited area of influence and other random failures which were required before core damage, all remaining fire scenarios were found to be less than the truncation limit (i.e., less than $1\text{E-}8$ per calendar year).

In all areas, additional random failures of equipment (damage not related to the fire itself) had to occur in order to obtain core damage. Adequate separation of equipment (and/or) cabling between redundant functions and the presence of automatic fire suppression systems reduced core damage frequency for those areas.

4.1.5.3 Insights from Internal Flooding Events Analyses

The overall conclusion of this work is that internal floods do not pose a significant core damage threat to the Grand Gulf Nuclear Station for POS 5 during a refueling outage.

The core damage frequency of $2.3\text{E-}8$ per calendar year due to internal flood events is approximately two orders of magnitude lower than the core damage frequency of $2.0\text{E-}6$ for traditional internal events. Thus, internal flooding would make only a minor contribution to the total core damage frequency at Grand Gulf during POS 5. This is principally because of the low frequency of fluid boundary component breaks that could result in a flood and a separation of systems that would be available to mitigate the effects of such an accident.

The two conservative assumptions affecting flow rates and flood volumes included in these analyses (i.e., fully guillotined catastrophic breaks and full hour undetected breaks) did not significantly affect the results of this study. For completeness, it should be noted that the assumed undetected break time for the single surviving sequence was 15 minutes. This time, while a departure from the 1-hour assumption, was sufficient to cause a loss of all Class 1E ac

and dc power, and probably represents a more realistic estimate of the undetected break time for POS 5 during a refueling outage.

4.1.5.4 Insights from Seismic Events Analyses

The mean core damage frequency of $7.1\text{E-}8$ per calendar year (i.e., the maximum estimate obtained by using the LLNL (1993) hazard curves) is also low relative to the $2.0\text{E-}6$ frequency for traditional internal initiators. Two reasons for this are

1. Grand Gulf's seismic capacity in responding to earthquakes during shutdown is excellent, well above its design basis.
2. The Grand Gulf site enjoys one of the least seismically active locations in the United States.

4.2 Level 2/3 Results

4.2.1 Core Damage Frequency

For discussion purposes, the core damage scenarios identified in the Level 1 analysis can be combined into the following three PDS groups (12 PDSs were actually evaluated in the accident progression analysis): loss of coolant accidents, Station Blackouts (SBOs), and Other Transients. The total core damage frequency and the fractional contributions to the core damage frequency for these three groups are provided in Table 2. The LOCA PDS group is the dominant contributor to the core damage frequency, followed by the SBO PDS group and the Other Transients PDS group.

4.2.2 Accident Progression

A simplified representation of the APET that addresses the major aspects of the accident is shown in Figure 7. (The actual APET included 59 top events or questions). Figure 7 combines the results from all the accidents and is conditional on the occurrence of core damage; the values displayed are mean conditional probabilities. From the simplified tree presented in Figure 7, it can be seen that in the most likely accidents in POS 5 the containment is open, the suppression pool is bypassed, and the vessel fails. For the cases where the vessel fails, there is a significant probability that the core debris will either be quenched in a flooded cavity or the interactions between the core debris and the concrete structures beneath the vessel, the core-concrete interaction (CCI), will occur in a flooded cavity. For the cases where the vessel fails, there is a significant probability that the core debris will either be quenched in a flooded cavity or the interactions between the core debris and the concrete structures beneath the vessel, the core-concrete interaction,

will occur in a flooded cavity. For the former, the releases associated with CCI are prevented. In the latter case, the radioactive releases are scrubbed by the water in the flooded cavity, which helps reduce the source term to the environment. If the containment is closed prior to core damage, it is predicted to either fail or to be vented after core damage because containment heat removal is not available in these accidents. Venting the containment late in the accident is the most likely scenario. For the accidents identified in POS 5, the containment sprays were never available after the onset of core damage.

4.2.3 Aggregate Risk

Table 3 presents the offsite risk results for the following six measures: early fatalities, total latent cancer fatalities, population dose within 50 miles of the site, population dose within 1000 miles of the site, average individual early fatality risk within 1 mile of the site, and average individual latent cancer risk within 10 miles of the site.

Many factors can affect the magnitude and severity of the release and in turn affect risk. Factors associated with POS 5 accidents that tend to increase risk include the following:

- In many of the accidents the containment equipment hatch was open during the entire accident. An open equipment hatch provides a path for radionuclides to escape from the containment to the auxiliary building and then out into the environment.
- Two plant features that can be used to attenuate the release of radioactive aerosols are the suppression pool and the containment sprays. In both the LOCA and the SBO PDSs, the radioactive material released from the damaged fuel bypassed the suppression pool. The containment sprays were not available in any of the POS 5 accidents.
- In many of the accidents, core cooling was not restored early in the accident, thus precluding any possibility of arresting the core damage process before vessel failure. When the vessel fails, the core debris in the vessel is released into the reactor cavity, allowing for possible CCIs. Significant amounts of radioactive material can be released during this ex-vessel phase of the accident.

A number of factors associated with these POS 5 accidents also tend to decrease risk. These factors are listed below:

- Although in many of the accidents the containment equipment hatch is open, the suppression pool is bypassed, and the containment sprays are unavailable, the releases pass through the auxiliary

Table 2 Core Damage Frequency for POS 5 and Fractional Contributions to the Core Damage Frequency for the LOCA, SBO, and Other Transients Plant Damage State Groups

Plant Damage State Groups	Descriptive Statistics*			
	5th	Percentiles 50th	95th	Mean Standard Deviation
Total	4.1E-07	Total Core Damage Frequency (per calendar year)		
		1.4E-06	5.6E-06	2.1E-06 2.7E-06
		Fractional Contribution to Core Damage Frequency		
LOCA	0.10	0.50	0.93	0.51 0.27
SBO	0.03	0.24	0.80	0.33 0.26
Other	0.01	0.09	0.58	0.17 0.18

* Statistics based on a Latin hypercube sampling (LHS) sample size of 200 observations.

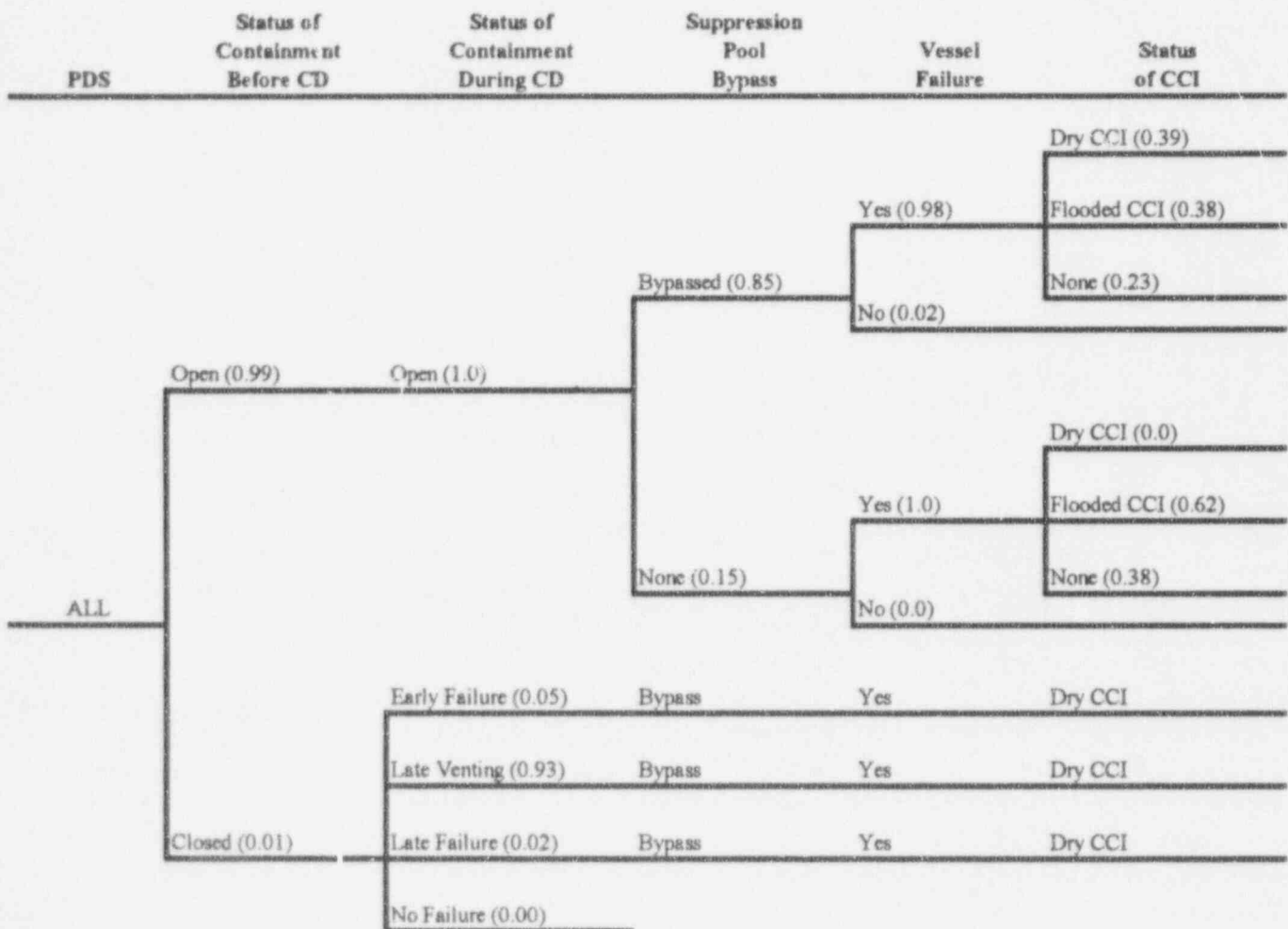


Figure 7 Simplified Representation of POS 5 Accident Progressions

Table 3 Distributions for Aggregated Risk for POS 5
(all values are per calendar year; population doses are in person-rem)

Consequence Measures	Descriptive Statistics*				
	Percentiles			Mean	Standard Deviation
	5th	50th	95th		
Early Fatality Risk	3.7E-11	2.8E-09	3.9E-08	1.4E-08	5.4E-08
Total Latent Cancer Risk	4.3E-04	1.9E-03	1.2E-02	3.8E-03	7.7E-03
Population Dose within 50 miles of the plant	1.3E-01	5.3E-01	3.1E+00	9.9E-01	1.9E+00
Population Dose within 1000 miles of the plant	9.9E-01	4.4E+00	2.8E+01	8.7E+00	1.8E+01
Individual Early Fatality Risk-- 0 to 1 mile	4.2E-13	2.7E-11	3.0E-10	9.6E-11	3.4E-10
Individual Latent Cancer Risk-- 0 to 10 miles	2.5E-10	9.4E-10	4.9E-09	1.6E-09	2.4E-09

* Statistics are based on a LHS sample of 200 observations.

building before escaping into the environment. Because of its large volume and surface area, the auxiliary building provides a location for the radionuclides to be attenuated by deposition and thereby reduce the source term to the environment.

- The accidents delineated for these shutdown conditions progress sufficiently slowly that there is typically a considerable amount of time available for the public to respond to the accident and evacuate before exposure to the release. This is primarily important for the early health effects consequence measures, which are more strongly affected by the time available for evacuation.
- Radioactive decay has reduced the radioactive potential of these shutdown accidents relative to the inventory that is present immediately after the reactor is shut down. This factor is primarily important for early health effects, which are more strongly affected by the shorter lived radionuclides. This effect is much less noticeable for latent health effects, which are more strongly affected by the longer lived isotopes.
- The population around the Grand Gulf plant is relatively low. Although many factors influence the magnitude of the consequences, in general, for a given release, a smaller population correlates with a smaller number of fatalities. Of the four Mark III plants in the United States, Grand Gulf has the fewest number of people living within 50 miles of the plant, according to the 1990 census data. The Mark III plant with the greatest number of people living within 50 miles of the site has a population that is more than an order of magnitude greater than the Grand Gulf 50 mile population.

Table 4 provides the fractional contributions to the early fatality risk and the total latent cancer risk for the following three PDS groups: LOCAs, SBOs, and Other Transients. The fractional contributions to the population dose risk measures (not shown in Table 4 for brevity) are similar to the fractional contributions to the total latent cancer risk measure. From Table 4 it can be seen that, on average, the SBO PDS group is the dominant contributor to the total early fatality risk.

Because a large amount of overlap exists among the three distributions, as is evident from the descriptive statistics provided in Table 4, on any given observation (an observation is one particular trial in the many trials made in a Monte Carlo type analysis) the contribution from the three groups can vary. That is, for one observation the SBO group may be dominant, whereas for another observation the LOCA group may be the dominant group. On average, however, the SBO is the dominant contributor. The SBO PDS group's large contribution to early fatality risk can be attributed to its relatively high contribution to the core damage frequency coupled with the fact that the containment equipment hatch is open, the suppression pool is bypassed, and the auxiliary building fails early in these accidents. Combined, these factors cause the SBOs to have relatively high risk values.

The LOCA PDS group, however, is not a dominant contributor to early fatality risk even though it is a dominant contributor to the core damage frequency. This situation occurs primarily because the dominant contributors to the LOCA core damage frequency are LOCA accidents that are initiated while the plant is in time window 3 (i.e., PDS3-1). Numerous factors can potentially reduce the number of early fatalities that occur when the accident is initiated in time window 3 relative to the other time windows. These factors include the following conditions: (1) Radioactive decay has reduced the inventory of short-lived radionuclides that are

Table 4
Fractional Contributions to Aggregate Risk for the LOCA, SBO, and Other Transients Plant Damage State Groups

Plant Damage State Groups	Descriptive Statistics*				
	Percentiles			Mean	Standard Deviation
	5th	50th	95th		
	Fractional Contribution to Early Fatality Risk				
LOCA	0.001	0.04	0.72	0.16	0.24
SBO	0.08	0.87	1.00	0.73	0.30
Other	0.001	0.04	0.61	0.12	0.18
	Fractional Contribution to Total Latent Cancer Fatality Risk				
LOCA	0.04	0.38	0.90	0.42	0.27
SBO	0.04	0.41	0.90	0.45	0.28
Other	0.01	0.06	0.55	0.13	0.17

* Statistics are based on a LHS sample of 200 observations.

important to early health effects. (2) Because of the lower decay heat the accidents progress more slowly, allowing more time for the population to evacuate. (3) The release is spread out over a longer time which, helps reduce the concentration of radionuclides in the environment. For these reasons time window 3 is a negligible contributor to early fatality risk.

For latent cancer health effects, the LOCA and SBO PDS groups are, on average, the dominant contributors to risk. Because the radionuclides that are important to the latent health effects tend to have long half lives, these risk measures are not particularly sensitive to the time of accident occurrence relative to shutdown. Latent cancers primarily depend on the total amount of radioactive material released, not on the time it was released (i.e., early in the accident versus late in the accident). Because latent cancers are not strongly dependent on the timing characteristics of the accident (i.e., start of release or release duration), the latent cancer risk will depend on the likelihood of the accident and on the total amount of radioactive material released. In all of the core damage accidents delineated in this study, the containment is either open at the start of the accident or fails during the accident, and in most of the accidents the core damage process is not arrested in the vessel. Thus, although the timing of the accident may vary, when the uncertainty in the source term is considered, all the accidents will result in roughly similar releases of radioactive material to the environment. Thus, as can be seen in Tables 2 and 4, the mean fractional contribution to latent cancer risks tends to be roughly similar to the mean fractional contribution to the core damage frequency for each of the PDS groups. The fractional contributions from the LOCA and Other Transient groups tend to be less than their fractional contributions to the core damage frequency because for these PDSs portions

of the release are scrubbed by either the suppression pool or the pool formed by flooding the containment. The fractional contribution from the SBO PDS group tends to be greater than the fractional contribution to the core damage frequency because for these accidents the containment is open at the start of the accident, the auxiliary building fails early in the accident, the vessel nearly always fails, CCI nearly always occurs, and the releases are rarely scrubbed by water. Therefore the releases associated with the SBO tend to be large relative to the other accidents analyzed in this study.

See Section 5 for a comparison of the results of this study with those from NUREG/CR-4551.

4.2.4 Qualitative Issues and Cautions

The results presented here for the Level 2/3 analysis are for a single POS (namely POS 5) and, as such, only assess the risk associated with this POS. While the Phase 1 Screening Study and other qualitative insights suggest that POS 5 is the risk-dominant mode of shutdown, no detailed study has been performed on the other POSs to confirm this conclusion.

Only accidents initiated from traditional internal events were analyzed in this study. Hence, the risk calculated for POS 5 is not complete in the sense that it does not include accidents initiated by internal fires and floods; it also does not include accidents initiated by seismic events.

It is important to realize that by changing the risk in one POS, for example by changing when equipment is available and unavailable, can shift the risk to another POS. Since this study only addresses the risk associated with one POS, the effect of such a change on overall risk (i.e., risk across all the POSs) cannot currently be quantitatively assessed.

Since only a single plant was analyzed, these results cannot be considered generic and applicable to a population of plants. The plant and system models used in this study are based on the Grand Gulf plant as it operates in a selected mode of operation. Thus, while some insights may be applicable to other plants, in general, the results from this study should not be arbitrarily applied to other plants or conditions. The model used to develop the progression of the accidents after the onset of core damage is, in part, based on the Grand Gulf Emergency Operating Procedures and other procedures and practices at the plant. Changes in these procedures and practices can certainly affect the progression of the accident and the ultimate risk of the POS. Similarly, since the offsite consequences are sensitive to the site characteristics and surrounding region (e.g., weather, population, land use), for a given release of radioactive material, the consequences can be expected to vary from one site to the next.

5. Comparison with Full Power

This section presents a comparison of POS 5 results with results from the NUREG-1150 (NUREG/CR-4550 and NUREG/CR-4551) full-power analyses as documented in SAND94-2949 [Whitehead et al., 1994b]. In Section 5.1 results are presented on a calendar-year basis, taking into account the fraction¹ of time on average the plant spends in each state during any one year. In Section 5.2 the results are presented on a per hour basis, conditional on being either at full power, in POS 5, or in a particular time window during POS 5.

5.1 Per Year Basis

By providing information on a calendar-year basis, results from all the different POSs can be added together to get a total CDF or risk measure for the plant as information becomes available.

Figure 8 presents a comparison of mean CDF percentages for the major classes of accidents from both the NUREG-1150 full-power [Brown et al., 1990] and the LP&S analyses [Brown et al., 1995]. From this figure one can see that there are points of both similarities and differences. The major similarity observed from the figure is that in both analyses the SBO class is important. SBOs showed up as dominant in full power because nothing else could cause loss or degradation of multiple systems and be above the truncation limit. In POS 5, SBOs also show up because they still cause loss or

degradation of multiple systems; however, now there are additional accidents (e.g., LOCAs) that can cause loss or degradation of multiple systems because of considerations unique to POS 5 (e.g., isolation of the automatic actuation of the suppression pool makeup system for safety reasons thereby requiring manual operator actions for continued use of ECCS pumps during a LOCA). Nonetheless, there are differences in the accident progression associated with the SBOs. These are (1) almost all the LP&S SBO sequences lead to an interfacing system LOCA and the full-power sequences do not; (2) the containment is always open at the start of the LP&S accidents whereas it is isolated at the start of full-power accidents; and (3) the probability of arresting the core damage process in the vessel is higher for full-power accidents than for LP&S accidents.

The makeup of the remaining accident classes provides a major difference between the two analyses. In the full-power analysis, the anticipated transient without scram (ATWS) class is the second most important class while in the LP&S analysis the most important class is the LOCA. Given the plant conditions analyzed in each of the two studies, the first point that can be made is that ATWS sequences were simply not possible in the LP&S analysis since the plant was already subcritical; therefore, one should not be surprised by this apparent difference. On the other hand, since LOCAs were possible in both analyses, why did this class show up in the LP&S results but not in the full-power results? While no detailed examination of this phenomenon was undertaken, the most likely reason for the appearance of LOCAs in the LP&S results is the intentional disabling of the automatic actuation of the suppression pool makeup system. This actuation is defeated for safety reasons. As a result, the continued use of injection systems during a LOCA requires operator intervention. The difference in reliability between automatic actuation and operator action generally accounts for the fact that LOCAs survived in the LP&S analysis but not in the full-power analysis.

Figures 9, 10, and 11 present a comparison on a calendar-year basis of the CDF, early fatality risk, and total latent cancer fatality risk for the three time windows² in POS 5, POS 5 in total, and in full power. Distribution information for POS 5 in total and for full power is displayed in Table 5. From Figure 9 one can see that while the POS 5 total mean core damage frequency is about a factor of two lower than the full-power value, there is overlap between the two

¹The fraction of time the plant is in POS 5 is 0.031 and the fraction of time the plant is at full power was taken as 1.0, since this has been traditionally used and thus can be compared with past results. In reality, the fraction associated with full-power operation would be something close to 0.8. This small difference is not expected to significantly affect any comparisons made in this section and is therefore ignored.

²A time window is a subdivision of the time spent in any one POS. Each subdivision allows a more realistic estimate of the decay heat load, equipment unavailabilities, and radionuclide inventories to be used during subsequent analyses. For POS 5, the three time windows used were time window 1—starting 14 hours after shutdown and having a duration of 10 hours; time window 2—starting 24 hours after shutdown and having a duration of 70 hours; and time window 3—starting 40 days after shutdown and having a duration of 10.4 days.

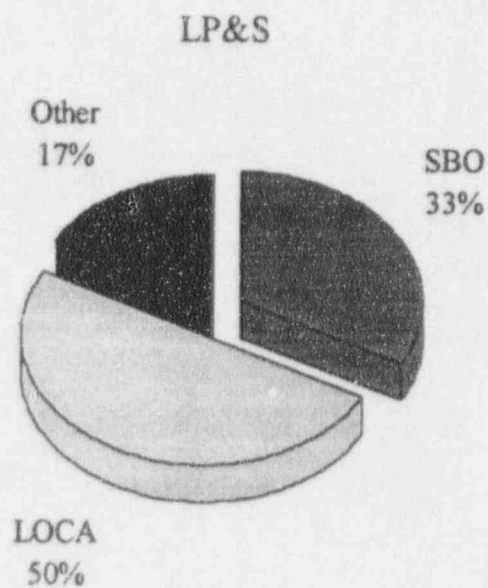
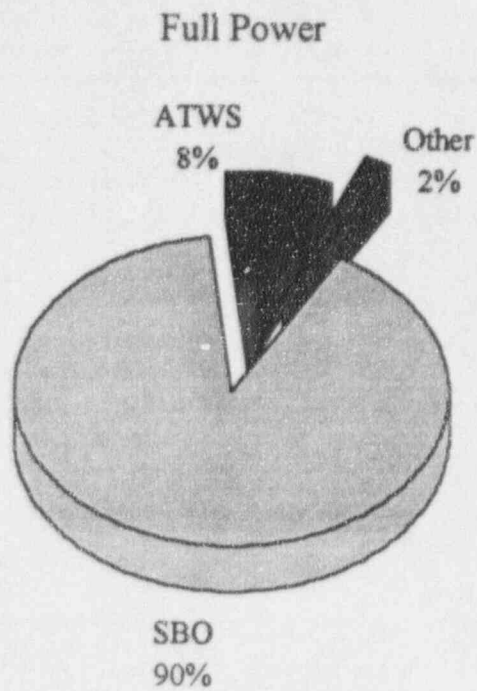


Figure 8 Percentage comparison of major accident sequence classes from full power and LP&S results.

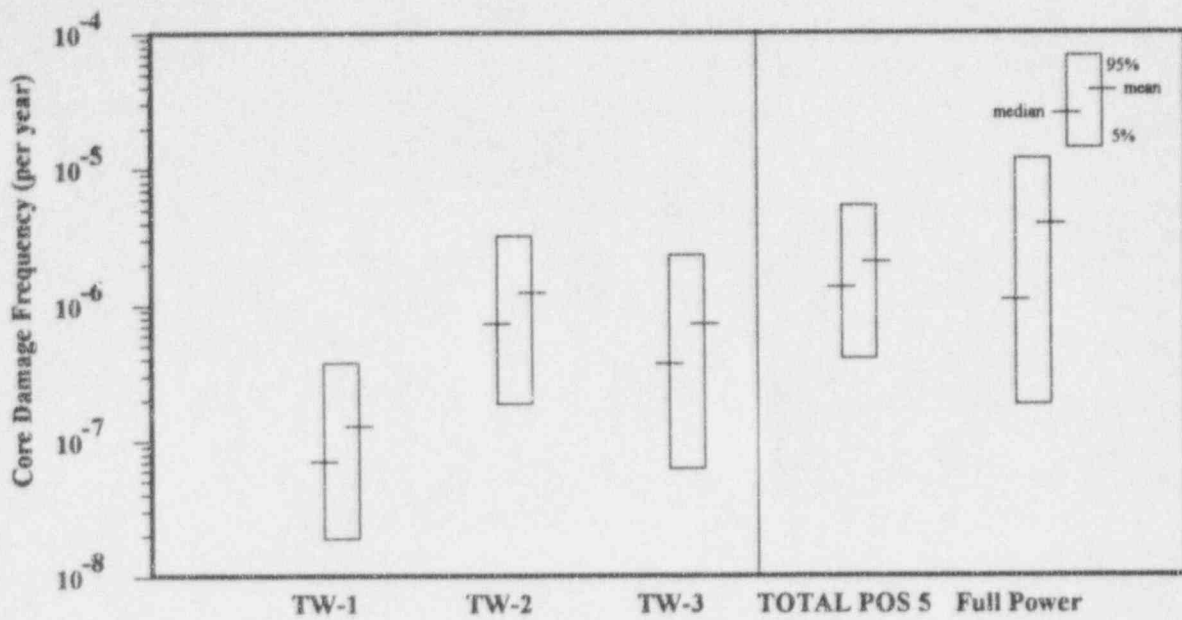


Figure 9 Core damage frequency per year for time windows 1, 2, and 3; total POS 5; and full power.

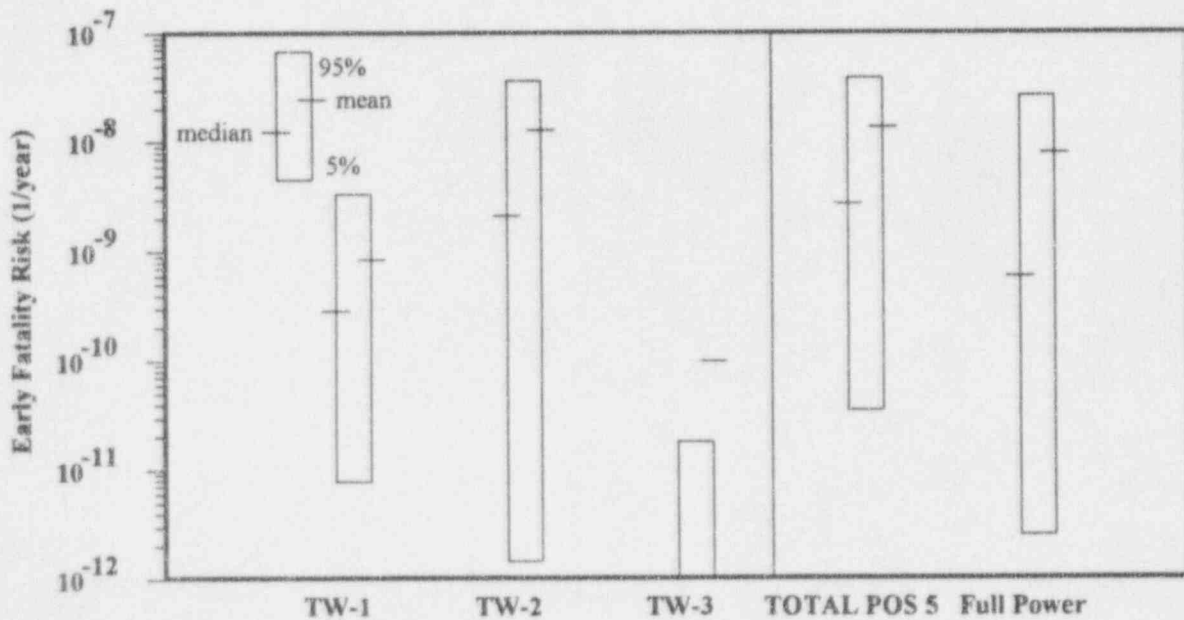


Figure 10 Early fatality risk per year for time windows 1, 2, and 3; total POS 5; and full power.

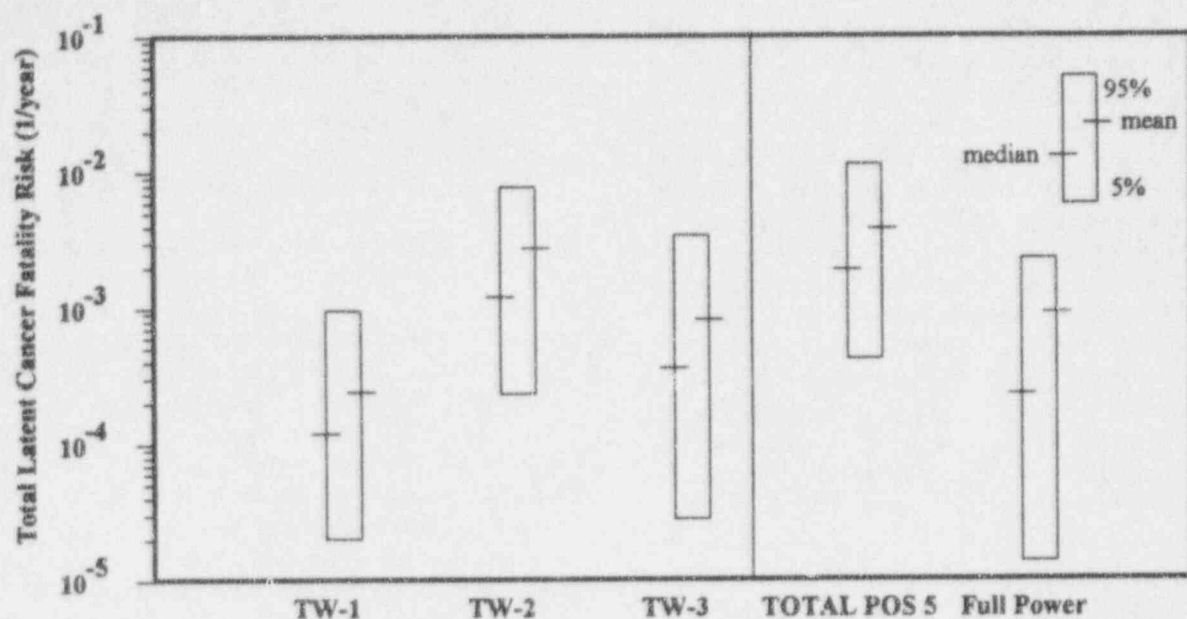


Figure 11 Total latent cancer fatality risk per year for time windows 1, 2, and 3; POS 5 in total; and full power.

Table 5 Distributions for Core damage frequency and aggregate risk for POS 5 and for full power*
(All values are per calendar year)

Analysis	Descriptive Statistics			
	Percentiles			Mean
	5th	50th	95th	
	Core Damage Frequency			
POS 5	4.1E-07	1.4E-06	5.6E-06	2.1E-06
Full Power	1.8E-07	1.1E-06	1.4E-05	4.1E-06
	Early Fatality Risk			
POS 5	3.7E-11	2.8E-09	3.9E-08	1.4E-08
Full Power	2.5E-12	6.1E-10	2.6E-08	8.2E-09
	Total Latent Cancer Fatality Risk			
POS 5	4.3E-04	1.9E-03	1.2E-02	3.8E-03
Full Power	1.4E-05	2.4E-04	2.3E-03	9.5E-04

* Full-power results were extracted from Volume 6 of NUREG/CR-4551 [Brown et. al, 1990]

contributions. Within POS 5, the least important time window appears to be time window 1. The other two windows have approximately the same importance, with window 2 being slightly more important from a mean CDF viewpoint. The primary reason window 1 is the least important is the small fraction of time the plant spends in window 1 compared with windows 2 and 3—0.03 in window 1 and 0.212 and 0.758 in windows 2 and 3, respectively. From Figure 10 one can see that the mean early fatality risk of POS 5 is only a factor of 1.7 greater than the full-power risk even though the containment is open during most of the accidents in POS 5. Within POS 5, the least important time window is window 3. The main reason for this is that time window 3 starts about 40 days after shutdown; thus, the radioactive material released during an accident in window 3 will have undergone decay, reducing the inventory of short-lived radionuclides that are important to early health effects. The decay associated with windows 1 and 2 is less; thus one would expect the risk associated with these windows to be higher. From Figure 11 one can see that the mean total latent cancer fatality risk of POS 5 is about a factor of 4 greater than the corresponding full-power risk. One reason for this is that in POS 5 the containment is always open and in full power the containment is always isolated at the start of an accident. Some of the difference results from the different models used in the MACCS calculations for the two studies. MACCS version 1.5.11.1 [Chanin et al., 1993] was used to estimate offsite consequences in the POS 5 probabilistic risk assessment; an earlier version was used in the NUREG-1150 plant studies. Cancer risk coefficients implemented in MACCS version 1.5.11.1 are two to three times greater than those utilized in earlier versions of the MACCS code. The total latent cancer fatality risk measure is directly affected by these risk coefficients. Within POS 5, the total latent cancer fatality risk associated with each time window tracks with the CDF associated with each time window since this risk measure is affected less by the decay of radionuclides than is the early fatality risk measure.

5.2 Per Hour Basis

By providing information on a per hour basis, results from different POSs can be compared, allowing the identification of the most important POS from a conditional CDF or risk viewpoint. However, before providing the CDF and risk information on a per hour basis, the following caution is made. Per hour results from the POS 5 analysis are not directly scalable to results based on being in POS 5 for 1 year. In other words, one *cannot simply multiply* the per hour results by the number of hours in a year and have the correct estimation of either CDF or risk for a POS 5 year. Such a process would overestimate the CDF and risk for POS 5 for the following three reasons:

1. The decay heat load would continue to decrease during the year, resulting in additional time for the operators to respond to any undesired event.
2. The unavailability associated with the systems would change as the year progressed, generally getting smaller, and thus reducing the likelihood of an accident progressing to core damage as a result of equipment unavailability.
3. Decay would reduce the radiological inventory that is available to cause health effects.

Figure 12 presents a comparison of the CDF on a per hour basis for each of the three POS 5 time windows, for POS 5 in total, and for full power. From the figure one can see that POS 5, generally speaking, is more important than full power when the CDF is considered on a per hour basis. While initially this might seem counterintuitive given the lower decay heat associated with operation in POS 5 and the subsequent increase in response time for the operators, the reasons for this become clear after examination. While the lower decay heat does provide the operators with more time to deal with events given an initiating event, some of these operator actions are more involved and/or complicated than those at full power. In addition, the operators must usually deal with the events that occur in POS 5 with a reduced set of equipment that results from the required test and maintenance activities associated with the various systems during the POS.

Within POS 5, the least important time window appears to be window 3. This window, generally speaking, is less important than the other two windows, even though the plant spends more time in this substate than it does in the other two combined. Time windows 1 and 2 have relatively the same importance, with time window 2 being slightly more important from a mean CDF viewpoint. However, as can be seen, the distributions for these two time windows have considerable overlap, and thus, for any given Latin hypercube sample (LHS) observation, either could be the most important. Possible explanations for these observations are as follows:

1. The decay heat load associated with time window 3 is the smallest of all three time windows. Since the water level in time window 3 is at least as high as it is in the other two windows, the operators will have more time to deal with events if they happen in time window 3. In addition, many accidents were eliminated from the analysis because the time to core damage was greater than the 24-hour mission time used in the analysis. These two in combination offset the larger fraction of time spent in this window.

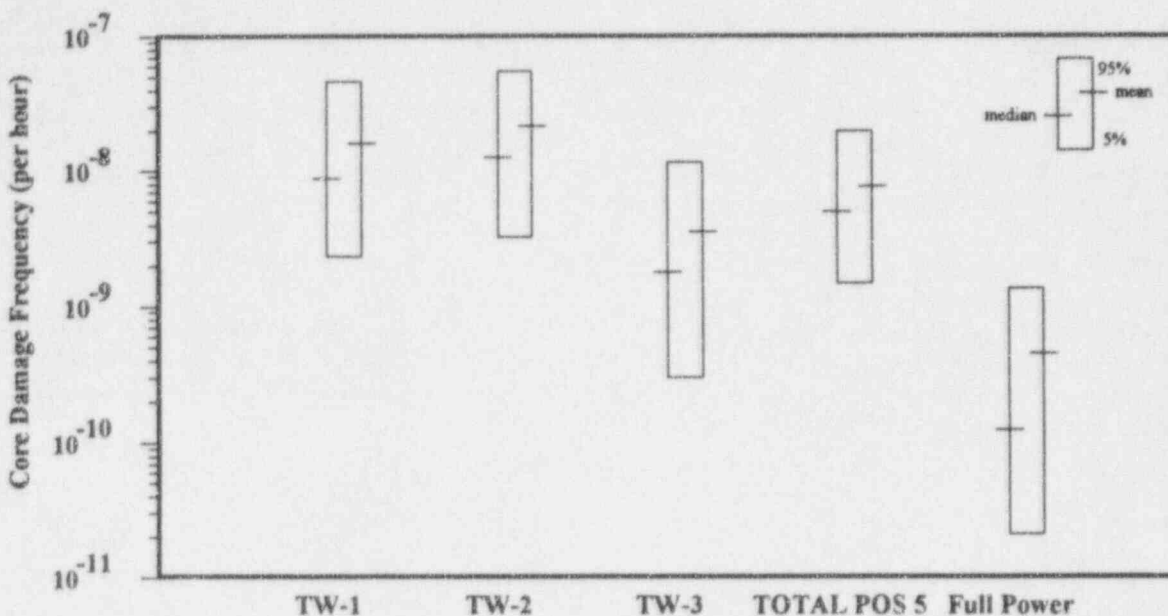


Figure 12 Core damage frequency per hour for time windows 1, 2, and 3; total POS 5; and full power.

2. The relatively equal importance of time windows 1 and 2 can be explained as follows:
 - The higher decay heat load in time window 1 implies that the operators will have less time to deal with events in window 1 than in window 2.
 - Generally speaking, the availability of equipment to respond to an initiating event is greater in window 1 than in window 2.
 - In combination, these two factors tend to balance each other, resulting in relative equality for both time windows.

Figure 13 presents a comparison of the early fatality risk on a per hour basis for each of the three POS 5 time windows, for POS 5 in total, and for full power. From this figure one can see that, generally speaking, the risk due to early fatalities is more important for the total POS 5 than for full power. However, as can be seen, there is overlap between the distributions such that for some LHS observations either might be the most important. One explanation for the generally higher importance of POS 5 is that, even though the plant has been shut down and the inventory of short-lived

radionuclides has decreased, the containment is generally always open and there are fewer means of removing radioactive materials from the atmosphere than is the case during an accident at full power. Also, on a per hour basis, the CDF is high for POS 5 relative to full power.

Within POS 5, the least important time window appears to be window 3, just as with the CDF. While window 2 is slightly more important than window 1 from a mean viewpoint, the overlap between the two distributions clearly indicates that for given LHS observations either window may be the more important. One reason window 3 is generally less important than the other two is that time window 3 starts 40 days after shutdown—which is enough time for many of the short-lived radionuclides important in early fatality risk to have decayed. This, in combination with the beneficial effect of the overlying pool of water (i.e., quenching the core debris and scrubbing releases for situations where the core debris is not cooled) associated with the accidents in this time window, tends to reduce the importance of early fatalities.

Figure 14 presents a comparison of the total latent cancer fatality risk on a per hour basis for each of the three POS 5 time windows, for POS 5 in total, and for full power. From this figure one can clearly see that the risk due to total latent cancer fatality is more important for the total POS 5 than for full power. The most likely reasons for this are the open

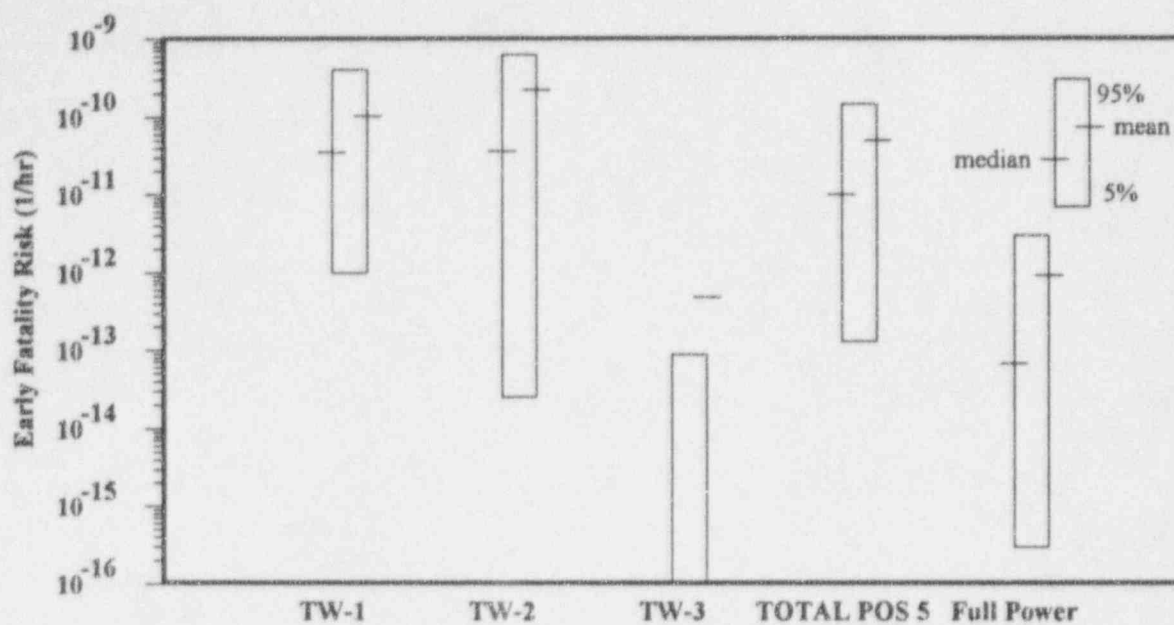


Figure 13 Early fatality risk per hour for time windows 1, 2, and 3; total POS 5; and full power.

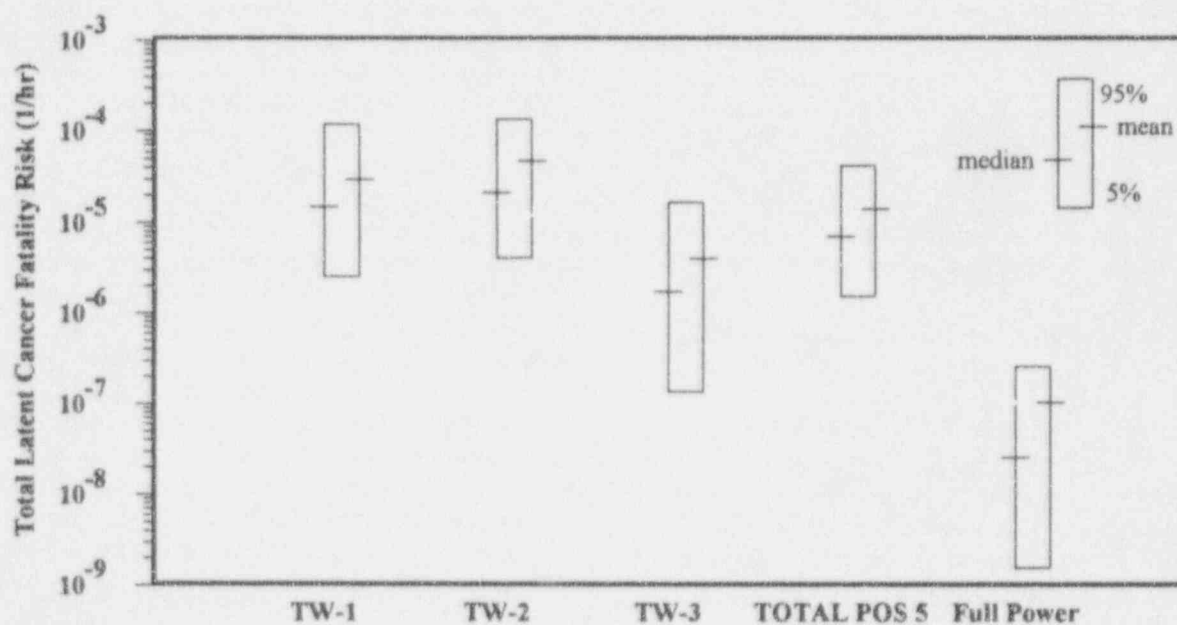


Figure 14 Total latent cancer fatality risk per hour for time windows 1, 2, and 3; total POS 5; and full power.

containment associated with many of the accidents in POS 5; there are fewer features of the plant to mitigate the release; and radioactive decay does not have a significant impact on the long-lived isotopes that are important in latent health effects. Also, on a per hour basis, the CDF is high for POS 5 relative to full power.

Within POS 5, the least important time window appears to be window 3. Given the overlap in the distributions associated with windows 1 and 2, either could be the most important for any given LHS observation. However, from a mean risk viewpoint, window 2 is slightly more important. The most likely reason window 3 is the least important is the beneficial effect of the overlying pool of water associated with the accidents in this time window.

6. General Conclusions and Insights

6.1 Level 1 Conclusions

The conclusions drawn from the Level 1 study can be grouped into three categories. They are

- (1) methodological,
- (2) plant specific, and
- (3) generic.

Methodological

This study was successful in developing a methodology to estimate the risk (i.e., the core damage frequency) associated with the operation of a BWR during low power and shutdown conditions. The methodology developed and the lessons learned from its application provide the NRC with new tools that could be used in subsequent analyses.

The event tree models developed for the analysis of POS 5 were more complicated than the full-power models because, given the lower decay heat load, there are many options for removing heat and keeping the core covered. In addition, more initiating events must be considered because of the systems that are normally operating to keep the plant within desired temperature and pressure parameters.

The mean CDF for each of the internal and external analyses presented in this report includes the fraction of time the plant is in POS 5 during a refueling outage. If one wanted to present the results as a conditional CDF (i.e., conditional on the plant being in POS 5), then the results should be divided by the value assigned to the POS 5 event. Thus, for example, for the Total Plant Model for the traditional internal events analysis, the conditional CDF is $(2E-6)/0.031 = 6.5E-5$ per year in POS 5. However the conditional CDF on a per year

basis is not recommended since plant conditions (e.g., system unavailabilities and decay heat loads) would change dramatically during a year. A more appropriate measure of conditional CDF would be one based on a per hour basis as described in Section 5.2 of this report. As was shown in Section 5.2, the conditional CDF is higher in POS 5 than at full power.

Plant Specific

There are three major aspects of the specific Grand Gulf plant model used in this analysis that significantly affected the results. These are

- (1) Grand Gulf's requirement for automatic isolation of low pressure components in the shutdown cooling system, given an increase in pressure and/or a decrease in water level in POS 5.
- (2) Grand Gulf's requirement that at least two safety relief valves be available in POS 5 allows the operators to use portions of their inadequate decay heat removal procedure, which would otherwise be inaccessible.
- (3) Grand Gulf's additional system for removing decay heat (i.e., the alternate decay heat removal system) affects the estimated core damage frequency during two of the three POS 5 time windows.

Generic

The results from this study appear to indicate that the core damage frequency associated with operating in POS 5 during a refueling outage is less than that from operating at full power. While this should be true for Grand Gulf, generalizations to other BWRs should be performed with care.

Two factors that should be considered during any generalization are

- (1) Does the other BWR have a motor-driven high-pressure pump? The availability of such a pump provides a mechanism for injecting water at high pressure, if necessary, and also provides an alternative means of injecting water at low pressure should the low-pressure pumps fail.
- (2) Does the other BWR have procedures in place to deal with the loss of the normal decay heat removal system? If the procedures do exist, does the utility require that the systems and components necessary for the procedure be available?

6.2 Level 2/3 Conclusions

The following conclusions can be drawn from this study:

- With many plant features unavailable to mitigate a release, the potential exists for a large release of radioactive material should core damage occur. For the most likely accidents, the containment is open, the suppression pool is bypassed, and the containment sprays are not available.
- In the event that the containment is closed prior to the onset of core damage, it is always predicted to fail since containment heat removal was not available in the accidents analyzed.
- The risks from POS 5 are not insignificant compared with the risks from full power operation. Hence the full-power risk distributions by themselves do not completely characterize the risks associated with the operation of this plant. To accurately characterize the plant's results from this study suggests that it may be necessary to include other modes of operation in addition to the full-power mode. This can have important implications for assessments that rely on the total risk from a plant, such as when comparisons are made with the safety goals.
- Although only a simplified scoping study of the onsite consequences was performed, the possible onsite consequences of an accident during shutdown could be significant, particularly since in many of the accidents the containment remains open allowing for an early release of radioactive material.

6.3 Insights from POS 5

This section presents insights for POS 5 as documented in SAND94-2949 [Whitehead et al., 1994b]. All insights presented here are derived from observations made on the specific results in the traditional internal events, the specific models and assumptions used in the LP&S analyses, selected sensitivity studies that made modifications to the models and assumptions, selected results from the full-power analysis, and the experience of the analysts who performed the original LP&S study. These insights will be discussed from both a vertical (i.e., within a specific observation) and horizontal (i.e., across many observations) viewpoint.

The reader should be aware that these insights are for POS 5 at Grand Gulf. As such, this information should not be generalized to other nuclear power plants without first considering all relevant factors. Complete details on how

these insights were developed and/or identified can be found in SAND94-2949.

6.3.1 Insights from LOCAs

6.3.1.1 CDF Insights

The LOCAs that were analyzed in the LP&S project can be grouped into two categories. These are LOCAs during nonhydro conditions (i.e., at atmospheric pressure) and during hydro conditions (i.e., at approximately 1000 psi). The accident sequences in the LOCA class are driven by events that result in the loss of multiple systems:

1. Failure of the operators to dump the suppression pool makeup (SPMU), resulting in loss of all ECCS, and
2. Flooding in the auxiliary building as a result of the operators failing to close the lower personnel lock.

From a CDF viewpoint, concerns about the value used for the initiating event frequency of the nonhydro LOCAs in POS 5 would be immaterial if the automatic actuation of the SPMU system were functional. Automatic actuation of the SPMU would most likely eliminate all LOCAs. However, given that this system is deactivated in POS 5 for the physical safety of the workers, elimination of the low-pressure LOCAs reduces the fractional contribution of the LOCAs to the total CDF by about a factor of 2. More important, elimination of the low-pressure LOCAs reduces the early fatality risk attributed to LOCAs by about a factor of 80. This is expected because low-pressure LOCAs occur in time windows 1 and 2, and, generally speaking, these affect early fatality risk more than time window 3 sequences.

Changes in procedures that would allow more credit to be given to the operators during the human reliability analysis for controlling injection systems, specifically the high pressure core spray (HPCS) system, would provide some reduction in the importance of the LOCA class from a CDF and total latent cancer risk point of view; however, since this change affected only accidents in time window 3 (sequences in time windows 1 and 2 were unaffected because the HPCS system is unavailable in the cut sets that survived the phase 2 analysis), no significant change would be expected in the early fatality risk measure.

6.3.1.2 Risk Insights

When considered as a group, LOCA accidents are not on average the most important contributor to early fatality risk. They are, however, an important contributor to the total latent cancer risk.

- The LOCA group is not on average the most important contributor to early fatality risk because the most probable LOCA accidents occur while the

plant is in time window 3 (approximately 40 days after shutdown), by which time radioactive decay has significantly reduced the inventory of short-lived radionuclides that are important in early health effects.

- The LOCA group is an important contributor to total latent cancer fatality risk because it is an important contributor to the core damage frequency and because accidents from this group release a considerable amount of radioactive material into the environment (primarily long-lived radionuclides that are important in latent health effects).

The releases associated with the LOCA have the potential to be relatively large because all of the accidents progress to full core damage and vessel failure, and many of the plant features that can be used to mitigate the release are unavailable or bypassed.

- The containment is ineffective as a barrier to the release of radioactive material during accidents initiated by a LOCA because the equipment hatch and personnel airlock remain open. This occurs when the operators fail to recognize the need to close the lower personnel airlock before core damage starts.
- The radioactive material released from the damaged fuel in the vessel bypasses the suppression pool and, hence, is not attenuated by the scrubbing properties of the pool. The containment spray system fails and cannot be recovered during the accident.

Two features of the accident that can mitigate the release to the environment are the flooded containment and the passage of the release through the auxiliary building.

- While the vessel is always predicted to fail and release core debris into the pedestal cavity, there is a significant probability that the core debris will be quenched in the cavity. For those accidents in which the core debris is not quenched, the releases that accompany the interactions between the core debris and the concrete structures will be scrubbed by an overlying pool of water. Hence, the flooded containment can attenuate the late release of radioactive material by either preventing core-concrete interactions or by scrubbing the releases that accompany CCI in the event that the core debris is not quenched.
- The passage of the release through the auxiliary building will also attenuate the release; owing to its size, the auxiliary building can act as a large holdup volume, allowing time for natural processes to remove airborne material from the building atmosphere.

6.3.2 Insights from Station Blackouts

6.3.2.1 CDF Insights

The SBO sequences are driven by events that result in the loss of multiple systems. For example, the loss of onsite ac power prevents the use of all systems except the diesel-driven firewater pumps. The use of firewater was unsuccessful in these accidents which were grouped into the following three classes:

1. Insufficient time for the operators to align and use the pumps before battery depletion.
2. Sufficient time for operators to use the pumps, but they fail.
3. Operators successfully align and begin use of the pumps, but the batteries deplete, the SRVs close, and injection is lost as the reactor vessel pressure increases.
 - In this third class, a distinction was made between those sequences where the pumps ran long enough to allow either the #8 or #9 isolation valves to be closed. If either of the valves was closed, then no interfacing system LOCA occurred. If the isolation valves were not closed, the decay heat removal system failed on overpressure, resulting in an interfacing systems LOCA in the auxiliary building.

Results from the sensitivity calculation where it was assumed that the SRVs could be either opened or kept open given a loss of dc power show that the ability to open or keep open the SRVs is relatively important—the mean fractional contribution of the SBO sequences changes from 0.33 to 0.10, indicating that approximately 23% of the total CDF comes from SBO sequences involving dependence on the SRVs and thus dc power. In addition, the sensitivity calculation gives an indication of the importance of the operator action associated with use of the diesel-driven firewater pumps. Failure to successfully align and use the firewater system contributes about 10% to the total mean CDF.

6.3.2.2 Risk Insights

Station blackout accidents, when considered as a group, are an important contributor to both early fatality risk and to total latent cancer fatality risk. The SBO group is an important contributor to these risks because:

- The group is a major contributor to the core damage frequency.
- The probability that core cooling is restored and the core damage process arrested is fairly small. The factor that is primarily responsible for this low probability is the relatively low probability of recovering offsite ac power before significant core damage has occurred. Hence, the most likely situation is that the accident progresses to full core damage and vessel failure.
- Since the containment is not flooded, the core debris released from the vessel will almost always interact with the concrete structure below the vessel and continue to release radioactive material. In these SBO accidents, these interactions rarely occur under a pool of water and, therefore, the releases that accompany these interactions will typically not be scrubbed by an overlying pool.
- There are very few plant features available to attenuate the release of radioactive material from the damaged fuel and core debris.
 - The containment remains open during the entire accident. Closing the containment requires offsite ac power, and ac power was not available before core damage started.
 - Owing to the configuration of the plant and the nature of the accident, radioactive material bypasses the suppression pool. The unisolated break in the decay heat removal system and the open reactor vessel head vent both allow material released from the damaged fuel in the vessel to bypass the suppression pool. The open drywell equipment hatch and personnel airlock allow airborne radioactive material in the drywell to bypass the suppression pool.
 - Owing to system failures and the fact that containment pressure control is not an issue when the containment is open, the containment sprays were not used during the accident and, hence, airborne radioactive material was not scrubbed by the sprays.
- All of the SBO accidents occur while the plant is in time windows 1 and 2 when there is still a significant inventory of radionuclides that are important in early health effects.

One of the few plant features available to attenuate the release of radioactive material in these accidents is the auxiliary building. Since the containment is open and the break in the decay heat removal system is located in the auxiliary building, all of the release passes through the building before escaping to the environment.

6.3.3 Insights from *Other*

6.3.3.1 CDF Insights

Accident sequences in the *Other* class were grouped into three classes:

1. Flooded containment,
2. Open main steam isolation valves (MSIVs), and
3. Loss of all standby service water (SSW).

Within each of these classes, some event (or assumption) causes failure of several of the systems that might be used to respond to the accident.

- For the first two, the water coming out of the flooded containment or the open MSIV is assumed to fail the remaining core cooling systems.
- For the third, the loss of all SSW fails all emergency core cooling systems.

All of the sequences in the first class involved an empty suppression pool—based on an assumption made in the LP&S analysis. If the sequences in this class are eliminated from the *Other* class, then the mean CDF for the *Other* class decreases by about a factor of 10. However, the total POS 5 mean CDF decreases by only a factor of 1.2 since the mean contribution of this class to the total CDF is only 17%. This implies that while water in the suppression pool is important to the *Other* class, it is not important to a change in the total mean CDF for POS 5.

6.3.3.2 Risk Insights

When considered as a group, the *Other* accidents are not on average the most important contributor to either early fatality risk or to the total latent cancer fatality risk. This stems primarily from the fact that the *Other* group is not on average the most important contributor to the core damage frequency and the consequences from these accidents are not large enough to compensate for the relatively low core damage frequency. This is not to say that the releases are negligible. While many different types of progressions can be found in this group, some common characteristics of these accidents include:

- Core cooling is never restored and, hence, all of the accidents progress to full core damage and vessel failure.
- The containment is either open during the entire accident or if it is closed before the core is damaged, it is either vented or fails during the accident. The principal reason the containment becomes pressurized is that containment heat is not removed; hydrogen combustion and loads that accompany failure of the reactor vessel at high pressure also contribute to containment failure.
- All the accidents occur while the plant is in time windows 1 and 2 and, thus there are enough short-lived radionuclides to cause early fatalities.

6.3.4 General Insights

6.3.4.1 CDF Insights

The sequences that survive generally contain some event or events that result in the failure of multiple systems. Examples of this are

- For LOCAs
 - The failure of the operators to dump the suppression pool makeup system results in the loss of all ECCS injection systems.
 - Flooding of the auxiliary building, as a result of failure to close the personnel lock, is assumed to result in loss of all remaining core cooling systems.
- For SBOs
 - The failure of onsite ac power eliminates all injection sources except for the diesel-driven firewater system.

Requiring the availability of SRVs during POS 5 allows for additional cooling options and thus requires that something cuts across multiple system boundaries to result in core damage. This is in part what contributes to the relatively small CDF estimate obtained by the LP&S project.

The presence of a motor-driven high-pressure system also contributes to the relatively small CDF estimate. This is true even after taking into account this system's sometimes significant unavailability, primarily because the motor-driven pump, when available, can be used in many situations where a turbine-driven pump could not be used.

Automatic isolation of the low-pressure piping on high pressure during POS 5 effectively eliminates the problem of

an interfacing system LOCA except when power is lost to the isolation valves, as is the case during the SBO sequences.

Operator actions play a significant role both in the progression of the accident sequences and in the estimate of the CDF associated with the sequences. Generally speaking, operator actions in POS 5 tend to be more involved and/or complex from a diagnosis viewpoint.

Only sequences that resulted in a loss of vessel inventory survived in time window 3. The major reason is that the decay heat load in time window 3 is small enough that only with a loss of water from a break or diversion will core damage occur within the 24-hour mission time used in the LP&S project.

6.3.4.2 Risk Insights

In these accidents, the containment is not effective as a barrier against the release of radioactive material. While it is a common characteristic of all of these accidents that the containment is an ineffective barrier, the reasons that lead to this characteristic vary across the accidents.

- For accidents where the containment is flooded (LOCAs and some of the *Other* accidents), it is the failure of the operators to close the lower personnel lock that leads to the containment being open during the entire accident.
- For SBO accidents, it is the fact that offsite ac power is not available prior to core damage that causes the containment to remain open for the duration of the accident.
- For the few accidents where the containment is isolated before the onset of core damage, it is the lack of containment heat removal that is the principal cause for containment venting or failure; hydrogen combustion events and loads that accompany failure of the reactor vessel at high pressure also contribute to containment failure.

Isolation of the containment is not sufficient to prevent a release to the environment. Owing to the relatively low design strength of the containment and the venting procedures used at the plant, containment heat removal and hydrogen control must also be provided to minimize the likelihood of a release to the environment. Even with these systems available, energetic loads accompanying the failure of the vessel at high pressure and energetic fuel-coolant interactions can threaten the containment. Nevertheless, without heat removal and hydrogen control, containment failure is only delayed; it is not prevented.

As illustrated by SBO accidents, in addition to the difficulty of closing the containment equipment hatch, it may be difficult to isolate all penetrations when ac power is not available because valves that were open before the initiating event may "fail" in the open position. This concern is not as acute during full-power operation since all of the low-pressure components are already isolated from the primary system. Furthermore, the remaining valves that must be isolated during power operation will typically "fail closed" on complete loss of power.

By the time the core is damaged, enough failures have occurred (both hardware and operator) that the probability of recovering core cooling and arresting the damage process in the vessel is relatively low. The most likely situation is that core cooling is not restored and the accident progresses to full core damage and vessel failure.

Many of the plant features that can mitigate the release of radioactive material that accompanies a severe accident are either bypassed or unavailable. In all these accidents, the containment is ineffective as a barrier to the release of radioactive material; the containment sprays are either unavailable or are not used; and in most of the accidents the radioactive material released from both the damaged fuel in the vessel and the core debris released from the vessel bypass the suppression pool. Both the containment sprays and the water pools have in the past been demonstrated to be effective devices for scrubbing radionuclides from the containment atmosphere.

The auxiliary building can play an important role in attenuating the release and reducing the risk to the offsite population. Since in most of these accidents the containment is either open or bypassed, and many of the plant features used to mitigate a release are bypassed, the auxiliary building is one of the few plant features available to reduce the release.

Compared with accidents that occur while the plant is in time windows 1 and 2, accidents in time window 3 result in relatively few early fatalities since by this time radioactive decay has significantly reduced the inventory of short-lived radionuclides that are important in early health effects.

Owing to the sparse population around the plant, relatively rapid evacuation, and the slow progression of the accident, nearly all of the early fatalities will occur in the fraction of the population that does not leave the evacuation zone. This parameter—the fraction of the population that does not evacuate—can have a significant impact on the number of early fatalities; it has a relatively minor effect on the number of latent cancer fatalities.

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This document contains a summarization of the results and insights from the Level 1 accident sequence analyses of internally initiated events, internally initiated fire and flood events, seismically initiated events, and the Level 2/3 risk analysis of internally initiated events (excluding fire and flood) for Grand Gulf, Unit 1 as it operates in the Low Power and Shutdown Plant Operational State 5 during a refueling outage. The report summarizes the Level 1 information contained in Volumes 2 - 5 and the Level 2/3 information contained in Volume 6 of NUREG/CR-6143.

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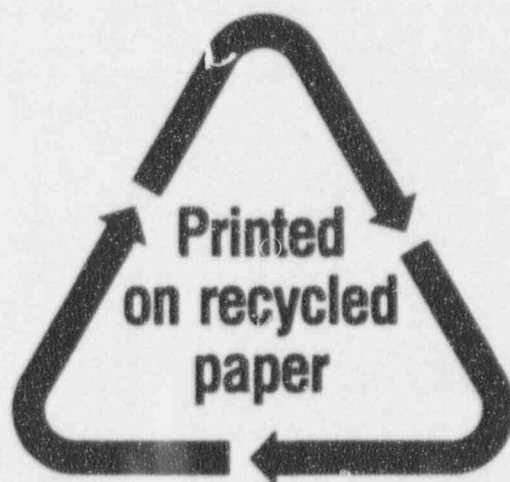
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