SUMMARY REPORT OF:
GRAND GULF LOW POWER AND SHUTDOWN ABRIDGED RISK ANALYSIS

POS 6: Early Refueling

FINAL LETTER REPORT

Thomas D. Brown[1]
LeAnn A. Miller[1]
Lubomyra N. Kmetyk[1]
Lanny N. Smith[2]
Donnie W. Whitehead[1]
John Darby[3]
John Forester[2]

October 1993

Sandia National Laboratories
Albuquerque, NM 87185
Operated for the
U.S. Department of Energy
Contract DE-AC04-76DP00789

Prepared for
Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN L1679

[1]Sandia National Laboratories
[2]Science Application International Corporation
[3]Science & Engineering Associates, Inc.

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ACKNOWLEDGMENTS

We wish to thank the many people who worked in various capacities to support this analysis: Fred T. Harper (SNL) who provided many helpful technical suggestions in the development and application of methods and who provided a much needed sanity check on the analysis and results; Jay Johnson (GRAM) who performed the MACCS calculations for the offsite consequences; David I. Chanin (Technadyne Engineering Consultants) who provided helpful ideas for assessing onsite consequences; and Ann W. Shiver (SNL) who provided support in the data preparation for the uncertainty analysis.

We would also like to thank Chris Ryder (NRC) for his program and management support and for his words that form the introduction of this report.

Finally, we would like to thank the Source Term Advisory Group, which consisted of John E. Kelly (SNL), Hossein P. Nourbakhsh (BNL), Dana A. Powers (SNL), and Trevor Pratt (BNL), for their review of the source term analysis and their many helpful suggestions.

ACRONYMS AND INITIALISMS

ADHR	Alternate Decay Heat Removal System
APET	Accident Progression Event Tree
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCI	Core Concrete Interactions
CDS	Condensate System
CF	Containment Failure
CNT	Containment
CRD	Control Rod Drive System
ECCS	Emergency Core Cooling System
FW	Firewater System
HIS	Hydrogen Ignition System
HPCS	High Pressure Core Spray System
HRA	Human Reliability Analysis
LHS	Latin Hypercube Sample
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LP&S	Low Power and Shutdown
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PSW	Plant Service Water
PWR	Pressurized Water Reactor
RFO	Refueling Outage
RHR	Residual Heat Removal System
RPV	Reactor Pressure Vessel
SBGT	Standby Gas Treatment System
SBO	Station Blackout
SDC	Shutdown Cooling System
SPC	Suppression Pool Cooling System
SPMU	Suppression Pool Makeup System
SRV	Safety Relief Valve
SSW	Standby Service Water
TAF	Top of Active Fuel
TBCW	Turbine Building Cooling Water
VB	Vessel Breach

1.0 INTRODUCTION

1.1 Study Objectives

The Office of Nuclear Regulatory Research at the U. S. Nuclear Regulatory Commission established programs to investigate postulated accidents during low power and shutdown (LP&S) operations of a BWR (Grand Gulf) and a PWR (Surry). One such program is a risk study of accident progressions and consequences.

The objective of this study is to make a preliminary risk determination of the progressions (Level 2 analysis) and the consequences (Level 3 analysis) of accidents during low power and shutdown operations in the Grand Gulf plant. The study was designed to obtain results for regulatory decisions. This letter report documents the methods, findings, and implications of the study done under NRC FIN L1679. A sister study of the Surry plant is reported separately by the staff at Brookhaven National Laboratory (BNL) under NRC FIN L1680.

1.2 Scope of the Study

The abbreviated risk analysis took place from January through April 1992. The study has been referred to as an ***abridged risk*** analysis. The term ***abridged*** means that simple event trees (about nine top event questions) were developed and used with assumptions and other approximate methods to compute rough estimates. The term ***risk*** means conditional consequences (probability of the various events during the accident progressions multiplied by the consequences), given that core damage has occurred. Traditional risk estimates, computed by multiplying the conditional consequences and the frequency of the sequences, could not be made at this time because the core damage frequencies have yet to be determined in companion Level 1 and HRA studies. Uncertainty has been taken into account in a manner consistent with the detail of the abridged study.

This study investigated the possible accident progressions and the associated consequences of a single plant operating state, POS 6, an early stage of refueling, where the reactor vessel head is removed, the steam dryers and separators are removed, the drywell is open, and the containment is open. The sister study at BNL investigated mid-loop operation. The scope of both studies is illustrated in Figure 1-1.

1.3 Methods

The abridged process of computing conditional consequences is shown in Figure 1-2. In general, both the study reported here and the study done at BNL follow this scheme. Some differences in the details of the procedure exist and are noted at the end of Section 1.3. The process used here is an abbreviated form of the NUREG-1150 study [1].

Accident progressions

The calculations begin with the assumption that core damage has occurred. Given core damage, the reasonable accident progressions are delineated with the accident progression event tree (APET). Much of the delineation is based on information obtained from PRAs of full power operation, knowledge of severe accident phenomena, and deterministic calculations with codes used to compute source terms, such as MELCOR [2]. The likelihood of the various accident progressions is reflected vis-a-vis branch point probabilities.

Branch point probabilities were assigned to reflect the likelihood of various pathways thought to exist. In large scale risk studies, the assignment can be done by groups of *experts* knowledgeable in severe accident issues. Here, because of resource limitations, most of the assignments were done by the project staff. The probabilities are not as rigorous as they could be but this is one of many limitations of the study to be discussed. Some lack of rigor in determining the probabilities is taken into account by repeating the calculations with other possible probabilities; taken together, the repeated calculations constitute an uncertainty analysis.

Through the uncertainty analysis, distributions, instead of point values, were assigned to selected branch points. The distributions are subjective but account for many possible values of the branch points. Point values are selected from the distributions with a form of Monte Carlo sampling known as Latin Hypercube Sampling (LHS) [3]. After making sets of inputs, each set, consisting of point values, is assigned to the branch points and multiplied through to the ends of the APET. The calculations are repeated using the sets of inputs, building a probability distribution at the end of each pathway.

Source terms

Having delineated accident progressions with the APET, the source terms of the progressions were calculated with a parametric code [4]. The parametric code is a collection of simple mass-balance equations designed to mimic detailed source term codes. The parametric approach is not meant to be a substitute for detailed, mechanistic computer simulations codes. Rather, it is a framework for integrating the results of these codes together with experimental results and expert judgment.

The parametric code determines source terms, given the characteristics of the accident progression and other inputs (e.g., fraction of the inventory a) leaving the reactor vessel; b) involved in core concrete interactions; c) entering the containment). Because these other variables are imprecisely known, many reasonable values can be assigned to the inputs. As in the APET calculations, distributions are assigned to the variables and sampled with LHS to form many sets of input values for repeated calculations. The result is a distribution of source terms for each accident progression pathway.

Because the estimation of the source terms is a critical component of this study, an internal advisory group, call the Source Term Advisory Group, was formed to support this study. The members of the advisory group included: John E. Kelly (SNL), Hossein P. Nourbakhsh (BNL), Dana A. Powers (SNL), and Trevor Pratt (BNL). The role of the Source Term Advisory Group was to 1) provide guidance on the identification of phenomena that may be important to the formation of the source term during these modes of operation, and 2) assess the adequacy, relative to the study's objectives and scope, of the assumptions, methods and data used in this study. The results of the accident progression and source term analysis were presented to and discussed with the advisory group in two meetings during the course of this analysis.

Consequences

Three sets of radiological consequences were determined: building dose, onsite dose (so called *parking lot* dose), and offsite consequences.

- o Building dose was determined based on source terms derived from the parametric source term expressions. Doses in the containment and auxiliary building were estimated.

- o Parking lot dose was based on relative concentrations computed with the Ramsdell model [11], in which the release concentration is somewhat proportional to wind speed, and a combination of the Wilson model [12] and the model in Regulatory Guide 1.145 [13], in which the concentration is inversely proportional to wind speed.

- o Offsite consequences were computed using the MACCS code [5,6,7].

Uncertainty was not propagated through the consequence analysis as it was through the APET and the source term calculations. While a sample size of 100 was used in the onsite analysis to propagate accident progression and source term uncertainties, a reduced sample size of 12 was used in the determination of offsite consequences.

Conditional offsite consequences

Conditional risk was computed by multiplying the offsite consequences by their associated accident probability that was determined with the APET. This product of probability and consequences was computed for each accident progression pathway. The products of the pathways were summed. This process was repeated for each of the few samples of the source terms. Then, high, medium, and low results were reported.

Differences

This study differs slightly from its sister program at BNL in three ways. (1) Here, one hundred samples from the uncertainty distributions were propagated through the accident progression and source term analyses whereas, in the BNL study, two hundred samples were taken. (2) Here, twelve samples were propagated through the APET to offsite consequences whereas, in the BNL study, twenty samples from the source term distributions were used in consequence calculations and traced back through the APET for the probabilities needed to compute conditional risk.

(3) Here, doses in the containment and auxiliary building were calculated whereas, in the BNL study, these calculations were thought unnecessary since the releases were assumed to pass from the containment directly into the environment.

1.4 Limitations and Strengths of the Study

In order to place the calculations in proper context, it is necessary to understand the strengths and limitations of the study.

Limitations

- o The subject of the study is one POS, early refueling. This POS was selected for study because it was identified in a preliminary Level 1 study, known as a coarse screening analysis [8], as potentially occurring at a relatively high frequency. Also, the POS had characteristics (i.e., reactor vessel head removed) of interest to the staff in the Office of Nuclear Reactor Regulations at the NRC.

- o The abridged study is based on the coarse screening analysis where accident sequences potentially having high frequencies were identified. The consequences of these sequences were determined in the Level 2 and 3 abridged study reported here. The frequency is not merged with the Level 2 and 3 calculations to determine risk because the numerical value of the frequency estimate is believed to be too rough for such use.

- o The simple APET accounts for a limited number of factors. The APET consisted of nine top event questions, compared to about one hundred questions in a large scale PRA.

- o The onsite dose estimates stem from simple equations yielding rough estimates.

- o Variables were selected and assigned distributions for the uncertainty analysis by the project staff.

- o Because of gaps in knowledge of the plant configuration and operator actions, assumptions were necessary. The assumptions are documented in the sections to follow.

Strengths

- o Even with the limitations noted above, the abridged study is a systematic evaluation, which includes a limited treatment of the uncertainty in severe accident progressions.

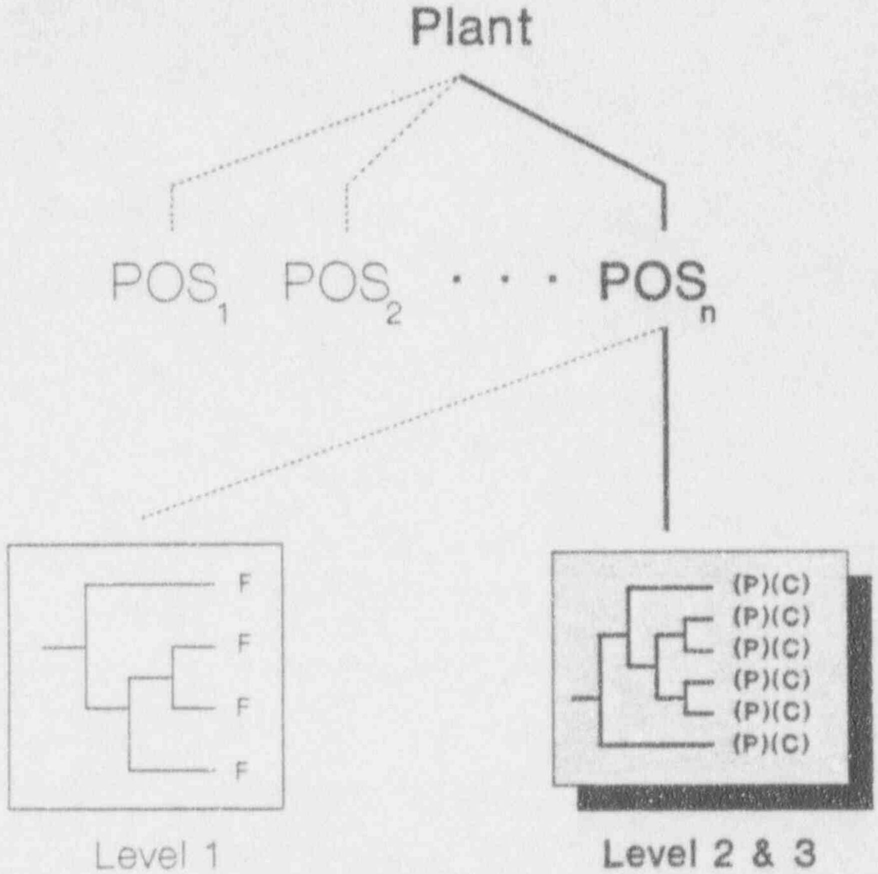
- o The source term analysis was reviewed by an internal advisory group.

- o The project staff and the NRC project staff believe that the APET represents the occurrence of key events during accident progressions.

- The relationship and timing of accident progression events and factors have been determined to at least a first approximation.

- Estimates of both onsite and offsite conditional consequences were made.

The sections to follow document the abridged study of the Grand Gulf plant. The discussion above is expanded, providing important details and results.



$$R = [F]\,[P]\,[C]$$

R = risk
F = sequence frequency
P = accident progression probability
C = consequences

Figure 1-1. Scope of abridged study

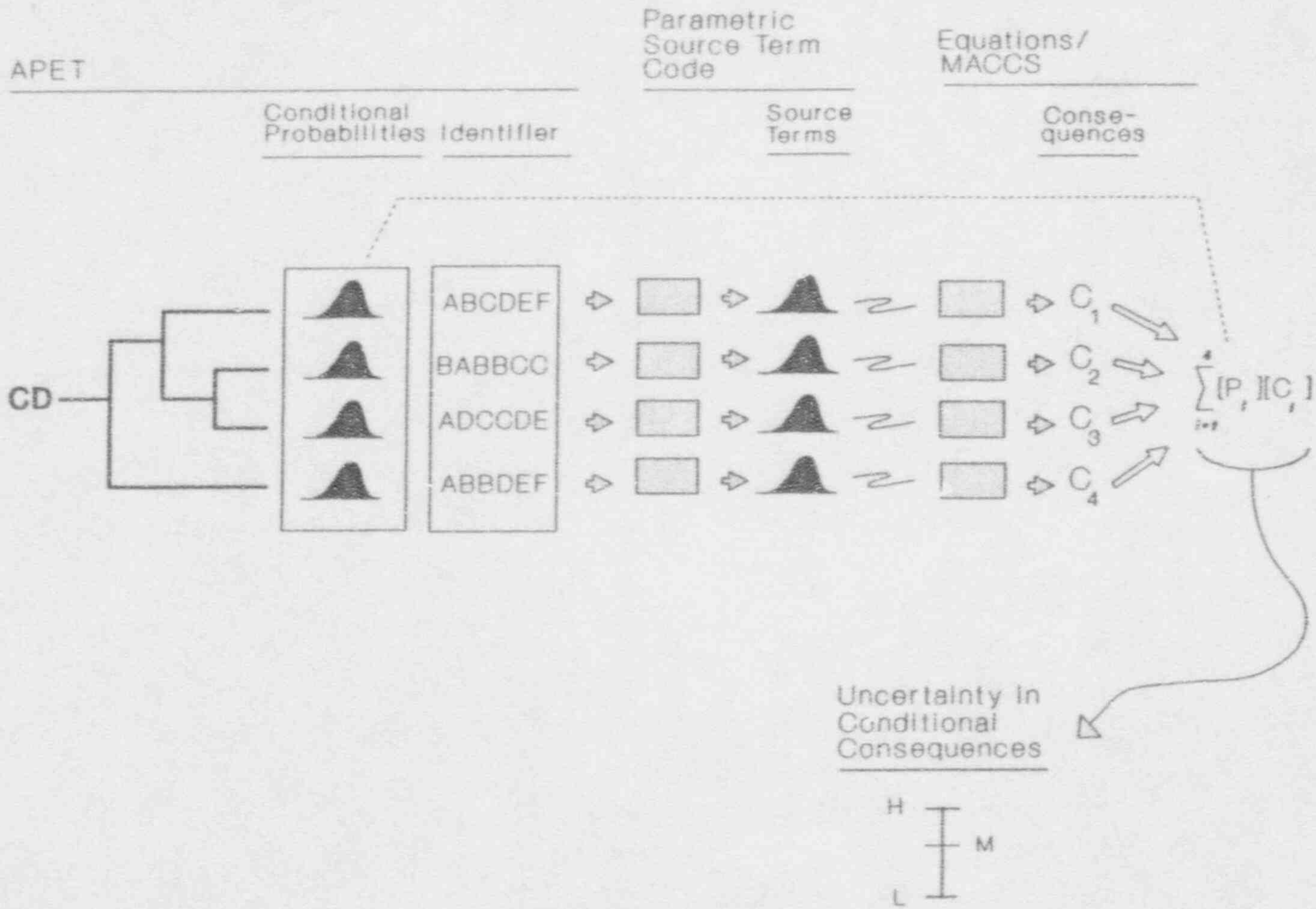


Figure 1-2. Summary of abridged methodology

2.0 ACCIDENT PROGRESSION ANALYSIS

2.1 Approach

The progression of accidents following core damage are analyzed in the Level 2 portion of the PRA. In this chapter the development and quantification of the accident progression scenarios will be presented. The input to the accident progression analysis is the core damage sequence definitions developed in the Level 1 analysis [8]. The core damage sequences define the successes and failures of equipment and human actions that have resulted in the loss of core cooling and the onset of core damage. The sequence definitions provide information on the status of core cooling systems, containment cooling systems, and containment integrity at the time of core damage. From this information the possible accident progressions, which identify the response of the core and the containment following core damage, are determined. These accident progressions are developed and displayed using an event tree. In this abridged analysis only the most important events that affect the timing and the magnitude of the radionuclide release are addressed. The outputs from the accident progression analysis are the accident progression path definitions and the likelihood, conditional on core damage having occurred, of each path. In the source term analysis, the fission product release associated with each path is estimated. The estimation of the source term is addressed in Chapter 3 and the resulting consequences are presented in Chapter 4.

In the following subsections the configuration of the plant during POS 6 will be presented, the important characteristics of the Level 1 core damage sequences will be identified, and the development of the accident progression paths will be discussed.

2.2 POS 6 Plant Configuration

The configuration of the plant at the onset of core damage is important because it will determine the framework within which the accident will unfold. That is, the plant configuration will define the boundary conditions for the analysis. For example, it will define the mitigative features of the plant that will be available during the accident (e.g., containment, suppression pool, containment sprays).

The abridged risk analysis was performed on the early portion of the refueling mode of operation, referred to as plant operating state 6 (POS 6). During a refueling outage the plant will enter POS 6 prior to loading fresh fuel (i.e., going down) and then following fuel transfer on the way back up to power conditions (i.e., going up). In the Level 1 analysis, the sequence definitions are based on the "going down" phase because (1) more systems are likely to be unavailable (i.e., on the way back up, maintenance and repairs may already have been performed on many systems) and (2) the decay heat levels are higher and, therefore, there is less time to respond to events in the going down phase versus the going up phase. Thus, in this study only the "going down" phase is analyzed. POS 6 begins when the vessel head is detached and ends when the upper reactor cavity has been filled with water. During this POS the following tasks are performed:

1. Steam dryers are removed,
2. Vessel water level is lowered to the bottom of the steam lines and the steam lines are plugged,
3. Water level is raised and the steam separators are removed, and
4. Vessel water level is raised to flood the upper reactor cavity.

Prior to this mode of operation, the containment equipment hatch and personnel locks have been opened, the drywell head has been removed, and the drywell equipment hatch and personnel locks have been opened. Thus, the suppression pool is effectively bypassed both from the vessel and from the drywell (i.e., steam lines are plugged and the drywell is open).

Timing information for the initiation of the accident in POS 6 is based on Grand Gulf refueling outage (RFO) data. Information was available for the first four RFOs. However, because of the number of special tests that were conducted during the first refueling outage, RFO-1 was considered atypical and, therefore, data from this outage was excluded from the analysis. Thus, only RFO 2,3, and 4 data were used in this study. Based on this data the fastest the plant will enter POS 6 from full power is approximately four days after shutdown and the longest the plant has been in POS 6 (in the "going down" phase) is approximately 12 days (i.e., 16 days from shutdown). In the Level 1 analysis the time window from the initiating event to core damage was based on the decay heat at four days. This assumption is carried through the Level 2/3 analyses.

2.3 Level 1 Sequence Description

2.3.1 Sequence Description

The initial conditions for the accident progression analysis are the core damage sequence descriptions from the Level 1 analysis [8]. That is, a list of attributes that describe the status of systems that can be used to mitigate the accident and the configuration of the plant at the time of core damage. In the Level 1 coarse screening analysis the sequences were placed into three groups: potentially high likelihood group, potentially medium likelihood group, and potentially low likelihood group. Only sequences from the high likelihood group were analyzed in this study. Fourteen different initiating events are associated with these sequences. A list of these 14 initiating events is presented in Table 2.3-1. The initiating events can be divided into four major groups: Loss of Offsite Power (LOSP) Transients, Loss of Support System Transients, Loss of Coolant Accidents (LOCAs), and Decay Heat Removal Challenges. The accident sequences that form the input to this study all progress to core damage in the following manner. The initiating event leads to the loss of the operating shutdown cooling system, subsequent random failures and unavailabilities complete the loss of core cooling and injection. Without a means to keep the core cool, the vessel inventory is lost via boiling and core damage ensues.

In the Level 1 screening analysis both the emergency core cooling system (ECCS) and Makeup (i.e., CRD and CDS) were assumed to be unavailable or unable, due to some postulated failure, to prevent core damage. Thus, only the firewater system (FW) and the standby service water (SSW) cross-tie were considered as potential injection systems.

In POS 6 the suppression pool can be either at its normal level, partially drained, or empty. Furthermore, the suppression pool makeup system (SPMU) is not available. Because a supply of water to the SP is not available, ECCS systems that draw water from the SP could not be used in a continuous mode and, therefore, it was assumed in the Level 1 analysis that these systems were not available to cool the core. Because the containment spray system is one mode of the residual heat removal system (i.e., part of ECCS) and draws water from the SP, it is also unavailable during these postulated accidents.

The CRD system has insufficient capacity to prevent the core inventory from boiling and, therefore, was not considered as a means to cool the core in the Level 1 screening study. (It should be noted, however, that if this system was used, the energy removed from the core via steaming would be sufficient to prevent core damage.) While CDS has more than enough capacity to cool the core, its unavailability due to random failures and maintenance precludes its use as a means to cool the core.

A general description of the core damage sequences for each class of initiators is presented below.

LOSP Transients

The LOSP initiating event leads directly to the loss of the alternate decay heat removal system (ADHR). Subsequent random failures lead to the complete loss of shutdown cooling (SDC), makeup, the standby service water and the firewater system. With ECCS unavailable in this POS, as a result of support system failures, the accident proceeds to core damage because of the lack of core cooling.

Loss of Support System Transients

In these sequences the initiating event leads directly to the loss of ADHR, makeup, and the firewater system. Subsequent random failures lead to the complete loss of SDC and the SSW system.

Decay Heat Removal Challenges

In these sequences the initiating event leads to the loss of the operating shutdown cooling system. In some of these sequences this system is recovered. However, subsequent random failures lead to the complete loss of SDC, the firewater system, and SSW.

LOCAs That Can Be Isolated

In these sequences the isolation of the LOCA also isolates the SDC systems. Subsequent random failures lead to the loss of both the firewater system and the standby service water cross-tie system.

Table 2.3-1 Grand Gulf LP&S POS 6 Initiating Events

Initiating Event Group	Initiating Event Nomenclature	Description
LOSP	T1	Loss of Offsite Power (LOSP) Transient
Loss of Support System Transient	T5B	Loss of all TBCW
	T5C	Loss of all PSW (includes Radial Well)
	TIA	Loss of all Instrument Air
Decay Heat Removal Challenge	E1B	Isolation of SDC Loop B only
	E2B	Loss of SDC Loop B only
	E1D	Isolation of ADHRS
	E2D	Loss of ADHRS only
	E1T	Isolation of SDC Common Suction Line
	E2T	Loss of SDC Common Suction Line
	E1V	Isolation of Common Suction Line for ADHRS
	E2V	Loss of Common Suction Line for ADHRS
Isolated LOCAs	H1	Diversion to Suppression Pool via RHR
	J2	LOCA in Connected System (RHR)

2.3.2 Plant Damage State Description

The Level 1 sequences were divided into two plant damage state (PDS) groups: LOSP and nonLOSP. This distinction is made because of the effect that the LOSP has on injection recovery and containment closure. In the analysis of the nonLOSP PDS it is assumed that if injection is not recovered prior to core damage, it will not be recovered during core damage. The reason for this assumption is that there is a considerable amount of time from the initiating event to core damage for the operators to align and use injection systems to cool the core. If this has not been done by the time of core damage, there is no reason to believe that they will recover core cooling during core damage. Recovery of injection is considered in the LOSP PDS. In these sequences offsite power is unavailable and, therefore, non-emergency systems are unavailable to provide injection to the core. Thus, for the LOSP PDS it is assumed that if offsite power is recovered, injection can be recovered. The availability of ac power also affects the likelihood that the containment is closed prior to core damage. The crane that is used to position the equipment hatch is powered with offsite ac power and, therefore, without offsite ac power the containment cannot be closed. If offsite power is available during the accident, closure of the containment prior to core damage is addressed in the event tree analysis. The key attributes associated with these two PDSs are presented in Table 2.3-2.

Table 2.3-2 Grand Gulf LP&S POS 6 Plant Damage State Attributes

PDS Attributes	Plant Damage States (PDS)	
	LOSP	nonLOSP
Offsite Power	Not Available	Available
Vessel Head	Off	Off
Containment Integrity	Open	Open
Drywell Integrity	Open	Open
Suppression Pool Makeup	Not Available	Not Available
Containment Sprays	Not Available	Not Available
Containment Closure Possible?	No	Yes
Injection Recovery Possible?	Yes	No

From Table 2.3-2 it can be seen that the main differences between the LOSP and nonLOSP PDSs are 1) the containment can be closed only in the nonLOSP PDS and 2) injection can be recovered only in the LOSP PDS. Because sequence frequencies are unavailable from the Level 1 screening analysis, the relative likelihood of the two PDSs is unavailable. The remaining analysis that is presented in this report is conditional on the occurrence of these PDSs.

2.4 Event Tree Analysis

A simplified APET was used in this analysis to delineate and quantify the likelihood of the possible accident progression paths. The selection of events to include in the accident progression analysis was based on (1) insights gained from the NUREG-1150 full power PRAs [1,9], (2) results from MELCOR calculations specifically performed for this analysis, and (3) the plant configuration during POS 6. Events deemed important for inclusion in the APET were events that related to containment performance and the estimation of the radionuclide release.

The APET addresses three general time regimes: prior to core damage, during core damage, and following vessel failure. In the first time regime the issue of containment closure is addressed. Injection recovery, core damage arrest, in-vessel steam explosions and early containment failure are all addressed in the second time regime. The characteristics of the interaction between the core debris release from the vessel and the reactor pedestal are addressed in the last time regime.

The times associated with these time regimes are based on results from a series of MELCOR calculations that were performed to support this analysis. The timing of key events in the accident progression analysis is presented in Table 2.4-1.

Table 2.4-1 Accident Progression Timing

Calculation	Timing of Key Events from Initiation of Accident (hours)				
	Time to TAF	Core Damage	Vessel Failure	Aux. Bldg Failure	Contain. failure
PRA MODEL INPUT					
PRA Model: Containment Open	13.0	18.3	25.4	21.1	Cnt Open(2)
PRA Model: Containment Fails	13.0	19.4	28.6	No Fail. (3)	30.
MELCOR RESULTS					
Base Case (BC)-No Aux Bldg	12.7	18.3	25.4	(1)	(2)
BC w/ Small Aux. Bldg	13.0	18.8	24.5	21.6	(2)
BC w/ Big Aux. Bldg	13.0	18.8	28.6	28.6	(2)
BC w/ Containment Closed	13.6	19.4	28.6	(1)	22 - 80.
BC initiated 15 days after SD	19.7	28.3	39.8	(1)	(2)

Notes:
1. Auxiliary building model not included
2. Containment is open during the accident
3. Containment failure bypasses the auxiliary building
4. MELCOR POS 6 BC Calculation:
   - Accident Initiated 4 days after shutdown
   - Containment is open (i.e. equipment hatch and both personnel locks)
   - Injection, shutdown cooling, and containment sprays are all unavailable
5. Core damage is defined as the first gap release
6. TAF = Collapsed water level at the top of the active fuel

In this table both the times estimated with MELCOR and the times assumed in this PRA are presented. From this table it is apparent that the timings of these accidents are quite different from accidents initiated at full power. For example, it takes approximately 18 hours to progress from the initiation of the accident to the onset of core damage. In comparison, a fast station blackout initiated from full power progresses to a similar point in approximately 1 hour. Another notable entry in this table is the predicted time of auxiliary building failure for cases with the containment open. The building is predicted to overpressurize and fail from the accumulation of steam and noncondensibles during core damage. The exact timing of building failure depends on the volume assumed for the auxiliary building (i.e., various rooms in the building can isolated) and the building failure pressure. For this abridged study, the auxiliary building is estimated to fail approximately half way through the core damage process.

Nine events are used to characterize the accident progression. A graphical depiction of the APET is presented in Figure 2.4-1. The first nine paths are associated with the LOSP PDS and the remaining 7 paths (i.e., paths 10 through 16) are associated with the nonLOSP PDS. The mean probability for each path is also presented in this figure. The path probabilities for each PDS sum to 1.0. The nine events and a brief description of each event are presented below.

1. **Is the containment closed prior to core damage?**

   The containment equipment hatch has been removed prior to entry into POS 6. For the LOSP PDS the lack of offsite ac power precludes containment closure prior to core damage. However, for the nonLOSP PDS it is possible that the plant personnel will close the containment after the initiation of the accident but prior to core damage. The containment can be closed if the operators recognize that a problem exists early in the accident and decide that containment closure would be prudent. Because it takes between 8 to 12 hours to completely close the hatch, it is necessary that the operators begin the closure tasks within the first few hours of the accident. The equipment hatch is a pressure seating hatch which requires the personnel closing the hatch to be in the containment. Thus, the environment in the containment during the boiloff is an important parameter that will affect the personnel's ability to close the containment. MELCOR calculations performed for this analysis indicate that the temperatures in the containment during this phase of the accident will be high (i.e., range from 100 to 140 degrees F) but not so high that it would preclude the personnel from carrying out their tasks. It was also assumed that the radiological environment in the containment will not preclude the closure tasks from being performed. These assumption will have to be verified in future analysis. In this analysis it was assumed that the containment was habitable up until the time of core uncovery (i.e., approximately 13 hours).

2. **If the containment is closed prior to core damage, does it fail prior to vessel failure?**

   The Grand Gulf plant utilizes a Mark III containment to house its BWR-6 reactor. The containment has a volume of 1.6 million cubic feet and a design pressure of 15 psig. The mean estimated failure pressure is 56 psig [9]. Since the containment has a relatively low failure pressure, the pressure rise from the accumulation of steam and noncondensibles can pose a threat to the containment integrity. Actions must also be taken to prevent the combustion of large quantities of hydrogen. Containment venting was not considered in this analysis as a means to control pressure because venting would still result in an open containment. In POS 6 the suppression pool is bypassed and, therefore, the steam and noncondensibles are released directly into the containment atmosphere. Furthermore, the containment sprays are not available. Thus, the containment will pressurize during the core damage process. The peak pressure during this phase of the accident depends on the steam generation rate, the condensation rate in the containment, and the presence and magnitude of hydrogen burns. MELCOR calculations indicate that the containment pressure can exceed the lower range of the containment failure pressure distribution if a burst of steam occurs at the time of vessel failure or if discrete hydrogen burns (not diffusion flames) occur during the core damage phase of the accident. Because steam and hot hydrogen are released directly into the containment in this POS, the effectiveness of the HIS to control the accumulation of hydrogen is uncertain. Thus, it is possible that for some accident scenarios the containment will fail early in the accident. If the containment does not fail early, calculations indicate that it will take several days to reach the mean estimated failure pressure of 56 psig. Therefore, it was assumed that if the containment does not fail early, it will not fail in the time frame of this analysis.

3. **If the containment fails, is the failure in the form of a leak or rupture?**

   The failure size will determine how fast the radionuclides are released from the containment and the amount of radionuclides deposited within the containment.

4. **Is the auxiliary building bypassed?**

   This question distinguishes the accidents in which the releases pass through the auxiliary building from those accidents which result in a release from the containment directly into the environment. The release path is important because it will affect the amount of mitigation that the release experiences before entering the environment. Accidents in which the containment equipment hatch is off will result in a release that passes through the auxiliary building; accidents in which the containment fails bypass the auxiliary building. Based on previous structural analysis of the Grand Gulf containment, it was concluded that the most likely location for failure is the region near the junction of the dome and the cylindrical wall [9]. A failure in this location will result in a release to the enclosure building that surrounds the containment dome. The enclosure building has virtually no pressure retaining capability and is essentially isolated from the auxiliary building. Therefore, it is assumed that following containment failure, the release goes directly from the containment into the environment. The retention in the containment will be fairly small in this case because the containment fails early in the accident. The result will be essentially an unmitigated release. If the containment is open to the auxiliary building, the majority of the radionuclides will quickly enter the auxiliary building and the retention in the containment will be small. For these accidents, the only significant mitigation feature will be the auxiliary building which acts as a large holdup volume allowing time for natural processes to remove radionuclides from the building atmosphere before being released into the environment. The auxiliary building is predicted to overpressurize and fail from the accumulation of steam and noncondensibles during core damage. The exact timing of building failure depends on the volume of the auxiliary building that will pressurize (i.e., some rooms within the building can be isolated and therefore will not pressurize) and the building failure pressure. For this abridged study, the auxiliary building is estimated to fail approximately halfway through the core damage process.

5. **Is injection recovered prior to vessel failure?**

This question is used to identify those accidents in which injection is restored to the vessel during the core damage process. The recovery of injection allows for the possibility that the core damage process will be arrested in the vessel (i.e., prevent vessel failure). Injection can only be recovered for the LOSP PDS. The probability that injection is recovered is based on the probability that offsite ac power is recovered during core damage.

6. **If injection is recovered, when is it recovered?**

The timing of injection recovery during core damage affects the likelihood that the core damage process will be arrested before the vessel fails. For this analysis, the in-vessel phase of the accident (i.e., core damage) has been divided into three time regimes: very early, early, and late. The very early time regime ranges from the initiation of core damage to the onset of autocatalytic oxidation. If injection is recovered during this phase of the accident the core damage process will be arrested in the vessel and the releases will be limited to the inventory in the gap. The early time regime ranges from onset of autocatalytic oxidation to 30% core damage. Based on extrapolation of analysis performed in NUREG-1150, if injection is recovered before 30% of the core has been damaged, it is very likely that the core damage process can be arrested. Because MELCOR calculations indicate that core damage progresses rapidly from 30% to full core damage, the late time regime is defined as 30% core damage to vessel failure. Recovery of injection during this phase of the accident will not prevent vessel failure. The time windows for each of these time regimes is based on results from MELCOR calculations. The possibility of the reactor going critical following the restoration of injection was not addressed in this abridged analysis.

7. **Does an in-vessel steam explosion occur during core damage?**

In-vessel steam explosions are treated in a very limited fashion in this abridged analysis. A primary motivation for including this question in the APET is to highlight the fact that in-vessel steam explosions are possible. The effect of the steam explosion on the accident progression can be quite different from in-vessel steam explosions that occur at full power because the steam and radionuclides that are generated during this event are released directly into the containment atmosphere. In this analysis the treatment of in-vessel steam explosions was limited to the estimation of the source term that is associated with the debris that participates in the steam explosions. Neither the pressure loading from in-vessel steam explosions nor the relocation of intact fuel from the steam explosion was addressed in this study. Both issues were beyond the scope of this abridged study. Ex-vessel steam explosions were not considered in this analysis because the pedestal cavity below the vessel will be essentially dry at the time of vessel failure.

8. **Is the core damage process arrested in the vessel?**

   This question addresses the coolability of the core debris following injection recovery. If the core damage process is arrested before the vessel fails, the core debris will remain in the vessel and core-concrete interactions (CCI) will be prevented. Because only a portion of the core is damaged and CCI is prevented, the source term associated with recovered accidents is typically less than the source term associated with full core damage accidents. If injection is not restored during core damage, the accident always progresses to vessel failure and the core debris relocates to the pedestal cavity below the vessel. The likelihood that the core damage process is arrested before vessel failure depends on when injection is restored during the core damage process (see question 6). If injection is restored during either the very early or early time regimes, analysis indicates that it is very likely the core damage process will be arrested. If, on the other hand, injection is not restored until the late time regime, it is very likely that the vessel will fail and the core debris will relocate to the pedestal cavity.

9. **Do core-concrete interactions occur following vessel failure?**

   Core-concrete interactions consist of the thermal and chemical interactions between the core debris and the concrete pedestal. During this process the concrete is eroded and gases and radionuclides are released from the core/concrete mixture. For the accidents analyzed in this study, the vessel will fail and the core debris will enter the cavity if 1) injection is not restored to vessel during core damage or 2) injection is restored during the late time regime. The presence of water can affect CCI in two different ways. First, water can quench the debris and prevent CCI. Second, if the debris is not quenched, the overlying pool of water will retain some of the radionuclides released during CCI and thus, tend to mitigate the release. Thus, for the accidents in which injection is restored but the vessel still fails, there is some probability that the core debris will be quenched and CCI will be prevented. The probability of this occurring is based on information from the NUREG-1150 study [9]. If injection is not restored during core damage, CCI will always proceed in a dry cavity.

From inspection of Figure 2.4-1 it can be seen that there are several important differences between the LOSP PDS and the nonLOSP PDS. In the LOSP PDS injection can be recovered allowing for the possibility to arrest the core damage process in the vessel. If the vessel does fail, it is still possible to quench the core debris in the cavity (i.e., no CCI). Thus, in many of the LOSP accidents the ex-vessel radionuclide release is prevented. The containment, however, cannot be closed in the LOSP PDS and, therefore, the releases always pass into the auxiliary building and then out into the environment. In the nonLOSP PDS, the containment can be closed, however, injection cannot be recovered. Thus, all of the nonLOSP accidents identified in the APET progress to full core damage, vessel failure, and involve CCI. In some of the scenarios the containment is closed. However, because containment cooling (i.e., containment sprays) is unavailable and the suppression pool is bypassed, even if the containment is closed it is possible that it will fail early in the accident from pressure transients associated with events accompanying vessel failure and hydrogen combustion. Based on information from NUREG-1150, it is expected that the containment will fail above the auxiliary building roof. Thus, the releases from the containment will enter the environment without first going through the auxiliary building. Because so many of the mitigative features of the plant are bypassed in this POS (e.g., suppression pool, containment sprays, containment), the auxiliary building plays an important role in reducing the amount of radionuclide material that is released into the environment. Thus, for the nonLOSP accidents there are two extremes: 1) if the containment is closed and remains intact, the releases to the environment are expected to be very small and 2) if the containment fails, the releases to the environment are expected to be quite large because all of the accidents involve full core damage and CCI and the releases bypass the auxiliary building. The scenarios in which the containment is not closed are very similar to the LOSP accidents in which injection is not recovered.

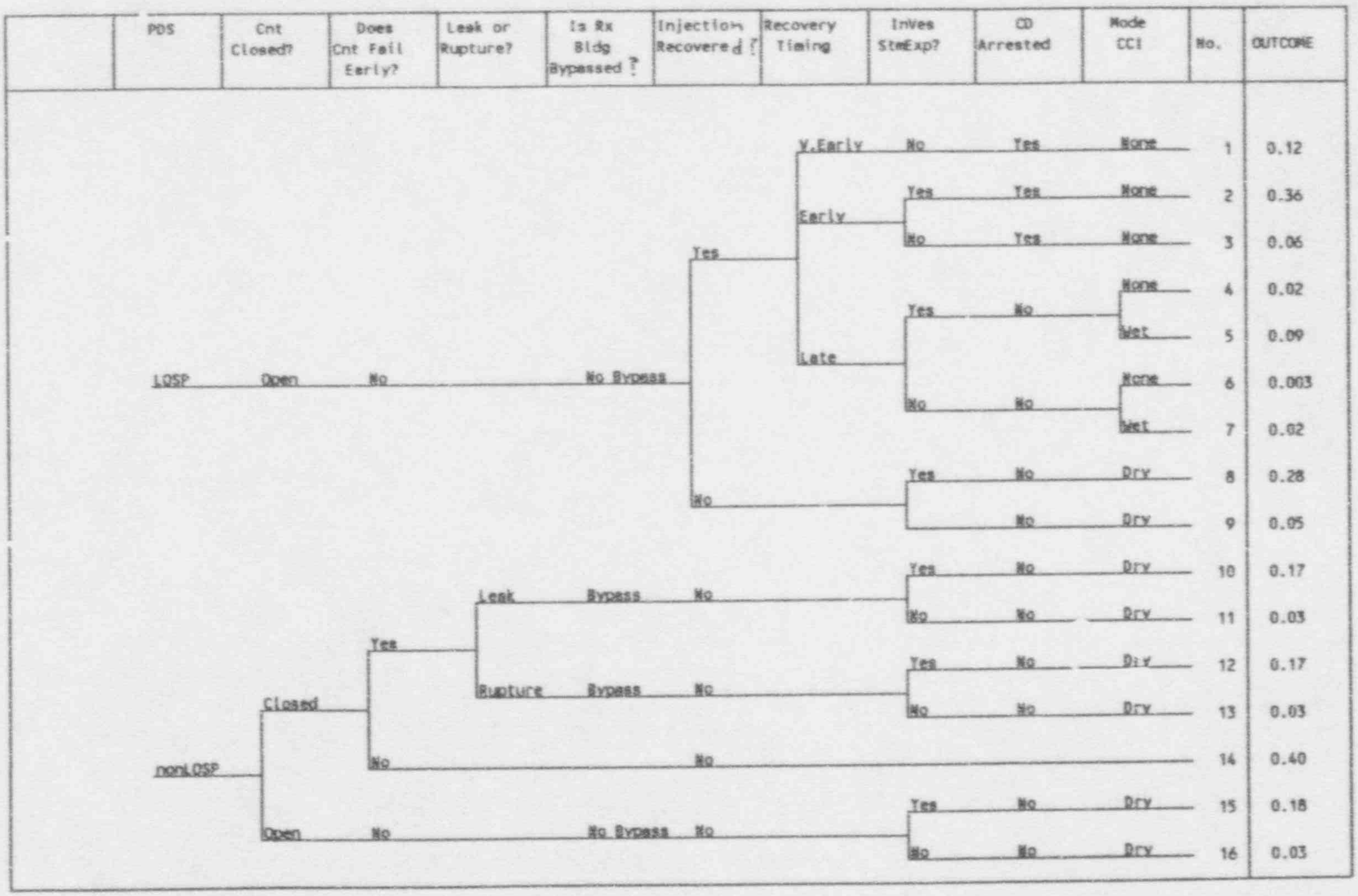


Figure 2.4-1. Grand Gulf POS 6 abridged accident progression event tree (APET)

3.0 SOURCE TERM ANALYSIS

3.1 Approach

A source term is estimated for each accident progression path identified in the APET (see Figure 2.4-1). The simple parametric source term approach that was used in NUREG-1150 to estimate source terms is used in this study. The parametric source approach is used because 1) information from a wide variety of sources can be used in the model, 2) it is easily incorporated into uncertainty analysis, and 3) thousands of source terms can be estimated with this model in a very efficient manner. The parametric source term code GGSOR that was developed in NUREG-1150 [9] was modified for this analysis. The modified parametric code is called GGLPSOR. Modifications were made to the code to incorporate the unique plant configuration associated with accidents initiated in POS 6. Wherever possible, data from NUREG-1150 was used to quantify the model. Results from MELCOR were compared with both the input distributions and the final source terms to verify that distributions developed for full power accidents could be applied to shutdown accidents.

A limited uncertainty analysis was performed in this section of the analysis. For each accident progression path, the model was repeatedly exercised with different combinations of selected input variables. The distributions for these input variables were obtained, when applicable, from NUREG-1150.

3.2 Description of Parametric Model

The parametric source term model GGSOR that was developed in NUREG-1150 was modified to account for unique features of POS 6 that have a strong impact on the source term. In POS 6 both the drywell head and the vessel head have been removed and the steam lines have been plugged. Thus, during the core damage process radionuclides released from the core debris will bypass the suppression pool and directly enter the containment. Furthermore, because most of the internal structures above the core (e.g., steam dryers and separators) have been removed and the steam lines are plugged, there is very little deposition of radionuclides in the vessel. Thus, the mitigative features of both the vessel and the suppression pool, which are present in many full power accident scenarios, are absent in this POS. For scenarios in which the containment hatch is open, the residence time of the radionuclides in the containment atmosphere is fairly short and, thus, there will be limited deposition (i.e., from gravitational settling) of radionuclides in the containment. In this POS the drywell is open to the containment (i.e., the drywell hatch is open) and, therefore, an ex-vessel release will also bypass the suppression pool. For these accidents, the containment sprays are not available and cannot be used to scrub the releases. Thus, the mitigative features of the vessel, suppression pool, containment sprays, and possibly the containment, are bypassed or unavailable.

For scenarios with the containment open, the only major mitigative feature of the plant is the auxiliary building. The auxiliary building encompasses a very large volume and, therefore, acts as a hold up volume for the radionuclides which allows time for the radionuclides to deposit on surfaces within the building. The auxiliary building can play an important role in POS 6 because so many of the other mitigative features of the plant are absent and the characteristics of the radionuclide transport to the auxiliary building are different from the transport associated with full power accidents. In full power accidents the containment pressurizes to the ultimate failure pressure and then blows down into the auxiliary/reactor building (i.e., Peach Bottom analysis in the NUREG-1150 study). Following containment failure the auxiliary building rapidly pressurizes and fails (the failure pressure of the auxiliary building is only a few psi). Thus, the releases are swept through the auxiliary building fairly rapidly. In the POS 6 accident scenarios the steam and radionuclides are released to the auxiliary building much more slowly allowing more time for condensation and deposition. The scenarios that involve containment closure followed by containment failure will not benefit from the auxiliary building because the containment failure location is assumed to be above the roof of the auxiliary building [9]. Thus, the releases will bypass the auxiliary building essentially resulting in an unmitigated release.

Neither the normal ventilation system nor the standby gas treatment system (SBGT) were modeled in this analysis. The filters and charcoal beds in the SBGT system could act to mitigate the release or at least delay the release of radionuclides. Before credit can be given to this system, the capacity of the system and the performance of the filters under severe accident conditions will have to be addressed. The analysis of this system was beyond the scope of this study.

3.3 Results

A source term is estimated for each path through the APET. In addition, because an uncertainty analysis was performed, a distribution of source terms is available for each path. For the sake of brevity, only the mean source terms, expressed as fractions of the core inventory, that enter the environment are presented in Table 3.3-1. When reviewing this table, it must be remembered that the initial inventory of radionuclides four days after shutdown is different from the inventory typical of full power accidents.

Inspection of Table 3.3-1 confirms that many of the releases are essentially unmitigated and, therefore, are quite large. Table 3.3-1 also highlights some of the differences between the various accident scenarios (i.e., paths). Paths 1 through 3 correspond to accidents in which injection is recovered early in the accident and the core damage process is arrested in the vessel. Thus, because only a portion of the core is damaged and there are no ex-vessel releases (i.e., no CCI), the source terms associated with these accidents are relatively small compared to the other source terms presented in this table. The notable exception is Path 14 which corresponds to the scenario in which the containment is closed prior to core damage and remains intact throughout the accident. Because the containment remains intact, only nominal leakage occurs and the resulting source term is quite small. Paths 4 through 9, on the other hand, correspond to full core damage accidents that have the containment open to the auxiliary building. The source terms associated with Paths 4 and 6 tend to be lower than the other full core damage source terms because the core debris is quenched in the pedestal cavity and, therefore, there are no releases associated with CCI. This difference is fairly minor, however, and the fact still remains that these are large source terms. Paths 10 through 13 are nonLOSP accidents in which the containment fails around the time of vessel failure. All of these accidents progress to full core damage and CCI. The containment fails via a leak in Paths 10 and 11; the containment ruptures in Paths 12 and 13. In all four of these scenarios the containment fails directly to the environment (i.e., the auxiliary building is bypassed). The source terms associated with the leak failure mode are similar to the source terms when the release passes through the auxiliary building. In the leakage cases, the radionuclides are held up in the containment for a period of time thus allowing a fraction of the radionuclides to settle out of the containment atmosphere. For the rupture cases, however, the containment quickly depressurizes following containment failure and considerably less deposition occurs. Thus, the source terms associated with the rupture cases are quite large. Paths 15 and 16 correspond to the nonLOSP cases where the containment is not closed prior to core damage and the radionuclides pass through the auxiliary building. These source terms are essentially the same as the LOSP full core damage source terms.

Table 3.3-1 Mean Source Terms for Accident Progression Paths (Total Release)

Path	Radionuclide Release Classes									Timing of Release (hr.s)			
	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba	TW	T1	DT1	DT2
	LOSP PDS												
1	0.015	0.002	5.9E-3	1.2E-5	0.0	0.0	0.0	0.0	1.2E-7	16.3	21.1	24.0	0.0
2	0.072	0.012	0.011	6.3E-3	2.1E-3	3.3E-4	1.4E-4	6.7E-4	2.2E-3	16.3	21.1	4.3	0.0
3	0.072	0.012	0.011	6.3E-3	2.1E-3	3.3E-4	1.4E-4	6.7E-4	2.2E-3	16.3	21.1	4.3	0.0
4	0.79	0.17	0.15	0.085	0.027	0.012	3.0E-3	8.7E-3	0.033	16.3	21.1	4.3	10.0
5	1.0	0.25	0.19	0.11	0.042	0.012	4.0E-3	0.011	0.047	16.3	21.1	4.3	10.0
6	0.74	0.15	0.13	0.075	0.022	4.9E-3	1.5E-3	7.1E-3	0.026	16.3	21.1	4.3	10.0
7	1.0	0.25	0.18	0.11	0.041	4.9E-3	2.7E-3	9.6E-3	0.042	16.3	21.1	4.3	10.0
8	1.0	0.25	0.25	0.16	0.08	0.012	6.9E-3	0.012	0.084	16.3	21.1	4.3	10.0
9	1.0	0.25	0.25	0.17	0.088	5.4E-3	6.3E-3	0.011	0.089	16.3	21.1	4.3	10.0
	nonLOSP PDS												
10	1.0	0.27	0.27	0.18	0.094	0.013	6.8E-3	0.011	0.083	17.4	30.0	2.0	10.0
11	1.0	0.28	0.28	0.19	0.10	6.1E-3	6.2E-3	0.011	0.086	17.4	30.0	2.0	10.0
12	1.0	0.62	0.63	0.40	0.22	0.029	0.016	0.027	0.19	17.4	30.0	0.05	10.0
13	1.0	0.62	0.63	0.43	0.24	0.013	0.014	0.025	0.20	17.4	30.0	0.05	10.0
14	5.0E-3	4.1E-7	4.1E-7	2.9E-7	1.4E-7	9.4E-9	9.7E-9	1.9E-8	1.4E-7	16.3	21.1	4.3	10.0
15	1.0	0.25	0.25	0.16	0.08	0.012	6.9E-3	0.012	8.4E-2	16.3	21.1	4.3	10.0
16	1.0	0.25	0.25	0.17	0.088	5.4E-3	6.3E-3	0.011	0.089	16.3	21.1	4.3	10.0

Notes:

1. TW = Warning Time
2. T1 = Timing of first release
3. DT1 = Duration of first release
4. DT2 = Duration of second release (start immediately after first release ends)

4.0 CONSEQUENCE ANALYSIS

The consequences of a severe accident during POS 6 were calculated as part of the abridged study. As is typically done, the offsite consequences were estimated. The onsite doses were also estimated, which is not typically done.

An important difference between this analysis and those previously performed for full power accidents is that the radionuclides in the fuel have had at least four days to decay resulting in a different inventory than that present at shutdown. ORIGEN2 [10] was used to calculate the inventory in three different fuel assemblies, one which had been irradiated for three fuel cycles, one which had been irradiated for two fuel cycles, and one which had been irradiated for one fuel cycle. All fuel assemblies were then allowed to decay for four days. Based on information from plant personnel, a fuel cycle consisted of 540 days of irradiation and 55 days of decay. The inventory for the whole core four days after shutdown was then summed. This inventory, which was reduced to include only the sixty radionuclides currently available in the MACCS code [5,6,7], was then used as the basis for both the onsite and offsite consequence calculations. This inventory, which does not include short-lived radionuclides, is appropriate for both the onsite and offsite analyses since the reactor has been in shutdown for at least four days at the beginning of the accident thus allowing decay of the short-lived radionuclides.

The following sections detail the methodology and results for the onsite consequences, both in the buildings and in the parking lot, and the offsite consequences.

4.1 Onsite Consequences

Onsite consequences have seldom been considered in the analysis of severe accidents at nuclear power plants. During shutdown there will be hundreds of onsite personnel and, thus, onsite consequences could be large. For this reason a method for estimating the potential doses to onsite personnel had to be developed as part of this study. The primary simplifying assumption of the analysis was that radioactive decay was neglected during the exposure time. This assumption is justified by the fact that the accident under analysis typically occurs no earlier than four days after shutdown by which time the decay heat curve is fairly flat. Other assumptions were employed in the two aspects of the onsite consequences: (1) in building doses and (2) parking lot doses. The method, assumptions, and results of the analyses are discussed in the following two sections.

4.1.1 Building Doses

The onsite consequences for POS 6 were estimated based on the source terms to both the containment and the auxiliary building that were determined with the parametric source term code, GGLPSOR. However, since GGLPSOR calculates integral releases, the time dependence of the two release segments of the source terms was determined from MELCOR calculations. Three different sets of residence times (i.e., estimated time airborne material spends in the building) were used based on the status of the containment. The first set of residence times was used if the containment was open to the auxiliary building at the time of the accident. The residence times through both buildings were based on a MELCOR calculation modeling this scenario. The residence time of the radioactive material in each building was directly proportional to the volume of that building. The second set of residence times was used if the containment ruptured directly to the environment. For this case, the same residence times were used as in the previous scenario, however, the residence time in the auxiliary building was set to zero. In other words, the amount of time the material spent in the containment was the same for both of these scenarios, but in the latter scenario the material did not pass through the auxiliary building. The third set of residence times was used if the containment leaked directly to the environment. In this case, the residence time for the first release was increased by two hours, and again the residence time in the auxiliary building was set to zero. The residence times used under the various conditions are summarized in Table 4.1.1-1.

Table 4.1 1-1. Residence times through the containment and auxiliary building for Grand Gulf POS 6.

Accident Progression Path Number	Containment Residence Time: First Segment (hours)	Containment Residence Time: Second Segment (min)	Auxiliary Building Residence Time: First Segment (hours)	Auxiliary Building Residence Time: Second Segment (hours)
	LOSP PDS			
Paths 1-9	3.4	47	6.1	1.4
	nonLOSP PDS			
Path 10	4.1[1]	47	0.0	0.0
Path 11	4.1	47	0.0	0.0
Path 12	3.4	47	0.0	0.0
Path 13	3.4	47	0.0	0.0
Path 14[1]	NC	NC	NC	NC
Path 15	3.4	47	6.1	1.4
Path 16	3.4	47	6.1	1.4

[1] Building doses were not calculated since the containment is not open.

To estimate the doses in the buildings, the average release fraction of each chemical group was determined for each building. The integrated concentration of each radionuclide in the buildings was then based on the average release fraction of its chemical group and the amount of time spent in that building. Using the integrated concentration for each radionuclide, the immersion and 50 year committed inhalation dose were calculated over the entire exposure time. In addition, the immersion and 50 year committed inhalation dose were calculated for the first 30 minutes of exposure. These doses should be viewed with caution since the integrated concentration in the building was based on an average concentration in the building and therefore the time dependence of the dose is not well represented. The final result estimated in the buildings was a dose rate. These results should also be viewed with caution since they are also based on average concentrations in the building. In addition, the dose rates were calculated by dividing the total dose during a release segment by the transit time through the building. This results in a conservative estimate of the inhalation dose rate. The mean dose due to the entire release, the first 30 minutes of exposure, and the mean dose rates during the first and second release segments in the containment are shown in Table 4.1.1-2 for each of the paths through the APET. Similar estimates are shown in Table 4.1.1-3 for the auxiliary building.

Table 4.1.1-2. Grand Gulf POS 6 mean containment doses and dose rates.

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Total Dose (rem)	30 minute Dose (rem)	Dose Rate First Segment (rem/hr)	Dose Rate Second Segment (rem/hr)
		LOSP PDS			
Path 1	0.10	1.81E+6	2.69E+5	5.38E+5	0.0
Path 2	0.48	4.27E+7	6.35E+6	1.27E+7	0.0
Path 3	0.08	4.27E+7	6.35E+6	1.27E+7	0.0
Path 4	0.02	5.38E+6	7.95E+7	1.59E+8	0.0
Path 5	0.09	5.69E+8	7.95E+7	1.59E+8	4.04E+7
Path 6	0.003	4.27E+8	6.35E+7	1.27E+8	0.0
Path 7	0.01	4.66E+8	6.35E+7	1.27E+8	5.05E+7
Path 8	0.18	6.16E+8	7.95E+7	1.59E+8	1.01E+8
Path 9	0.03	5.25E+8	6.35E+7	1.27E+8	1.26E+8
		nonLOSP PDS			
Path 10	0.16	5.78E+8	6.35E+7	1.27E+8	7.74E+7
Path 11	0.02	4.88E+8	5.05E+7	1.01E+8	9.68E+7
Path 12	0.16	6.16E+8	7.95E+7	1.59E+8	1.01E+8
Path 13	0.02	5.25E+8	6.35E+7	1.27E+8	1.26E+8
Path 14[1]	0.37	NC	NC	NC	NC
Path 15	0.23	6.16E+8	7.95E+7	1.59E+8	1.01E+8
Path 16	0.04	5.25E+8	6.35E+7	1.27E+8	1.26E+8

[1] Building doses were not calculated since the containment is not open.

Table 4.1.1-3. Grand Gulf POS 6 mean auxiliary building doses and dose rates.

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Total Dose (rem)	30 minute Dose (rem)	Dose Rate First Segment (rem/hr)	Dose Rate Second Segment (rem/hr)
		LOSP PDS			
Path 1	0.10	9.41E+5	7.70E+4	1.54E+5	0.0
Path 2	0.48	2.13E+7	1.74E+6	3.47E+6	0.0
Path 3	0.08	2.13E+7	1.74E+6	3.47E+6	0.0
Path 4	0.02	2.80E+8	2.28E+7	4.56E+7	0.0
Path 5	0.09	2.98E+8	2.28E+7	4.56E+7	1.31E+7
Path 6	0.003	2.20E+8	1.79E+7	3.59E+7	0.0
Path 7	0.01	2.43E+8	1.79E+7	3.59E+7	1.64E+7
Path 8	0.18	3.23E÷8	2.28E+7	4.56E+7	3.06E+7
Path 9	0.03	2.74E+8	1.79E+7	3.59E+7	3.83E+7
		nonLOSP PDS			
Path 10	0.16	0.0	0.0	0.0	0.0
Path 11	0.02	0.0	0.0	0.0	0.0
Path 12	0.16	0.0	0.0	0.0	0.0
Path 13	0.02	0.0	0.0	0.0	0.0
Path 14[1]	0.37	NC	NC	NC	NC
Path 15	0.23	3.23E+8	2.28E+7	4.56E+7	3.06E+7
Path 16	0.04	2.74E+8	1.79E+7	3.59E+7	3.83E+7

[1] Building doses were not calculated since the containment is not open.

To illustrate the uncertainty in the dose rate in the containment and the auxiliary building due to the uncertainty in the source term, the 5th, 50th, and 95th percentile dose rates as well as the mean dose rate for two pathways through the APET are shown in Figure 4.1.1-1. The first of these paths represents a scenario in which injection is recovered very early in the accident, thus arresting core damage. Note that in the recovered accident, CCI does not occur therefore the source term consists of only one segment and only one dose rate was calculated. The second path represents a scenario in which full core damage occurs.

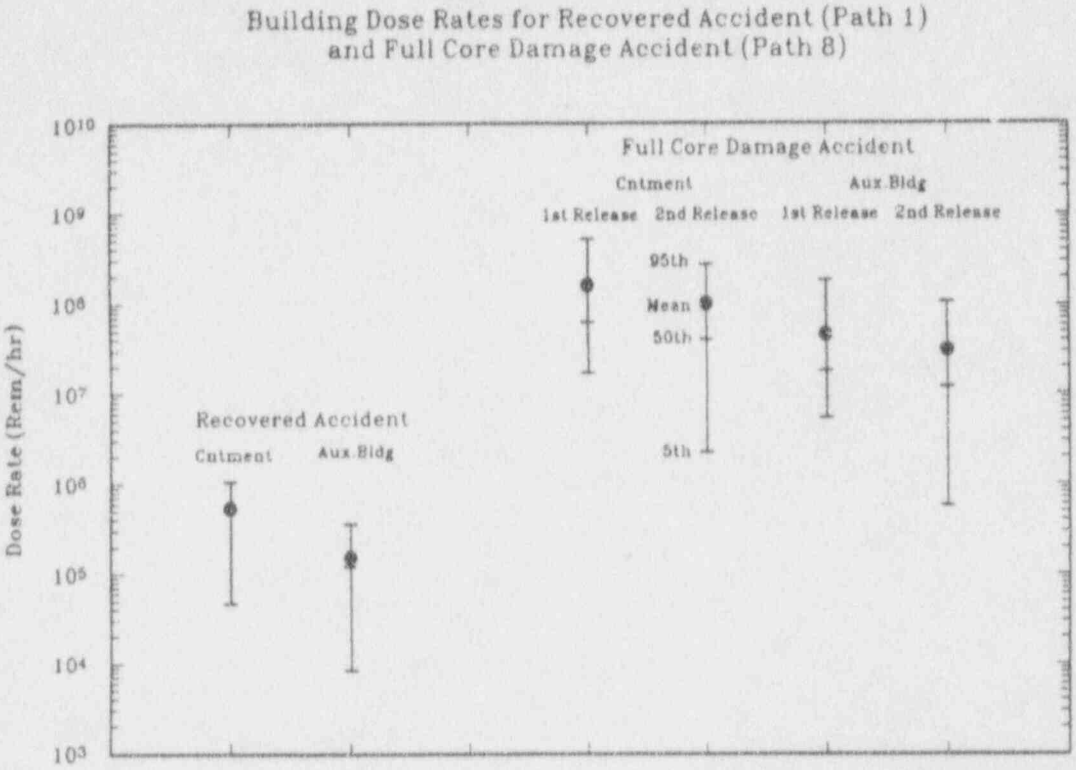


Figure 4.1.1-1. Containment and auxiliary building dose rates for selected paths

4.1.2 Parking Lot Doses

The dose due to immersion and inhalation was also estimated for several distances from the reactor. The source terms were obtained from the parametric source term code, GGLPSOR. In contrast to the building doses, the timing of the source terms was taken directly from GGLPSOR. For comparative purposes, three different wake effect models were used to estimate the relative concentrations downwind of the reactor. These models were developed by Ramsdell [11], Wilson [12], and the NRC [13]. For simplicity, the directional dependence of the weather was ignored and doses were calculated for several distances from the reactor. The weather used in each of the wake effect models was chosen to represent conservative values for the model. In the case of the Ramsdell model the relative concentration is somewhat proportional to the wind speed and the stability class. For this reason the highest wind speed and the corresponding stability class in a year of weather data at Grand Gulf was chosen as input to this model. In addition, the relative concentration is predicted to be somewhat inversely proportional to the area of the building, therefore, the minimum area was utilized. In the case of the Wilson and NRC models the relative concentration is predicted to be inversely proportional to the wind speed.

Therefore, a wind speed of 1 m/s and a stability class of F (i.e., moderately stable meteorological conditions) were used in these models. Using the integrated air concentrations for each building wake effect model, the dose and dose rate due to immersion and inhalation for the entire source term was determined for each of the unique accident progression paths. As with the building dose rates, the dose rates in the parking lot are very conservative since the inhalation dose rate was determined by dividing the 50 year committed dose by the exposure time. The dose due to 30 minutes of exposure was also estimated. Table 4.1.2-1 contains the mean total dose, 30 minute dose, and dose rates for each segment of the release based on the Ramsdell building wake effect model at 100 meters from the reactor. Similar estimates of the mean doses and dose rates at 100 meters based on the Wilson model is shown in Table 4.1.2-2.

Table 4.1.2-1. Grand Gulf POS 6 mean doses and dose rates at 100 m based on the Ramsdell building wake effect model.

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Total Dose (rem)	30 minute Dose (rem)	Dose Rate First Segment (rem/hr)	Dose Rate Second Segment (rem/hr)
		LOSP PDS			
Path 1	0.10	4.23E+2	8.80	17.6	0.0
Path 2	0.48	9.42E+3	1.09E+3	2.19E+3	0.0
Path 3	0.08	9.42E+3	1.09E+3	2.19E+3	0.0
Path 4	0.02	1.32E+5	1.53E+4	3.06E+4	0.0
Path 5	0.09	1.73E+5	1.53E+4	3.06E+4	4.08E+3
Path 6	0.003	1.04E+5	1.21E+4	2.43E+4	0.0
Path 7	0.01	1.55E+5	1.21E+4	2.43E+4	5.10E+3
Path 8	0.18	2.16E+5	1.53E+4	3.06E+4	8.44E+3
Path 9	0.03	2.10E+5	1.21E+4	2.43E+4	1.05E+4
		nonLOSP PDS			
Path 10	0.16	2.26E+5	3.41E+4	6.83E+4	8.94E+3
Path 11	0.02	2.19E+5	2.68E+4	5.37E+4	1.12E+4
Path 12	0.16	5.24E+5	3.20E+5	6.22E+6	2.13E+4
Path 13	0.02	5.10E+5	2.56E+5	4.88E+6	2.66E+4
Path 14	0.37	0.899	5.25E-2	0.105	4.50E-2
Path 15	0.23	2.16E+5	1.53E+4	3.06E+4	8.44E+3
Path 16	0.04	2.10E+5	1.21E+4	2.43E+4	1.05E+4

Table 4.1.2-2. Grand Gulf POS 6 mean doses and dose rates at 100 m based on the Wilson building wake effect model.

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Total Dose (rem)	30 minute Dose (rem)	Dose Rate First Segment (rem/hr)	Dose Rate Second Segment (rem/hr)
		LOSP PDS			
Path 1	0.10	9.48E+3	1.97E+2	3.95E+2	0.0
Path 2	0.48	2.11E+5	2.45E+4	4.91E+4	0.0
Path 3	0.08	2.11E+5	2.45E+4	4.91E+4	0.0
Path 4	0.02	2.95E+6	3.43E+5	6.86E+5	0.0
Path 5	0.09	3.56E+6	3.43E+5	6.86E+5	9.14E+4
Path 6	0.003	2.34E+6	2.72E+5	5.44E+5	0.0
Path 7	0.01	3.48E+6	2.72E+5	5.44E+5	1.14E+5
Path 8	0.18	4.84E+6	3.43E+5	6.86E+5	1.89E+5
Path 9	0.03	4.70E+6	2.72E+5	5.44E+5	2.36E+5
		nonLOSP PDS			
Path 10	0.16	5.06E+6	7.65E+5	1.53E+6	2.00E+5
Path 11	0.02	4.91E+6	6.00E+5	1.20E+6	2.50E+5
Path 12	0.16	1.17E+7	7.16E+6	1.39E+8	4.77E+5
Path 13	0.02	1.14E+7	5.72E+6	1.09E+8	5.96E+5
Path 14	0.37	20.1	1.17	2.34	1.01
Path 15	0.23	4.84E+6	3.43E+5	6.86E+5	1.89E+5
Path 16	0.04	4.70E+6	2.72E+5	5.44E+5	2.36E+5

Figure 4.1.2-1 contains the 5th, 50th, 95th percentile as well as the mean parking lot dose rates for the first release for both the Ramsdell and Wilson/Regulatory Guide models for distances of 10 - 500 meters from the reactor. A similar plot for the second release is shown in Figure 4.1.2-2. The uncertainty in both the building wake effect models and the source term is shown by the wide range of dose rates at each distance.

4.2 Offsite Consequences

The MACCS code [5,6,7] was used to estimate the consequences to the general public. MACCS models the transport and dispersion of plumes of radioactive material released from the plant. As the plumes travel through the atmosphere, material is deposited on the ground. Several of the pathways through which the general population can be exposed are considered. Emergency

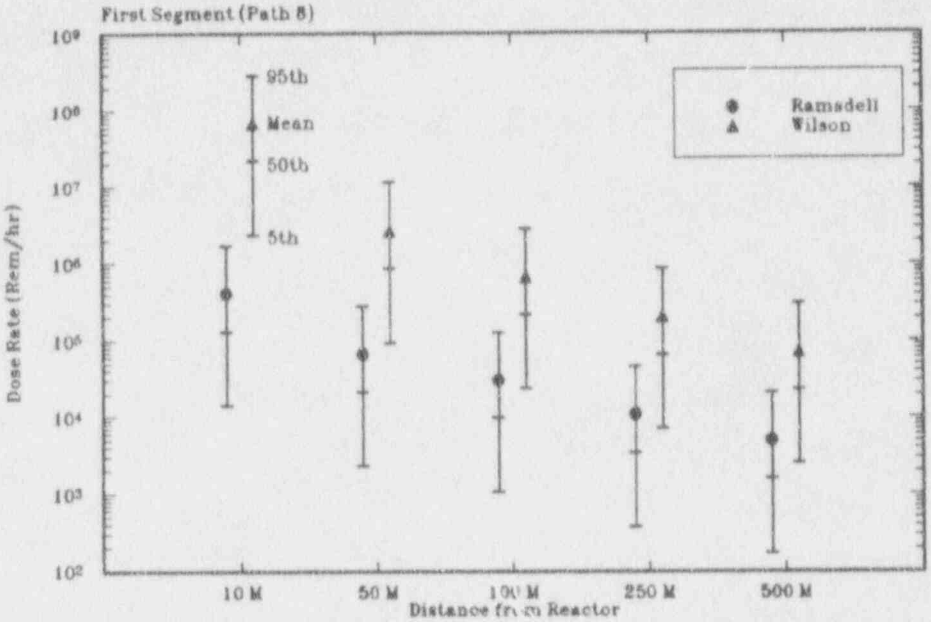


Figure 4.1.2-1. Parking Lot dose rates for Ramsdell and Wilson/Regulatory Guide models for distances from 10 - 500 meters from the reactor: First Release Segment

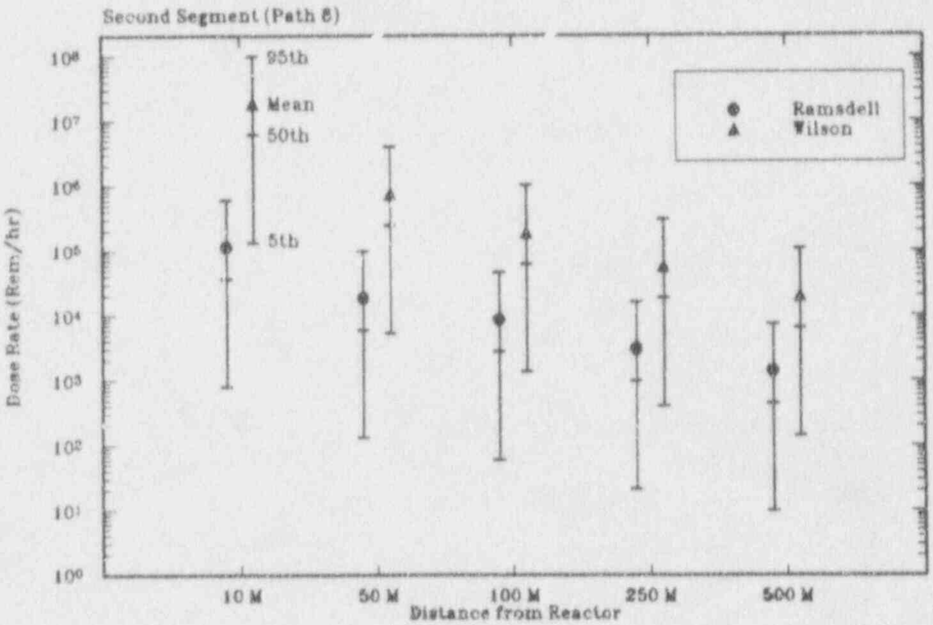


Figure 4.1.2-2. Parking Lot dose rates for Ramsdell and Wilson/Regulatory Guide models for distances from 10 - 500 meters from the reactor: Second Release Segment

response and protective action guides are also considered as means to mitigate the extent of the public exposure.

The input used in this study is identical to that used for Grand Gulf in the NUREG-1150 study [9] with the exception of the core inventory for which the inventory four days after shutdown was used and the source terms which resulted from GGLPSOR. The emergency response assumptions were not changed for this analysis.

Table 4.2-1 contains the estimated mean number of early fatalities, latent cancers, 50 mile population dose, and 1000 mile population dose for the sixteen paths through the APET along with the conditional probability of that path. The mean number of early fatalities ranged from 0 to 3.9×10^{-2} while the mean number of latent cancers ranged from 0 to 1940.

Table 4.2-1 Grand Gulf POS 6 Offsite Mean Consequences

Accident Progression Path Number	Path Conditional Probability	Consequence Measure			
		Early Fatalities	Total Latent Cancers	50 mile Population Dose[1]	1000 Mile Pop. Dose[1]
	LOSP PDS				
Path 1[2]	0.10	NC	NC	NC	NC
Path 2	0.48	1.3E-5	102	77,000	591,000
Path 3	0.08	1.3E-5	102	77,000	591,000
Path 4	0.02	4.8E-3	684	330,000	4,010,000
Path 5	0.09	4.8E-3	984	496,000	5,800,000
Path 6	0.003	4.0E-3	588	293,000	3,450,000
Path 7	0.01	4.0E-3	940	479,000	5,560,000
Path 8	0.18	5.2E-3	1270	652,000	7,480,000
Path 9	0.03	4.7E-3	1260	662,000	7,460,000
	nonLOSP PDS				
Path 10	0.16	8.9E-3	1190	624,000	7,090,000
Path 11	0.02	9.3E-3	1200	640,000	7,130,000
Path 12	0.16	3.7E-2	1920	939,000	11,300,000
Path 13	0.02	3.9E-2	1940	966,000	11,500,000
Path 14[2]	0.37	NC	NC	NC	NC
Path 15	0.23	5.2E-3	1270	652,000	7,480,000
Path 16	0.04	4.7E-3	1260	662,000	7,460,000

Table Notes:

[1] Dose is in Person Rem

[2] Offsite consequences were not evaluated for these paths because the offsite consequences associated with these paths were assessed to be negligible.

5.0 INTEGRATED RESULTS CONDITIONAL ON CORE DAMAGE

In the previous section the consequences associated with individual accident progression paths were presented. In this section the offsite consequences conditional on the occurrence of the LOSP PDS and the nonLOSP PDS are presented and are compared to full power PRA results extracted from the Grand Gulf analysis presented in NUREG-1150. Onsite consequences were not evaluated in NUREG-1150 and, therefore, an analogous comparison is not provided.

The consequences for a given PDS are calculated by taking a weighted average of the consequences for the individual paths. The weighted average is based on the conditional probability of each path. The PDS consequence is the sum of all of the "weighted" path consequences for the given PDS.

The offsite consequence distributions associated with the LOSP and nonLOSP PDSs are presented in Figure 5.1. Because a relatively small LHS sample was used in the evaluation of offsite consequences, the presentation of exact quantiles (i.e., 95^{th}) is inappropriate. Instead of quantiles, the high, low, median, and mean values are presented in this figure. From this figure it can be seen that the consequences associated with the nonLOSP PDS tend to be higher than the consequences associated with the LOSP PDS. This stems from the assumption that injection cannot be recovered in the nonLOSP PDS and, therefore, all of these accidents proceed to full core damage and CCI. Although the probability that the containment is closed during this PDS is significant, the lack of a means to control the containment pressure results in a significant probability of early containment failure. Containment failure bypasses the auxiliary building and results in, essentially, an unmitigated release.

Also presented in Figure 5.1 are the conditional consequences from the Grand Gulf full power PRA. The full power results are for internal events and are "averaged" over all of the accidents analyzed in the study. In addition to the global consequences, the mean consequences associated with a selected full power accident are also presented (i.e., triangle on the full power distribution). This selected accident is a fast station blackout that progresses to full core damage. The containment is ruptured during core damage; the containment sprays are unavailable throughout the accident. Thus, this accident is similar to the accidents analyzed in this abridged study in that many of the mitigative features of the plant (i.e., the containment and sprays) are unavailable. In this full power accident, however, the in-vessel releases are typically scrubbed by the suppression pool. From Figure 5.1 it can be seen that the number of early fatalities associated with POS 6 are very similar to the number of early fatalities associated with full power accidents. This may seem somewhat surprising at first because the inventory of radionuclides important to early fatalities during POS 6 is less than the inventory at full power. However, this difference is compensated by the lack of mitigative features in POS 6. In POS 6 the inventory has been reduced by decay but because of the lack of mitigative features, a significant amount of the radionuclides are released to the environment. In the full power accidents, on the other hand, there is a large inventory, however, mitigative features of the plant limited the size of the release. In full power accidents, for example, a considerable fraction of these radionuclides are retained in the suppression pool. The net effect is that the number of early fatalities is roughly the same. The number of latent cancers associated with POS 6 accidents is greater than the number of latent cancers associated with full power accidents. The radionuclides that are important to latent health effects are long lived isotopes and, therefore, four days of decay will not have a significant impact on the radiological potential of the release to cause latent cancer fatalities. Thus, the magnitude of the release is the driving factor for latent cancer fatalities. Because in POS 6 the releases tend to be higher than the full power accidents, the number of latent cancers associated with POS 6 are greater than the number of latent cancers associated with full power accidents. The factors that influence latent cancers also affect the population dose.

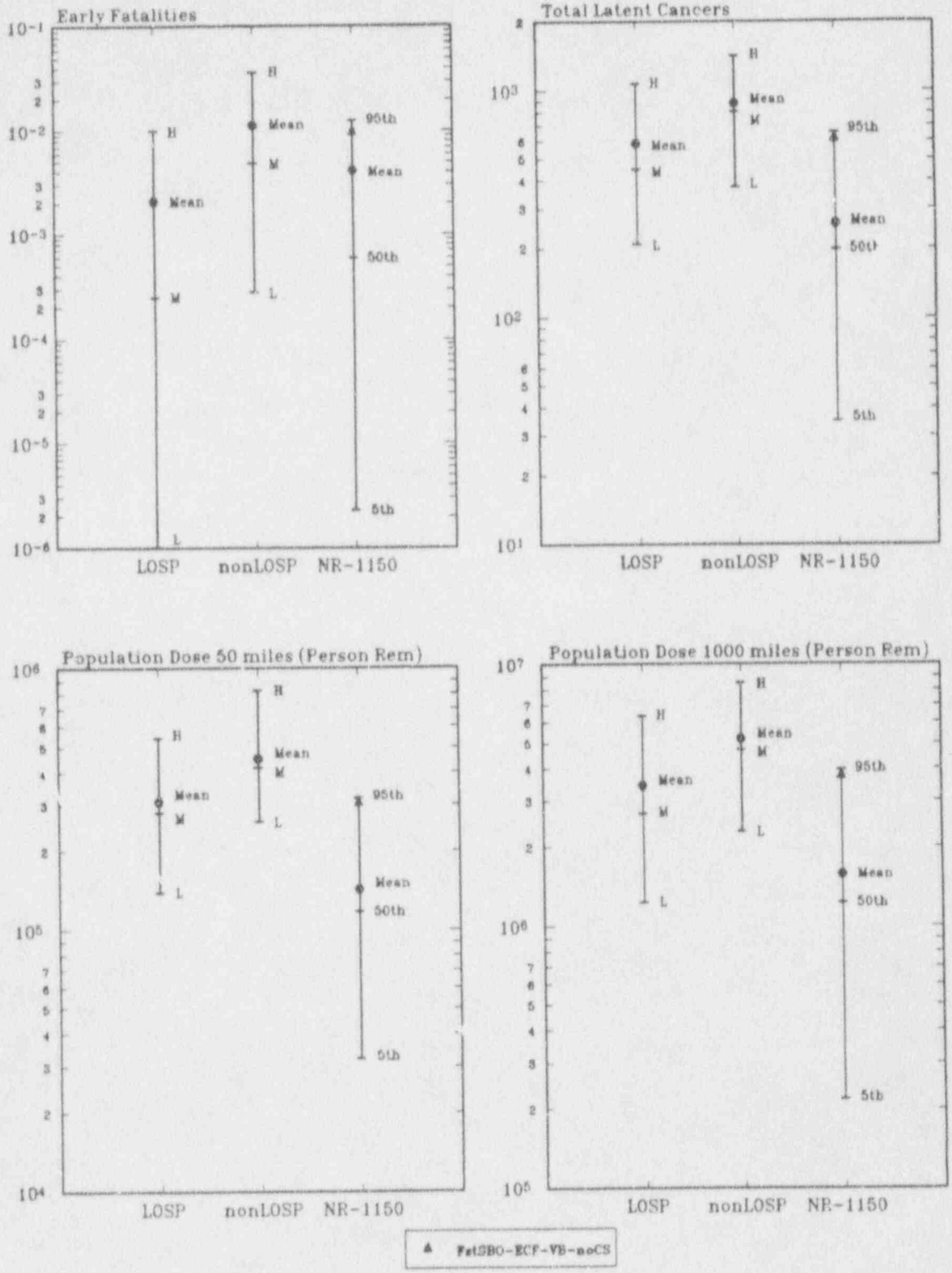


Figure 5-1. Grand Gulf POS 6 offsite consequences for LOSP and nonLOSP PDSs

6.0 INSIGHTS AND CONCLUSIONS

The results and insights presented in this study are ***conditional on the occurrence of core damage.*** Thus, this study gives no indication about the likelihood of these postulated accidents, but rather what could be expected given that core damage does occur. The input to this analysis is the core damage sequence definitions from the Level 1 coarse screening analysis. In this Level 1 scoping analysis conservative assumptions were made with regard to the availability of certain systems and the performance of the plant operators. These assumptions provided the necessary simplifications such that the dominant sequences could be identified and still keep the scope of the study manageable. While the calculated frequencies from the Level 1 study are used to rank the sequences, the absolute values of these frequencies were not reported due to the conservative nature of many of the necessary simplifications. Thus, frequencies were not propagated through to the Level 2 and 3 analyses. It is within this framework that the abridged study was performed. Therefore, when interpreting these results it must be remembered that frequency information is not available to indicate the likelihood of accidents and simplifying assumptions were made in both the Level 1 and the Level 2/3 studies.

The following is a list of insights obtained from this study:

o During POS 6 the majority of the mitigative features of the plant are bypassed or are unavailable. The vessel and drywell are open to the containment and, thus, the suppression pool is effectively bypassed. Furthermore, the containment spray system is unavailable during these accidents. Thus, steam and radionuclides are released directly into the containment atmosphere without being scrubbed by either the suppression pool or the containment sprays. For the accidents in which the containment hatch is removed, the only significant plant mitigative feature is the deposition that occurs in the auxiliary building. If the containment is closed but then fails during core damage, the auxiliary building is also bypassed.

o Because of the lack of mitigative features associated with these accidents, the source terms tend to be quite large.

o The consequences associated with these accidents are also significant. Offsite consequences are comparable with consequences associated with full power accidents. Onsite consequences are large.

o The time from the accident initiation to the onset of core damage is significant (i.e., from 18 to 28 hours). Thus, there is a considerable amount of time to restore core cooling and to close the containment. If offsite ac power is available, it is likely that the operators will close the containment prior to core damage.

o The pressure suppression features (i.e., suppression pool and containment sprays) of the Mark III design are bypassed during POS 6. Since the ultimate pressure capacity of the containment is fairly low, the plant is vulnerable to pressurization events accompanying vessel failure and associated with hydrogen burns. Failure to avoid or mitigate pressure excursions from these events could result in early containment failure.

o Because of the large recovery potential associated with these accidents (i.e., which were not fully accounted for in either the Level 1 analysis or this abridged analysis because of simplifying assumptions), POS 6 offsite risk *could* be significantly lower than the risk associated with full power accidents.

o Recovered accidents can pose a significant threat to onsite personnel.

o Because of the lack of mitigation features associated with accidents initiated in this POS, the auxiliary building and the SBGTs could play a significant role in the mitigation of the release, especially for recovered accidents.

There were many issues that were identified in this study that could affect the possible accident progressions and consequences. The resolution of many of these issues was beyond the scope of this abridged analysis and will have to be addressed in any more detailed analysis that is performed in the future. The following is a list of potentially significant issues:

o **Containment Closure.** The effects that the temperature, humidity, and radiation have on the plant personnel's ability to close the containment needs to be addressed in more detail. Containment closure is a critical issue that will affect the consequences associated with these accidents.

o **Containment Loading.** Hydrogen combustion phenomena associated with this plant configuration need to be investigated. In this plant configuration steam and hot hydrogen are released directly into the containment atmosphere. The amount of steam blanketing and air ingression and the availability of ignition sources will all affect the likelihood and magnitude of hydrogen burns. The effectiveness of the hydrogen ignition system in this plant configuration also needs to be investigated. The loading from in-vessel steam explosions is another issue that needs to be addressed. With the vessel head off in this POS and the relatively low failure pressure of this containment, in-vessel steam explosions could be a significant mechanism for early containment failure.

o **Source Term.** There are several events that can enhance the source term that were not included in the PRA model. First, the role that air ingression plays during core damage needs to be investigated. If significant air ingression does occur, the in-vessel phase of the core damage process could be significantly altered and the release of certain radionuclides enhanced. Second, the relocation of intact fuel from an in-vessel steam explosion could also result in the enhancement of an early source term. This issue was not addressed in this analysis. Third, for recovered accidents the embrittlement and failure of the clad could lead to a release earlier than what is currently modeled. This could be particularly important for onsite consequences.

- o **Auxiliary Building.** For accidents in which the containment is open during core damage, the auxiliary building could play a major role in mitigating the release. The radionuclide retention capabilities of this building need to be assessed in more detail than what was done in this abridged analysis. Furthermore, the effectiveness of the SBGT system to mitigate the release, especially for recovered accidents, also needs to be assessed.

- o **Onsite Consequences.** Only a scoping type analysis of onsite consequences was performed in this study. In the calculation of doses in the building, the integrated concentrations were based on average concentrations from GGLPSOR and on crude estimates of the residence times in the buildings. More detailed information on the concentration as a function of time and on the residence time would produce more realistic dose estimates.

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POLICY ISSUE

(Information)

October 1, 1991

SECY-91-309

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: DRAFT SAFETY EVALUATION REPORT ON THE GENERAL ELECTRIC BOILING WATER REACTOR DESIGN COVERING CHAPTER 19 OF THE STANDARD SAFETY ANALYSIS REPORT, "RESPONSE TO SEVERE ACCIDENT POLICY STATEMENT"

Purpose: To inform the Commission of the staff's intent to issue Chapter 19 of the draft safety evaluation report (DSER) on the General Electric Company's (GE's) advanced boiling water reactor (ABWR) design. The staff's DSER addresses open items needing closure as identified by the staff's review of Chapter 19 of GE's Standard Safety Analysis Report (SSAR).

Background: In SECY-91-153, "Draft Safety Evaluation Report on the General Electric Company Advanced Boiling Water Reactor Design Covering Chapters 1, 2, 3, 4, 5, 6, and 17 of the Standard Safety Analysis Report," the staff discussed the ABWR review process and the Commission guidance that it is following.

Discussion: The enclosed DSER addresses the ABWR Probabilistic Risk Assessment (PRA) discussed in SSAR Sections 19.1 through 19.6 and Appendices 19D, 19E, 19H, 19I and 19J. The issuance of this report, will facilitate the resolution of a number of open items identified by the staff's review. The staff is continuing its review of Chapter 19 and will issue supplements to this DSER to document the review and evaluation of the appendices not presently included.

This DSER focuses significant attention on the quality of the ABWR PRA rather than on insights developed from the PRA. The staff believes that knowledge of how PRA insights were employed in the ABWR design underscores the significance of design features

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OCTOBER 4, 1991

CONTACTS:
V. McCree, NRR
X21121

D. Scaletti, NRR
X21104

9110070046 XA

which eliminate dominant contributors to the estimated core damage frequency and offsite consequences, and facilitates a balancing of preventive and mitigative design features. In its remaining review of the SSAR, the staff will include an assessment of (1) how GE used PRA insights in the ABWR design process, (2) what ABWR design features, if any, GE included as a result of PRA insights to reduce risk significant sequences and phenomena, (3) how GE factored plant operating experience into the ABWR PRA, and (4) how GE used PRA insights to address severe accident phenomena.

The staff is currently engaged in dialogue with GE to reach closure on open items which have been identified from the staff's review of other ABWR SSAR chapters. The staff believes that resolution of some open items may be advanced by the PRA and has begun an examination of these open items using PRA insights. The staff also expects GE to employ PRA insights to support issue resolution.

The staff will provide copies of this DSER to the Advisory Committee on Reactor Safeguards.

Conclusion: The staff concludes that the enclosed DSER contains no new policy issues. However, GE has not provided sufficient information in many areas to allow the staff to assess the reasonableness of the ABWR risk estimates and the effect of certain severe accident phenomenological issues. In writing the final safety evaluation report for the ABWR, the staff will discuss the status of all issues including those issues previously open, but subsequently resolved.

Senior NRR technical staff plan to meet with GE at San Jose on October 8, 9 and 10, 1991 to discuss the issues identified in this DSER. The staff plans to issue this DSER by October 4, 1991 to facilitate those discussions. The staff would also place the enclosed DSER in the NRC Public Document Room at that time.

Coordination: The Office of the General Counsel has reviewed this paper and has no legal objection.

James M. Taylor
Executive Director
for Operations

Enclosure:
DSER Chapter 19

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ENCLOSURE

DRAFT SAFETY EVALUATION REPORT

ON

CHAPTER 19 OF

THE GENERAL ELECTRIC COMPANY'S APPLICATION

FOR CERTIFICATION OF THEIR

ADVANCED BOILING WATER REACTOR DESIGN

prepared by the

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

September 1991

19. PROBABILISTIC RISK ASSESSMENT

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19.1 INTRODUCTION

1.1 Background

As part of the ABWR Final Design Approval (FDA) application, General Electric has performed a Probabilistic Risk Assessment (PRA) in response to the Commission's Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants dated August 8, 1991 and the ABWR Licensing Review Bases, and has submitted the PRA for the staff's review.

Unlike deterministic evaluations of plant design, where an applicant's proposed design is evaluated against regulations in the manner described in the Standard Review Plan and is found "acceptable" or "not acceptable," the review of the PRA is not governed by explicit formal criteria. PRAs and their evaluations are used to assess, in a realistic rather than conservative manner, the safety profile of the proposed design as expressed in terms of the frequency of severe core damage accidents, the consequences of a spectrum of such accidents of varying severities, and the integrated risk to the public; the uncertainty in these parameters; and insights into the safety profile. In addition, a PRA and its evaluation can be used to make deterministic judgments of the safety of the proposed design.

1.2 Licensing Review Bases

The licensing review bases for the ABWR design were documented in a letter dated August 7, 1987, from T. E. Murley of U.S. NRC to R. Artiga of General Electric Company (GE), "Advanced Boiling Water Reactor Licensing Review Bases" (Reference 19.1). A summary of these review bases is as follows:

A. The licensing review bases include applicable portions of the balance of plant (BOP) and an enveloping site for the approved ABWR design. GE, as part of the SAR and PRA submittals, is expected to provide documentation related to the interface requirements of various BOP features, including the characteristics of site envelope parameters. As part of the ABWR FDA approval, the staff is expected to review and make a finding (with respect to risk significance) as to whether the proposed plant-specific design parameters and site-specific envelope parameters are within design interface requirements and site envelope requirements, as applicable.

B. Because GE is expected to provide its PRA submittals in the form of magnetic media (in addition to the hard copies), the staff's review of GE's risk submittals will be documented in the form of magnetic media also.

C. As part of the ABWR FDA review, GE is required to provide adequate resolutions to a list of unresolved open items and/or issues (such as unmodeled severe accident phenomenological issues applicable to the ABWR design) with respect to safety goals, if any, for the staff's review. The staff will review them with respect to their risk significance in the context of safety goals.

D. GE is required to submit, in addition to the level 3 PRA submittal, bounding risk analyses of external events (including seismic events, fires, internal floods and tornados). The staff will review them with respect to the overall risk significance of applicable external events, including seismic events, in the context of safety goals.

E. GE's method to calculate the containment response and the source term estimates will be based on the IDCOR-developed Modular Accident Analysis Program (MAAP). If the staff's review of GE's MAAP analyses finds significant deviations in containment loadings and source term estimates, additional sensitivity analyses will be performed by the staff to incorporate these deviations into the overall risk estimates and to gain insight into the uncertainty estimates on these critical parameters.

F. The results of the PRA will be compared with the Commission's safety goals, including the quantitative health objectives, and the quantitative design goals proposed by the Electric Power Research Institute (EPRI) in its Advanced Light Water Reactor (ALWR) Requirements Document. The ALWR design goals cover core damage prevention, containment performance, and severe accident mitigation concepts (discussed in detail in the following sections). The staff's review will evaluate the ABWR design against these goals.

G. The ABWR PRA will be applicable to all ABWR plants to be built within the ABWR design and site envelopes. That is, as part of the operating license, individual ABWR applicants will not have to submit a separate PRA for the staff's review. However, the licensee of an ABWR plant should submit a revision to the approved ABWR PRA within 2 years after an ABWR plant is licensed.

GE has agreed to follow, to the extent possible, the design requirements documented in the Electric Power Research Institute, "Advanced Light Water Reactor Requirements Document," dated December 1987, for the ABWR design (Reference 19.2). The staff's resolution of the applicabilities of these design requirements and the Commission's guidelines regarding deviations of the ABWR design requirements from those documented in Reference 19.2, are documented in the U.S. Nuclear Regulatory Commission, "Resolution Process for Severe Accident Issues on

Evolutionary Light Water Reactors," Commission Paper SECY-89-311, dated December 15, 1989 (Reference 19.3). The staff's review of the ABWR PRA has followed these guidelines, as applicable. The staff has also incorporated into its review, the issues and acceptance criteria outlined in the U.S. Nuclear Regulatory Commission, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," Commission Paper SECY-90-016, dated January 12, 1990, to gain insights into the acceptability of the ABWR design (Reference 19.4).

1.3 Review Objectives

The overall objective of this project is to assess the reasonableness of the risk estimates documented in the PRA and other risk related documents submitted as part of the FDA application package. In addition to this overall objective, there are several secondary objectives:

1. To assess the reasonableness of the accident frequency estimates of the major sequences (for both internal events and external events), identify strengths and weaknesses of certain design features, and identify major contributions to the uncertainty in the core damage frequency.

2. To assess the reasonableness of the proposed ABWR containment failure probabilities for early and late failure modes, identify failure mechanisms for various potential failure modes consistent with staff-developed core melt phenomenological knowledge, and provide design-specific risk results along with uncertainty estimates.

3. To compare the ABWR risk results with the Commission's safety goal and the "safety margin basis design requirements" provided in the EPRI ALWR Requirements Document.

4. To provide an integrated perspective on the overall risk estimates with respect to the impact of certain severe accident prevention and mitigation features applicable to the ABWR design.

1.4 Review Process

GE initially submitted on January 27, 1989, Amendment 4 to Chapter 19 of the ABWR Safety Analysis Report," Docket 50-605, the risk analyses of the ABWR design (Reference 19.5). The staff, with the help of Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL), completed a preliminary review of Amendment 4 of the ABWR PRA. As part of this review, a letter dated November 28, 1989, from D. S. Scaletti, NRC, to P. W. Marriott, General Electric Company, "Request for Additional Information (RAI) regarding the General Electric Company Application for Certification of the ABWR Design" (Reference 19.6); and a meeting with GE was conducted on September 12 and September 13, 1989 at GE Headquarters (San Jose, California) to discuss and resolve matters presented in the staff's RAI. GE submitted its response to the staff's RAI on January 9 and January 11, 1990 (References 19.7 and Reference 19.8). As a result of the ongoing development of the revised PRA documentation and incorporation of additional severe accident prevention and mitigation systems (such as a gas turbine-generator, an AC-independent firewater addition capability, an improved version of the containment vent system, and a passive ex-vessel corium flooder system) to the reference ABWR design, GE submitted, on July 28, 1990, Amendment 8 to the Chapter 19 of the ABWR Safety Analysis Report," Docket 50-605 (Reference 19.9). This amendment made substantial modifications to the original PRA Version (Amendment 4), added additional analyses in the area of external events, and, most importantly, modified analyses of certain plant improvements important to public risk. The staff notes that Amendment 8 of the FDA application, including the PRA, responds only partly to the staff's RAI.

In order to understand the complex operating characteristics of the containment mitigation systems during a postulated core melt scenario, the staff initiated additional research to investigate the adequacy of GE's methods and assumptions (related to core-melt phenomena) using the staff-sponsored MELCOR code at Sandia National Laboratories by letter dated November 12, 1990, from M. Carmel (SNL) to Jae Jo (BNL) (Reference 19.10). As part of this work, the staff also developed an RAI and issued it to GE on November 28, 1989 (Reference 19.6), and conducted a meeting with GE in December, 1989, to discuss and clarify the RAI with GE. GE documented its response to the above RAI in Amendment 10 to Chapter 20 of the ABWR Safety Analysis Report, Docket 50-605, dated March 28, 1990 (Reference 19.11). This version of Chapter 19 reflects the design through Amendment 8, except where otherwise noted.

The staff's review of the seismic risk analysis documented in Reference 19.9 and Reference 19.12, "Amendment 9 to Chapter 19 of the ABWR Safety Analysis Report," Docket 50-605, dated November 17, 1989, resulted in an additional RAI, and this RAI was issued by letter dated May 1, 1990, from D. C. Scaletti, NRC, to P. W. Marriott, General Electric Company, "Request for Additional Information regarding the General Electric Company Application for Certification of the ABWR Design" (Reference 19.13). GE has documented its response in letters dated July 3, 1990, "Amendment 13 to Chapter 20 of the ABWR Safety Analysis Report," Docket 50-605 and October 2, 1990, "Amendment 14 to Chapter 20 of the ABWR Safety Analysis Report," Docket 50-605 (References 19.14 and 19.15).

In addition to the above, due to the advanced nature of the ABWR control room design, the staff identified additional review work related to the risk significance of critical human factor issues and human reliability issues, and initiated additional work with BNL to obtain technical assistance in this area. This human reliability review also resulted in an additional RAI which was issued to GE on February 28, 1990 (Reference 19.13). The staff also conducted a meeting with GE on

March 6 and 7, 1990, to discuss the staff's RAI and the summary details of the representative K6 and K7 control room designs (located at Tokyo Electric Power Company, Japan). Subsequently, GE provided, as appropriate, its response dated July 3, 1990, "Amendment 13 to Chapter 18 of the ABWR Safety Analysis Report," Docket 50-605 to the above RAI in Reference 19.16. The staff's review of Reference 19.16 resulted in an additional human factors related RAI, and this RAI was issued from D. C. Scaletti, NRC, to P. W. Marriott, General Electric Company, "Request for Additional Information regarding the General Electric Company Application for Certification of the ABWR Design" (Reference 19.17). GE documented its response to the above RAI dated August 31, 1990, "Amendment 14 to Chapter 20 of the ABWR Safety Analysis Report," Docket 50-605 (Reference 19.18).

The staff notes that, as part of most PRA reviews, a plant walk-down and/or plant walk-through of major systems, components, structures, the main control room and remote shutdown system panels, including operator interviews and simulator tests (if any), is a critical and useful step to obtain a full understanding of postulated accident scenarios and to harvest qualitative safety insights into potential accident vulnerabilities, if any. Because the licensing review of the ABWR design involves a plant design only, the staff could not perform such walk-down activities. However, GE has indicated that it will make use of most of the design features of the K6 and K7 control room (Japanese design) concepts in designing a control room for the ABWR plant to be built during the production phase in United States (Reference 19.16). Thus the staff reviewed the Japanese K6 and K7 control rooms (inspected mock-ups, observed operator performance tests in training simulators, and conducted walk-downs of tasks determined critical to risk). Data collected from these reviews provided a framework for evaluating the adequacy of the ABWR PRA, under the assumption that the ABWR control room instrumentation and workplace layout will be similar to those of the Japanese K6 and K7 control rooms.

The probabilistic risk review performed by the staff primarily involves examination of the ABWR PRA, as opposed to making reanalysis and recalculation of selected sequences and release categories. This approach was adopted for the review of the ABWR PRA and is consistent with the guidelines documented in a memorandum dated March 14, 1988, from E. S. Beckjord and T. E. Murley to V. Stello, "Memorandum of Agreement of the RES Role in the Review of the Standard Plant Design-NRR/RES" (Reference 19.19). Detailed technical findings in the above review areas have been documented in NUREG/CR-5676P, "A Review of the General Electric ABWR Probabilistic Risk Assessment" (Reference 19.20).

19.2 RISK-SIGNIFICANT ABWR DESIGN FEATURES

2.1 ABWR Safety System Features

The following are the frontline and support systems that have been explicitly modeled in the ABWR PRA by GE. During the course of the review, the staff found that, compared with earlier BWR designs, many safety systems have substantial design modifications which contribute to reductions in system unavailabilities and thereby a reduction in the core damage frequency for the ABWR design, compared to these earlier designs. Detailed deterministic reviews of these systems, and findings regarding their acceptability, can be found in Chapters 3 through 10 of this document. The following are some major highlights:

1. <u>Reactor Protection System (RPS)</u>

   The reactor protection system (RPS) refers to the overall complex of instrument channels, trip logics and signals, manual controls and trip actuators that are involved in generating a reactor trip (or scram) to bring the reactor subcritical. The RPS of the ABWR, which has four-division redundancy, is designed in such a way that the failure of any single element will not hinder the actuation of a required trip. Although GE has significantly improved the RPS design compared to all earlier designs, GE did not make an attempt to quantify the RPS unavailability following a transient or a postulated LOCA event. Instead, an unavailability estimate of 1 E-7 per demand has been assigned based on the reliability analysis performed as part of the solid state RPS design for the Clinton facility.

   The staff noted that GE's use of an RPS unavailability estimate for the Clinton facility is contradictory since the staff has evaluated the adequacy of the ABWR PRA under the assumption that the control room layout will be similar to K6/7. Furthermore, the design of the Clinton RPS and the ABWR RPS are essentially different. The Clinton design uses analog trip modules and isolation devices, whereas the ABWR design uses microprocessors (software), multiplexors and fiber optics. Other design features such as the control room layout, operator interface, recirculation pump trip, and data transmission are also different. These design dissimilarities result in PRAs with fundamentally different failure mechanisms and common mode considerations.

   The study for Clinton indicated that the unavailability of the RPS is essentially dominated by common cause failures of the divisional multiplexors and the system logic. GE assumed that similar failures will dominate the ABWR RPS unavailability. The staff noted that this unavailability estimate is significantly lower than the estimate documented in the results of the staff's analysis dated April 1978, "Anticipated Transients without Scram for Light Water Reactors," NUREG-0460 (Reference 19.21). The staff also noted that GE used the GE NUMAC line of equipment, which is not used as the basis for either the Clinton or K6/7 designs, as the example of the type of equipment that will be used for ABWR I&C systems. The staff concluded that GE should justify the use of the Clinton reliability estimates in the ABWR PRA because the ABWR RPS design is significantly different from the Clinton design. This is an outstanding item.

2. Control Rod Drive System (CRDS)

The control rod drive system (CRDS) of the ABWR design differs significantly from that of GESSAR-II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment, 22A7007, dated March 1982 (Reference 19.22) or currently operating plants in that it utilizes electric-hydraulic fine motion control rod drive (FMCRD) mechanisms rather than locking piston mechanisms. The ABWR CRDS consists of FMCRD mechanisms and the CRD hydraulic system, which includes pumps, filters, hydraulic control units, instrumentation and electrical controls etc. The hydraulic power required for scram is provided by high-pressure water stored in the individual hydraulic control units (HCU). A single HCU contains a nitrogen-water accumulator charged to a high pressure and the necessary valves and components to power the scram action of two FMCRDs. Rod insertion can be alternatively achieved by driving all the rods in simultaneously with the FMCRD electric motors. The ABWR CRDS can be used, in conjunction with the rod control and information system (RC&IS) and the reactor protection system (RPS), to perform a number of important reactivity control functions. For example, upon receiving manual or automatic signals from the RPS, it can provide rapid control rod insertion (scram). Another function provided by the CRDS is alternate rod insertion (ARI), an alternate means of actuating motor-driven rod insertion in the event of electrical failures following a failure-to-scram event. The FMCRD mechanism used in the ABWR CRDS possesses several meritorious features which enhance both the reliability of the scram system and plant maneuverability. Some of these features are highlighted below:

1. The FMCRD permits insertion either hydraulically or electrically. Upon receiving a scram signal, the FMCRD is inserted hydraulically by the energy stored in the nitrogen-water accumulators of the hydraulic control units. At the same time, a signal is also sent to insert the FMCRD electrically via its motor drive. This enhanced design feature increases the diversity of the scram system.

2. The FMCRD does not employ a scram discharge volume (SDV). This enhanced design feature eliminates certain common-mode failures (applicable to other BWR designs) and the SDV LOCA.

3. Standby Liquid Control (SLC) System

The SLC system is a backup means to shut down the reactor to subcritical conditions by injecting sodium pentaborate solution into the reactor. This system consists of two 100 percent capacity trains, each containing a positive displacement pump (with a flow rate of 50 gpm). It is manually initiated by the operator if it is determined that the reactor has not successfully scrammed following an anticipated transient or a small LOCA event.

The staff notes that, except for the manual initiation feature, the design features of the SLC system are consistent with design requirements specified in the EPRI ALWR Requirements Document (Reference 19.2). GE has quantified the SLC system unavailability using fault trees and historical operating data. It is about 0.2 per demand. The system unavailability is dominated by the human failure to initiate the system on demand.

4. Reactor Core Isolation Cooling (RCIC) System

The RCIC system in the ABWR design is a system designed to provide coolant makeup to the reactor when the reactor is at high pressure following a transient or a postulated LOCA. It is also capable of providing coolant makeup to the reactor at high pressure during a station blackout.

GE has quantified the RCIC system unavailability using fault trees and historical operating data. It is about 0.04 per demand (without room cooling dependency) for the vessel isolation event (Table 19.3-3, page 19-44). The system unavailability is dominated by: (A) mechanical failure of the pump-turbine, (b) unavailability due to maintenance, and (C) pump failures.

5. High Pressure Core Flooder (HPCF) System

The HPCF system of the ABWR design is somewhat similar to the High Pressure Core Spray (HPCS) system of the GESSAR-II design except that it consists of two independent high-pressure trains (HPCF-B and HPCF-C) rather than one train. It is designed to provide coolant makeup to the reactor vessel under postulated LOCA events and anticipated transients. The staff notes that, although the ABWR design has two HPCF pumps, the flow rate per pump (800 gpm) is actually only about half of that provided by the single-train GESSAR-II HPCS system (1550 gpm) (Reference 19.22) and a still smaller fraction of the capacity of the single-train HPCI system (5600 gpm) of older plants, as described in the PRA for Limerick Generating Station, Philadelphia Electric Co., Docket Nos.50-352 and 50-353, dated September 1982 (Reference 19.23). The staff also makes note of the redundancy with respect to electrical, mechanical and physical separation characteristics of the two train HPCF system.

GE has quantified the HPCF system unavailability using fault trees and historical operating data, as applicable. It is about 2 E-3 per demand for transients, 3 E-3 per demand for LOCA events, and 5 E-3 for loss of AC power events (Table 19.3-3, page 19-44).

6. Reactor Depressurization Function

In the event of the failure of all high pressure coolant makeup sources, the reactor must be depressurized to a primary system pressure such that one of the three trains of the RHR system or the condensate transfer system or the AC-independent firewater system could provide low pressure coolant makeup to the reactor.

The purpose of the Automatic Depressurization System (ADS) is to depressurize the reactor pressure vessel to allow use of the RHR system (in the core flooding mode) for reactor water makeup in the event that the RCIC system and the HPCF system fail to provide coolant makeup to the reactor. The ADS of the ABWR design is similar to that of the GESSAR-II design (Reference 19.22).

GE has quantified the ADS system unavailability using fault trees and historical operating data, as applicable. It is about 0.002 per demand for transients and less than 1 E-6 for LOCAs. The system unavailability for transients is dominated by the human failure to depressurize the reactor in a timely fashion following the onset of a transient. In order to compensate for this dominant failure mode, operating BWRs have changed their actuation logic such that low reactor water level will initiate the system following some time delay. However, no change has yet been proposed for the ABWR.

7. Residual Heat Removal (RHR) System

The ABWR RHR system consists of three closed independent loops, and each loop has one RHR pump and one RHR heat exchanger. The purpose of the RHR is to provide coolant makeup to the reactor, containment cooling, and heat transport from the suppression pool to the RCW system for the complete spectrum of LOCAs and transients. Each of the RHR loops is equipped with the necessary piping, valves, pump and heat exchanger to inject water into the reactor vessel and/or remove heat from the reactor vessel or containment to the ultimate heat sink.

GE has quantified the RHR system unavailability using fault trees and historical operating data, as applicable. For Low Pressure Flooder (LPFL) mode followed by a successful scram event, it is about 5 E-5 per demand. For suppression pool cooling mode followed by a successful scram event, it is about 5 E-4 per demand

(Table 19.3-3). The three-train system unavailability is dominated by: (A) miscalibration failures of the flow transmitters, and (b) cavitation failures of the RHR pumps due to a drop in suction pressure.

8. Containment Overpressure Relief

In the event of failure of the RHR system, the ABWR containment pressure will be expected to increase due to in-vessel steaming. At 80 psig the overpressure protection system (OPS) will automatically open to relieve the excessive containment pressure in order to prevent gross structural failure of the containment. The OPS can help prevent core damage for some accident sequences and help mitigate the consequences of other sequences. This section will briefly mention the preventative role of the OPS. The mitigative capability is discussed in Section 19.6.4.2.

For those accident sequences with successful core cooling but an unavailable RHR system, the containment pressure will increase and eventually fail the containment, allowing the possibility of damaging the core cooling systems and causing core damage. The OPS will help relieve containment pressure in a controlled manner and reduce the potential for containment-failure-induced failure of the core cooling systems.

9. Electric Power System

The ABWR design consists of a three train electrical system to provide power supply to onsite in-plant electrical loads. The ABWR design will be designed to take offsite AC power from a minimum of two independent offsite power sources. In the event offsite power sources are lost, the on-site emergency power sources, comprised of three emergency diesel generators and four DC batteries, are designed to fulfil the power requirements of the safety-related systems to achieve cold shutdown. The ABWR gas turbine generator will have a black-start capability during a postulated station blackout. The staff believes that this additional AC power source will significantly reduce the contribution of station blackout events to the core damage frequency and the likelihood of early containment failures.

GE developed fault trees to quantify the failure probabilities of 125 AC buses, 125 DC buses, associated 480V motor control centers (MCC), MCCs for service water system components, and 6.9KV buses (including the impact of failures of onsite emergency sources, and normal and preferred offsite power sources). When GE subsequently added the gas turbine generator to the ABWR design, it was incorrectly incorporated into the PRA model in an optimistic manner. This was corrected in the staff review quantification. The staff notes that these fault trees have been linked with the frontline system fault trees, constructed for various other safety systems, in the evaluation of their unavailabilities.

10. Service Water System

The ABWR reactor building cooling water (RCW) system, which consists of three independent divisions, is designed to remove heat from essential equipment in the reactor building, such as RHR heat exchangers, heating, ventilating and air conditioning (HVAC) emergency cooling water system refrigerators, diesel generators, and other equipment.

During normal operation, one RCW and one SW pump in each loop (primary and secondary) in each division and one RCW heat exchanger in each division are operating. The ABWR PRA assumes that, under these conditions, sufficient cooling capacity is available to provide seal and motor bearing cooling water for the core cooling pumps. It further assumes that, if all three loops are operated in this manner, sufficient cooling capacity is available to remove heat from the RHR heat exchangers during a postulated LOCA.

GE has developed fault trees to quantify the failure probabilities of the RCW trains and SW trains, including the impact of failures of support system dependencies (such as power failures and air system failures). The staff notes that these fault trees have been linked with the frontline system fault trees, constructed for various other safety systems, in the evaluation of their unavailabilities.

2.2 ABWR Human-System Interfaces

2.2.1 Introduction

Section 2.2 will:

- provide a description of the Advanced Boiling Water Reactor (ABWR) Control Room and other significant human-system interfaces (HSIs),
- identify the new and conventional human-system interface technologies with potential risk significance, and
- identify how the General Electric (GE) design approach has (or will) managed the risk.

The reader is referred to Section 19.3.7 of this Chapter, "Human Reliability Analysis," and the previous Chapter 18, "Human Factors Engineering" for additional information related to ABWR human-system interfaces and potential risk related to those interfaces.

2.2.2 General Description of the ABWR Human-System Interfaces

The following description of the ABWR HSIs is based upon information presented in GE's Standard Safety Analysis Report (SSAR) Chapter 18 (Reference 19.24). It should be noted that the design of the HSIs is not complete. Further design development and testing may result in changes to the description provided below.

One of the most significant human factors differences in the design of the ABWR when compared with "traditional" Boiling Water Reactor (BWR) plants is the design of the human-system interfaces in the control room. A major driving objective of the control room design is the operational philosophy of single operator monitoring and control during normal plant operations, including startup, power operations, and shutdown. However, during normal plant operations, the control room staff will also include an assistant control room supervisor, a control room shift supervisor, and two auxiliary equipment operators. The proposed operational philosophy leads to two significant design requirements. First, all controls and displays need to be located in a compact workstation so a single operator can perform all required tasks. Second, increased automation is required to relieve the operator from tedious, labor-intensive, and repetitive tasks. Thus, rather than having direct control over components, the operator acts as a supervisory controller monitoring and authorizing automated task performance.

The ABWR will have a control room (CR) whose main elements, from an operations standpoint, are a centralized command and control workstation and a wide screen display panel. The CR provides for single operator monitoring and control from the centralized workstation during normal operations. During emergencies, the workstation can accommodate additional operators. The main control board is a "compact," computer-based console where all the information (displays) and controls needed by the operator are available at the board and/or from the large-screen display. The detailed design of the control room has not yet been finalized. The control board may make use of many advanced technologies, including color graphic displays on high resolution CRTs, flat panel displays (e.g., electroluminescent technology), touch screen input devices, data display location flexibility (e.g., capability of locate displays on different cathode ray tubes (CRTs), and a variety of dedicated controls. Touch panels may be used for the control of non-safety system components such as valves, pump motors, etc., as well as other functions. Not all controls will be accomplished via computer input devices, however. For example, many safety system functions (such as Standby Liquid Control (SLC) injection, Emergency Core Cooling System (ECCS) initiation/reset, manual scram, turbine trip, and MSIV controls) will be controlled with hardwired switches.

The wide screen display panel is used for the display of top-level plant status information, important parameter, and important alarms. This is information that will be available to the entire control room crew. The wide display will consist of three panels. One non-safety grade, process-driven panel will contain summaries of important plant information such as displays for reactor start-up and load changes. The other two panels are used for the display of top-level alarms and fixed mimics of important NSSS and BOP systems. These displays will utilize safety-grade equipment driven independently of the process computer.

The alarm system design description includes discussions of basic design concepts, alarm classifications, configuration of alarm systems and alarm system implementation (wide display device vs. CRTs) and suppression of alarms. Compared with other aspects of the control room, the alarms are discussed in considerable detail. Of particular note is the critical parameter alarm display, which is part of the wide display device hardware alarm group along with plant trip sequence and safety-system status displays. It is noteworthy that the critical parameter alarm display is intended to annunciate entry conditions to the symptomatic emergency operating procedures (EOPs).

As for the safety parameter display system (SPDS), "the ABWR is not to have a separate SPDS, but rather, the principal functions of SPDS are to be integrated into the overall control room display capabilities" and displayed on the "wide screen display panel." This approach is consistent with NRC expectations for new plant designs. SPDS functions should be integrated into the overall display design. Assistance functions will also be available to select appropriate summary displays based upon plant mode.

In addition to providing a major design driver to control board and CR design, the single operator control philosophy also increases the requirement for automation to assist the operator with traditionally difficult operations characterized by heavy workload (such as plant startup and operations for the various RHR operating modes). The ABWR non-safety systems are coordinated via the power generation control system (PGCS) during normal operations. Neither the PGCS nor any other automated non-safety system can automatically change the status of any safety system. The PGCS is a non-safety system, which provides automatic plant startup and shutdown, and which automatically alerts the Senior Reactor Operator (SRO) of specific abnormal conditions being detected. If a change in a safety system is required, the PGCS notifies the operator and the change is made manually. When appropriate, the PGCS automatically disengages its automatic mode of operation. Therefore, any required changes in the operational status of any ABWR safety system must be performed by the SRO or an automatic safety-related initiation.

According to Chapter 18 of the ABWR SSAR, the remote shutdown system (RSS) will "use conventional, hardwired controls and indicators to maintain diversity from the main control room." GE's Request for Additional Information (RAI) Response 620.32, dated November 2, 1990, (Reference 19.25) provides a rationale for the diversity which includes protection "against the improbable event of common mode hardware or software failure in the plant instrumentation and control systems" and that it is "typical of all BWRs."

Regarding other local control station (LCS) designs, RAI Response 620.33 (Reference 19.25) provides that their "man-machine interface design...will be defined as part of the ABWR design implementation equipment activities." As for the design of local valve operations, RAI Response 620.34 (Reference 19.25) states in part "The ABWR design philosophy regarding local valve operations is similar to previous BWR designs" and that "local position indication will be provided along with parallel control room position indication."

2.2.3 Risk Significant Control Room Innovations

The design of the ABWR control room includes many features which have <u>potential</u> risk significance. The ABWR operational philosophy and methods of operator interface are quite different from more "conventional" U.S. BWRs and employ approaches for which the U.S. nuclear industry has very little experience. These differences between the ABWR and conventional BWRs increase the degree of uncertainty with respect to risk implications. This is not meant to imply that the ABWR design is associated with greater risk; however, evaluating the risk is more difficult and uncertain.

A preliminary identification of the aspects of the ABWR design that are most potentially risk significant is presented below.

1. <u>Single Operator Philosophy:</u> It appears to the staff that the goal of single operator control for normal operations is a major design driver influencing decisions for increased automation as well as control console and control room layout and design. Yet, the linkage of this goal to overall safety and reliability of operators was not provided. Nor was the desirability of this approach adequately supported by analysis, tests, and evaluations. It is unclear, for example, why the high level goals of safe, efficient, and reliable operator performance are fostered by single operator control. RAI Question 620.20, dated December 17, 1990, (Reference 19.26) addressed the importance of single operator control and its rationale. GE's response to the question offered three

points: elimination of communication errors, elimination of coordinating activities across operators, and the low workload levels resulting from increased automation (i.e., there will not be enough work at the main console for two operators). With respect to the first two points, while communication and coordination can contribute to human error as indicated, communication and coordination also provide an important check on the control process and check on the performance of the other operators. The net effect on reliability of the drawbacks and benefits of communication/coordination would have to be investigated. Also, the shift from normal to emergency operations may be problematic with only one operator. Under emergency conditions, the operator will receive additional assistance. Precisely what this assistance will consist of and how the tasks will be allocated and coordinated between operators is unclear. The second operator will be "coming in cold" and will have very poor situation awareness; thus, the communication/coordination burden on the first operator may be excessive at a time when workload is already high (the emergency condition). The second operator's effectiveness may be limited for an extended period of time.

Considering the several reported instances of operators in U.S. nuclear plants not being alert and attentive to their duties, thereby potentially compromising plant safety, it is the staff's opinion that an appropriate analysis should be provided to justify how one operator, the senior reactor operator (SRO), at the main console will remain attentive to his duties. In addition, the RAI 620.20 Response (Reference 19.26) about one-person operations during normal conditions does not mention that operator communication between several licensed operators monitoring plant conditions has historically provided a system of "checks and balances" to compensate for inactivity during extended periods of monitoring without any required control.

In essence, the net effect on operator and system reliability needs to be evaluated for normal operations and for the shift from normal to emergency operations. However, the documentation in the SSAR does not provide this information.

With respect to the third point, operator workload, GE indicated that ABWR workload analyses indicate that "because the high degree of plant automation which is available during normal operations reduces the operator workload to a level easily sustained by a single operator but one which may provide a lower level of stimulus if divided between two operators" thus affecting alertness, etc. The staff agrees with concerns over low workload levels. However, without documentation of the cited studies, we do not know if the concern is warranted in the ABWR. How was workload defined in the cited studies? Was the <u>cognitive</u> workload associated with system monitoring evaluated? The studies of workload presented in the Japan briefings defined workload in terms of the number of tasks performed per unit of time. This approach is typically insensitive to the cognitive workload associated with supervisory control tasks and monitoring activities. Long-term system monitoring is difficult for operators, and perhaps having two operators would be preferable to one in such a situation. For each operator, monitoring duties could be shared with other more actively-oriented tasks to achieve acceptable workload levels for both operators, thus removing the heavy monitoring burden from a single operator. This issue also relates to the first two points in that higher workload perhaps increased by communications and coordination might provide more stimulation for the operators and a more reliable control room.

Also, the workload argument is somewhat circular. A single operator control approach leads to increased automation so one operator can perform all needed tasks. Then, when a second operator is considered, it is rejected by indicating that due to extensive automation, there is only enough work at the main console for one operator. The more appropriate question to address is what level of staffing, automation, and allocation of function will meet the goals of safe and reliable performance of the operating crew and the overall system.

2. <u>High Degree of Automation:</u> The increases in automation (automation of tasks traditionally performed by an operator) and enhanced decision aiding in the ABWR results in a shift of the operator's function in the system from a direct manual controller to a supervisory controller and system monitor (largely removed from direct control). The shift in a human operator's role away from direct control is typically viewed as positive from a reliability standpoint since the human operator is considered one of the more unpredictable components in the system. It is generally presumed that automation will enhance overall system reliability by removing or reducing the need for human action. The operator's performance in the system is believed to be improved by freeing him from tasks which are routine, tedious, physically demanding, or difficult. Thus, the operator can better concentrate on supervising the overall performance and safety of the system. However, rather than removing error, such a change has frequently been associated with a shift of human error to higher levels in the system which are more difficult to detect and quantify. For example, evaluations of increased automation in civilian aviation has led to the identification of several new categories of error that were introduced. The potential for "new" types of errors that can occur in an advanced system should be reflected in the risk analysis.

This shift in role has implications for a wide range of factors of concern in HRA including operator selection, training and procedures, and human-system interface design to adequately support the new role. Since these effects are not well understood, it may be difficult to assess them.

3. Display Formats: Much of the data on plant performance will be presented to the operators on computer-based CRT screen displays. These displays will replace conventional indicators such as gauges and meters. Thus, the methods by which these data are presented is very important. Yet, design requirements were notably absent for the display of data and information (human-software interface). Much detail is presented on the hardware aspects of the HSI, e.g., use of CRTs, hardware switches, and console design; however, the methods and formats by which information is displayed is not discussed. Nor is the operator-interface transaction methodology discussed beyond indicating that a direct manipulation interface is planned. The staff considers this a major limitation since, in a computer-based control room, display methodologies are at least as (and probably more) significant to safety and reliable operator performance as the hardware design.

4. Advanced Technology Human System Interfaces: As indicated in item 3 above, almost no information was provided regarding the display of plant data and information. For those aspects of the control room requirements that were described, many represent innovative approaches for U.S. BWRs. As noted in the introduction, their significance with respect to risk mainly lies in uncertainty about their impact on human performance and reliability. Since the control room design is not final at this time, the technologies indicated below were only described at a "requirements" level, not a design implementation level.

- Compact Workstation Console - The workstation console has been designed for a single operator control during normal operations and more than one during off-normal conditions. However, it remains to be validated that multiple operators can perform their tasks at the compact console without interfering with each other.

- CRT Displays - The primary display devices at the workstation are the seven CRTs. Few, if any, traditional indicators will be used. Thus, not all data is presented to the operator at all times. The acceptability of providing "glass windows" on the process rather than conventional indicator displays will have to be evaluated.

- Overview Summary Displays - Since only seven CRTs will be used at the console to provide plant status information, summary or overview displays will have to be used (in contrast to individual indicators on conventional boards). The success of these displays to present hierarchical status information will impact the operator's ability to maintain adequate situation awareness.

- Soft Switches - The primary means of control will be through the use of software generated controls (soft controls) presented on the CRTs and flat panels and activated through touch screen input. The impact of this mode of control on operator performance will have to be evaluated.

- Computer-Based Alarms - While some alarms will be presented on the wide panel in a conventional tile format, the predominant display of alarms will be on CRT. The methods by which alarms are presented (e.g., lists or graphic tiles) and the way in which the operators interact with the computer-generated alarms may affect safety. The impact of alarm suppression technology on safety will also require evaluation.

- Large Wide-Screen Display - The large overview displays represent an innovative technology now to the U.S. nuclear industry. The allocation of display information to the wide-screen display versus the console CRTs and the way in which the wide-screen displays are formatted may impact safety and will have to be validated.

There may be further design potentially safety-significant innovations as the control room design proceeds to final implementation.

.3 Control Room Technology Innovations

The introduction of advanced and innovative technologies into the control room may be accomplished at various points in the design process, including:

- Thorough well-documented, top-down system analysis assuring appropriate allocation of function to system and operator control.

- Technology assessments and iterative design testing to evaluate operator and system performance.

- Use of human factors engineering guidelines and standards to assure that the design conforms to currently accepted human engineering principles.

- Explicit design objectives for developing error resistant and error tolerant design.

- Verification and validation (V&V) testing of the final design in "full-mission" scenarios.

Based upon the information provided in SSAR Chapter 18 "Human Factors Engineering" describing the control room design, it is difficult to determine the extent to which these risk management elements are being addressed. The design process is discussed in detail, but no results of system analyses, technology assessments, trade studies, tests and evaluations, etc., are provided. Further, since the design is at the stage of requirements only (and not final design), no check on the final design is possible.

The only aspect of the risk management activities elaborated in SSAR Chapter 18 (Reference 19.24) are those associated with the recommended V&V approach in Section 18.5. However, V&V activities related to the ABWR control room are identified as the applicant's responsibility. They are centered on basically three related sets of activities. First, the design is evaluated with respect to the general design criteria of 10 CFR Part 50 - Appendix A (Reference 19.27) and the NRC requirements and guidelines as reflected in U.S. Nuclear Regulatory Commission, "Standard Review Plan," NUREGs-0800, Washington, DDC, Revision 1, 1984 (Reference 19.28), U.S. Nuclear Regulatory Commission, "Guidelines for Control Room Design Reviews," NUREG-0700, Washington, DC, 1981 (Reference 19.29), and U.S. Nuclear Regulatory Commission, "Clarific-ation of TMI Action Plan Requirements," NUREG-0737, Supplement 1, Washington, DC, 1980 (Reference 19.30).

Second, system/operations analyses are performed for normal and emergency situations. The objective is to evaluate plant operation with a specified crew size and specific control room design including interface design, procedures, etc. The validation activities are to be performed on a functional prototype or simulator, or by walkthrough where appropriate. The acceptance criteria are somewhat vague but address reasonable high-level performance dimensions.

The third aspect to evaluation is a human reliability analysis (HRA) requirement. For each "primary operation action" modeled in the PRA, specific reference to the action (1) will be clearly identified in the EOP, (2) the associated controls and displays will be evaluated by an independent control room design review team to be free from any significant Human Factors Engineering (HFE) discrepancies, and (3) the HEPs assumed in the PRA will be evaluated as to their reasonableness by an independent HRA review team. Primary operator actions are those expected to minimize the adverse consequences of an event modeled in the PRA. While (3) above should not impact the human engineering design, and (2) should be done for all interfaces, it is a good practice to pay special attention to significant human actions. (A similar analysis is postulated for other human actions modeled in the PRA.)

While the overall plan is reasonable, it is unclear why (or how) all of these activities can be the responsibility of the applicant, since test and evaluation activities such as these are integral to the iterative design and analysis process depicted in Section 18.3 Many of the test activities described in this section should not be performed as part of a final test. For example, it seems inappropriate for GE to evaluate the adequacy of the HSI design for a specified crew size at the final design implementation validation or to wait until final design implementation to assure compliance with NRC guidance such as NUREG-0700 (Reference 19.29). Crew size validation is critical to control room and procedure development and should be determined only in the design. Yet, these important design considerations are to be validated by the licensee applicant at the validation phase. In the staff's opinion, these design features should be evaluated much earlier.

19.3 CALCULATION OF CORE DAMAGE FREQUENCY DUE TO INTERNALLY INITIATED EVENTS

3.1 Introduction

Internally initiated events are those which originate within the plant itself, as opposed to earthquakes and other events generally considered "external." Internally initiated events include transients and LOCAs. In addition, loss of offsite power events are considered internal events for PRA purposes. Accidents initiated during full power operation were included in the submittal and this review.

3.2 Initiating Event Frequency

GE's analyses of various initiating events, including unplanned manual shutdowns, are provided in Appendix 19D.3 of the PRA. The frequency estimates of the various initiating events are provided in Table 19.3-1 of this report, along with the staff's comments. The detailed findings are as follows:

1. The frequency of manual shutdowns (planned and unplanned) is based on the results of the analyses documented in NUREG/CR-3862 developed by the Idaho National Engineering Laboratory (INEL). The staff noted that this frequency (one manual shutdown event per reactor-year) is based on operating reactor experience (through calendar year 1985) in the U.S. However, the new GE ABWR design will have more redundancy in safety systems and may have better operating characteristics (with respect to improvements in calibration and maintenance procedures affecting human reliability issues) than existing BWR plants. Thus, it is possible that the actual experience for the ABWR will be somewhat better than that estimated in the PRA. The staff finds GE's estimate to be appropriate at this design stage.

2. GE's estimate for the frequency of vessel isolation (including loss of feedwater events) is about 0.2 per reactor-year, based on the EPRI ALWR requirements document (Reference 19.2). This estimate included contributions due to MSIV closure events, losses of condenser vacuum, and pressure regulator failure events. The frequency of vessel non-isolation events (i.e., a reactor trip with bypass valves available) is about 0.68 per reactor-year. GE claims that these frequency estimates are consistent with the predicted scram frequency and corresponding design requirements documented in the EPRI ALWR Requirements Document (Reference 19.2). The staff noted that current operating reactor experience indicates a value close to about 2.4 per reactor-year. GE has provided neither highlights of the ABWR design improvements in the balance of plant (BOP) systems nor applicable references to such BOP improvements in the ABWR PRA to support the estimate of only one reactor trip per year, which is lower than current experience in the U.S. The staff also noted that, due to lack of design details at this stage, the staff has used one event per

year for the loss of feedwater frequency and one event per year for the MSIV closure event frequency in its review of the ABWR PRA. Unless GE can provide more justification for its estimates, the staff's estimates were considered to be appropriate at this design stage.

3. GE's estimate for the inadvertent open relief valve (IORV) frequency is about 0.01 per reactor-year. The staff notes that this estimate is substantially lower than the value (0.07 per reactor-year) used for the Limerick plant (Reference 19.23). GE has not provided detailed documentation regarding any design improvements made to the multi-stage relief valves to be installed in the future to support this lower unreliability value. In the absence of evidence to the contrary, the staff has used a higher value (0.1) for the IORV event for its independent assessment.

4. The staff's review of the ABWR PRA indicates that GE did not document the details of the contribution of support system failures (such as loss of DC power, loss of service water system) as initiating events. This is an outstanding item. GE should consider the impact of the partial and/or total failure of the support systems (such as the AC power system, the DC power system, the heating and ventilation system, the service water system, the reactor building cooling water system, the reactor service water system, and other applicable auxiliary systems) on plant trips as applicable, including subsequent dependent failures of the mitigating systems needed to provide a vessel coolant makeup function and/or containment heat removal function. The frequency of failure of the support systems (as an initiating event) should be estimated based on the historical frequency of one support system train failure in combination with the failure of the other independent trains of the support systems (including common cause failures of the rest of the support systems), and operator recovery of the initially lost support system train. In developing an event tree for the support system failure as an initiating event, attention should be given to the dependent faults of various components of the mitigating systems to be modeled in the accident sequences.

5. GE's estimate of the loss of offsite power frequency is about 0.1 per reactor-year. This estimate is based on the values for an average site documented in the NSAC report, "Loss of Offsite Power at U.S. Nuclear Power Plants - All Years through 1986," NSAC-111, EPRI NSAC, dated May 1987 (Reference 19.31). However, for the purpose of frequency estimation of the loss of AC power event, GE should consider site-specific parameters (as indicated in the staff's licensing review basis document), such as specific causes (e.g., a severe storm) of the loss of power, and their impact on recovery of AC power in a timely fashion). The staff has considered the impact of site variations (parametric variation in the loss of grid power frequency) on the core damage frequency by

including it in the uncertainty analysis, as reported in Section 8 of Reference 19.20 (see Table 19.3-1). This is an interface requirement.

6. GE's estimates for various postulated LOCA events are also shown in Table 19.3-1. These frequency estimates are the same as those documented and accepted in the GESSAR PRA (Reference 19.22). The staff finds GE's estimate to be appropriate at the design stage.

7. The staff noted that GE did not provide results of accident analyses of postulated interfacing LOCA events as applicable to the ABWR design. This is an outstanding item. Special attention should be paid [illegible] investigating various ways of obtaining an interfacing LOCA event. Items to be considered should include, as a minimum, the following: the number of valves, if any, connected at high an[illegible] [illegible]w pressure boundaries of the reactor primary system; the types o[illegible] [illegible]lves; provisions of the design-specific technical specification[illegible], "Amendment 9 to Chapter 16 of the ABWR Safety Analysis Report," Docket 50-605, dated November 17, 1989, (Reference 19.32) with respect to testing and maintenance intervals, and postulated post-testing and maintenance errors; valve position indication and/or its equivalent in the main control room; pressure rating of the downstream piping, provisions of the reactor primary system geometry with respect to conservation of mass of the primary system; and continued core cooling with the unaffected system, if any, during an interfacing LOCA event. GE must estimate the frequency of interfacing LOCA events to account for the above considerations and historical data (such as the event which occurred at the Hatch facility).

8. The staff also noted that GE did not document the results of the accident analyses of postulated LOCA events outside the containment (in particular, steam line breaks in the RCIC steam piping and the RWCU lines) in combination with failure of the isolation valves. This is an outstanding item.

3.3 Success Criteria

GE's core cooling success criteria for transients, postulated LOCA events, and failure-to-scram events are provided in Sections 19.2 and 19.3 of the ABWR SAR. The staff's review indicates that GE has determined design-specific core cooling success criteria which are based on realistic thermal-hydraulic (T-H) calculations and assumptions, and has documented this as part of its revised submittal to the staff's RAI (References 19.9 and 19.11). For example, following the IORV and stuck open relief valve events, GE has determined that the RCIC system alone cannot provide sufficient coolant makeup to the reactor (due to lack of sufficient steam) and has modeled the system characteristics accordingly for the IORV event and station blackout events (based on availability of battery power for only eight hours). However, the text that describes the IORV event is inconsistent with the IORV event trees (Table 19.3-3). This is a confirmatory item.

As part of the risk review, the staff raised questions (Reference 19.13) regarding the adequacy of one of the three RHR trains to remove heat from the suppression pool following a failure-to-scram event in combination with a vessel isolation event and the failure of boron injection. As part of the staff's reactivity accident analysis efforts, detailed calculations were performed in NUREG/CR-5368, "Reactivity Transients," dated January 1990, (Reference 19.33) to predict the amount of heat produced following a vessel isolation event and the additional demand on suppression pool cooling systems. The staff thereby determined that two (rather than one) of the three RHR trains will be required to provide adequate pool cooling following a failure-to-scram event with a failure to provide poison injection. In a response to the staff's RAI (Q725.65), GE provided the results of its thermal-hydraulic calculations for this ATWS scenario. This response confirms the staff's finding regarding the minimum suppression pool cooling requirements of the ABWR RHR system.

The staff also notes that additional investigation is currently underway to determine the logical minimum injection flow to the vessel needed to avoid core damage following a vessel isolation event coupled with failure to scram and failure to provide poison injection. Preliminary calculations indicate that a flow rate of 800 gpm from a HPCF train alone may not be sufficient to keep the water level above the top of the active fuel for the above scenario. Meanwhile, the staff has used GE's success criteria for the MSIV closure event in its requantification of the ATWS-induced sequence frequencies. However, if the final thermal-hydraulic calculations demonstrate a need for two or more trains of the high pressure injection systems (that is, more than an 800 gpm flow rate) to avoid core damage for the above scenario, then the overall ABWR core damage frequency and risk could increase significantly. This is a confirmatory item.

GE should provide further documentation in the area of success criteria, as described in this section.

3.4 Accident Sequence Definition

GE's discussion of accident sequence definition is provided in Section 19.2.3 of the ABWR SAR. The assumptions used to define and develop accident sequences are the same as those documented in Appendix A of Reference 19.2. Basically, all of the accident sequences are assumed to occur when the reactor is at normal full power operation, and a transient and/or a postulated LOCA challenges the safety systems. GE has made use of traditional (WASH-1400, NUREG-75/014, "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," dated October 1975 - see Reference 19.34) event trees to develop core damage accident sequences following an anticipated transient or a postulated LOCA. This event tree method used to develop sequences is acceptable to the staff. The following are the staff's general observations on GE's application of event trees:

1. The development of event trees for various initiating events (anticipated or postulated) has reflected operational characteristics of the reactor primary system as it is intended by the ABWR NSSS design and has also reflected realistic core cooling criteria (as previously discussed). The staff notes that the timing of various demands for safety functions has been reflected appropriately. The staff also notes that the modeling of the present state of a particular branch point (in particular, system unavailability and failure of human recovery or restoration) in any given event tree, has reflected the previous states (success or failure) of the preceding branch points, including the initiating event. The staff did, however, find some minor errors in the GE event trees. These errors are addressed later in this section and in Reference 19.20. All staff calculations have been based upon the corrected event trees.

2. The staff notes that the sequence development has been terminated when the failure or success of achievement of hot shutdown conditions (as defined in the ABWR design-specific Technical Specifications documented in Reference 19.32) occurs. In other words, the sequence development (like that of the RSS analysis documented in Reference 19.34) has not been carried out to cold shutdown conditions.

   The staff also notes that (as in most PRAs) GE has not applied event trees or equivalent methods to develop sequences that could occur during operating modes other than full power, such as the startup and refueling modes. The staff believes that risk (due to certain scenarios) could be incurred during the refueling operational conditions as permitted by existing regulations. The evidence for this belief is from detailed sequence analysis performed (Reference 19.33) for the refueling mode of a typical BWR plant. For example, one of the critical sequences involves inadvertent loading of fuel assemblies with two (minimum logical) adjacent control rods withdrawn from the core with the vessel top head and the reactor enclosure being in an open condition. GE has submitted a separate evaluation, Chapter 19, Appendix L, "ABWR Shutdown Risk," to address these issues. This appendix is currently under staff review and will be addressed in a supplement to this SER. An adequate treatment of shutdown risk will be required prior to design certification. This is an outstanding issue.

3.5 System Modeling

GE has used the traditional fault tree method to develop various system failure models (combinations of component failures) which are used to develop an estimate of various system unavailabilities. A condensed version of these fault trees (17 trees in total) for various safety systems is documented in Appendix 19D.6 of the ABWR SAR. A summary description of system design features affecting unavailabilities, and the staff's evaluations of them, are provided in Section 19.2 above. The staff notes that these system models are grouped based on four basic safety functions (described in Section 19.2.1), namely, reactor coolant makeup, containment heat removal, reactivity control, and other auxiliary supports. The staff finds GE's method of employing fault trees to develop functional failure models to be acceptable. The system model developed by GE also includes combinations of train level components or subcomponents for systems such as the High Pressure Core Flooder (HPCF) and the Low Pressure Core Flooder (LPCF) system. The staff also finds that the limit of resolution employed in the system failure model is consistent with the availability of failure data for a component or a subcomponent.

The staff notes that GE has considered various operational (normal operating and standby) characteristics of all the front line and support systems as part of the development of the system failure models. In particular, the LPCF system failure models have reflected design-specific capabilities (e.g., unique RHR room cooling design requirements), and emergency procedures to characterize the available minimum recovery time (a critical parameter in containment heat removal analyses) to be modeled in the system model.

The staff requested that GE document all critical assumptions affecting the system failure models. GE provided, as part of the ABWR SSAR Amendments (References 19.11 and 19.14), its responses to the staff's RAI in this area. The staff finds these responses acceptable.

As part of the staff's review, the system failure models (documented in the form of fault trees) were requantified for various safety systems and system combinations (such as HPCF train B and train C) and the requantified results were documented in Section 4 of Reference 19.20. These results, with staff comments, are provided in Table 19.3-3 of this report.

It was noted during the review that common mode failures were incorporated into the GE model at the train level of each system. Due to the fact that the calculated core damage frequency for the ABWR is quite low due, in part, to the existence of multiple redundant systems, common mode failure probabilities could possibly dominate the results. Therefore, the staff requires further justification from GE that its train-level common mode failure analysis was able to capture the full contribution to common mode failure probability had it been calculated at the component level. This is an outstanding item.

Except where explicitly noted above, GE has adequately and appropriately generated fault trees for the various systems, and has calculated estimated unavailabilities with which the staff has found no reason to disagree.

In developing the ABWR PRA, GE has had to make assumptions about the design and reliability of systems outside of the ABWR design certification. Since the design of these interface systems are outside of the scope of the certified design and are the responsibility of the utility/applicant, it is important that the reliability assumptions and risk-significant insights used by GE in developing the ABWR PRA are transferred to future applicants. The staff requires that GE provide a list of these interface systems, the assumed reliability for each interface system, and any safety significant insights GE believes are important to designing the interface systems to meet the assumptions of the PRA. This is an Outstanding Item. Future applicants must demonstrate how their design for interface systems outside of the ABWR certified design meets the reliability assumptions and design insights provided by GE. This is an Interface Requirement.

3.6 Data Analysis

3.6.1 Hardware Reliability Data Analysis

GE's reliability data for various components are provided in Appendix 19D.6 of the ABWR SAR. GE's systematic documentation of the ABWR design-specific data for various safety systems includes: (A) General Electric document 22056, Rev. 2, "Failure Rate Data Manual" (Reference 19.36), (B) GESSAR II SAR (Reference 19.22), (C) DOE ABWR ECCS Instrumentation Fault Trees, 1987 (Reference 19.37), and (D) IEEE Standard 500, 1984 (Reference 19.38). The staff notes that, wherever the design-specific data for certain ABWR component are not readily available, GE has employed GESSAR II data and IEEE 500 data for similar ABWR components. This method is acceptable with respect to system unavailability quantification purposes. However, GE should provide documentation on the justification regarding the applicability of certain generic common cause/mode failure data to ABWR design-specific components (such as the diesel generators, the HPCF pumps, the LPCF pumps, and the RHR heat exchangers) involved in the system unavailability modeling. Such justification should also include the conditions under which generic common cause/mode failure data were evaluated, and the ABWR design-specific component data as modeled in the PRA. This is a confirmatory item.

Our review also indicates GE's use of the reliability data for the individual components of the ABWR design seems reasonable with the exception of the adequacy of use (in the ABWR design) of the following component data:

1. The RHR pump mechanical failure.
2. The HPCF pump (no experience on such a component).

GE should provide justification for the use of its data for the above components. This is an outstanding item.

The staff, as part of its review, performed an uncertainty analysis of certain critical parameters, but based only on system level and human failures, and documented the results in Section 7 of Reference 19.20. Therefore, the staff's uncertainty estimates on the ABWR core damage frequency do not include the impact of large variations, if any, in the above critical components.

3.6.2 Test and Maintenance Data Analysis

GE's estimates on unavailability due to test and maintenance are provided in Appendix 19D.6 of the ABWR SAR. GE's systematic documentation of the ABWR design-specific test and maintenance data for various safety system components (the HPCF pump, RCIC turbine lubrication system components) includes: (A) General Electric document 22A6278, Rev. 2, "HPCF Technical Specifications" (Reference 19.39), (B) GESSAR II SAR (Reference 19.22), (C) General Electric Document NEDC 30936P (Reference 19.40), (D) BWROG Technical Specifications Improvement Methodology, Part I, 1985 (Reference 19.41), and (E) the ABWR design-specific Technical Specifications (Reference 19.32). GE has also documented the methods of analysis and results of data on unavailability due to test and maintenance for certain critical ABWR components. It is also noted that, for certain ABWR components (the RCIC turbine pump, the HPCF pumps, the RHR pumps), GE has employed the use of GESSAR II design information to obtain data on unavailability due to test and maintenance. However, GE has not provided justification regarding the applicability of GESSAR II design information to the ABWR design (on a train basis). Such justification should also include the differences in design features, if any. This is a confirmatory item.

3.7 Human Reliability Analysis

The purpose of this portion of the review is to assess the ABWR Probabilistic Risk Assessment (PRA) related Human Reliability Analysis (HRA) on 12 review items. These items relate to HRA documentation, the material available to support the HRA, the types of analyses performed, the quantification methods and performance shaping factors utilized, the completeness and types of human actions modelled, and the sources of generic data used. In addition, since the ABWR will include more automation and advanced human-system interface technology than previous nuclear power plant designs, the review also focuses on how the effects of the advanced technology (on the operator's [illegible]e/tasks) are addressed in the HRA.

The review methodology, results and conclusions are presented in the following sections.

3.7.1 HRA Review Methodology

3.7.1.1 ABWR PRA/HRA Review Criteria

1. Adequacy and Completeness of the Documentation - The documentation of the HRA should provide a description of the analyses, an audit trail for each analysis performed and each human error probability (HEP) derived, supporting rationale, and source materials.

2. Material Available to Support the HRA - The materials (such as procedural guidance and control room panel design information) available to the HRA team should provide a clear understanding of human involvement in the ABWR.

3. Human-System Analyses Performed - The human-system analyses performed (such as detailed task analyses) should provide a clear understanding of the task requirements and demands on the operating staff, their interfaces with plant equipment, and the time constraints within which critical tasks must be accomplished. Also, the human-system analyses should provide a clear understanding of how this knowledge was used to support the PRA model development for the inclusion of human actions in the event and fault trees. Finally, the human-system analyses should demonstrate how state of knowledge techniques were used to evaluate the utilization of screening analyses and other techniques to identify important human actions.

4. Types of Human Task Actions Analyzed - The extent to which the variety of human interactions with the plant systems and components were considered and how they were modelled. As per the PRA Review Manual, NUREG/GR-3485, dated 1985 (Reference 19.42), the human actions should include "operating, calibrating, testing, monitoring, communicating, responding, inspecting, deciding, and managing." Attention was directed to the following types of actions:

   - Pre-accident and during-accident human actions,
   - Errors of omission and commission,
   - Miscalibration and misrestoration (component restoration errors),
   - Cognitive errors, and
   - Recovery errors.

5. Adequacy of the Human Action Modelling - Human actions should be modelled within the event and fault trees.

6. Quantification Metnods Used to Estimate HEPs - The PRA should use HRA methods (such as Technique for Human Error R e Prediction [THERP] or Human Cognitive Response [HCR] to quantify the errors in terms of HEPs and specify what types of behavioral/performance models (such as action dependency) were utilized.

7. Performance Shaping Factors (PSFs) Evaluated - PSFs should be used to identify human errors, how they are characterized, and how their effects are quantified and used in the analysis.

8. Treatment of Advanced Technology - The influences of the advanced technology aspects of the ABWR should be accounted for in the analysis. In addition, the PRA model should reflect the changes in the operator's tasks and role in the system resulting from the increases in system automation. The HRA methods and data should analyze any advanced technologies.

9. Generic Human Error Data Sources - The HRA should use generic source data for HEP estimates.

10. Generalization from Earlier PRAs - The analyses and data from earlier PRAs should have a rationale to justify any generalizations, and if/why/how the values were modified for use in the ABWR.

11. Sensitivity Modelling Approach - Sensitivity or uncertainty analyses should be performed on the HEP values; and should state how error factors were determined, and what criteria were used for performing the analysis.

12. Insights Gained from the Analyses - Human actions impacting plant risk and insights should be factored into system/operational design. Unlike most PRA/HRAs, which are performed after the plant is designed, the ABWR PRA/HRA can be utilized to provide information on significant or sensitive human actions which can be used as an input to design of hardware, software, and procedures.

3.7.1.2 Documentation Sources

The main source of information on the HRA was the ABWR Standard Safety Analysis Report (SSAR) (Reference 19.24). The pertinent sections in the SSAR were from Chapter 18, "Human Factors Engineering," and Chapter 19, "Response to Severe Accident Policy Statement."

In addition to the SSAR, several other sources of information were used:

- GE's responses to the Request for Additional Information (RAI) questions 620.6, dated October 9, 1990 (Reference 19.43), GE response to RAIs 620.7 and 620.13, dated October 9, 1990 (Reference 19.44), and GE response to RAIs 621.1 through 621.11, dated December 17, 1990 (References 19.24 and 19.45).

- Information obtained on advanced technology aspects of the ABWR obtained in the "Foreign Travel Trip Report - Japan" in October 1990 dated December 12, 1990 by the review team to Hitachi and Toshiba (Reference 19.46).

- GESSAR II PRA (Reference 19.22).

- Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, Draft Report for Interim Use and Comment dated 1980 and Final Report dated 1983 (References 19.47 and 19.48).

- Systematic Human Action Reliability Procedure (SHARP), EPRI NP-3583 dated 1984 (Reference 19.49).

- Post Event Human Decision Errors: Operator Action Tree/Time Reliability Correlation, NUREG/CR-3010 dated 1982 (Reference 19.50).

3.7.2 HRA Review Results

3.7.2.1 Adequacy and Completeness of the Documentation (Item 1)

Review Item 1 is found to be an outstanding item,

The HRA-related documentation provided in the ABWR Standard Safety Analysis Report (SSAR) was found to be incomplete and, therefore, not adequate in terms of providing the information needed to evaluate the approach taken to human action modelling in the PRA. Detailed rationale or discussion for the HEP estimate was provided for only six of the human actions modelled in the PRA. In general, the documentation did not identify the type of analyses used, how the HRA analysis methods were implemented (such as specific reference to parts of Swain's Handbook of Human Reliability Analysis - NUREG/CR-1278, Reference 19.48 and

19.43), what performance models and performance shaping factors were considered, or how HEPs were quantified. Several human actions identified in "component failure rate data" tables were not identified in fault trees and human actions found in the fault trees that were not listed in the tables. For example, Fault Tree Figure 19D.6-16a, "Reactivity Controls" lists "Operator Fails to Inhibit ADS" (ADSIN) but was not listed in Failure Data Table 19D.6-4, "ADS Failure Data," which lists or should list all failures associated with ADS.

3.7.2.2 Material Available to Support the HRA (Item 2)

Review Item 2 is found to be an interface requirement.

Based upon the information contained in SSAR Chapters 18 and 19 and GE's response to RAI Question 621.2, it was not possible to determine whether the material available to the HRA team was adequate for a detailed evaluation of human action or an estimate of the HEPs. While RAI Question 621.2 specifically asked for this information, GE's response did not directly address the request. In Section 18.5.3.1 and the response to RAI Question 621.2, GE indicated that the HEPs are to be validated by an independent HRA team after additional design detail is available.

3.7.2.3 Human-System Analyses Performed (Item 3)

Review Item 3 is found to be an interface requirement.

The available documentation provided little evidence that thorough human-system analyses were performed in support of ABWR PRA/HRA activities. RAI Question 621.11 was a direct request for information related to the human-systems analysis approaches including use of task analysis, HEP estimation methods, screening analyses, and HEP modification for the ABWR. GE's response to the question did not address task analysis. In response to Question 620.6, it was indicated that system-level operating procedures and emergency procedure guidelines were used as a basis for task analyses (in support of man-machine interface design). However, as indicated in GE's response to Question 620.13, the sample task analysis which was provided for review was not to a suitable level of detail to support the HRA. In response, GE indicated that the task analyses will proceed and become more detailed in an iterative fashion as the design becomes better developed. Further, it is indicated that task analyses for transient and accident scenarios estimated will be performed.

3.7.2.4 Types of Human Task Actions Analyzed (Item 4)

Review Item 4 is found to be acceptable.

Based on an analysis and classification of the human actions identified in the fault and event trees (Reference 19.20), there appears to be a good mix of types of human interactions in the ABWR.

3.7.2.5 Adequacy of the Human Action Modelling (Item 5)

Review Item 5 is found to be an interface requirement.

The ABWR human action modelling appears to be reasonable in terms of conventional control boards. However, since the ABWR has an advanced main control board, there is a concern about the adequacy of human action modelling with regard to its increased automation and advanced technology. Also, there were event tree branch points depicted as hardware failures that should include important human actions, but apparently do not. For example, the failure to recover offsite power or one diesel in either two- or eight-hour branch points, described in SSAR Table 19D.4-7 for loss of offsite power and station blackout (SBO) event trees, should have a documented human action component.

3.7.2.6 Quantification Methods Used to Estimate Human Error Probabilities (Item 6)

Review Item 6 is found to be an outstanding item.

The ABWR human action modelling appears to be reasonable in terms of conventional control boards. However, since the ABWR has an advanced main control board, there is a concern about the adequacy of human action modelling with regard to its increased automation and advanced technology. Also, there were event tree branch points depicted as hardware failures that should include important human actions, but apparently do not. For example, the failure to recover offsite power or one diesel in either two- or eight-hour branch points, described in SSAR Table 19D.4-7 for loss of offsite power and station blackout (SBO) event trees, should have a human action component which is not documented.

3.7.2.7 Performance Shaping Factors Evaluated (Item 7)

Review Item 7 is found to be an interface requirement.

Based upon the available documentation, it did not appear that performance shaping factors (PSFs) were considered in the HRA. PSFs for ABWR human actions were only described within the context of three errors. Even for these three errors, the treatment of PSFs in the documentation was incomplete. For errors taken from the GESSAR II PRA, PSFs were only considered for the calibration of sensors actions, and this was limited to stress and dependence. Due to immaturity of the design, important PSFs such as emergency operating procedures (EOPs) and human-system interface (HSI) design were not analyzed.

3.7.2.8 Treatment of Advanced Technology (Item 8)

Review Item 8 is found to be an interface requirement.

As presently documented, the HRA is limited in its treatment of the advanced technology aspects of the ABWR and little evidence exists that the changing role of the operator due to increased automation was analyzed for its HRA implications. RAI Questions 621.9 and 621.10 specifically requested information on these issues, however, GE's responses were sparse and did not address the questions in a direct manner. In the response to RAI Question 621.10, GE indicated that an independent HRA team will validate the ABWR HEPs (see also Section 19.3.7.2.3) and that this team will also analyze potential "new" operator errors. The identification of new operator errors could potentially require PRA remodelling and/or cause significant changes to the HRA/PRA results.

3.7.2.9 Generic Human Error Data Sources (Item 9)

Review Item 9 is found to be an outstanding item.

In general, the documentation in the SSAR provided general information on the source data used for HEP estimates. While methods were identified, such as NUREG/CR-1278 (Reference 19.48) which contains such data, references to specific data tables were generally not in the documentation. In addition to NUREG/CR-1278, four other generic sources were identified. However, two of these turned out to be secondary sources, both of which identified the same time-reliability correlation (TRC) figure originally published in J. Wreathall's "Operator Action Trees: An Approach to Quantifying Operator Error Probability During Accident Sequences," NUS-4159, dated 1982 (Reference 19.51). GE should also justify the use of human error databases which are largely based on simple manual control tasks (such as is provided in NUREG/CR-1278) for estimation of (monitoring and supervisory control) operator tasks in an advanced reactor.

3.7.2.10 Generalization from Earlier PRAs (Item 10)

Review Item 10 is found to be an interface requirement.

As indicated in the SSAR, most of the HEPs were taken from the GESSAR II PRA (Reference 19.22). The use of these values was judged acceptable by GE because of improved human-system interface design and greater automation. In general, the HEPs were not modified for use in the ABWR. The staff has conclude that GE should justify the use of GESSAR II HEPs since a detailed analysis of human actions in the ABWR has not been completed and the design and procedural detail has not been completed.

3.7.2.11 Sensitivity and Uncertainty Modelling Approach (Item 11)

Review Item 11 is found to be an outstanding item.

The available documentation did not indicate that HEP sensitivity analyses or HEP uncertainty bounds (or error factors) were analyzed or calculated for the effect on core damage frequency.

3.7.2.12 Insights Gained from the Analyses (Item 12)

Review Item 12 is found to be an confirmatory item.

The staff has concluded that GE has developed a reasonable plan to use information and insights gained from the HRA to support the system/operational design. The acceptability of any insights realized from the HRA however, must await further design development.

3.7.3 HRA Review Conclusions

Although there are several strong points in the ABWR PRA/HRA process, the HRA documentation is generally lacking: there is little evidence of detailed analyses, many HEPs are assimilated into the ABWR HRA from an earlier PRA with little objective analysis-based justification, and at present, there does not appear to be much consideration of the advanced technology aspects of the ABWR control room. With respect to the latter two issues, GE plans to "validate" its HRA with the independent analysis of an HRA review panel, but this has not been accomplished. In this sense, the HRA is still incomplete. The identification of significant "new" operator errors could potentially require PRA remodelling and/or cause significant changes to the HRA/PRA results.

In summary, 11 of the 12 review items were classified as either "Outstanding Items" requiring additional information (Items 1, 6, 9, and 11), "Interface Requirements", (Items 2, 3, 5, 7, 8, and 10) or "Confirmatory Items (Item 12). Item 4 was found to be "Acceptable."

3.8 Quantification of Accident Sequence Frequencies

GE's quantification approach used in combination with its design-specific and generic data to quantify the sequence frequency estimates is provided in subsection 19D.4. These traditional and conventional methods are acceptable to the staff.

The staff notes that this process was carried out by developing a single set of branch point probabilities for the various systems (and combinations of systems that appeared side by side) in the event trees. This is an acceptable but somewhat cumbersome approach with respect to assuring that there has been no double counting of failure probabilities (i.e., an underprediction of overall sequence failure frequency). Based upon the review, the staff believes that GE took sufficient precautions within the modeling to minimize the possibility of double counting. The staff did, however, find some minor errors in the GE event trees. In the staff's requantification effort, these errors were corrected. This is discussed in detail in Reference 19.20.

As part of the staff's review, an uncertainty analysis using NUREG-1150 type methods was performed on the corrected model to evaluate the impact of the variations of certain critical system failures, human actions, and initiating events on the ABWR core damage frequency. These results are discussed in detail in Section 7 of Reference 19.20. The results are provided in Table 19.3-4 along with a summary of mean frequency estimates for various accident classes. Table 19.3-4 also lists a relative ranking of dominant sequence frequency estimates.

3.9 Quantification of Accident Sequence Class Frequencies

As is done in most PRAs, GE has grouped postulated accident sequences into a small set of classes of accident sequences. GE's analysis of the classification of postulated accident sequences is provided in Subsection 19D.5.2 of the ABWR SAR. An itemization of the definitions used t naracterize these accident classes is provided in Table 19.3-2. The staff's review of these accident classes indicates that the classification of accident sequences is based on the suppression pool conditions (subcooled or saturated) and the timing of the containment failure due to loss of decay heat removal systems following a postulated accident sequence. These definitions seem reasonable.

The staff notes that these definitions are somewhat consistent with those used in the Limerick PRA, although some accident sequences have been regrouped into other classes. For example, sequences involving failure-to-scram events followed by the failure of boron injection and loss of high pressure coolant makeup to the reactor, have been grouped into the loss of coolant inventory makeup (with successful scram) sequences for which a subcooled suppression pool condition is expected for a longer time. The staff notes (Reference 19.13) that the amount of heat dumped into the suppression pool for these two groups of accident sequences will be completely different and will result in a completely different containment response and different demand on the suppression pool cooling systems. Further discussion of containment states will be found in Section 19.6.

3.10 Summary of GE's Estimates of Core Damage Frequency Due to Internally Initiated Events

3.10.1 Initiating Events and Principal Contributors

GE's PRA has estimated the relative contributions of various initiating events (transients and LOCAs) to the total core damage frequency. An estimate of the relative contributions of these initiating events to the overall core damage frequency (transients and LOCA events only) is provided in Table 19.3-5. Because GE has not performed an uncertainty analysis, the relative contributions are based on point estimates. This is an outstanding item. However, as part of the staff's review, an uncertainty analyses was performed (using the staff's NUREG-1150 methods) of major sources (critical system failures, critical human failures, and major initiating events) of uncertainties contributing to postulated core damage events. Thus, the staff has obtained estimates of the relative contributions of various initiating events which, in addition to using revised failure probabilities, are based on arithmetic mean estimates of the core damage frequency. The staff's estimates of the relative contributions of initiating events are also provided in Table 19.3-5. This table indicates that failure-to-scram events following anticipated transients are the largest contributor (about 31.5 percent) to the point estimate core damage frequency. The second dominant contributor (about 26.4 percent) is the loss of the main feedwater system. Turbine trip events, reactor isolation events, and inadvertent open relief valve events contribute equally (about 3 to 4 percent) to the core damage frequency. The collective contribution of all postulated LOCA events is found to be very small (about 3 percent).

It should be noted that, although GE has provided substantial design improvements to the scram system, failure-to-scram events still contribute significantly to the total core damage frequency. This is primarily due to the staff's upward revision in the demand failure probability of the overall scram system. In view of recent experience (a failure event in one of the West European nations) applicable to a similar scram system discussed in the October 12, 1991, edition of Nucleonics Week (Reference 19.52), the staff judges this revision of the scram demand failure probability to be appropriate. The staff also finds that, unlike currently operating BWRs, station blackout events do not contribute significantly to the core damage frequency. This is primarily due to incorporation of the onsite gas-turbine generator as part of the ABWR design. With respect to station blackout events, the staff notes that the decrease in reliability due to removal of a steam-driven high pressure system (such as the HPCI system in earlier designs) in the ABWR design is well compensated by a substantial reliability improvement due to the addition of a gas-turbine generator in the ABWR design. Incorporation of an additional motor-driven high pressure system train (the C train of the HPCF system) in the ABWR is found to have an insignificant impact on the contribution of LOCA events to the overall core damage frequency.

3.10.2 Accident Sequences

GE's discussions regarding accident sequence descriptions are provided in Appendix 19D.4 of the ABWR SAR. As previously indicated, this section provides only descriptions of groups of accident sequences, that is, accident sequence classes. The same information is also provided in Section 19.3 of the ABWR SAR. A description of these accident sequence classes along with the frequency estimates used in the PRA and in the staff's review is provided in Table 19.3-4. The staff notes that both the PRA and the staff's review did not attempt to develop a ranking of all individual accident sequences. This is primarily because neither GE nor the staff attempted to develop sequence level Boolean equations which would have yielded the detailed information required.

However, the staff's revised frequency estimates provided in Table 19.3-4 indicate that the high pressure core melt sequences (Class IA) involving loss of all high pressure coolant makeup to the reactor, dominate the overall core damage frequency. The second dominant accident class consists of the high pressure core melt sequences (Class IV) involving failure-to-scram events coupled with boron injection system failures. The third dominant accident class consists of the low pressure melt sequences (Class ID) involving loss of high pressure and low pressure coolant makeup to the reactor. Both Class II sequences (sometimes referred to as "TW" sequences as in the WASH-1400 nomenclature) involving loss of the containment heat removal function (prior to the failure of loss of coolant inventory makeup to the reactor), and station blackout sequences (Classes IB-1, IB-2, and IB-3), are found to be insignificant contributors to the overall core damage frequency. The staff notes also that the most dominant accident class (that is, Class IA sequences) contributing to the overall core damage frequency, is the same for both GE's risk analyses a d the staff's review of them.

3.10.3 Observations

1) There appears to be substantial improvement in the reliability of safety systems. The reliability improvements include enhancements in redundancy and diversity applied to the design

of the safety systems. Lack of detailed information in the areas of instrumentation and control systems precludes the staff from drawing similar conclusions in those areas.

(2) Our review of the probabilistic analysis of the SLC system (needed to mitigate failure-to-scram events) indicates that this system is manually initiated. Relative to currently operating BWR designs (such as Limerick), the above feature is not a design enhancement, considering the very short time available for the operator to initiate the SLC system following a failure-to-scram event. GE claims that the addition of the FMCRD along with the independent electrically driven control rod insertion feature minimizes the demand on the SLC system, because of the lower likelihood of a need for SLC initiation. In a meeting with the staff on August 6, 1991, GE indicated that the SLC system design has been changed for automatic initiation. The staff has requested that GE docket this information. The acceptability of SLC initiation is currently under staff review and is discussed in Chapter 4 of the SER.

(3) When core damage frequencies on the order of 10^{-7} per reactor-year are calculated, the question of the physical significance of such an extremely low number naturally arises. To understand such a number, it is necessary to understand the context in which it is presented.

- The main reason that the number is so low is that the designers have intentionally done their best to address all known accident scenarios. Thus, it is expected that the core damage frequency estimate would be low. It would be far more disturbing if it were not low.

- Like most other disciplines, it is a limitation of probabilistic analysis that it can only address known accident scenarios and failure modes. Thus, any new generic issue or plant-specific issue that may be discovered in the future, once added into the probabilistic analysis, could revise the core damage frequency upward by perhaps an order of magnitude or more. Of course, the same new issue, adding the same number to the CDF, might change the core damage frequency of a present day plant by only a few percent, since the present day plant would start out with a higher base value of the CDF.

- Realistically, one must allow for this possibility of some new issue changing the core damage frequency once it is discovered. Nevertheless, until it is discovered, the core damage frequency estimate properly remains at a very low value.

- The entire PRA is based upon the assumption that the plant will be built exactly as described in the SAR. Thus, the low core damage frequency estimate is contingent upon there being no significant deviations during construction, and no surprises discovered during subsequent system walkdowns.

- The core damage frequencies produced by all PRAs inherently have large uncertainties. Therefore, comparisons of frequencies between PRAs or with absolute limits or goals are not simply a matter of comparing two numbers. If a probability distribution is available, it is more appropriate to observe how much of the probability distribution lies below a given point, which translates into a measure of the probability that the point has not been exceeded. Thus, although the central tendency of a calculation may be very low, there is still a finite probability of a higher core damage frequency. Even in the case of the ABWR PRA, where GE did not calculate a probability distribution, it must be recognized that the uncertainty range will extend significantly above GE's estimate.

All of these points should be considered when using core damage frequency estimates such as these.

3.10.4 Conclusion

A review of the findings documented in previous sections of this report indicates that no unique highly dominant scenario exists with respect to core damage frequency (that is within the enhanced design umbrella as documented in Reference 19.2). This finding is based on the level of design detail applicable to the ABWR NSSS and associated BOP systems as documented in the ABWR SAR.

Resolution of the outstanding items in the PRA (as discussed above and summarized in Section 19.10 at the end of this evaluation report) may result in a significant increase in the estimate of core damage frequency.

Table 19.3-1 A Summary of Initiating Event Frequency Estimates For The ABWR Design

Initiating Event	Frequency (Per Reactor-Year) PRA	Review	Comments
Manual Shutdown	1	1	1/
Isolation Event	0.1	1.	2/, 3/
Loss of Feedwater	0.1	1.	2/, 3/
Turbine Trip	0.68	1.	2/, 3/
Inadvertent Open Relief Relief Valve	0.01	0.1	2/, 4/
Loss of Offsite Power	0.1	0.1	5/
Small Loca event	1.2 E-3	1.2 E-3	
Medium LOCA Event	6.7 E-4	6.7 E-4	
Large LOCA Event	2.1 E-4	2.1 E-4	
Anticipated Transient	0.99	3.2	2/, 3/

Notes:

1/ The staff has used the same value in its accident sequence requantification efforts.

2/ For review comments, refer to Section 4.1 of Reference 19.20.

3/ GE's estimates are not based on historical data.

4/ GE's low estimate (relative to historical data) could be justified through documentation of the improvements made to the ABWR multi-stage relief valves.

5/ GE's estimate does not consider the characteristics of all applicable grids in the U.S.

Table 19.3-2 A Summary of GE's Assignment of Accident Classes for Various Accident Sequences

Accident Class	Description
IA	Transients followed by failure of the high pressure coolant makeup to the reactor and a failure to depressurize the reactor in a timely fashion.
IB-1	Short term Station Blackout (SBO) events with RCIC failure, onsite power is recovered in eight hours.
IB-2	SBO events with RCIC available for core coolant makeup for approximately eight hours.
IB-3	SBO events (more than eight hours) with RCIC failure.
IC	ATWS events without boron injection with failure of coolant makeup to the reactor.
ID	Transients followed by failure of high pressure coolant makeup to the reactor, successful depressurization of the reactor, and failure of low pressure coolant makeup to the reactor.
IE	ATWS events followed by failure of high pressure coolant makeup to the reactor and failure to depressurize the reactor.
II	Transient, LOCA, and ATWS (with boron injection) events, with successful coolant makeup, but with potential prior failure of containment.
IIIA	Small and medium LOCA events, followed by failure of high pressure coolant makeup to the reactor and failure to depressurize the reactor.
IIID	All LOCA events followed by failure of high pressure coolant makeup to the reactor, successful depressurization of the reactor, and failure of low pressure coolant makeup to the reactor.
IV	ATWS events followed by failure to provide boron injection and successful high pressure coolant makeup to the reactor; ATWS events followed by successful boron injection, but failure to keep the vessel at high pressure, resulting in boron dilution.
V	All core damage events followed by failure to prevent suppression pool bypass.

Table 19.3-3 A Summary of Staff Review Findings on GE's System Unavailability Estimates

System Combination	Transients	LOCA Events	Loss of Offsite Power	Comments
Reactivity Control				
Scram & ARI	1 E-8	1 E-8	1 E-8	2/
High Pressure Coolant Makeup				
RCIC & HPCFB & HPCFC	1 E-4	3 E-3	7 E-4	1/
HPCFB & HPCFC	2 E-3	E-3	5 E-3	1/, 3/
HPCFB or HPCFC	4 E-2	6 E-2	4 E-2	1/, 4/
RCIC	4 E-2	4 E-2 5/	4 E-2	1/
Low Pressure Coolant Makeup				
ADS or Manual Depr.	2 E-3	Negligible	2 E-3	6/
RHRA & RHRB & RHRC	5 E-5	1 E-4	E-4	1/
RHRA or RHRB or RHRC	3 E-2	4 E-2	[illegible]-2	1/
Suppression Pool Cooling Mode				
RHRA & RHRB & RHRC (Start and Run)	5 E-4 7/	5 E-4	2 E-3	1/

Notes:

1/ The staff's requantification of the GE fault tree has provided similar results.

2/ The staff's review has provided somewhat lower credit for the ABWR design improvements and resulted in a conditional probability of 1E-6 per demand.

3/ The staff's requantification of the same fault tree identified a modeling error and corrected it. This resulted in a probability of 7.95E-3 per demand.

4/ The staff's requantification of the same fault tree identified a modeling error and corrected it. This resulted in a probability of 7.33E-2 per demand.

5/ GE has taken credit for the RCIC system for small LOCA events only. The staff's review agrees with this.

Table 19.3-3 (Continued)

6/ The staff's review provided an alternate estimate of 2E-2 per demand.

7/ This estimate is reasonable for transients with successful scram. However, this unavailability could be higher depending on minimum cooling requirements for ATWS events (in particular, isolation events followed by failure to scram). This is an open item for GE to resolve.

Table 19.3-4 A Summary of GE Results and the Staff's Review Findings on Dominant Sequence Frequency Estimates

Class 1/	Frequency Estimates GE 2/	Point (per RY) 4/ Staff 3/	Staff's Mean Estimate 6/	Ranking of Staff's Estimate 4/
IA	4.3 E-8	2.4 E-7	3.4 E-7	1
IB-1	1.9 E-9	1.9 E-8	1.8 E-8	5
IB-2	1.6 E-9	7.9 E-9	6.1 E-9	7
IB-3	6.4 E-11	6.9 E-10	6.4 E-10	9
IC	2.6 E-13	6.5 E-10	8.0 E-10	10
ID	1.5 E-8	9.5 E-8	1.1 E-7	3
II	2.5 E-10	2.5 E-8	2.8 E-8	4
IIIA	4.4 E-9	4.4 E-9	5.3 E-7	8
IIID	1.3 E-8	1.3 E-8	1.3 E-8	6
IV	2.7 E-9	1.8 E-7	2.3 E-7	2
V	5/	5/	5/	
Total	8.1 E-8	5.9 E-7	7.5 E-7	

Notes:

1/ For a description of accident class definitions, refer to Table 19.3-2.

2/ These frequency estimates are the same as those documented in Table 19.3-9 of the ABWR SAR.

3/ These frequency estimates are those documented in Table 6.8 of Reference 19.20. These numbers are point estimates rather than means, to permit meaningful comparisons with GE's numbers.

4/ The staff's estimates are based on GE's reference design, including an onsite gas-turbine generator and an AC-independent firewater (low pressure coolant makeup to the reactor) system.

5/ With respect to frequency estimates, the ABWR PRA did not document a quantitative evaluation of these sequences. Although the staff has qualitatively evaluated these sequences, a quantification of them was not performed.

6/ These mean estimates are those documented in Table 7.2 of Reference 19.20

Table 19.3-5 A Summary of Relative Contributions of Various Initiating Events to the Overall Core Damage Frequency

Event designator	Description of Event	Relative Contribution (percent) based on Staff Point Estimate 1/ 2/	Mean 2/
T(M)	Normal Shutdown	1.9	2.0
T(T)	Turbine Trip	4.4	5.1
T(ISO)	Reactor Isolation	3.4	3.3
T(FW)	Loss of Feedwater	26.4	29.8
T(L/P)	Loss of Offsite Power (With Partial Onsite Power)	20.2	19.1
T(L/S)	Loss of Offsite Power (Without Onsite Power)	5.7	4.1
T(I)	Inadvertent Open Relief Valve	3.5	3.4
T(FS)	Failure-to-Scram Events	31.5	30.2
A	Large LOCA Events	0.5	0.4
S(1)	Medium LOCA Events	1.6	1.2
S(2)	Small LOCA Events	0.9	0.3

Notes:

1/ The point estimate for the total core damage frequency is about 6 E-7 per reactor-year. These estimates are the same as those provided in Table 6.10 of Reference 19.20.

2/ The mean estimate for the total core damage frequency is about 8 E-7 per reactor-year. These estimates are the same as those provided in Table 7.3 of Reference 19.20.

19.4 CALCULATION OF CORE DAMAGE FREQUENCY DUE TO EXTERNALLY INITIATED EVENTS

4.1 Introduction and Review of the Scope of External Event Analyses in the ABWR PRA

In the ABWR PRA, quantitative treatment of external events was performed only for tornado strikes and earthquakes. These two external accident initiators have been identified by EPRI in the PRA Key Assumptions and Ground rules Document (Reference 19.2) as events that may require quantitative assessment for each ALWR. Other external events are considered not to be important contributors to ALWR core damage based on improved design, proper siting, or low probability of occurrence. Tornado strike and seismic analyses are evaluated in Sections 19.4.2 and 19.4.3.

The staff is currently reviewing Reference 19.2, and may not necessarily conclude that only earthquakes and tornado strikes need quantitative evaluations. At the staff's request, GE has provided two documents (References 19.53 and 19.54) which were prepared to support positions recommended in Reference 19.2. These references, one of which is an interim report, attempt to identify external events that will be integrated into the external event PRAs, and those that can be excluded from the detailed analyses. This identification was carried out using a screening analysis which relied upon the review of existing PRAs for the current generation of plants, and then assessing whether the severe accident vulnerabilities found in the existing plants have been eliminated in the ALWR design. The other screening criterion used was whether the event had a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. The conclusions drawn in these references are that the seismic event needs to be analyzed in detail, and a limited evaluation of a tornado strike at the plant resulting in a prolonged loss of off-site power is also needed. GE concludes that all other external events (including fire and flooding events) can be excluded from detailed quantitative analysis in the PRAs based on improved plant design and siting criteria. However, these references further conclude that exclusion of many external events requires that a design and site verification be performed after a site and design are selected to ensure that the design and site do indeed meet the criteria used to exclude these events.

The staff, in developing the draft guidance for the Individual Plant Examination for External Events (IPEEE) (Reference 19.55), had also used a similar approach to identify the external events which needed some quantitative analyses for the operating plants. The following five external events were identified as requiring some examination at all plants (internal floods have been included with the internal events evaluation): (1) fire; (2) external floods; (3) seismic; (4) high winds; and (5) transportation and nearby facility hazards.

Given the above studies and their findings, the following observations are made in the context of the ABWR PRA review. The argument of the low frequency of occurrence or low frequency of consequences needs to be examined in conjunction with the overall results. For example, the mean core damage frequency (CDF) due to internal initiators is estimated to be 7.5 E-7/ry; based on the current external flood design criteria, the staff has estimated the frequency of the occurrence of the design basis flood to be 1.0 E-5/yr, with this initiator frequency it is not clear why the CDF from the external floods could not be in order of 1 E-7 or greater, comparable to the internal events. Thus, even though the estimated mean frequencies may be low, insights with respect to contributors to the overall results may be significantly affected by including these "low frequency" external events. Therefore, the staff, consistent with the recommendations of Reference 19.53, pending the staff review of the ALWR Requirements Document, recommends that a site and design verification be performed when a specific site is selected for the external events, such as external floods and transportation hazards, for which no analyses can be performed at this stage. This is an interface requirement.

The staff, based on its past experience, does not agree with GE that no severe accident examination is needed with respect to the fire hazard (response to the staff Q.725.74, Reference 19.56). This hazard is not truly "external," and can be evaluated to some extent at the design stage. This is an outstanding item.

From past PRA review experience (e.g., unique design features of the Shoreham facility), the staff also believes that various combinations of human failures and hardware failures could yield a relatively significant core damage frequency due to room-specific floods. The staff also believes that the large release goal may not be affected significantly due to the requirement of additional failures needed to fail the containment. Because the details of the plant and equipment layout are needed for such an evaluation, the probabilistic analysis for internal floods should be performed when details of the plant design are more complete. This is an interface requirement.

The staff's review of the ABWR tornado strike and seismic PRA was assisted by BNL (as a primary contractor to the staff) and its subcontractor, EQE Engineering, Inc. (EQE), with BNL mainly responsible for reviewing system analysis and developing alternative Boolean equations for seismic accident sequences. The staff has reviewed, accepted, and adopted the revie. s by its contractors as its own.

The remainder of the task, including a critical review of both the seismic hazard analysis and the equipment and structural fragility analysis presented in the ABWR seismic PRA, re-quantification of seismic core damage frequency using different, staff-provided seismic hazard curves, and an uncertainty analysis was performed by EQE. The detail of the results of BNL's and EQE's review and independent estimates of accident sequence frequencies as well as the conclusions drawn from the study are documented in Reference 19.20.

The following SER sections on tornado strike and seismic events are based on the technical evaluation conducted by BNL and EQE. Results of this evaluation, insights, and safety findings are highlighted here.

4.2 Tornado Strike Analysis

The tornado strike analysis performed in the ABWR PRA essentially follows the EPRI PRA Key Assumptions and Groundrules (KAG) position and approach (Reference 19.2). (As noted earlier, the staff is separately reviewing Reference 19.2). EPRI, with the technical assistance of the Advanced Reactor Severe Accident Program (ARSAP), has assessed the ALWR vulnerability to tornado-induced events and concluded that the dominant effect of a tornado strike is likely to be a prolonged loss of offsite power (Reference 19.54). Most of the vulnerabilities found in past PRAs are not expected to occur in the ALWR design. The EPRI position, therefore, is such that it deems a simplified model sufficient for the assessment of tornado strike impact, provided that it addresses random failures in combination with loss of offsite power.

As a support to EPRI's effort in developing the ALWR Requirements Document, ARSAP carried out the evaluation of ALWR designs to identify their tornado vulnerability and developed a method to quantitatively estimate ALWR tornado strike core damage frequency. Expected tornado strike frequencies were calculated based on regional historical data summarized in an EPRI report on tornado missile risk assessment. The frequencies of tornado strikes with intensities large enough to lead to core damage events were combined to generate total regional frequencies per square miles per year. The regional value is, then, multiplied by the plant area, assumed to be about 0.14 square miles, to yield the expected tornado strike frequency. The staff has not evaluated the adequacy of 0.14 square miles assigned for the plant area. The plant area consisting of critical safety systems, components, and structures (e.g., ultimate heat sink) may exceed this assigned estimate of 0.14 square miles. However, because of the low CDF estimated for the tornado initiating events, even increasing the plant area substantially is not likely to make the tornado events major contributors.

Since the resulting regional site strike frequencies were found to be relatively insensitive to the region specified, the maximum assessed regional value of 2.86 E-05 tornado strikes per year was conservatively chosen as the basis for the ABWR tornado strike analysis.

This value was used as the initiating event frequency in the loss of offsite power and station blackout event trees developed for the internal events in Section 19D.4 of the ABWR PRA for estimating the core damage frequency attributable to tornado strikes. In calculating the core damage frequency, the following assumptions, resulting from the ARSAP qualitative evaluation of the expected ALWR tornado strike vulnerabilities, were further imposed by properly modifying the event trees: (i) no credit is given to either the condensate storage tank or the main condenser due to their vulnerability to tornado effects; (ii) both the power conversion system and the feedwater system are unavailable due to loss of offsite power; and (iii) offsite power restoration is not expected within 24 hours following a tornado strike.

All other assumptions and conditions remain the same as those used in the internal events analysis. Quantification of these event trees on this basis yielded a total core damage frequency attributable to tornado-initiated events of 1.1 E-08 per year. Due to its relatively insignificant contribution to the overall core damage frequency, no further detailed analysis was carried out for tornado-induced events.

4.3 Seismic Events

4.3.1 Introduction and Overview

The ABWR seismic event analysis, described in Section 19.4.3 of ABWR SSAR, was performed in part to satisfy a requirement set forth in Appendix A to the Advanced Light Water Reactor Requirements Document (Reference 19.2). Such an analysis is called for to assure that the standardized plant at the certification stage has a balanced design from a seismic risk standpoint as well as to demonstrate that the Commission's safety goals (including quantitative health objectives) can be fulfilled.

The main objectives of the seismic event analysis are stated as follows in the ABWR PRA:

(1) To assure that the ABWR standard plant meets the intent of the NRC policy statement on severe accidents which includes consideration of seismic events as requirements for plant certification.

(2) To gain insights and understanding of the relative contribution to seismic risk of the individual components and structures of the plant.

(3) To understand, within the uncertainty limits, the relative degree of risk contribution from seismic events in comparison with other events.

(4) To identify the most probable sequences of events following a seismic event as well as any vulnerabilities (if any) to seismic events.

GE's approach to carry out seismic event analysis is described in the next section.

4.3.1.1 ABWR PRA Approach and Assumptions

The general approach and methods used in the ABWR PRA seismic event analysis are essentially identical to those used in the GESSAR II PRA (Reference 19.22). They are generally consistent with the guidelines provided in the PRA Procedures Guide (NUREG/CR-2300, Reference 19.57) and the PSA Procedure Guide (NUREG/CR-2815, Reference 19.58). GE has also assessed that the seismic PRA approach meets the requirements set forth in the EPRI ALWR Requirements Document (Reference 19.2). Note that, as was the case with internal event analysis, the ABWR seismic PRA only analyzes core damage sequences from power operation up to hot shutdown condition. Also, no explicit uncertainty analysis was performed by GE.

Four major tasks are involved in the assessment of seismic-initiated core damage frequency. They are: (i) the establishment of a seismic hazard curve; (ii) the determination of the seismic capability of critical components and structures; (iii) system modeling; and (iv) quantification of accident sequence frequencies. Each of these major tasks are discussed and evaluated in separate sections of this report.

The ABWR seismic event analysis has been conducted making use of the several ground rules and assumptions as follows:

(i) No credit is given to recovery of offsite power when lost due to seismic events.

(ii) No credit is given to repair or recovery of mechanical failure of components caused by seismic events.

(iii) Structural failure of a building containing important equipment results in functional failure of all contained equipment.

(iv) Seismic failure of identical redundant components at similar locations are treated as dependent failures, i.e.,, all components fail together.

The following lists some of the key assumptions described in the ALWR Requirements Document (Reference 19.2) which are also used in the ABWR PRA:

(v) It was assumed that the primary seismic hazard is due to ground shaking and that there is no soil failure potential in the range of ground motions considered.

(vi) The seismic hazard for potential ABWR sites in the future was characterized by a singl hazard curve.

(vii) Seismic fragilities for a few structures and components were estimated using specific design information. For the rest of the structures and components, fragilities were assigned on a generic basis with the assumption that they are achievable in light of the ABWR evolutionary seismic design criteria.

GE's general approach and assumptions (i) through (iv) are consistent with the-state-of-the art approaches and the past PRA practices. Assumptions (v) through (vii) necessitate from the fact that the ABWR seismic PRA is being conducted for a standard design which has not been built or located at a specific site. These assumptions have several implications, mainly in the area of interface requirements, and are discussed later.

4.3.1.2 Comparison with the ALWR Requirements Document

Based on a preliminary review and a conference call with GE on January 31, 1991, the staff has noted the following three differences among the requirements outlined in Reference 19.2 and those used in the ABWR PRA.

(i) The seismic hazard curve used in the ABWR PRA is different than that recommended in the ALWR Requirements Document.

(ii) Seismic induced fires have not been addressed in the ABWR PRA.

(iii) Seismic induced floods have not been addressed in the ABWR PRA.

As will be discussed in Section 19.4.3.2, the staff has used alternate hazard curves in its quantification. The impact of using different curves on results is also discussed.

GE is in a process of performing a screening analyses to address items (ii) and (iii) above. The staff will review this information when it becomes available.

4.3.1.3 Overview of ABWR PRA Results

The seismic core damage frequencies calculated for various accident classes in the ABWR PRA are summarized in Table 19.4-1. Roughly speaking, Class I events are transients with loss of core cooling, Class II events are events with successful core cooling, but with loss of containment cooling, and Class IV events are anticipated transients without scram (ATWS) without boron injection, but with core cooling available. The total seismic core damage frequency was calculated to be 2.5 E-7 per year in the ABWR PRA. The largest contribution (about 95 percent) to seismic core damage frequency comes from Class I sequences, which have a total seismic CDF of 2.4 E-7 per year.

The staff does not agree with the treatment of Class II sequences in the ABWR PRA and this issue is discussed further in Section 19.4.3.5.

4.3.1.4 Review Approach

The seismic hazard curve used in the ABWR PRA was reviewed as to its applicability to potential ABWR sites in the central and eastern United States in light of recent seismic hazard study results. Three representative sites with high seismic hazard were selected to estimate the changes in seismic accident frequencies from the ABWR PRA values.

The methodology used in estimating the seismic fragilities of structures and components was reviewed. The calculations of fragilities of specific structures and components performed by GE were reviewed. The reasonableness of the assignment of generic fragilities for structures and components was assessed in light of the ABWR seismic design criteria.

In the system modeling area, accident sequences, random failure rates and human actions reported in the ABWR PRA were reviewed, and modified appropriately to be consistent with the findings from the internal events evaluation and other seismic related findings discussed later in this chapter. Boolean equations for different accident sequences were developed and used in requantification. In the review, it was assumed that the system cutsets developed in the ABWR PRA accurately represent the systems. The staff did not develop the fault trees and cutsets independently.

The results of this review and requantification include seismic accident class frequencies estimated using different seismic hazard curves, seismic margins for different accident classes and identification of dominant contributors in terms of component failures and accident sequences. Uncertainty analysis using families of hazard and fragility curves were also carried out. Further sensitivity studies were conducted to assess the impact of alternative seismic fragilities for some selected components.

The staff review of the ABWR PRA has focused on both the numerical results and other insights such as the plant capacity to withstand a large seismic event. Consistent with the staff guidance (Reference 19.55) to conduct individual plant examinations for external events (IPEEE), the staff has developed margin (High Confidence of Low Probability of Failure - HCLPF) information for the various accident classes.

During the review process, the staff and its contractors have also focused on identifying the interface requirements which will have to be addressed on a specific application when a plant is built.

4.3.2 Hazard Analysis

4.3.2.1 ABWR PRA Hazard

The seismic hazard curve used in the ABWR seismic risk analysis is taken from the GESSAR II seismic event analysis, except that the effective peak ground acceleration used in the GESSAR hazard curve was converted to peak ground acceleration (PGA). The PGA was used in the ABWR analysis in order to be consistent with the ground motion definition for seismic fragility.

This hazard curve (Fig. 19.4-1) was shown in the GESSAR II seismic event analysis to be a bounding curve of the best-estimate hazard curves for the Limerick, Indian Point, Zion, and Oyster Creek sites based on the information available at that time. For the ABWR application, this curve was further compared to the median hazard curve of the Oconee site and found to be bounding (see GE's response to staff question 725.68, Reference 19.56). The soil-structure interaction effect on seismic risk is not included in the hazard curve but is treated in the seismic fragility estimate.

No uncertainty estimates were made for the use of a single best-estimate seismic hazard curve for the ABWR. Since the ALWR top-tier requirement states that the mean annual CDF should be compared with the goal, this single best estimate curve is inferred to represent the mean value of seismic hazard. The hazard curve does not reflect the more recent studies, results and scientific opinion concerning seismicity and seismic hazard for the Eastern United States (EUS).

4.3.2.2 Hazard Review Approach

To understand how representative the ABWR seismic hazard curve is of EUS sites, and to study the effect of site variations on the calculated mean CDF, three different sites, Pilgrim, Seabrook and Watts Bar, were selected and the hazard curves developed by both the Lawrence Livermore National Laboratory (LLNL) (Reference 19.59) and the Electric Power Research Institute (EPRI) (Reference 19.60) for these sites were compared with the ABWR hazard. These three locations in the EUS were selected because of their relatively high seismic hazard. A comparison among these various hazard estimates, including that of GE's, is shown in Fig. 19.4-2.

4.3.2.3 Evaluation of ABWR PRA Hazard

In Fig. 19.4-2, all LLNL curves indicate a much larger seismic hazard than the values used in the ABWR analysis. Generally, the EPRI seismic hazard for sites in the EUS is an order of magnitude below the LLNL hazard. However, even the EPRI mean hazard for Pilgrim and Seabrook was found to be larger than the ABWR best estimate hazard. Therefore, the ABWR seismic hazard cannot be considered to be a conservative estimate of seismic hazard for the eastern United States, and does not appear to account for the large uncertainties which exist in the hazard estimation. The ABWR seismic hazard curve also indicates a different slope characteristic at accelerations greater than 1g. The impact of different hazard curves on CDF and identification of dominant components/sequences is discussed in Section 19.4.3.7.

4.3.3 Fragility Analysis

4.3.3.1 ABWR Approach

The seismic fragility of components in the ABWR is modeled using a lognormal distribution with the parameters as median peak ground acceleration capacity (A_m) and logarithmic standard deviation (β_c) representing randomness in capacity and uncertainty in the median capacity. Note that this representation is equivalent to using a mean fragility curve. These parameters are generally estimated using the design information, qualification analyses and test results. Seismic fragilities of structures in the reactor building complex were evaluated following the methods employed in previous seismic PRAs (References 19.57 and 19.58) for:

- Reactor building shear walls
- Containment

- Reactor pressure vessel pedestal

Detailed fragility calculations for other structures such as the control building and turbine building could not be made at this time. These fragilities were assigned by comparison with similar structures in past seismic PRAs.

Seismic fragilities of safety-related components were assessed for the following two categories of components:

- ABWR specific components whose fragility evaluation was made according to existing design information.
- Generic components whose fragilities are based on data compiled in the "Compilation of Fragility Information from Available Probabilistic Risk Assessments," dated September 1988 (Reference 19.61).

ABWR Specific Components

Detailed seismic fragility evaluations were performed for the following ABWR specific components:

- Reactor pressure vessel (RPV)
- Shroud support
- Control rod drive (CRD) guide tubes
- CRD housings
- Fuel assemblies

Generic Components

Detailed fragility evaluations for safety-related components other than those specific components presented above could not be made by GE at this stage. The fragilities for generic components recommended in the Advanced Light Water Reactor (ALWR) Requirements Document (Reference 19.2) were adopted for the ABWR standard plant. These generic fragilities were chosen based on a review of prior PRAs and fragility data. These are considered by GE to be achievable for the ABWR with an evolutionary improvement in the seismic capacities of the components designed to a 0.3g Safe Shutdown Earthquake (SSE). Both design specific and generic fragilities used in the ABWR PRA are summarized in Table 19.4-2. (Table 19.4-2 of the ABWR PRA). Demonstration that the plant equipment/structures have the assumed capacities is an interface requirement.

4.3.3.2 Review and Evaluation

Specific Structural Fragilities

In the fragility evaluation, structures are considered to fail functionally when inelastic deformations of the structure under seismic load increase to the extent that the operability of the safety-related components attached to the structure cannot be assured. The ductility limits chosen for the structures were estimated as corresponding to the onset of significant structural damage. These definitions of failures modes are consistent with the seismic PRA practice.

The potential of seismic-induced soil failures such as liquefaction, differential settlement, or slope instability is not evaluated at this time since these are highly site dependent. These modes should, however, be considered when an ABWR location is fixed. This is an interface requirement.

The calculations for the following structures were reviewed:

	Median Capacity (g)	β_c	HCLPF (g)
Reactor Building Shear Walls	2.8	0.45	0.98
Containment	4.3	0.44	1.54
RPV Pedestal	7.9	0.44	2.83

The HCLPF capacity stated above is calculated by the staff using an approximate relationship for HCLPF as equal to A_m times exp $[-2.33\ \beta_c]$. This approximation is consistent with that recommended by the staff in the draft IPEEE guidance document (Reference 19.55).

These structural fragility parameters appear to be reasonable; especially, the median and HCLPF capacities of the reactor building shear walls and containment are judged to be achievable. The high capacity of the RPV pedestal would make its failure contribute negligibly to seismic the CDF.

Specific Component Fragilities

Detailed seismic fragility evaluations were performed by GE for the following ABWR specific components:

- Reactor pressure vessel (RPV)

- Shroud support
- Control rod drive (CRD) guide tubes
- CRD housings
- Fuel assemblies

Reactor Pressure Vessel:

The median capacity and HCLPF capacity of the RPV are 5.3g and 2.4g, respectively. The value of β_c = 0.33 used in the ABWR PRA is judged to be low. A later sensitivity study by the staff assigned a lower median capacity and a higher value of β_c.

RPV Internal Components:

The internal components examined for seismic fragilities include the shroud support, CRD guide tubes, CRD housings, and fuel assemblies. Failure of these components could potentially result in inability to insert the control rods to shut down the reactor.

The critical failure modes and seismic capacities of these components are:

Component	Failure Mode	Median Capacity, g	HCLPF Capacity, g
Shroud Support	Buckling	1.90	0.82
CRD Guide Tubes	Buckling	1.70	0.88
CRD Housing	Plastic Yielding	3.90	1.34
Fuel Assemblies	Channel Buckling	1.30	0.58

Although these capacities are generally higher than those reported for these components in past Boiling Water Reactor (BWR) seismic PRAs, review of GE's calculations did not reveal any reasons for revising the above capacities except for fuel assemblies.

The seismic capacity of the fuel assemblies was calculated by GE as corresponding to a center deflection of 55 mm, at which scram can be achieved. However, the moment corresponding to this deflection is not the collapse moment as used in the calculations. It is some value between the yield moment and the collapse moment. Therefore, the median ultimate capacity of the fuel assemblies is less than the median value of 1.3g. In a sensitivity study, the staff has used a value of 0.92g median capacity to estimate the accident sequence frequencies.

Generic Components

Detailed fragility evaluations for many safety-related components could not be made; therefore, GE has assigned generic seismic fragilities for these components consistent with design requirements of the ALWR Requirements Document (Reference 19.2). Table 19.4-2 shows these generic assignments. These capacities are considered by GE to be achievable for the ABWR with evolutionary improvements in the seismic capacities of components designed to SSE of 0.3g.

The review of these fragilities focused on the reasonableness of these estimates in light of the ABWR seismic design criteria and based on actual performance of similar equipment in tests and real earthquakes.

The seismic capacities assigned to the following five components differed from the ALWR Requirements Document: cable trays, large flat-bottom storage tanks, accumulators, air-operated valves, and heat exchangers. These capacities are generally much higher than those reported in past seismic PRAs. If credit is taken for these higher capacities in the PRA, their values should be proved later when the design and installation are completed. When considering the impact on CDF and risk, it is notable that only large tanks, discussed further below, are considered to be important. This is due primarily to the relatively low seismic capacity of large tanks and their past contribution to risk in PRA accident sequences. Details regarding evaluation of other components can be found in Reference 19.20.

As discussed in Section 19.4.3.3, the staff has judged that the generic fragilities for the following components and structures used in the ABWR PRA are optimistic. Although, the staff sensitivity analysis did not indicate a major impact on the CDF or HCLPF estimates, demonstrating that these components do have assumed capacities may be a difficult task on a specific application. This is an interface requirement. To assist GE in its response, the staff has included a list of components for which specific information should be developed now.

Component/Structure Fragility Estimate

	Review	ABWR
Diesel Generators	1.5	2.5
480V MCC (also SW MCC)	1.5	2.5
Batteries and Racks	1.5	3.0
ABWR Fuel Assemblies	0.9	1.3
RHR Heat Exchangers	1.4	2.0
Fire Water Tank	1.4	2.8

The issue of "achievability" should also take into account the fact that, for some systems, the fragilities of components at various locations is represented by a single value.

The standardization of all Category I structures needs to be confirmed so that the applicability of structural fragilities calculated in this PRA to all future ABWR plants can be assessed.

Large Flat-Bottom Storage Tanks:

Although the ABWR PRA report (Table 19H.4-6) states that the median capacity of these tanks is 2.1g with β_c of 0.45, the only tank used in the seismic system analysis (see GE's response to Question 725.72, Reference 19.56) is the fire water tank, with a generic assigned median capacity of 2.8g and a β_c of 0.45. This makes the HCLPF capacity equal to 0.98g. Experience with design and actual performance (Reference 19.61) of these large yard tanks is that this high capacity is not generally achieved. Therefore, use of a median value of 1.43g with a HCLPF capacity of 0.50g is made in a sensitivity analysis.

Some other specific components of interest are discussed below.

Diesel Generators:

GE has assigned a median capacity of 2.5g with a β_c of 0.45 to the diesel generators. This means that the HCLPF capacity is about 0.88g. Although diesel generators by themselves have high seismic capacities, the peripheral equipment required for the diesel generators to operate can have low capacities. These include diesel oil day tanks, control cabinets, air receiver tanks, accumulators, compressors and motors, lube-oil coolers, fuel oil transfer pumps, heat exchangers, heating and venting equipment, etc. Some of these components have been modeled in system "PW" (defined as loss of offsite power, loss of emergency power and loss of service water) in the systems analysis; however, other components such as the diesel day tank, the air receiver tank are not modeled. These components could have lower capacities. Therefore, the staff concludes that the diesel generator fragility is rather optimistic. In a later sensitivity study, the staff has assigned a much lower capacity (median of 1.5g and HCLPF of 0.47g) to diesel generators in acknowledgement of lower capacity components in the system.

Electrical Equipment:

Electrical equipment includes switchgear, motor control centers, inverters, battery chargers, instrumentation racks, load sequencers, control system cabinets, and other cabinets containing electrical sensors, switches, or control instruments. Potential failure modes include relay chatter, breaker trip, and structural failure. Relay chatter is addressed below. The seismic capacities assigned to the electrical equipment in the structural failure mode are generally higher than the specific capacities calculated in previous seismic PRAs. For example, the HCLPF capacities of motor control centers (0.88g), relay switches (0.62g) and battery and battery racks (1.05g) appear to be too high. In the sensitivity analysis the staff has assigned different fragility values to these components. Demonstration of these capacities is an interface requirement.

Relay Chatter

GE has stated that the potential for relay chatter was treated in the following manner: Only the scram system function is required during a seismic event. This function is fail-safe, so relay chatter would cause a safe-state failure (scram) even if relays were employed. For the ABWR, the scram actuating devices are solid-state power switches with no failure mode similar to relay chatter. The scram function is supplemented by an alternate scram method (energizing the air header dump valve) to provide diversity. This method uses relay actuation, but no credit was taken for this capability in the seismic analysis.

Switchgear and motor control centers do include relays whose failure could prevent safety actions after the seismic event. It was assumed that the indicated capacity for this equipment (2.5g) was more representative than the specific relay chatter value (2g). Also, the type of auxiliary relays used tend to be the most rugged of relay types and would have a capacity above 2g. The multiplexor output devices for ECCS and RHR operation have been assumed to be solid-state devices (rather than relays), so that the relay chatter failure mode does not apply. Demonstration of these capacities is an interface requirement.

Heat Exchangers:

Because the seismically-induced failure of the RHR heat exchanger could result in a suppression pool drainage scenario and loss of melt-release scrubbing capability, the seismic capacity of this exchanger is very important to risk estimates. It is the staff's understanding, based on discussions in a conference call, that GE may increase the median capacity of this component to 2.8g from the current median capacity of 2.0g.

Other Fragility Related Issues

The potential of seismic-induced soil failures such as liquefaction, differential settlement, or slope stability is not evaluated at this time since these are highly site dependent. However, these modes should be considered when the ABWR is sited. This is an interface requirement.

No analyses were conducted in the ABWR PRA for the potential failure of non-safety related structures and equipment which could affect safety-related functions. GE stated in response to Question 725.70 (Reference 19.56) that a walkdown of the final constructed plant as well as a review of construction drawings and documents will be performed to verify that the assumed seismic capacities are met or exceeded. The staff concluded that the plant specific walkdown is one of the most important interface requirements and should address the potential for failure of non-safety as well as safety related components. The walkdown should focus on potential seismic vulnerabilities such as system interactions, marginal anchorage of equipment and gross deviations from the design documents, and identification of failure modes not analyzed at this time. This is an interface requirement.

GE indicated that ABWR fragilities are achievable and designed to withstand a Regulatory Guide 1.60 design response spectrum with a zero period acceleration of 0.3g SSE. However, the staff finds that this design ground motion may not envelope the site-specific spectra for sites near some areas in the eastern and central United States (such as sites near the New Madrid Seismic Zone and Charleston, South Carolina) and sites in the western U.S., in addition to sites along the California coast. Included in the staff finding is the significant role that soil amplification plays at sites where the bedrock is overlain by a layer of soil.

It will be necessary for applicants to submit a plant-specific probabilistic seismic hazard analysis in conformance with Section 2.5.2 of the Standard Review Plan. At sites where the design spectra or the probabilistic hazard curve in the ABWR PRA are exceeded by the corresponding site-specific parameters, the adequacy of the structural design must be demonstrated by the applicant and submitted for review and approval by the NRC staff. This is an interface requirement.

The operator action probabilities (discussed in 19.4.3.4) assume that the operators are not injured in a seismic event to prevent them from performing these actions. The one failure mode that may disable the operators is the failure of the suspended ceiling in the control room. In response to Question 723.69, (Reference EE.6), GE has stated that the design of the ceilings will be made to ensure that the ceilings are properly braced and equipment above the ceilings is adequately anchored against the design SSE. The plant walkdown should focus on this aspect of the spatial systems interaction. This is an interface requirement.

4.3.3.3 Summary Evaluation of Fragilities

GE's general approach used in developing design specific fragilities is consistent with past PRA practices. The staff has only identified one component (fuel assemblies) where alternate values are suggested for the sensitivity study.

The staff review finds that for several components in the generic component category, the ABWR fragility estimates are optimistic, and special attention will be required on a site specific application to assure that these fragilities are achieved. As a result, the staff has identified several interface requirements discussed in Section 19.4.3.8, and also conducted a sensitivity analysis using alternate fragility values. Table 19.4-3 compares the alternate fragility parameters for some components used to study the effect of fragility assignment on the seismic-induced CDF. Results of this sensitivity study are discussed in Section 19.4.3.6.

4.3.4 System Modeling

4.3.4.1 Seismic Fault Trees

Seismic fault trees were developed in the ABWR PRA for the following seven important frontline and support systems that contain components having relatively low fragilities: HPCF, RCIC, LPFL, RHR (suppression pool cooling mode), fire water, service water and electric power. Additionally, a fault tree is presented to show structural failures that could contribute to seismic core damage frequency. No fault tree was constructed for the automatic depressurization system (ADS), the standby liquid control system (SLCS) or the reactor scram system. The ADS is needed to depressurize the reactor so that the AC-independent fire water could be used to provide coolant makeup to the reactor. The staff notes that the structural failures of the ADS components could also play a role in addition to the human failure of the ADS system, which only has been considered in the PRA at this time. Because the PRA has assigned a higher probability for the human failure of the ADS system, incorporation of the structural failures via a system fault tree will not significantly increase the overall seismic CDF. Feedwater and condensate systems, which require offsite power, were also not modeled because offsite power is conservatively assumed to be lost to these systems as a result of an earthquake.

Since these fault trees were specifically developed for evaluation of seismically-induced failures, only those components potentially vulnerable to seismic failure are included. Random failures were not explicitly shown as basic events in the fault trees, although they were included in the quantification of seismic core damage frequency. Also, the assumption of complete dependence, i.e., when one component fails, all like components fail, made it unnecessary to carry out multi-divisional analysis or common-cause failure analysis.

Critical human errors are identified and included in the analysis; however, they appear mainly in the event trees. The important operator actions modeled in the front-end seismic analysis include the following:

(i) Operator fails to inhibit ADS during an ATWS

(ii) Operator fails to initiate SLCS during an ATWS

(iii) Operator fails to control flow during an ATWS

(iv) Operator fails to depressurize the reactor in a controlled manner, to permit use of low-pressure injection

(v) Operator fails to inject firewater into the RPV

(vi) Operator fails to isolate failed RHR heat exchangers

The staff's detailed evaluation of fault trees, in general, is described in Section 19.3.5, Calculation of Core Damage Frequency Due to Internally Initiated Events. Several seismic specific observations with regards to fault trees are described below.

At this time it is not possible to include any spatial-interaction type of failure modes (e.g., valve stem impacting a near by piping or wall) or flow diversion failure modes which may have an impact on the total system availability. The use of a single fragility value to represent, say all valves in the system, does not take into account location effects. In other words, when a fragility is characterized by a failure probability conditioned on the occurrence of the peak ground acceleration; theoretically, the same component on different locations should have different fragility curves as different locations will experience different responses to the same ground motion. The staff assumes that the generic fragility values used in the ABWR PRA represent the component location which will experience the most adverse response; fragilities for other locations should be lower. This observation is important in light of the concept of "achievable" fragilities discussed in Section 19.4.3.3.

The above two observations highlight the need for interface requirements and development of guidance to implement these requirements. Some interface requirements are discussed in Section 19.4.3.8. The staff has requested GE to provide a discussion on the "design come true (DCT)" principle to assure the achievability of the estimated fragilities based on the current design practices.

4.3.4.2 Seismic Event Trees

The seismic event trees consist of a seismic support state event tree and three seismic front line trees. These trees are described in Appendix 19I of the ABWR PRA. The seismic support state event tree starts with seismic events (low, moderate to high intensity). The first event tree top event inquires about whether structural failure has occurred, the second top event considers whether or not offsite power is lost and so on. Loss of structural integrity is assumed to lead to core damage based on the groundrules of the analysis and the relative values of seismic fragilities, and survival of offsite power results in a successful event termination. The three front line trees represent the diverse entry conditions from the support state tree and various operator actions as follows: (1) loss of offsite power (LOOP) with scram; (2) LOOP without scram with several operator actions to initiate the standby liquid control (SLC), and initiate low pressure injection at a later time in some sequences when the high pressure injection fails; and (3) LOOP without scram with no initiation of SLC, but with operator action to achieve shutdown by controlling flow to prevent reactivity increase by boron dilution.

The seismic containment event trees, per se, are not evaluated in this section. However, in certain sequences in which there is initial successful core cooling, the likelihood of core damage depends on whether the containment heat removal functions are available or not. To this extent, the evaluation of the containment event trees is described in the next section, Accident Sequence Definition.

4.3.4.3 Summary Evaluation of System Modeling

As discussed earlier, the fault tree modeling essentially includes only the seismic-induced hardware related failures (both structural and functional); no spatial interaction failures are incorporated at this time.

The non-seismic failures are integrated in the quantification at the system level. The human actions are incorporated as top events into the event trees.

4.3.5 Accident Sequence Definition

Accident sequences are developed using the event trees described above. The accident sequences are classified into three basic classes and several subclasses as shown in Table 19.4-1.

Roughly speaking, Class I events are transients (or ATWS) with loss of core cooling, Class II events are events with successful core cooling, but with loss of containment cooling, and Class IV events are anticipated transients without scram (ATWS) without boron injection, but with core cooling available.

The evaluation of Class II sequences is discussed below. As was the case with internal events, the Class II events frequency obtained from the event tree analysis was further processed through a seismic containment event tree (see Fig. 19J.5-6 of the ABWR PRA) to give credit to RHR recovery for containment heat removal (failure probability = 0.66), continued core cooling (failure probability = 0.01) and fire water (failure probability = 0.1).

In the GE seismic analysis, the frequency of Class II events input to the CET is 4.8 E-06. Through consideration of the potential for recovery of containment heat removal, continued core cooling, containment venting, and firewater system operation, the core damage frequency for this accident class is estimated by GE to be 3.2 E-09. (Note: On page 19.4-11 of the ABWR PRA, the Class II total seismic CDF is shown to be 4.8 E-9 per year, which appears to be somewhat inconsistent with that shown in Fig. 19J.5-6 of the ABWR PRA). The staff believes that the benefits of firewater addition are overstated in the GE Class II CET, and that a release frequency of E-07 is more appropriate. Specifically, the Class II events entering this seismic containment event tree consist of three types of events, i.e., (i) station blackout sequences with RCIC failure at eight hours and low pressure makeup with the fire water system; (ii) station blackout sequences with initial failure of RCIC but with successful operation of firewater; and (iii) loss of offsite power (D.G. available) ATWS sequences with successful high and low pressure cooling but with failure of suppression pool cooling. The bulk of the Class II sequences are type (i) events, which have already employed fire water as the only available means of core cooling. Accordingly, for these Class II events, it is inconsistent to give credit to firewater as a means of containment cooling given the failure of continued core cooling.

An essentially similar seismic containment event tree (see Fig. 19J.5-7 of the ABWR PRA) was also constructed for all the Class IV events, including Classes IV, IV-1 and IV-2,3,5. Credit was given to RHR recovery for containment heat removal (failure probability = 0.93), continued core cooling (failure probability = 0.17) and fire water (failure probability = 0.1). If continued core cooling fails, however, the sequence is considered to lead to core damage regardless of whether fire water is successful. By processing through this seismic containment event tree, the total Class IV events frequency (including Classes IV, IV-1 and IV-2,3,5) was reduced from 7.33 E-8 to 1.16 E-8. (Note: On page 19.4-11 of the ABWR PRA, the total Class IV seismic CDF is shown to be 7.34 E-10 per year, which appears to be inconsistent with that shown on Fig. 19J.5-7 of the ABWR PRA). These changes are reflected in the seismic core damage frequency shown in Table 19.4-1. It should be remarked that more than 90 percent of the Class IV sequences entering the Class IV seismic containment event tree involve failure of injecting boron into the reactor due to either failure to initiate SLCS or failure of flow control/alternate boron. Reactor power will, therefore, remain high and, regardless of whether continued core cooling is available, core damage may ensue.

In the ABWR PRA, the largest contribution (about 95 percent) to the seismic core damage frequency comes from Class I sequences, which have a total seismic CDF of 2.37 E-7 per year. However, because of the different treatment of Classes II and IV sequences, the staff requantification discussed in the next section has resulted in two different contributions.

4.3.6 Quantification of Accident Class Frequencies and Margin Values

In this section, the results of the staff requantification of accident class frequencies are described from two perspectives: (1) The first requantification is based on the "system modeling" issues identified by the staff's review. The treatment of Class II and Class IV sequences is modified as discussed in Section 19.4.3.5 above, and some of the non-seismic failure probabilities ("random" failures) are revised to be consistent with the staff evaluation of the internal event analysis; (2) The second requantification discussed here is with respect to several alternate hazard curves based on the LLNL and EPRI Eastern Seismicity studies (References 19.59 and 19.60).

The staff approach to requantification is somewhat different than the ABWR PRA approach. Details of and differences between the two approaches are discussed in Reference 19.20. Both approaches give essentially the same results using the same data verifying the adequacy of the quantification approach. The staff has performed the requantification using the Boolean equations developed for various accident classes. The use of the Boolean equations has allowed the staff to develop accident class level fragilities to develop margin information.

4.3.6.1 Reassessment of the ABWR Seismic Core Damage Frequency - System Modeling Issues

The mean annual frequencies for the nine accident classes calculated by using GE's hazard curve (extended up to 2g) and fragility data are shown in Table 19.4-1, where GE's best-estimate values are also shown for comparison. The total seismic CDF obtained by the staff is 1.3 E-06/ry, about a factor of five larger than that obtained by GE (2.5 E-07/ry). The combined mean annual frequency of all Class I sequences is 6.2 E-07 compared to GE's value of 2.4 E-07. The larger values obtained in the staff study can be attributed, in most cases, to modifications of sequences and changes in random failure probabilities (Table 19.4-4) and contributions from earthquakes beyond 1.25g. The mean annual frequency calculated for Class II events is 5.7 E-06 compared to the ABWR PRA value of 4.8 E-06. The staff value was reduced by an order of magnitude to 5.7 E-07 by giving credit to containment venting. (The failure probability of

0.1 mainly includes functional failures of the venting system, such as the failure of the rupture disc, for which data is scarce. Note that the ABWR seismic PRA does not explicitly analyze containment venting). The ABWR PRA value was reduced to 3.2 E-09 by processing through a seismic containment event tree, giving credits to RHR recovery, continued core cooling and fire water. As explained in detail in Section 19.4.3.5, the staff believes that GE has overestimated the credit for firewater addition for Class II sequences and that a value of 5.7 E-07 is more appropriate.

A representative mean fragility curve for one accident class is shown in Fig. 19.4-3. A more detailed discussion of accident class fragilities is in the next subsection.

4.3.6.2 Mean CDF for Three Selected Sites Using LLNL or EPRI Hazard Curves

To study the effect of site variations on the calculated mean CDF, three different sites, Pilgrim, Seabrook and Watts Bar, were selected and the hazard curves developed by both LLNL and EPRI for these sites were applied. These three locations in the EUS were selected because of their relatively high seismic hazard. By convolving LLNL mean hazard curves with mean fragility curves for different sequences, the mean annual frequencies of the nine accident classes were calculated for the three chosen sites.

Table 19.4-5 shows the comparison of total and accident class core damage frequencies obtained from the use of various hazard curves (with the sequences modified as discussed above).

From the observations of results in Table 19.4-5, it is quite clear that the CDFs are greatly impacted by the choice of a hazard curve. The use of the LLNL hazard curves predict much higher CDFs than the EPRI hazard curves. Implications of these estimates on comparison with the ALWR Requirements Document CDF and the Commission's subsidiary CDF goal are discussed in the conclusion section. More important to point out here is the fact that the ranking of the sequences is robust for different hazard curves as highlighted in Table 19.4-6. (This may not be apparent at the first glance until it is realized that the mean CDFs for the first two classes differ by a factor of less than two. Given the range of uncertainty in the CDF estimates, this difference is insignificant). The dominant contributors to various sequences are listed in Table 19.4-7.

The staff further investigated the accident class frequencies to identify the ranges of acceleration that contribute most significantly to the overall frequency of occurrence of the accident class sequence. A representative example for one hazard curve and one accident class is shown in Fig. 19.4-4. Observations from this figure characterize the general trend; as seen the contribution from the acceleration ranges below 0.5g is very small. This indicates that very large earthquakes must occur in order for any significant damage to be done to the ABWR and that the design is capable of resisting earthquakes significantly larger than an SSE of 0.3g.

The above observations are not surprising in the light of HCLPF values for the accident classes. Table 19.4-8 lists the HCLPF values for various accident classes. The lowest value of the accident class HCLPF is 0.64g. In the margin sense, this can be interpreted to indicate that there is a very high confidence that core damage would not occur for acceleration levels as great as 0.64g. Further, the HCLPF values do not represent a cliff beyond which the capacity decreases sharply. In fact, for the fragility of the IB-2 sequence shown in Fig. 19.4-3, the median value is approximately 1.8g. It is also important to note that the proper accounting of the random failures in the combinations where both seismic and random failures are involved to cause an accident sequence is essential. For example, for Class II sequences, if containment venting is not combined with the fragility of seismic-induced failures, the inferred HCLPF value for the Class II sequences will be 0.43g rather than 0.73g. This point is also highlighted in the staff's draft IPEEE guidance document (Reference 19.55).

The above discussion should highlight the fact that the numerical CDF results for the ABWR design are controlled by much larger earthquakes which are most open to speculations because of the lack of recorded data. The mean CDF frequencies are dominated by uncertainties in the high hazard estimates. At the same time, the ABWR plant, with the assigned fragilities, is shown to be a rugged plant with respect to a 0.3g SSE. Therefore, the staff, consistent with the recommendations in the draft IPEEE guidance document, believes that the use of bottom line numbers should not be a sole governing criterion to determine the adequacy of a design.

4.3.7 Uncertainty and Sensitivity Analyses

4.3.7.1 Uncertainty Analysis

The ABWR seismic risk analysis used a single seismic hazard curve with no explicit consideration of uncertainty. Similarly, variability in the median seismic capacities of the components and structures was also not explicitly accounted for. The results from previous PRAs indicate that there is large uncertainty in seismic hazard and in some of the component median fragilities, often resulting in orders of magnitude variability in core damage frequency. The uncertainty in seismic core damage frequency was estimated in the following by explicitly treating the uncertainties in seismic hazard curves for the three sites and seismic fragilities of components. Variability in the capacity, β_c, for different components was split into randomness β_R and uncertainty β_U parts, assuming equal contribution from each. The representative results of seismic risk quantification for the Pilgrim site seismic hazard curves are given in Table 19.4-9. Note that the mean values in this table agree closely with the results obtained by a convolution of mean hazard and mean fragility curves, given in Section 19.4.3.6. This, in part, provides a confirmation for the staff recommendation made in Reference 19.55 regarding the use of mean hazard and mean fragility curves to approximately obtain mean CDF values.

In order to differentiate between the contributions of uncertainty in fragility from the uncertainty in hazard, seismic risk quantification was repeated using median (A_m) and β_c values for fragility with a full set of hazard curves. Comparison (Table 19.4-10) of the annual frequency values with the original results indicate that the contribution of uncertainty in fragility is negligible and most of the uncertainty in core damage sequence frequencies is due to uncertainty in the seismic hazard.

4.3.7.2 Sensitivity Studies

4.3.7.2.1 Specific and Generic Fragilities

In the ABWR standard plant seismic PRA, a limited number of structures and components were analyzed for specific fragilities; the rest of the components were assigned fragilities generically.

The structural fragilities specifically evaluated in the ABWR seismic PRA are for the reactor building shear walls, containment, reactor pressure vessel and pedestal. These appear in the system called SI, i.e., seismically induced structural failure. Only the Class IE frequency is affected by structural failures. The mean annual frequency of this sequence is calculated using the ABWR seismic hazard curve as 5.3 E-08; if only the SI system is retained in this sequence, this frequency is decreased slightly to 4.9 E-08. About 50 percent of this frequency comes from the control building and the rest comes from the reactor building.

Only Class IB-2 is directly affected by the RPV related failures, i.e., ABWR specific components such as the RPV, shroud support, CRD guide tubes, CRD housings, and fuel assemblies. When these components were removed from the Boolean equation of this Class, the mean annual frequency dropped from 4.1 E-07 to 3.9 E-07, demonstrating that the generic components provide the dominant contribution to the class frequency.

4.3.7.2.2 Alternative Fragilities

The accident class mean frequencies were calculated using different seismic hazard curves and alternative seismic fragilities from Table 19.4-3. Table 19.4-11 shows the results, including HCLPF values resulting from the use of different fragilities. It is seen that the mean annual frequency of Class IB-2 increased by about a factor of two; this is because of changes in the fragilities of the reactor internals and the fire water tank. The mean annual frequency of Class IE increased by a factor of 6 to 8. This increase is mainly due to the revised value of the control building fragility. The Class II frequency increased by about a factor of two. The Class IV frequency also increased by a factor of two.

4.3.8 Summary of Results, Interface Requirements, and Conclusions

4.3.8.1 Summary of Results

Table 19.4-12 summarizes the mean CDF of various accident classes obtained in the seismic requantifications using LLNL seismic hazard curves, along with the mean CDF estimated by the staff for internal events (with the original fragility values). Note that all of these results are obtained from uncertainty analysis. The total CDF of all accident classes due to seismic events ranges from 4.6 E-05 to 8.5 E-05. If they are combined with the frequency due to internal events, the total CDF from both internal events and seismic events would range from 4.7 E-05 to 8.6 E-05. However, the three sites chosen for the evaluation represent sites where higher hazard estimates have been predicted using the LLNL methodology. For the same three sites, the resulting seismic core damage frequencies using the EPRI hazard estimates range from 1.1 E-6 to 2.9 E-6, with the range of the total core damage frequency being 1.8 E-6 to 3.6 E-6. Also, for many other EUS sites, particularly those in the low seismic areas (e.g. Florida or the Gulf Coast Region), the seismically-induced core damage frequencies can easily be one or two orders of magnitude lower than that computed above. As an example, results are also shown for a midwestern site in Table 19.4-11. The calculated CDF for this site is 2 E-5, even with the alternative fragility values. Therefore, it is necessary to recognize that: (1) even for the same site, hazard predictions can be vastly different using different, but equally acceptable, methods; (2) mean hazard predictions are driven by the large uncertainties and outlier estimates; and (3) in a vast area, such as the EUS, seismic hazard varies a great deal from site to site. It is necessary that any conclusions regarding site suitability or design suitability not be governed by using numerical results in the absolute sense. Other insights, such as the plant ruggedness, the nature of predicted sequences, and identification of failures contributing to the sequences should also be taken into account.

It should be remarked that the annual frequencies of Class II seismic events shown in Table 19.4-12 are those after giving credit to containment venting by assuming the failure probability of venting to be 0.1. For Class II seismic events, about 79 percent of the contribution to the mean frequency comes from earthquakes, with peak ground acceleration less than 1g. Under these accelerations, the containment structural integrity is preserved since the HCLPF capacity of the containment is about 1g. It is, therefore, reasonable to assume that the operator would be able to vent the containment if the venting is to be done manually.

A review of the HCLPF capacities for different accident classes also reveals the importance of certain classes. From Table 19.4-12, it can be seen that the two classes with the lowest HCLPF capacities are Class IB-2 (HCLPF = 0.64 g) and Class II (HCLPF = 0.73g). With the alternative fragility estimates, these HCLPF capacities are changed to 0.63g and 0.7g respectively, which are still about twice the plant SSE. These capacities, therefore, appear to have considerable margins. It is important to note, however, that for sequences where the ambient containment pressure is not reduced sufficiently to allow the operation of steam-driven pumps, the HCLPF value for the plant reduces to 0.43g. The robustness of the ranking of sequences, and hence contributing failures, was discussed in the previous sections.

Finally, the ABWR seismic PRA considered isolation of a seismically induced RHR heat exchanger failure to prevent draining of the suppression pool. However, seismic events leading to an early failure of containment isolation are not explicitly analyzed. Although failures of the ECCS lines and related valves appear to be implicitly treated, failures of other containment penetration lines (e.g., inert lines, ILRT lines, and purge lines) or containment isolation valves due to seismic events are not addressed. This may have some impact on risk integration. The staff has concluded that GE should provide an evaluation of the probability and consequences of containment penetration lines or containment isolation valves failing during a seismic event. This is an outstanding item.

4.3.8.2 Seismic Review Conclusions

From this review, the following conclusions are drawn:

(1) The study identified the dominant sequences contributing to seismic risk and demonstrated that the high confidence-low probability of failure accelerations (seismic margins) for these sequences are 0.64g or greater. It is concluded, therefore, that the ABWR standard design for structures and equipment exhibit seismic capability significantly beyond Safe Shutdown Earthquake acceleration levels (approximately twice the SSE) as long as the assigned seismic capacities are achieved, and unanalyzed failure modes do not have adverse impacts.

(2) By convolving the accident class fragilities with the seismic hazard curves, the annual frequencies of the accident classes were obtained. The results using the LLNL seismic hazard curves demonstrate that the mean annual occurrence frequency of seismically induced core damage for the ABWR standard design is in the range of 4.9 E-05 to 7.5 E-05 for the three sites studied. The corresponding values using the EPRI hazard curves range from 1.1 E-06 to 2.9 E-06. For the sites chosen, the use of the LLNL hazard estimates indicates that the ALWR design goal of 1 E-5 for the CDF is exceeded; however, for the same sites the use of the EPRI hazard estimates indicates that the CDF estimates are quite below the goal. Even with the alternate fragility values and the higher of the two hazard estimates, CDF for high seismic sites in the Eastern United States is in the order of 2 E-04. For a more representative site, this estimate is in order of 2 E-5.

(3) An analysis of the contribution of different peak ground acceleration ranges to the core damage accident class frequencies demonstrated that the contribution of earthquakes up to 0.5g pga is not significant. This supports the conclusion that very large earthquakes must occur in order for any significant damage to be done to the plant.

(4) The influence of uncertainty in different assumptions made in the fragility modeling and seismic hazard on the core damage class frequencies was investigated. From this, it is concluded that the results are fairly insensitive to uncertainty in fragility and are driven by uncertainty in seismic hazard.

(5) The sensitivity of the accident class frequencies to seismic hazard was addressed by comparing the results obtained using the hazard curves developed by LLNL and EPRI for the Pilgrim, Seabrook and Watts Bar sites. The results showed that, while gross differences in hazard models can significantly affect the accident class frequencies, the importance or ordering of different classes in terms of their contribution to the total core damage frequency is not significantly affected.

(6) For the three alternative sites examined, the accident classes IB-2, IC and II were identified as dominant in their contribution to the total core damage frequency.

(7) The seismic margins expressed as the High Confidence of Low Probability Failure capacities for accident classes IB-2 and IC are 0.64g and 0.88g, respectively. For accident class II, the seismic margin is 0.73g.

(8) The ranking of the sequences, and hence contributing failures, are insensitive to the hazard selection and is, therefore, relatively robust.

In summary, pending the resolution of open items, the ABWR plant design from the perspective of seismically-induced severe accidents is demonstrated to have a significant capacity beyond the design basis. With the assumed fragilities, the computed range of seismic CDFs demonstrate that the plant design could be located at many of the EUS sites with the likelihood that the CDF will be less than 1 E-4. In order to demonstrate such a suitability, a number of interfacing requirements will have to be met, and a site/plant specific seismic PRA will have to be performed such that failure modes not considered at this stage do not invalidate the above conclusions. These conclusions are based on the core damage accident sequences induced by the seismic events, the review of consequence analysis and other deterministic design reviews contained in other parts of this SER may have separate requirements and conclusions.

4.4 Interface Requirements for Other External Events

1. The staff, consistent with the recommendations of Reference 19.53, pending the staff review of the ALWR Requirements Document, requires that a site and design verification be performed when a specific site is selected for the external events, such as external floods and transportation hazards, for which no analyses can be performed at this stage.

2. The probabilistic analysis for internal floods must be performed when a specific site is selected and the plant is built.

4.5 External Events Review Conclusions

Review findings for the tornado strike and seismic events are discussed in Sections 19.4.2 and 19.4.3.8.3, respectively. With respect to other events, GE has not conducted any quantitative analyses. It is concluded that some quantitative analyses should be performed for fire and internal flood hazards. For other external events, such as external floods and transportation accidents, site specific evaluations will have to be performed to demonstrate no adverse impact on the risk from these events. Walkdowns are the major interface requirements for the external events.

For the seismic and tornado events, the design, with the assigned fragility values, has been shown to be rugged for the beyond design basis events. The computed CDFs indicate that the design can be placed at most of the EUS sites. However, a considerable interface requirement evaluation will be needed on a site-specific application to demonstrate that assumptions made in the PRA are not grossly violated and the site specific features do not adversely affect the computed CDFs or other insights.

Table 19.4-1 Comparison of Seismic Core Damage Frequency Using G.E. Hazard and Fragility Data (ABWR PRA Best Estimate Values vs. Staff-Sponsored Mean Values)

Accident Class	Description	ABWR PRA Best-Estimate CDF (/ry)	Staff-Sponsored Requantification Mean CDF (/ry)
IA	Transients followed by failure of high pressure core cooling and failure to depress- he reactor.	2.6 E-9	4.8 E-9
IB-2	Station blackout events with RCIC avail.... for core cooling for approximately 8 hours.	7.2 E-8	4.1 E-7
IC	ATWS events without boron inject on, coupled with loss of core cooling. Vessel failure at low pressure.	9.0 E-8	9.5 E-8
ID	Transients followed by loss of h gh pressure core cooling, successful depressu-ization, but loss of low pressure core cooling.	2.3 E	5.5 E-8
IE	ATWS events without boron injection, coupled with loss of core cooling. Vessel failure at high pressure.	5.0 E-8	5.3 E-8
II	Transient, LOCA and ATWS (with boron injection) events with successful core cooling, but with possible failure of containment.	3.2 E-9***	5.7 E-7**
IV	ATWS events without boron injection, but with core cooling available.	2.	5.6 E-8
IV-1	ATWS with one injection pump running. Successful flow control.	9.9 E-10	1.3 E-8
IV-2,3,5*	ATWS with 2,3 or 5 pumps running. Operator fails to control flow.	7.9 E-9	4.1 E-8
Total		2.5 E-7	1.3 E-6

* IV-2 ATWS with RCIC failure. 2 HPCF pumps are running. Operator fails to control flow.
IV-3 ATWS with 3 pumps (RCIC + 2 HPCF, or 3 LPFL) running. Operator fails to control flow.
IV-5 ATWS with an ADS actuation or a stuck-open SRV. All HPCF and LPFL pumps are assumed to be in operation. Operator fails to control flow.

** This value reflects the failure probability of containment venting, which was taken to be 0.1. The frequency before giving credit to containment venting was 5.7 E-6.

*** The frequency before processing through seismic containment event tree was 4.8 E-6.

Table 19.4-2 ABWR Seismic Fragility Summary

(Table 19.4-2 of the ABWR PRA)

Structures/Components	Failure Mode	Capacity[1] (g)	Combined[2] Uncertainty
Reactor bldg shear walls	Shear	2.8	0.45
Containment	Shear	4.3	0.44
RPV Pedestal	Flexural	7.9	0.44
Control Building	Flexural	2.0	0.50
Turbine Building	Flexural	1.0	0.50
Reactor pressure vessel	Skirt anchor bolts	5.3	0.33
Shroud support	Buckling	1.9	0.36
CRD guide tubes	Buckling	1.7	0.36
CRD housings	Plastic yielding	3.9	0.46
Fuel Assemblies	Channel buckling	1.3	0.35
Cable trays	Support	2.0	0.60
Batteries and battery racks	Anchorage/LOF	3.0	0.45
Battery chargers/inverters	LOF	1.3	0.45
Electric equipment (chatter)			
function required during event	Relay chattering	0.8	0.50
function required after event	Relay chattering	2.0	0.50
Panelboards/Instrumentation panels	Functional/Structural	3.0	0.45
Switchgear/Motor control centers	Functional/Structural	2.5	0.45
Transformers	Functional/Structural	1.5	0.45
Diesel generators & support systems	Support	2.5	0.45
Turbine-driven pumps	Anchorage	2.0	0.45
Motor-driven pumps	Anchorage/Impeller defle	1.6	0.45
Heat exchangers/Small tanks	Anchorage	2.0	0.45
Air-operated valves	Stem binding/Air line	3.0	0.60
Motor-operated valves	Operator distortion	3.0	0.60
Safety relief, manual & check valves	Internal damage	3.0	0.60
Hydraulic control units	LOF	2.0	0.50
Accumulators	Support	2.0	0.45
Large flat-bottom storage tanks	Anchorage	0.9	0.45
HVAC ducting	Support	2.0	0.60
Air handling units	Blade rubbing	2.0	0.50
Piping	Support	3.0	0.60
Buried welded steel piping	Buckling/Support	2.0	0.40

[1] Capacities are in terms of median ground acceleration.

[2] Combined uncertainties are composite logarithmic standard deviations of uncertainty and randomness components.

Table 19.4-3 Alternative Fragility Values for Selected Components

Components	Alternative Values			Original Values		
	Capacity (g)	Combined Uncertainty	HCLPF(g)	Capacity(g)	Combined Uncertainty	HCLPF(G)
Control Building	1.5	0.50	0.47	2.8	0.45	0.98
Diesel Generators	1.5	0.50	0.47	2.5	0.45	0.88
Transformer (480 V AC)	1.1	0.45	0.39	1.5	0.45	0.53
Batteries and Racks	1.5	0.45	0.52	3.0	0.45	1.05
Battery Chargers Inverters	1.1	0.45	0.39	1.3	0.45	0.46
Relay Switches (120 V)	1.3	0.50	0.40	2.0	0.50	0.62
Transformers (480 V SW)	1.1	0.45	0.39	1.5	0.45	0.53
Motor Control Ctr (480 V SW)	1.5	0.50	0.47	2.5	0.45	0.88
Fuel Assembly	0.9	0.35	0.40	1.3	0.35	0.58
Heat Exchanger	1.4	0.45	0.50	2.0	0.45	0.70
Tanks (SLC)	1.1	0.45	0.39	1.5	0.45	0.53
Tank (fire water)	1.4	0.45	0.50	2.8	0.45	0.98
Heat Exchanger (SW)	1.4	0.45	0.50	2.0	0.45	0.70
Room Air Cond. Unit	1.2	0.50	0.37	2.0	0.50	0.62
Pump, Motor	1.6	0.45	0.56	1.6	0.45	0.56

NOTE: Reference 19.20 contains explanations for alternate fragility assignments.

Table 19.4-4 Random Failure Probabilities Used in Quantifying Seismic Core Damage Frequency (Staff Requantification)

Seismic Event Tree Top Event	Definition	Random Failure Probability (Mean Value)	Error Factor for Lognormal Distribution
C	Scram and ARI failure.	1.0 E-06	5
C4	Failure to initiate SLC.	0.2	3
C42	Failure to initiate SLC following failure to inhibit ADS.	0.3	2
FA	Failure of fire water.	0.1	3
FCTR	Flow control/alternate boron.	0.2	3
HX	RHR heat exchanger failure	6.0 E-03	3
LOP	Loss of offsite power.	0	-
LPL	SRVs fail to open.	1.0 E-02	3
PA	Failure to inhibit ADS.	0.1	3
PC	SRVs fail to reclose.	3.0 E-03	5
PC2	SRVs fail to reclose during ATWS.	0.1	3
PW	Emergency power/emergency service water.	0	-
SI	Structural integrity.	0	-
UH	Failure of HPCF (1 out of 2).	8.0 E-03	3
UR	Failure of RCIC.	4.0 E-02	3
UR2	Failure of RCIC (ATWS).	5.2 E-02	3
V	Failure of LPFL (1 out of 3).	7.4 E-02	3
W1	Failure of RHR (1 out of 3).	1.6 E-03	3
W2	Failure of RHR (2 out of 3).	3.9 E-02	3
X	Failure of manual depressurization.	2.0 E-03	3
X1	Failure of manual depressurization (station blackout).	2.0 E-02	5
X2	Failure of manual depressurization (ATWS).	1.0 E-01	5

Table 19.4-5 Advanced Boiling Water Reactor Design - A Summary of Mean Core Damage Frequency Based on BNL (Internal Events) and EQE-BNL (Seismic Events) Requantifications

Accident Class	Mean C.D.F. due to Internal Events (/ry)	Mean Core Damage Frequency Due to Seismic Events						
		Seismic Hazard Curve of ABWR PRA [1]	LLNL Mean Hazard Curve			EPRI Mean Hazard Curve		
			Pilgrim	Seabrook	Watts Bar	Pilgrim	Seabrook	Watts Bar
1A	3.4 E-7	4.8 E-9	2.3 E-7	1 5 E-7	2.6 E-7	9.9 E-9	6.3 E-9	3.8 E-9
IB-1[2]	1.8 E-8	-----	-----	-----	-----	-----	-----	-----
IB-2	6.1 E-9	4.1 E-7	2.2 E-'	1.5 E-5	2.3 E-5	9.5 E-7	6.0 E-7	3.6 E-7
IB-3[2]	6.4 E-10	-----	· ---	-----	-----	-----	-----	-----
IC	8.0 E-10	9.5 E-8	1.8 E-5	1.2 E-5	1.7 E-5	3.3 E-7	3.4 E-7	2.0 E-7
ID	1.1 E-7	5.5 E-8	3.5 E-6	2.3 E-6	3.7 E-6	1.2 E-7	8.6 E-7	5.1 E-8
IE	-----[3]	5.3 E-8	7.3 E-6	5.1 E-6	7.0 E-6	1.6 E-7	1.5 E-7	8.6 E-8
II	2.8 E-8	·5.7 E-7[4]	1.3 E-5[4]	9.9 E-6[4]	1.7 E-5[4]	1.1 E-6[4]	5.5 E-7[4]	3.4 E-7[4]
IIIA[5]	5.3 E-9	-----	-----	-----	-----	-----	-----	-----
IIID[5]	1.3 E-8	-----	-----	-----	-----	-----	-----	-----
IV	2.3 E-7	5.6 E-8	4.2 E-6	2.7 E-6	4.3 E-6	1.2 E-7	9.3 E-8	5.6 E-8
IV-1[6]	-----	1.3 E-8	7.1 E-7	4.5 E-7	7.5 E-7	2.5 E-8	1.7 E-8	1.0 E-8
IV-2,3,5[6]	-----	4.1 E-8	1.5 E-6	1.0 E-6	1.8 E-6	8.1 E-8	4.7 E-8	2.8 E-8
Total	7.5 E-7	1.3 E-6	7.0 E-5	4.9 E-5	7.5 E-5	2.9 E-6	1.9 E-6	1.1 E-6

(1) This is G.E.'s bounding hazard curve taken from GESSAR II PRA.

(2) For seismic events, core damage frequencies due to classes IB-1 and IB-3 (station blackout with RCIC failure) are considered negligible.

(3) For internal events, no distinction is made between class IC (ATWS, vessel fai'ure at low pressure) and Class IE (ATWS, vessel failure at high pressure).

(4) These values reflect the failure probability of containment venting, which was taken to be 0.1.

(5) Class III seismic CDF is negligibly small because of relatively low seismic fragility of the piping and other pressure boundary components.

(6) These subclasses of Class IV events were not considered in the ABWR PRA internal event analysis.

IV-1 ATWS with one injection pump running. Successful flow control.
IV-2 ATWS with RCIC failure. 2 HPCF pumps are running.
IV-3 ATWS with 3 pumps (RCIC + 2 HPCF or 3 LPFL) running.
IV-5 ATWS with ADS actuated or a stuck-open SRV. All HPCF and LPFL pumps are running.

For IV-2, IV-3 and IV-5, operator fails to control flow.

19.4.6 Comparison of Mean Annual Sequence Frequency Ordering for Different Seismic Hazard Curves

			LLNL Hazard						EPRI Hazard					
	ABWR Hazard		Pilgrim		Seabrook		Watts Bar		Pilgrim		Seabrook		Watts Bar	
Sequence	Frequency*	Rank	Frequency	Rank	Frequency	Rank	Frequency	Rank	Frequency	Rank	Frequency	Rank	Frequency	Rank
IA	4.8E-09	9	2.3E-07	9	1.5E-07	9	2.6E-07	9	9.9E-09	9	6.3E-09	9	3.8E-09	9
IB-2	4.1E-07	2	2.1E-05	1	1.5E-05	1	2.3E-05	1	9.5E-07	2	5.9E-07	2	3.6E-07	2
IC	9.5E-08	3	1.8E-05	2	1.2E-05	2	1.7E-05	3	3.3E-07	3	3.4E-07	3	1.9E-07	3
ID	5.5E-08	5	3.5E-06	6	2.3E-06	6	3.7E-06	6	1.2E-07	6	8.6E-08	6	5.1E-08	6
IE	5.3E-08	6	7.3E-06	4	5.1E-06	4	7.0E-06	4	1.6E-07	4	1.5E-07	4	8.6E-08	4
II**	5.7E-06	1	1.3E-05	3	9.9E-06	3	1.7E-05	2	1.1E-06	1	5.5E-07	1	3.4E-07	1
IV	5.6E-08	4	4.2E-06	5	2.7E-06	5	4.3E-06	5	1.2E-07	5	9.5E-08	5	5.6E-08	5
IV-1	1.3E-08	8	7.1E-07	8	4.5E-07	8	7.5E-07	8	2.5E-08	8	1.7E-08	8	1.0E-08	8
?,3,5	4.1E-08	7	1.5E-06	7	1.0E-06	7	1.8E-06	7	8.1E-08	7	4.7E-08	7	2.8E-08	7

Table 19.4-7 Dominant Contributors to Accident Class Frequencies Based on Calculations Using Mean LLNL Seismic Hazard - Curve for Pilgrim Site

Accident Class	Mean Annual Frequency	Rank Order of Accident Class	Dominant Contributors*
IA	2.3 E-7	9	Mostly high capacity doubles and triples
IB-2	2.2 E-5	1	Inverters, 480 V AC transformer, Service Water Pump
IC	1.8 E-5	2	Fuel Assemblies
ID	3.5 E-6	6	Motor driven pumps
IE	7.3 E-6	4	Reactor building, control building
II	1.3 E-5	3	Inverters, 480 V AC transformer, service water pump, motor driven pump
IV	4.2 E-6	5	Mostly doubles
IV-1	7.1 E-7	8	Mostly doubles
IV-2,3,5	1.5 E-6	7	Mostly doubles
Total	6.9 E-5		

* Loss of offsite power is assumed to occur at small earthquake acceleration values.

Table 19.4-8 HCLPF Values for Accident Classes

Classes	HCLPF(g)
IA	-
IB-2	0.64
IC	0.88
ID	1.01
IE	0.91
II	0.73
IV	0.86
IV-1	-
IV-2,3,5	-

NOTE: Failure probability of containment venting (0.1) is included in Class II analysis.

Table 19.4-9 Annual Core Damage Sequence Frequencies Calculated using LLNL Seismic Hazard Curves for the Pilgrim Site

Sequence	HCLPF(g)	Mean	Median	5% Confidence	95% Confidence
IA	-	3.6 E-07	1.1 E-08	5.6 E-11	8.0 E-07
IB-2	0.64	2.7 E-05	1.2 E-06	6.5 E-09	6.7 E-05
IC	0.88	2.3 E-05	6.8 E-07	8.5 E-10	5.1 E-05
ID	1.01	3.8 E-06	1.4 E-07	7.2 E-10	8.7 E-06
IE	0.91	9.5 E-06	1.8 E-07	2.0 E-10	1.9 E-05
II*	0.73**	1.3 E-05	8.6 E-07	1.1 E-08	3.7 E-05
IV	0.86	5.6 E-06	1.6 E-07	3.8 E-10	1.2 E-05
IV-1	-	6.6 E-07	1.8 E-08	5.5 E-11	1.5 E-06
IV-2,3,5	-	1.9 E-06	5.8 E-08	2.4 E-10	4.3 E-06

* Failure probability of containment venting (0.1) is included

** HCLPF without containment venting is 0.43g.

Table 19.4-10 Comparison of Core Damage Frequency for Different Sequences using (A_m, β_r, β_u) and (A_m, β_c) with Full Set of LLNL Seismic Hazard Curves for Pilgrim Site

	With (A_m, β_r, β_u)				With (A_m, β_c)			
Sequence	Mean	Median	5% Confidence	95% Confidence	Mean	Median	5% Confidence	95% Confidence
IA	3.6E-07	1.1E-08	5.6E-11	8.0E-07	2.42E-07	1.31E-08	8.80E-11	6.5E-07
IB-2	2.74E-05	1.15E-06	6.46E-09	6.65E-05	2.59E-05	1.24E-06	7.97E-09	6.61E-05
IC	2.28E-05	6.78E-07	8.51E-10	5.10E-05	2.29E-05	7.74E-07	1.29E-09	5.63E-05
ID	3.76E-06	1.37E-07	7.17E-10	8.74E-06	3.89E-06	1.86E-07	1.02E-09	1.00E-05
IE	9.45E-06	1.82E-07	2.04E-10	1.86E-05	1.03E-05	3.60E-07	9.50E-10	2.49E-05
II*	1.30E-05	8.58E-07	1.05E-08	3.71E-05	1.05E-05	8.82E-07	1.49E-08	3.29E-05
IV	5.57E-06	1.63E-07	3.75E-10	1.23E-05	4.42E-06	2.00E-07	6.48E-10	1.13E-05
IV-1	6.58E-07	1.84E-08	5.54E-11	1.46E-06	6.72E-07	3.43E-08	1.54E-10	1.78E-06
IV-2,3,5	1.88E-06	5.82E-08	2.43E-10	4.33E-06	1.51E-06	9.25E-08	6.63E-10	4.19E-06

Table 19.4-11 Accident Class Frequencies for Different Sites with Modified Fragilities (LLNL Hazard Curves)

Class	Pilgrim	Seabrook	Watts Bar	Zion	HCLPF
IA	1.88 E-07	1.38 E-07	2.39 E-07	4.10 E-08	-
IB-2	5.59 E-05	3.99 E-05	6.38 E-05	6.79 E-06	0.63
IC	2.14 E-05	1.42 E-05	2.18 E-05	1.89 E-06	0.85
ID	3.43 E-06	2.34 E-06	3.95 E-06	4.07 E-07	-
IE	5.24 E-05	3.71 E-05	5.90 E-05	5.44 E-06	0.66
II	1.96 E-05	1.59 E-05	2.79 E-05	3.20 E-06	0.70
IV	9.84 E-06	6.72 E-06	1.14 E-05	1.37 E-06	0.89
IV-1	5.21 E-07	3.87 E-07	6.76 E-07	7.34 E-08	-
IV-2	1.92 E-06	1.58 E-06	2.80 E-06	4.31 E-07	-
Total	1.65 E-04	1.18 E-04	1.91 E-04	1.96 E-05	

Table 19.4-12 Mean Core Damage Frequency Based on BNL (Internal Events) and EQE-BNL (Seismic Events) Uncertainty Analyses

Accident Class	Mean C.D.F. Due to Internal Events (/ry)	Mean Core Damage Frequency Due to Seismic Events Using LLNL Seismic Hazard Curves* (/ry)			HCLPF (g)
		Pilgrim	Seabrook	Watts Bar	
IA	3.4E-7	3.6E-7	2.1E-7	3.7E-7	———
IB-1	1.8E-8	———	———	———	———
IB-2	6.1E-9	2.7E-5	1.6E-5	2.6E-5	0.6
IB-3	6.4E-10	———	———	———	———
IC	8.0E-10	2.3E-5	1.2E-5	1.8E-5	0.9
ID	1.1E-7	3.8E-6	2.2E-6	3.6E-6	1.0
IE	———	9.5E-6	4.7E-6	7.5E-6	0.9
II	2.8E-8	1.3E-5**	6.9E-6**	1.2E-5**	0.7
IIIA	5.3E-9	———	———	———	———
IIID	1.3E-8	———	———	———	———
IV	2.3E-7	5.6E-6	3.1E-6	5.2E-6	0.9
IV-1	———	6.6E-7	3.9E-7	6.7E-7	———
IV-2,3,5	———	1.9E-6	1.2E-6	2.0E-6	———
Total	7.0-E-7	8.5E-5	4.6E-5	7.6E-5	

* All sites considered are enveloping sites with respect to postulated seismic events.

** For Class II seismic events, credit is given for containment venting by assuming the failure probability of venting to be 0.1. This reduces the Class II seismic CDF by a factor of ten.

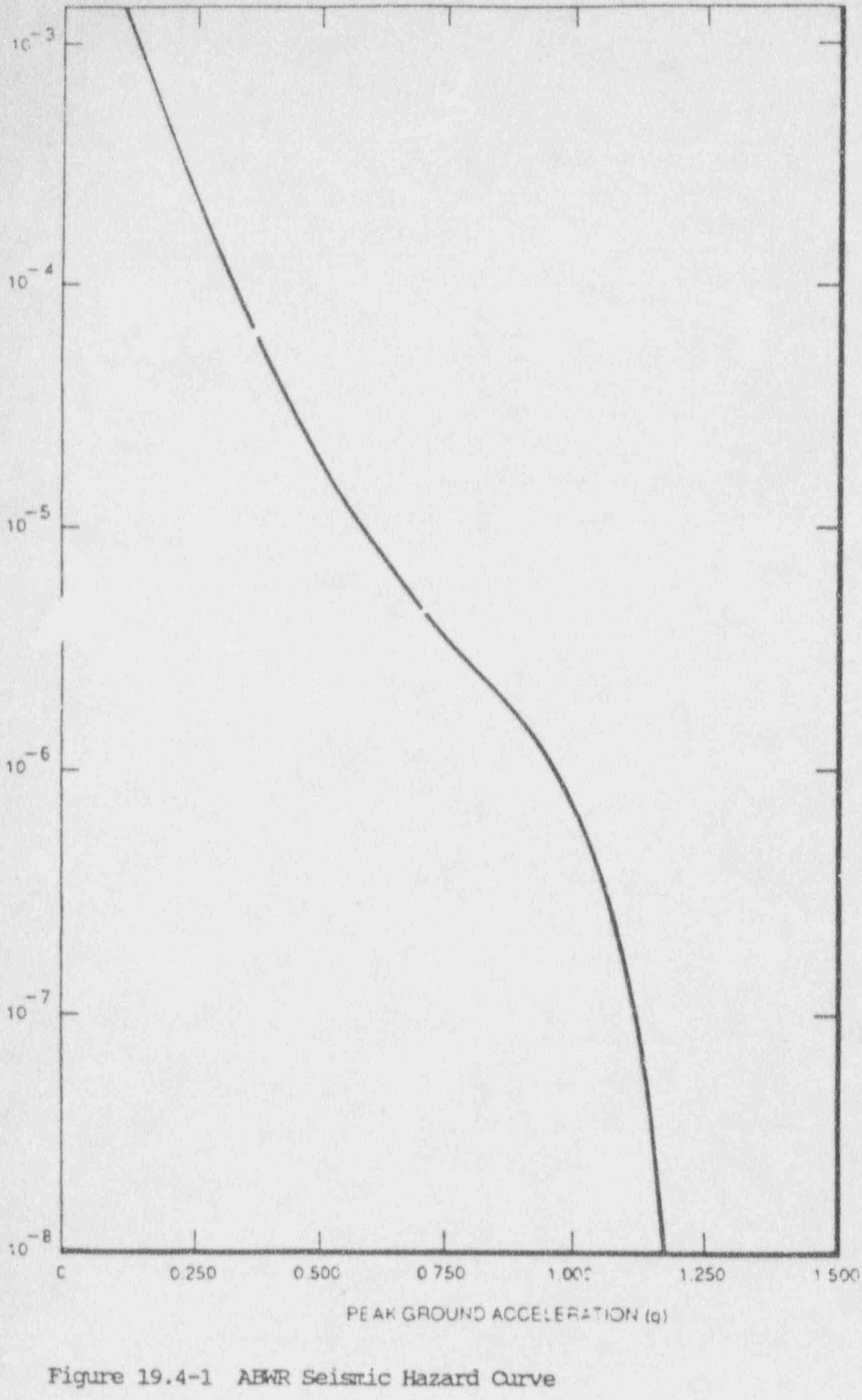

Figure 19.4-1 ABWR Seismic Hazard Curve

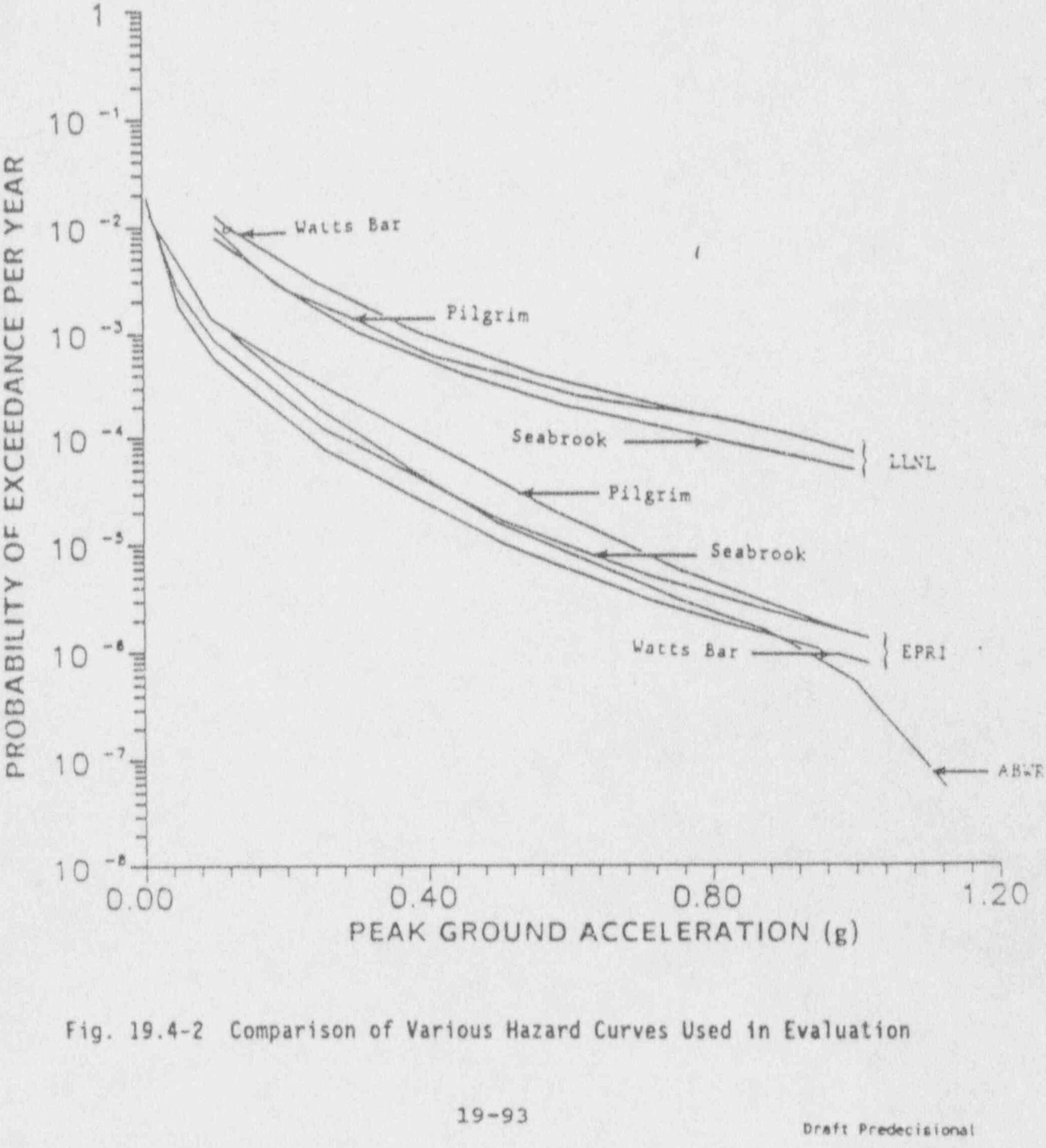

Fig. 19.4-2 Comparison of Various Hazard Curves Used in Evaluation

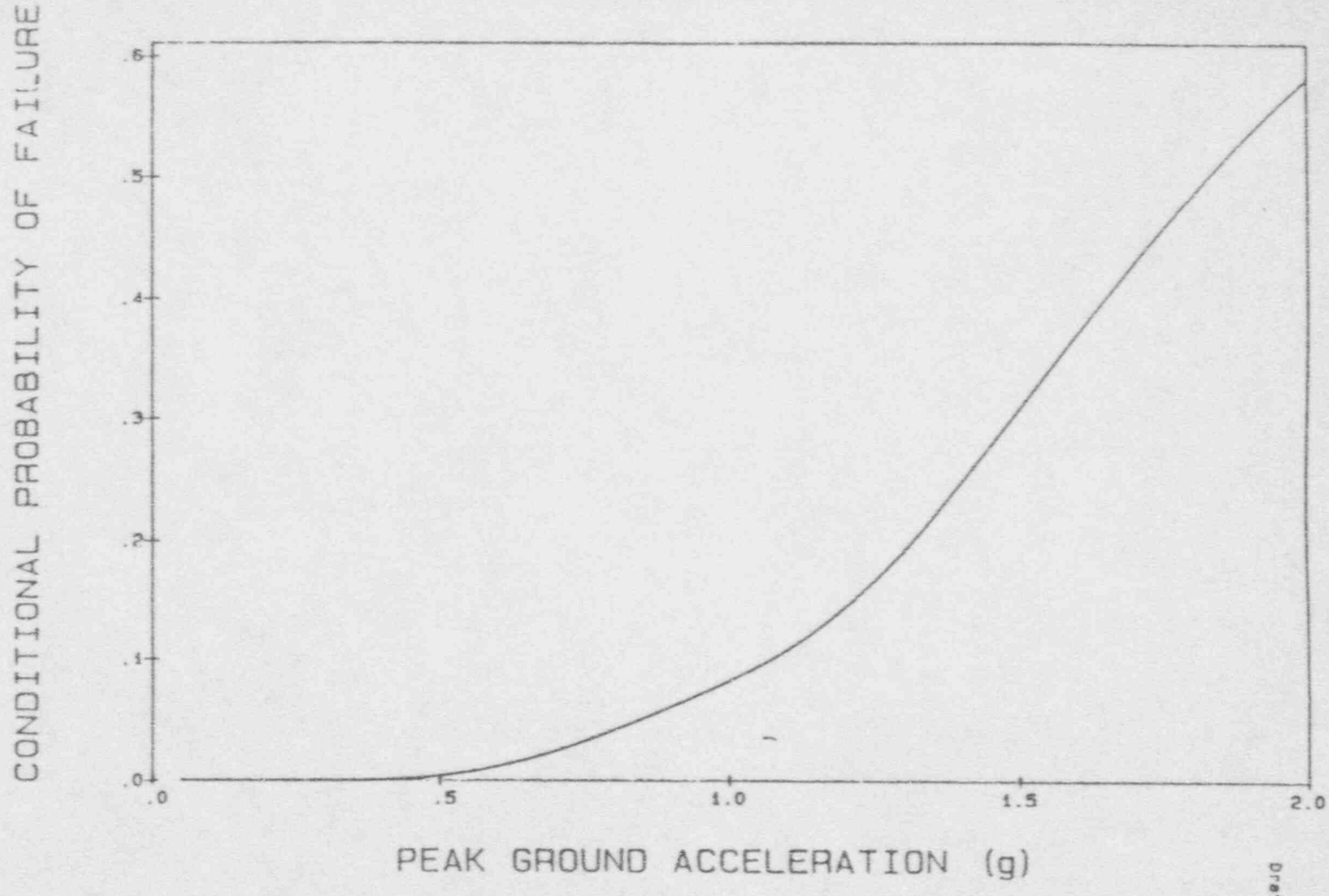

Fig. 19.4-3 Mean Fragility Curve for Sequence IB-2

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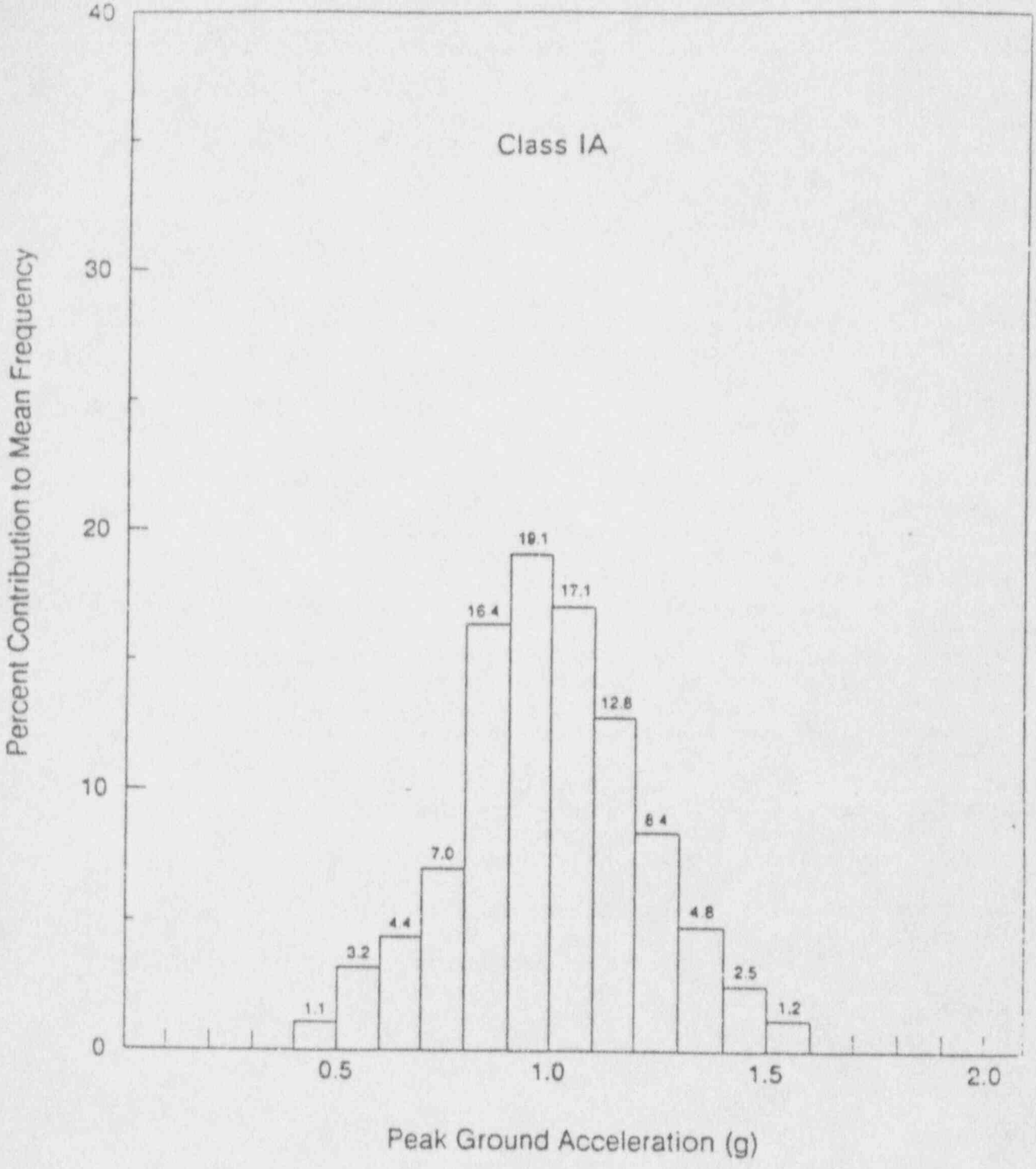

Fig. 19.4-4 An Example of Percent Contribution of Different Acceleration Ranges to Mean Frequency of Sequence IA (LLNL - Pilgrim Site)

19.5 INTRODUCTION TO THE LEVEL 2/LEVEL 3 REVIEW

The accident sequence event tree analysis, Level 1, primarily determines the various combination of system actions to shut down the reactor, cool the core, and prevent its damage. The accident progression event tree analysis, Level 2, potentially involving a core melt, vessel breach, and containment challenge, determines the various combinations of actions and systems that enter into the determination of the containment failure probability and source term releases.

GE's Level 2 analysis began with the development of a containment event tree for each of the accident sequence classes and subclasses from the Level 1 analysis. Accident progression pathways through each CET were determined along with their frequency and fission product releases. The numerous pathways were consolidated by grouping their outcomes according to the various factors, such as the mitigating systems involved in the events (i.e. passive flooder, firewater sprays system), the mechanism of the release to the environment (i.e. normal leakage, leakage through penetrations), the magnitude of the release (low, medium, high), and the timing of the release, into groups of source terms. Groups that had negligible frequencies were combined with other similar groups, instead of being discarded, to account for the entire core damage frequency. For each group, source terms were calculated with the MAAP code.

In GE's Level 3 analysis, off-site consequences were calculated with the CRAC2 code. Consequences were determined at five sites, each representing a geographic region of the U.S. The results of the five sets of consequence calculations were averaged and then compared to various safety goals.

GE's risk integration approach was to quantify the CETs with point estimates which were multiplied together to determine the frequency of each source term group. Averaged consequences for each such group were multiplied by the frequencies of the groups and summed to determine a point estimate of risk.

The staff's approach was different. At each stage of the analytical process (with the exception of off-site consequences), uncertainties in a few key parameters were estimated and combined to estimate the uncertainty in the risk estimates. These individual uncertainty estimates are described in the section on containment performance (Section 19.6) and source terms (Section 19.7) and compared to GE's relevant point estimates. For the risk integration (Section 19.9), an approach similar to that used in the NUREG-1150 study (Reference 19.62) was used.

GE's PRA is based on information as of Amendment 8. Interactions between the NRC staff and GE have resulted in design modifications discussed in subsequent amendments of the SSAR; however, the PRA was not appropriately updated. Specifically, the following features are to be added to the ABWR but not included in GE's PRA:

- o Strengthened drywell head such that the ultimate strength is increased from 100 psig to 134 psig.

- o Lower drywell wall and floor composition of basaltic concrete, instead of limestone concrete.

The analyses serving as the basis for the NRC staff's review were done at the Brookhaven National Laboratory and adopted by the NRC staff. These analyses were based on Amendment 8 of GE SSAR (Reference 19.63) and included information from the NUREG-1150 study (Reference 19.62); the findings are discussed in greater detail than appear here in "A Review of the Advanced Boiling Water Reactor Probabilistic Risk Assessment, Vol. 2: Internal and External Events, Containment, and Offsite Consequence Analysis," Brookhaven National Laboratory, dated 1991 (Reference 19.64). In addition, the staff's review was discussed with several staff members of Sandia National Laboratories who participated in the analyses of the NUREG-1150 study (Reference 19.62).

19.6 CONTAINMENT PERFORMANCE

6.1 Introduction

The review of the ABWR containment performance utilized several types of calculations. A stochastic assessment of the overall performance was done in the containment event tree analysis. Deterministic assessments were done with the MELCOR (Reference 19.65) and the STCP (Reference 19.66) codes.

6.2 Methods Discussion

A review of the ABWR CET is necessary to evaluate the robustness of GE's assessment of accident sequence progression characteristics following the onset of core damage and to evaluate the nature and the frequency of the threats to the containment and source term releases. The objectives of reviewing the CET then are to assess the reasonableness of the CET, assess the significance of ABWR features and operator actions to CET results, and assess GE's conclusions based on the outcome of the CET analysis. The analysis and the review are described, the findings are presented, and implications and relationships are discussed. The parts of the CET that were reviewed are the general approach, the structure of the CETs, the data, and the assumptions.

The CET should represent a logical and consistent way to ascertain the various accident progression sequences. There are four aspects to this, namely, characterizing accident progressions, determining the relationship between severe accident phenomena, quantifying the probability of accident progression groups, and quantifying source terms.

In the Level 1 portion of the PRA, both GE and the staff estimated the core damage frequencies (CDF) based on several assumptions regarding the availability of the gas turbine generator and the fire water system. Among the possible combination of assumptions, GE selected the case where only the gas turbine generator is assumed available, not the firewater system. In the staff's base case, both the gas turbine generator and the firewater system were assumed available. The significance of this is discussed in Section 19.6.4.1. Here, it suffices to say that the assumptions influence the contribution of various types of accident sequences and the output of the CETs.

6.2.1 GE Analysis

The results of the Level 1 portion of the PRA are grouped into similar outcomes, called accident classes or plant damage states (PDSs), that describe the condition of the plant at the onset of core damage. In GE's analysis, each PDS has a separate CET (all of which have a similar structure), and which includes recovery actions where appropriate. Calculations from the MAAP code

(Reference 19.67), an industry code, were used as the basis for developing and analyzing the CETs.

Compared to the CETs in the NUREG-1150 study (Reference 19.62), where there are 125 questions in the Grand Gulf analysis, the CETs built by GE are small and are composed of a minimal number (usually nine) of more general questions defining the branch points of the accident progressions. Most of the CETs have nine such questions relating to the following areas of an accident progression:

- o Depressurization of the RCS.
- o Availability of containment heat removal.
- o Core melt arrest in the reactor vessel as a result of recovery of one of the ECC systems.
- o Containment failure at the time of vessel failure as a direct and immediate consequence of vessel failure.
- o High temperature failure of the containment as a result of core debris in the upper drywell.
- o Prior to reaching the containment failure pressure, core melt arrest in the lower drywell as a result of recovering an ECCS function or the addition of firewater.
- o Quenching of the core debris in the lower drywell by the passive flooder system.
- o Recovery of containment heat removal prior to containment failure.
- o Venting.

The branch point probabilities of the CETs were quantified with point estimates which in turn were multiplied together to determine the likelihood of the accident progression.

GE did not analyze uncertainty. In contrast, the NUREG-1150 study (Reference 19.62), propagated probability distributions throughout the CETs to estimate risk.

In addition, several phenomena were not included in GE's CETs but which the staff considered important in containment performance. Such phenomena include in-vessel fuel/coolant interaction, ex-vessel fuel/coolant interaction, core/concrete interactions, direct containment heating, and drywell/wetwell suppression pool bypass.

6.2.2 Staff Review

The staff's review consisted of an audit calculation based on GE's CET. GE's CET structure was modified by making simplifications and additions to the CET. Modifications to the CETs were as follows:

- o Outcomes of the CETs were consolidated into fewer, slightly more general groups.
- o Unnecessary questions were eliminated, such as when certain phenomena or events always occur.
- o Questions were added and a few questions were reworded to account for missing phenomena and events.

More important than modifying the structure was the staff's attempt to account for, in an approximate and preliminary way, some phenomena known not to have been taken into account in GE's CETs, such as direct containment heating and ex-vessel fuel/coolant interaction.

The staff took into consideration some threats to the containment found in other studies, such as the NUREG-1150 study (Reference 19.62). Some phenomena, such as hydrogen combustion and liner melt-through, could be eliminated because the ABWR containment is inerted and the ABWR reactor cavity is configured to prevent core debris impingement, respectively. Another phenomena, design basis accident pressure load from a blowdown of the reactor coolant system following a failure of the reactor vessel, could be eliminated because the staff assessed that the design strength of the containment (134 psig) is capable of withstanding the peak pressure spike (40 to 50 psig) from this accident. Two phenomena that could not be dismissed were direct containment heating and ex-vessel fuel/coolant interaction. As described below, the staff briefly treated these phenomena differently than other phenomena in determining the fractional contributions to the accident progression group frequency.

The estimates of containment loads associated with direct containment heating and ex-vessel fuel/coolant interaction were determined with a Monte Carlo sampling procedure. The pressure loads were obtained as a distribution from the NUREG-1150 analysis (Reference 19.62) of the Grand Gulf plant, which has a similar power rating and drywell size as the ABWR. The uncertainty in the ABWR containment strength was estimated to be $\pm$ 20 psi of the mean ultimate containment strength, based on the NUREG-1150 study (Reference 19.62) and the staff's judgment. Both distributions were sampled using LHS (Ref 19.67), a type of Monte Carlo sampling. The staff's calculations were repeated with an uncertainty in the containment strength of ± 40 psi; little difference in the distribution of accident progression groups was observed.

Incorrect

added

The staff independently estimated the distribution of accident progression groups using the staff's ~~CETs and~~ assumptions regarding the integrity of the containment (intact or failed) and the status of the RHR system (available or unavailable) for each class of accident sequences within the accident progression pathways. The results of this determination are shown in Figures 19.6-1 and 19.6-2; these are point estimates, which, except in the treatment of direct containment heating and ex-vessel fuel/coolant interaction, reflect the staff's engineering judgment of a reasonable selection of inputs.

Simplified point estimates of the effects of direct containment heating and ex-vessel fuel/coolant interaction were determined by sampling the uncertainty in the pressure loading and the containment strength forty times. When the pressure loading was greater than the containment strength, the sample was counted as a failed containment and vice versa. The point estimate of the early containment failure probability was taken as the number of failure trials divided by the total number of trials. The staff's analysis indicated a containment failure probability, conditional on vessel breach, due to these mechanisms (after the containment design modifications) of 0.11 for the high pressure vessel failure case and 0.04 for the low pressure vessel failure case. These point estimates, which were factored into the staff's CETs, replaced GE's probability values of 0.001 and 0.0, respectively, for containment failure at the time of vessel failure. The results should be understood as providing only a rough estimate of the threat posed by direct containment heating and fuel/coolant interaction to the integrity of the containment because the staff's analysis is based in part on an analysis of another plant, Grand Gulf.

In the review, the complexity of some phenomena known from the NUREG-1150 study (Reference 19.62) to represent potentially significant threats to the containment precluded the staff from readily accounting for them directly in the CETs. These phenomena include drywell/wetwell bypass, the effect of in-vessel fuel/coolant interaction on in-vessel core recovery, impulse loads on the reactor pedestal and quasi-static loads on the drywell from ex-vessel fuel/coolant interaction, the effects of a core/concrete interaction on the integrity of the reactor pedestal, and the effect of venting on the accident progressions. Consideration of some phenomena, such as drywell/wetwell bypass and vent setpoint, was inferred but not directly factored into the staff's CET analysis.

This sentence is backwards

It is unclear how the design differences in total between the ABWR and Grand Gulf would modify the containment performance predicted for Grand Gulf in the NUREG-1150 (Reference 19.62). For example, the Grand Gulf plant has a standard atmosphere in the containment while the ABWR has an inerted atmosphere; this would act to reduce the pressure loads arising from direct containment heating, by eliminating the oxidation/combustion component of DCH load. The Grand Gulf plant has a larger wetwell airspace compared to the ABWR, which would affect the pressure loads as the drywell is rapidly pressurized and the downcomers, connecting the drywell to the wetwell, are cleared.

Muddled

Nevertheless, with these modifications and inferences, the staff's review showed that the outcomes of the CET appear to be strongly influenced by the probabilities of core melt arrest in the reactor vessel or in the containment and the availability of the RHR system. The staff's treatment of containment failure was further resolved into questions of high temperature degradation of the upper drywell seals due to debris entering the upper drywell, and early containment failure due to rapid pressurization from direct containment heating and fuel/coolant interaction. Subsequently estimated was the fraction of the CDF resulting in a particular group of accident progression pathways.

6.3 Assessment of the Methods

GE's approach to modelling containment performance with GE's CETs is a reasonable first attempt at examining the design and trying to identify some of the principal threats to containment integrity. The CETs portray an abbreviated description of how a core melt may be arrested in the reactor vessel or in the containment. Because the MAAP code was used to develop the CETs, the GE CETs reflect assumptions and views of those who developed the code. Some of the assumptions and views have a significant impact on the outcome of the CETs and differ from the views of the staff.

GE's CETs appear to be insufficient for delineating and characterizing accident progressions, assessing the importance of severe accident phenomena, quantifying the probability of accident progression groups, and quantifying source terms. The small CETs used for each PDS differ from the one large CET for all PDSs in the NUREG-1150 study (Reference 19.62). There are advantages and disadvantages to using the smaller CETs, i.e. small CETs are more manageable than large CETs, but interactions of various systems can be overlooked with the smaller trees. The staff believes that small CETs are only partially successful at identifying design weaknesses because important interactions between systems may be overlooked (see also Reference 19.63). But the number of questions composing the CETs is not the only indicator of the adequacy of the CETs. Sufficient delineation of the accident progression also requires a CET analysis having questions that allow for sufficient resolution (i.e. detail definition or description) of severe accident issues. The questions composing GE's CETs are high level questions, lacking much detail that would be necessary to model subtle interactions.

An example of this lack of resolution is at the portion of GE's CETs pertaining to core melt arrest in the reactor vessel. GE's question asks only if this occurs. A more detailed analysis would also factor in the possibility of in-vessel fuel/coolant interaction. In-vessel fuel/coolant interaction can alter the progression of an accident in two ways; it can affect the capability to arrest the core melt; it can have the transient effect of converting a low pressure sequence into a high pressure sequence.

The following important design features of the ABWR may make the construction of the CETs less complicated than CETs usually found in PRAs for existing LWRs:

- o Hydrogen combustion and detonation in the containment need not be modelled because the containment is inerted.
- o Direct attack of core debris on the steel shell of the containment is eliminated by the design.
- o Basemat failure is precluded by having a thick (5.5 meters) basemat. The staff has evaluated the phenomena of concrete erosion due to corium attack and, within the current computational capabilities, determined that the thick basemat would accomodate the expected corium penetration (1.0 meter).

However, the design features of the ABWR do not by themselves justify a simple CET. The purpose of the CET is to ascertain and evaluate subtle interactions among the features.

Some important phenomena relevant to containment response are missing from GE's CETs, mainly because of deliberate omissions based on engineering judgment. Larger CETs would address a more comprehensive set of possible outcomes before judging which outcomes are most important to risk. Examples of potentially important phenomena not addressed in GE's CETs are as follows:

- o GE did not address the effects of in-vessel fuel/coolant interaction on the in-vessel arrest of a core melt.
- o GE considers that a pressure pulse from direct containment heating is unlikely to damage the containment and assigns a low conditional probability of containment failure by this mechanism. The staff considers it an uncertain phenomenon which can potentially threaten the integrity of the containment.
- o GE considers that an ex-vessel steam explosion due to a fuel/coolant interaction sufficient to threaten the containment is

precluded by temperature/pressure conditions and energy transfers due to debris particle size (Section 19E.2.3 of Reference 19.63). The NUREG-1150 study (Reference 19.62) allows for the possibility of rapid steam generation due to a fuel/coolant interaction which potentially challenges containment integrity; the staff believes that this position is applicable to the ABWR.

- GE considers the effects of bypass areas between the drywell and the wetwell due to normal leakage and/or stuck open vacuum breakers to be precluded on the basis of low frequency and risk. The staff believes that containment threats due to drywell/wetwell bypass events need to be more fully addressed.

- GE did not consider that the integrity of the pedestal wall is threatened by an ex-vessel steam explosion. This threat was considered in the NUREG-1150 study (Reference 19.62); the staff believes that this position is applicable to the ABWR.

- GE considers the integrity of the pedestal wall is not threatened once core debris in the lower drywell is covered by water. However, calculations with MELCOR (Reference 19.65) and the STCP (Reference 19.66) indicate that a core/concrete interaction could continue to degrade the pedestal wall, even when the debris is covered by water. This leaves open the possibility that containment penetrations would be damaged, should the loss of the pedestal integrity allow the reactor vessel to tilt. This MAAP/MELCOR comparison is the subject of study in an ongoing code comparison being done by the staff.

- GE believes that the weakest point of the containment is the head of the upper drywell (Section 19.3.2.5 of Reference 19.9). It is thought that the principal failure mode is structural failure of the drywell head due to overpressurization. Another possible failure mode accounted by GE is high temperature degradation of the seals. However, other locations of a structural failure are possible, though perhaps less likely.

Because these phenomena are missing or are minimized in the CETs and there is no analysis of uncertainty within the context of the CETs, the abbreviated GE CET analysis is considered to be incomplete.

6.4 CET Results

6.4.1 Presentation of CET Results from the GE PRA and the Staff Review

Sequences coming out of the CETs were grouped to show the fraction of the conditional probability, given core damage, of accident progression groups giving rise to the expected containment response. Figures 19.6-1 and 19.6-2 show the fractional contribution of the core damage frequency groups for internal events (GE's results are presented with arithmetic errors corrected) and for seismic events, respectively. Both the GE and the staff's estimates of the frequency of the accident progression groups and the associated containment responses are shown. Table 19.6-1 describes the groups in detail. The alphabetical naming of the groups links the table with the figures.

Differences between the staff's and GE's results from the containment event tree analyses arise because of various assumptions made in performing the calculations. These assumptions include the following:

- o The staff took credit in both the Level 1 and the Level 2 analysis for a capability to inject firewater into the reactor vessel to prevent core damage whereas GE took credit only in the Level 2 analysis (discussed later in this section).

- o The staff attempted to account for uncertainty in phenomena potentially threatening to containment integrity whereas GE did not account for such uncertainty (discussed in Section 19.6.3).

- o GE's PRA was based on the design up through Amendment 8 of GE's SSAR (Reference 19.9), i.e. an ultimate containment strength of 100 psig and limestone concrete forming the lower drywell, while the staff's analysis included the design changes mentioned in a letter from P. W. Mariott, General Electric Corporation, to C. L. Miller, NRC, dated August 9, 1990, regarding response to NRC/GE May 16 - 17, 1990 Meeting Discussing Topics (Reference 19.69), i.e ultimate containment strength of 134 psig and basaltic concrete forming the lower drywell.

- o For seismic events the major analytical difference is the seismic hazard curves (annual probability of exceeding a specified peak ground acceleration). GE used a seismic hazard curve developed as the bounding curve of a few selected plant sites. The staff's analysis was based on three sites having the highest seismic hazard in the eastern and central U.S. (Pilgrim, Seabrook, and Watts Bar), and used two seismic hazard curves, one developed by Lawrence Livermore National Laboratory (LLNL) and the other developed by the Electric Power Research institute (EPRI). The LLNL hazard curves generally provide a much higher core damage frequency than either the GE or the EPRI curves, while the later two curves give rise to core damage frequencies of about the same magnitude. However, the uncertainty ranges of the LLNL curve are large and encompass the other two curves.

In Figure 19.6-2, the frequency of accident progression groups of the staff's analysis is the result of using the Pilgrim site with its LLNL hazard curve, selected because this gives the highest core damage frequency of the staff's six seismic analyses.

The following points can be made from Figures 19.6-1 and 19.6-2:

- Groups (A), (B), and (H) represent the sequences which do not result in containment failure or vent actuation. In these sequences, the core melt is arrested either in the reactor vessel (Group (B)) or in the lower drywell (Groups (A) and (H)) since the containment cooling function (RHR system) is recovered before the containment pressure reaches the vent actuation pressure. Group (A), which appears in the staff's analysis, is essentially similar to Group (H), which only appears in the GE analysis, the difference arising from assumptions about drywell/wetwell bypass (Section 19.6.4.2.1) and pedestal integrity (Sections 19.6.4.2.3 and 19.6.4.2.4).

- Groups (C) and (G) are similar to Groups (A), (B), and (H) in that there is steaming, either in the reactor vessel or the lower drywell; the difference is that in the former, the containment cooling function of the RHR system is unavailable leading to a rise in containment pressure resulting in vent actuation or containment failure.

- Groups (D), (E), and (F) represent early containment failures. Group (F) appears only in the staff's analysis because it represents direct containment heating. GE does not consider direct containment heating to be a credible phenomenon (see Section 19.6.3).

In the staff's analysis, a Group (C) situation does not arise because of assumptions regarding the availability of various equipment. The staff credited the firewater addition system for preventing core damage in Level 1. If core damage occurs (Level 2), then the firewater system could not have been available to <u>prevent</u> core damage, hence, it unlikely to be available to <u>arrest</u> a core melt in the reactor vessel. However, later in the accident progression, the staff took credit for arresting an ex-vessel core melt progression because there is more time available to restore the firewater system than for the in-vessel situation. In contrast, GE calculations did not take credit in the Level 1 portion of the PRA to prevent core damage using the firewater system, but they take credit in the Level 2 portion to arrest core damage in the reactor vessel using the firewater system.

The treatment of ATWS sequences (Class IV) in GE's internal events analysis differs from the treatment in GE's seismic events analysis. GE considered the Class IV sequence to result in an early containment failure in the internal events analysis and as late containment failure in the seismic events analysis. The staff's treatment of Class IV sequences for internal events is similar to GE's seismic events treatment of these sequences. As above, this accounting has an effect on the results of the CET analysis.

To begin, the staff estimated the frequency of the Class IV accidents to be higher than GE's estimate. The larger frequency warranted further study of the progression of these accidents.

The staff reasoned that the Class IV accidents are primarily caused by a failure of reactivity control due to failure of adding boron, or flushing or diluting boron subsequent to scram failure. This implies that the lower plenum of the reactor vessel is filled with water. As the core melt progresses, the heat generation is eventually reduced to the decay heat level, and since the lower plenum is filled with water, the core debris is covered with water, allowing for the potential to arrest the core melt progression in the reactor vessel. Such an accident sequence would be grouped in Group B of Table 19.6-1 and Figures 19.6-1 and 19.6-2.

GE treatment was apparently done because the frequency of the Class IV accidents is small, less than 2 percent in GE's internal events analysis. Thus, they were conservatively grouped with other sequences which resulted in the largest release of fission products (early containment failure). However, since its frequency in the seismic events was significantly higher, 60 percent, GE performed a more thorough analysis, where most of the sequences resulted in late containment failure.

According to GE, water in the lower plenum of the reactor vessel does not necessarily quench core debris, hence, the core debris can eventually fail the bottom head penetrations. Once the vessel fails, the water in the lower plenum flows into the lower drywell along with water from actuation of the passive flooder system. The core debris steams in the lower drywell. Since the availability of the RHR system in these sequences is high, most of these accidents would not result in containment failure. These accidents are classified as Group (A) of Table 19.6-1 and Figures 19.6-1 and 19.6-2.

added
Not true

In the presentation of GE's results for internal and external events and the staff's PRA results for internal events, the effect of controlled venting on the conditional containment failure probability (CCFP) is not specifically addressed. Until this change is made, the staff's results (for external events only, since this is the only place the OPS was credited) will show containment failure whenever the overpressure protection system is actuated within the first twenty four hours.

Given the assumptions noted in the Group (A) description of Figure 19.6-1 and the above definition of containment failure, the point estimate of the total conditional containment failure probability (CCFP) for internal events is 16 percent in the staff's analysis and 12 percent in the GE analysis (i.e. Groups (C), (D), (E), and (G) in Figure 19.6-1); the difference is mainly due to the consideration of direct containment heating in the former analysis. The conditional containment failure probability for early failures in the internal events analysis is predicted to be 13 percent by the staff and 3 percent by GE (Groups (D), (E), and (F) in figure 19.6-1). For external events, the point estimate of the total conditional containment failure probability is 78 percent in the staff's analysis and 82 percent in GE's analysis. The higher conditional failure probabilities in the seismic events analysis relative to the internal events analysis are largely due to the assumption in both the GE and staff analyses that the loss of power is non-recoverable after a seismic event. Note that the GE values reported above incorporate corrections to a number of arithmetic and logic errors identified through the staff's review. Hence, these values differ from the (uncorrected) GE point estimates presented in Table 19.9-1.

added

Table 19.6-1 Description of Sequence Groups in Figures 19.6-1 and 19.6-2.

Accident Progression Group in Figure 19.6-1 and Figure 19.6-2	Description of Accident Progression Groups
A	Sequences result in an uncertain response to the containment in the staff's analysis when the passive flooder system introduces water to the core debris. Although the RHR system is operable to remove heat delivered to the suppression pool as a result of the steaming from of the core debris, containment response is uncertain as a result of two factors. First, drywell/wetwell bypass flow may circumvent the suppression pool. Unless the wetwell sprays are manually aligned and available, the containment will pressurize at a rate that depends on the extent of the bypass flow (Figure 19.6-4), and vent actuation could occur. Also contributing to the uncertainty in the containment response is the uncer' ainty in the integrity of the pedestal wall due to core/concrete interaction, even given the operation of the passive flooder system. Should the wall fail, the reactor vessel would tilt and could damage penetrations. The damaged penetrations constitute a containment failure.
B	Sequences result in arrest of a core melt in the vessel by the recovery of some form of in-vessel cooling. The RHR system is operable to remove decay heat from the suppression pool. The containment does not pressurize because of an operable RHR system. Hence, the containment remains intact.
C	Sequences result in arrest of a core melt in the vessel by the recovery of some form of core cooling. Steaming from the core debris goes through the SRVs and the T-quenchers into the suppression pool. Because of a failure to recover the RHR system, the suppression pool heats, allowing the containment to pressurize. In a vented containment, the overpressure protection system would actuate. In this group of sequences, the overpressure protection system is of no consequence because the suppression pool is along the release path, whether or not the system is present.
D	Sequences result in core debris being ejected out of the reactor vessel at high pressure. Some of the debris settles in the upper drywell and heats the penetration seals and pressurizes the containment. When the temperature reaches about 500 degrees F and the pressure is 52 psig, the seals are assumed to fail. The leakage is sufficient enough to constitute a containment failure but may be insufficient.

Table 19.6-1 (continued)

Accident Progression Group in Figure 19.6-1 and Figure 19.6-2	Description of Accident Progression Groups
	enough to relieve the pressurization. Should the containment continue to pressurize, the overpressure protection system would actuate
E	Sequences result in core cooling being maintained, but without the RHR system, the containment pressurizes. A failure to vent is assumed leading to containment failure. The debris from the damaged containment disables the systems maintaining core cooling. With the loss of cooling, core damage results. Because the containment fails prior to core damage and vessel failure, the containment failure is considered an early failure.
F	Sequences result in a rapid pressurization of the containment. The pressurization comes from the blowdown of the reactor vessel, direct heating of the containment atmosphere (decay heat and exothermic chemical reactions), and fuel/coolant interaction. In the staff's review, as in the NUREG-1150 study (Reference 19.62), a fuel/coolant interaction is not necessarily a shock wave; it may be rapid pressurization. The pressure rise in the drywell is too rapid for the drywell/wetwell connecting vents (downcomers) to clear. A structural failure of the containment in the drywell results.
G	Sequences result in arrest of a core melt in the containment when the passive flooder system introduces water to quench the core debris. Unlike in the Group (A) or Group (C) sequence, the RHR system is inoperable. Hence, the steaming from the core debris heats the suppression pool and pressurizes the containment. Eventually, the overpressure protection system actuates. NOTE: For these sequences, early action of the overpressure protection system depends on the extent of bypass flow.
H	Sequences result in arrest of a core melt in the containment when the passive flooder system introduces water to quench the core debris. The RHR system is recovered to remove heat delivered to the suppression pool as a result of the steaming from the core debris. The containment does not pressurize, hence, it remains intact. NOTE: For these sequences, the overpressure protection system could be actuated early, depending on the extent of bypass flow.

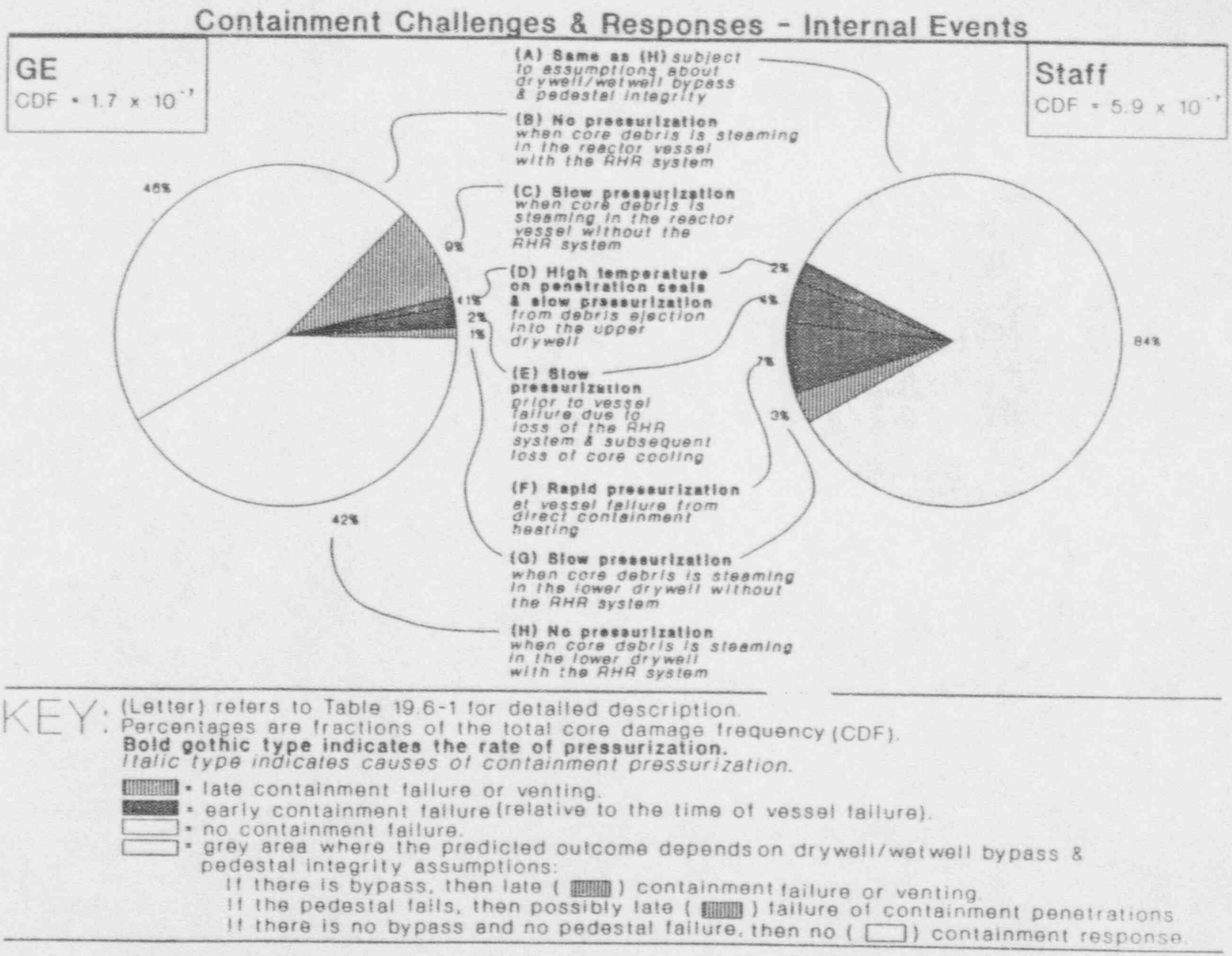

Figure 19.6-1 GE's and the staff's breakdown of the conditional probability of accident progression groups given core damage for internal events.

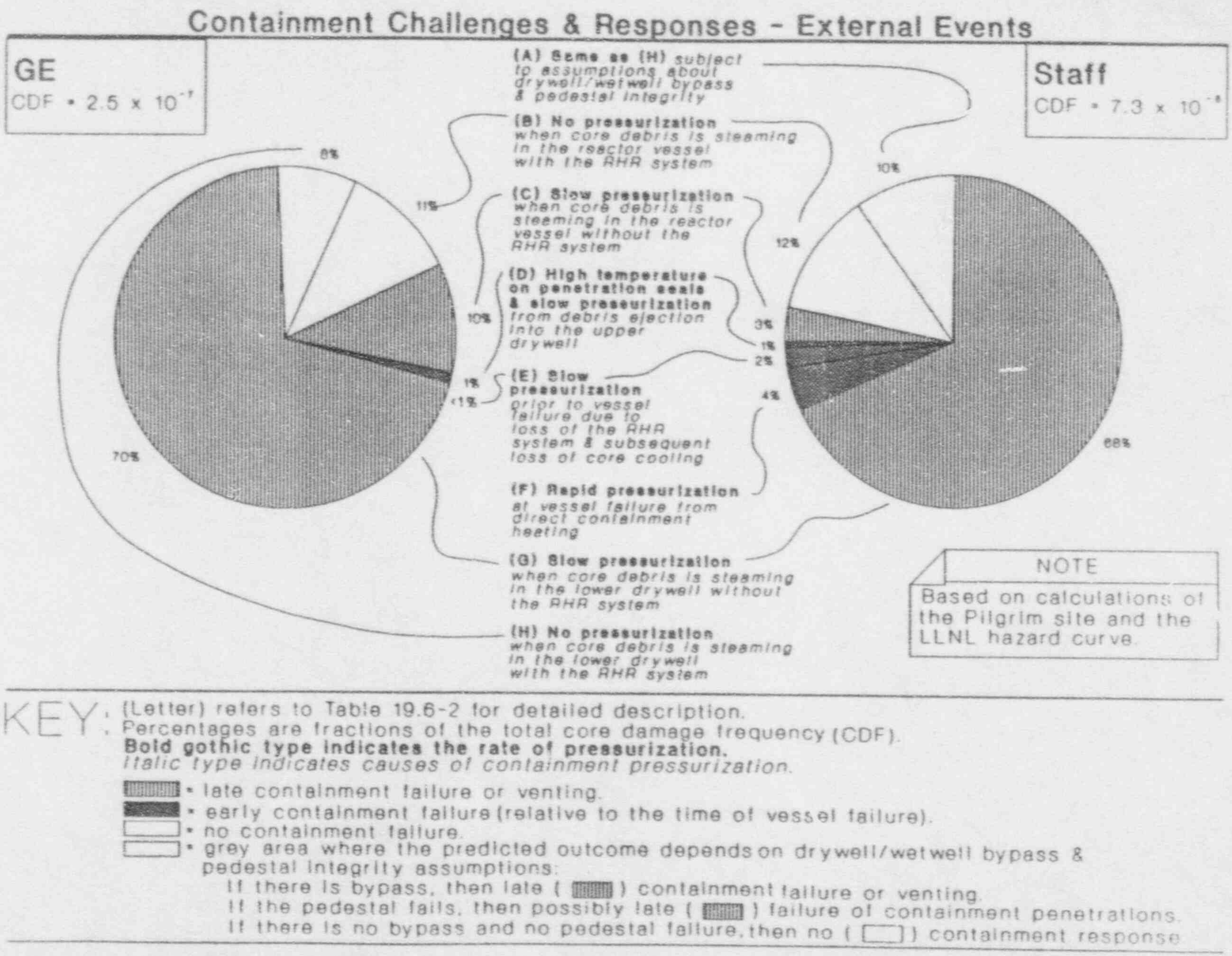

Figure 19.6-2 GE's and the staff's breakdown of the conditional probability of accident progression groups given core damage for seismic events.

6.4.2 Discussion of the CET Results

In light of the low core damage frequencies coming into the CET, e.g. 10^{-7}/year (internal events), there are two positions that can be taken with respect to reviewing the CETs and the subsequent plant risk. The first position is that the very low frequencies are believable, which removes any concern over containment performance and plant risk. For example, the containment could be construed as of little benefit since the two NRC Quantitative Health Objectives, individual early fatality risk of $\leq 5x10^{-7}$/yr and individual latent cancer fatality risk of $\leq 2x10^{-6}$/yr are met, even without the containment. The second position is that even with such low frequencies, the CETs take on importance in the context of balancing prevention and mitigation as well as maintaining defense-in-depth. The balance of prevention and mitigation is achieved in part through the NRC's conditional containment failure probability goal of 0.10. It is in regard to this latter position that the staff pursued its CET evaluation and as such have identified aspects of the ABWR design that have a major influence on the CET results. Each of these factors is discussed in turn.

6.4.2.1 Drywell/Wetwell Bypass

A certain amount of drywell-to-wetwell leakage is allowed for in BWR suppression containment as stated in NRC's Standard Review Plan. In its deterministic analysis, GE addressed certain aspects of this drywell/wetwell bypass (page 19E.2-28 of GE's SSAR (Reference 19.9)). GE's results supported no further consideration of suppression pool bypass flow effects in the CETs based on low estimated frequencies and risk.

Allowed bypass areas ($A/K^{1/2}$) in a plant's technical specifications have historically been set at 0.10 of the $A/K^{1/2}$ assumed in the containment DBA. Since a large $A/K^{1/2}$ provides for a robust containment pressure design but an increased plant risk due to suppression pool bypass, the need to understand all of the implications of a range of possible $A/K^{1/2}$ values were considered in the staff evaluation.

In GE's SSAR (Reference 19.9), potential bypass paths between the drywell and the wetwell are identified and discussed. Included among those were the eight 20-inch diameter vacuum breakers (Table 19E.2-1 of Reference 19.9) designed to prevent a negative drywell pressure (relative to the wetwell) from occurring following a design basis reactor coolant pipe rupture. As noted above, GE did not include any consideration of bypass effects in its CETs; however, since there is an allowance for such leakage bypass flow, the staff thought it prudent to investigate this potential problem affecting containment integrity. This investigation was motivated in part by operating experience at an existing BWR with a Mark III containment that has shown bypass flows ranging from 800 to 2500 cfm, which are considerably greater than the design value for the ABWR. Calculations were performed (Figure 19.6-4) to investigate the effect of increased ABWR bypass leakage on risk. Results of these calculations indicate that this matter warrants further investigation.

Although an assessed high bypass flow increases the robustness of the containment vis-a-vis the design basis calculation, it appears to have the following negative effects on risk:

1. Potentially increases the number of sequences where the overpressure protection system is actuated by reducing the time available for RHR recovery.

2. Potentially increases the amount of radioactive release to the environment because the bypass flow would not be scrubbed by the suppression pool.

3. Reduces the time available for fission product decay, aerosol settling, and evacuation.

4. Places additional reliance on operator action to successfully initiate containment spray. Currently, these sprays are manually operated (Section 19D.6.3.3 of Reference 19.5).

5. Places additional reliance on the firewater addition system and on the capability to connect a fire truck to the system when the RHR system is unavailable,

   Figure 19.6-3 is a schematic diagram showing features of the ABWR containment relevant to drywell/wetwell bypass. Figure 19.6-4 illustrates the bypass issue by showing the containment pressure as a function of time for various bypass flow rates in the ABWR. The figure was generated with equations that yield only a first approximation to the expected pressures and suggests that bypass flow rates must not be allowed to exceed those values that could compromise the integrity of the containment system, i.e., early vent actuation followed by frequent manual opening thereafter. The relationship between flow rate and $A/K^{1/2}$ shown in Figure 19.6-4 was obtained from data and analysis performed in the NUREG-1150 study of the Grand Gulf plant (Reference 19.62). Figure 19.6-4 is based on

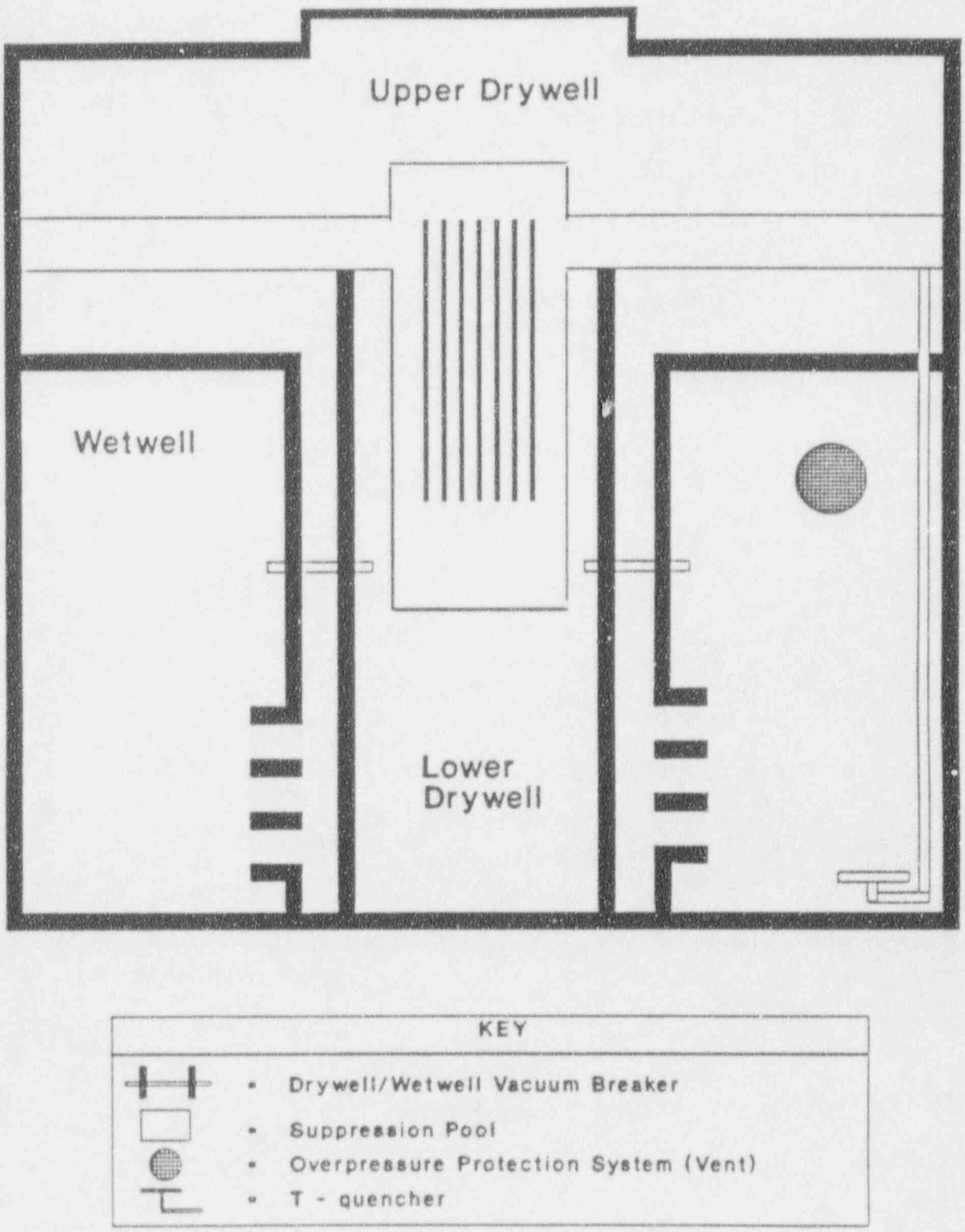

Figure 19.6-3 Schematic diagram of the ABWR containment

simple hand calculations using the ideal gas equation and the volumes of the drywell and the wetwell of the ABWR.

Missing points to be gained from Figure 6-4

added then but these seem like our words

The staff has concluded that GE should provide a comprehensive assessment of the risks associated with drywell-to-wetwell bypass leakage. Such an assessment should include complete consideration to such matters as (1) the basis to support an allowed leakage area ($A/K^{1/2}$), and (2) the basis to support an expected leakage area during the course of a severe accident when the vacuum breakers would be required to perform several times in a severe environment. This is an outstanding item.

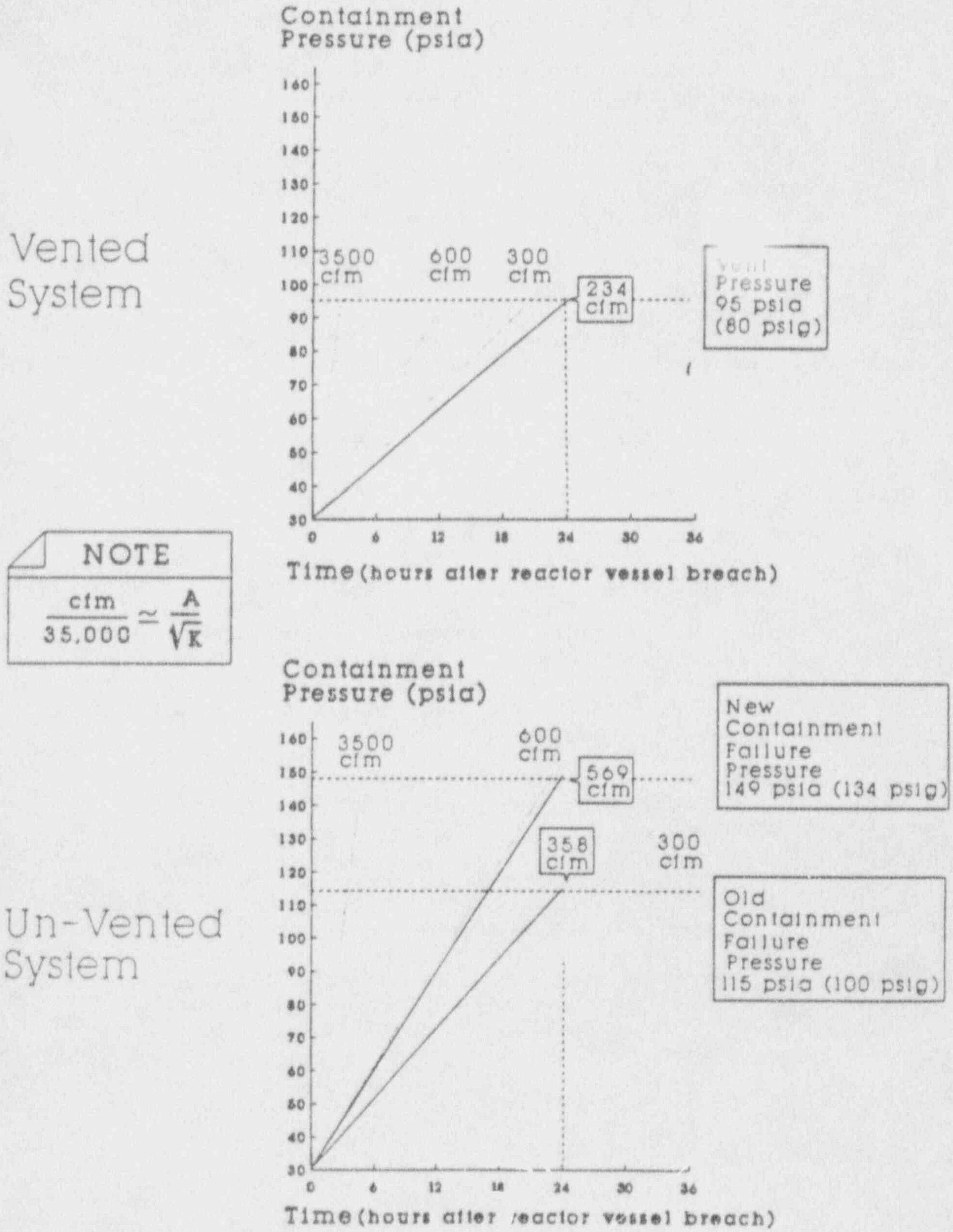

Figure 19.6-4 Wetwell pressure as a function of bypass flow (CFM) and time (hours).

6.4.2.2 Overpressure Protection System (OPS)

added

A containment overpressure protection system was not included in the original ABWR design, but was added to the design in Amendment 8. GE modified their containment event trees to include the OPS, but in baseline calculations assigned a failure probability of 1.0 for the system, thereby taking no credit (or penalty) for venting. The bulk of the staff's analysis reported herein were already completed by the time of the OPS modification and do not account for the effect of the OPS on containment performance and risk. Although the effect of the venting was addressed in a number of additional staff calculations, as discussed below, no attempt was made to revise the earlier analyses to address the effects of the venting system.

The proposed overpressure protection system (OPS) for the ABWR represents a significant departure from previous BWR design submittals in that the design automatically activates at a containment pressure above the design value setpoint (currently 80 psig) by venting the wetwell to the environment. On the basis of GE's analysis, this containment pressure would be reached only in the event that the RHR system was unavailable to remove decay heat from the suppression pool. The system is intended to provide protection against rare sequences where containment integrity is challenged by overpressurization. The staff's review supports this view, however, the staff has the following observations that warrant consideration before accepting such a system:

1. The system would be effective only in a small percentage of internally initiated severe accident sequences.

2. Pressures could develop to actuate the system as a result of drywell/wetwell bypass flows in the absence of the above mentioned spray operation.

3. Uncertainty considerations complicate the prediction of how this system would operate, i.e., determining the proper setpoint for vent actuation, in view of adverse effects.

Each of these concerns is discussed below:

(1) Frequency of Actuation

Table 19.6-2 shows a Level 2 perspective of the relative importance of the OPS. The table was derived from Figures

19.6-1 and 19.6-2 and shows the frequency of the accident progression groups in terms of the operation of the OPS. The table is a rough approximation, with the other considerations such as plant specific features, drywell/wetwell bypass, and uncertainty aside. Several points are emphasized:

Internal events. Because of the high reliability of the RHR system, the OPS would actuate in only 3 percent of the accident progression groups sequences. According to staff analysis, in about 1 percent of the accident progression sequences, the system actuates unnecessarily as a result of pressure challenges that are sufficient to actuate the vent but insufficient to fail the containment had the system not been present.

In 13 percent of the accident progression group frequency, the drywell is predicted to fail due to rapid overpressure. The OPS is not expected to have a significant effect for these scenarios.

Seismic events. The OPS is actuated in 71 percent of the accident progression group frequency, largely due to unavailability of the RHR system.

Overall, from a risk perspective, GE's analysis does not appear to make a strong case for needing an automatic OPS, at least for internal events. The overall intent is to provide a "last-ditch" mitigation effort for rare accidents. For seismic events the relative importance of this system appears greater, due to the unavailability of the RHR system.

In the staff's view, GE should provide a comprehensive determination of the positive and negative risks associated with the operation of the OPS. Such a determination would involve a thorough consideration of plant specific design features and a coupling of both the Level 1 and the Level 2 analyses, first with the overpressure protection system and then without the system. The determination should answer the following questions:

1. How does the overpressure protection system reduce the core damage frequency, and eventually, the containment failure frequency?

2. Given core damage, how are the source terms affected?

This is an outstanding item.

Table 19.6-2 The Staff's Venting Outcome Frequencies for Internal and Seismic Events.

Venting[1.] Outcome	Staff's CDF %[2.] Int.	Seis.	Conditions	Consequences
Successful venting when venting is necessary	3%	71%	The pressurization rate is slow and without the RHR system and containment sprays.	Routes releases from the drywell through the suppression pool.
Successful venting when venting is unnecessary	1%	1%	Containment pressure becomes high enough to open the vent but would not have been high enough to fail the containment if the vent had not opened because of RHR system recovery before containment failure.	Unnecessary release through the suppression pool.
Potentially no impact of venting	13%	7%	The containment (drywell) pressurizes rapidly due to energetic event such as DCH, and approaches or exceeds its ultimate pressure capacity. Vent actuation may or may not occur depending on rate of drywell pressurization and pool dynamics. The vent is not expected to be effective in these scenarios.	Unscrubbed release through the drywell failure, assuming that the vent is ineffective in preventing overpressure.
Insignificant pressure challenge to the containment	83%	21%	Operable RHR system.	None.

1. Values presented assume no significant wetwell/drywell bypass flow. Significant bypass, if it were to occur, would result in an increased frequency of venting (rows 1 and 2 in Table) and a corresponding decrease in no-vent scenarios (row 4 in Table).
2. Int. = internal events. Seis. = seismic events.

(2) Drywell/Wetwell Bypass

A few sentences deleted regarding the importance of the OPS.

The effectiveness of the vent can be severely compromised unless drywell/wetwell bypass is controlled. Because the setpoint of the vent actuation is lower than the ultimate strength of the containment (Figure 19.6-4), the overpressure protection system may merely open the containment earlier than had it not been present. For more discussion about drywell/wetwell bypass, see Section 19.6.4.2.1.

(3) Uncertainty

GE's analysis in the vent system actuation pressure does not adequately consider uncertainties. That is, if the vent setpoint pressure is set at a particular value, GE claims that the OPS will operate only at that setpoint value. GE states (Pages 61-62 of Reference 19.70) that the vent actuation pressure setpoint is based on the pressurization from a DBA LOCA and 100 percent metal/water reaction as specified in 10 CFR, Part 50.34(f). The resulting DBA pressure of 75 psig combined with a tolerance of $\pm$ 3 to 4 psi in the vent actuation pressure, therefore, appears to be the basis for the current rupture disk setpoint of 80 psig; no consideration of severe accidents or uncertainty is evident. The model of the system in the CET is also simplistic, consisting of a branch point where there is either complete success (100 percent reliability) or complete failure of the vent actuation to occur. GE does not take credit for any operation of the overpressure protection system in its sequence development, thereby claiming that its PRA is more conservative with this treatment of the overpressure protection system.

The staff considered uncertainties in its review of the overpressure protection system. The staff finds that considering uncertainty in the actual failure pressure of the containment makes a model of the overpressure protection system complicated. Figure 19.6-5 shows a more comprehensive view of the overpressure protection system, illustrating inherent uncertainties, for a slow pressurization of the containment. The overpressure protection system is represented by a narrow distribution to reflect the small uncertainty in the vent actuation. The staff infers a small uncertainty because the overpressure protection system is presumably designed within the realm of well established and tested engineering practices, allowing its response to be reasonably and accurately characterized.

The containment structural response is represented by a wide distribution to reflect the large uncertainty in its response above the design pressure. Intuitively, the failure probability immediately above the design pressure is low and increases as some function of the pressure. This distribution is assumed to peak at some ultimate containment failure pressure and decrease to zero according to the best understanding of that particular containment structure.

Setting the overpressure protection system above the containment design pressure allows for a possibility of containment structural failure. The lower tail of the containment failure distribution can be superimposed on the distribution of the vent actuation. This allows for the possibility, although remote, of a containment failure rather than vent operation. Competing risks develop, where either a vent actuation or a containment failure may occur. If the containment is weaker than the setpoint of the vent, then the containment will fail without the vent actuating. Conversely, if the containment is stronger than the vent setpoint, then the vent will actuate.

add: misleading. Not a summary

(4) Summary

The staff has qualitatively identified the following OPS associated risks that are not modelled by GE in its PRA:

1. There is a possibility of unnecessary venting, where the vent may actuate in response to a pressure challenge that the containment could have accommodated had the vent not been present. However, for such events the releases should be small (given that suppression pool bypass is not an issue) and the design of the OPS would allow the operator to manually isolate the vent.

2. In the event there is significant bypass between the drywell and the wetwell, the following can result:

   (a) a potentially significant increase in the frequency of actuating the overpressure protection system, and

   (b) reduced time to overpressure protection system actuation and fission product decay and aerosol settling.

Deleted pt.

In its analysis, the staff has identified the trade-offs assuming an all-or-none behavior of containment structural failure and an initial attempt to quantify the effects of the bypass on the efficacy of the overpressure protection system. Based on the review, the staff believes that GE should justify the setpoint of the overpressure protection system, taking into account downside risks, and carry out the necessary analysis of the effects of drywell/wetwell bypass on risk before conclusions can be reached that the system has a net benefit from a risk perspective.

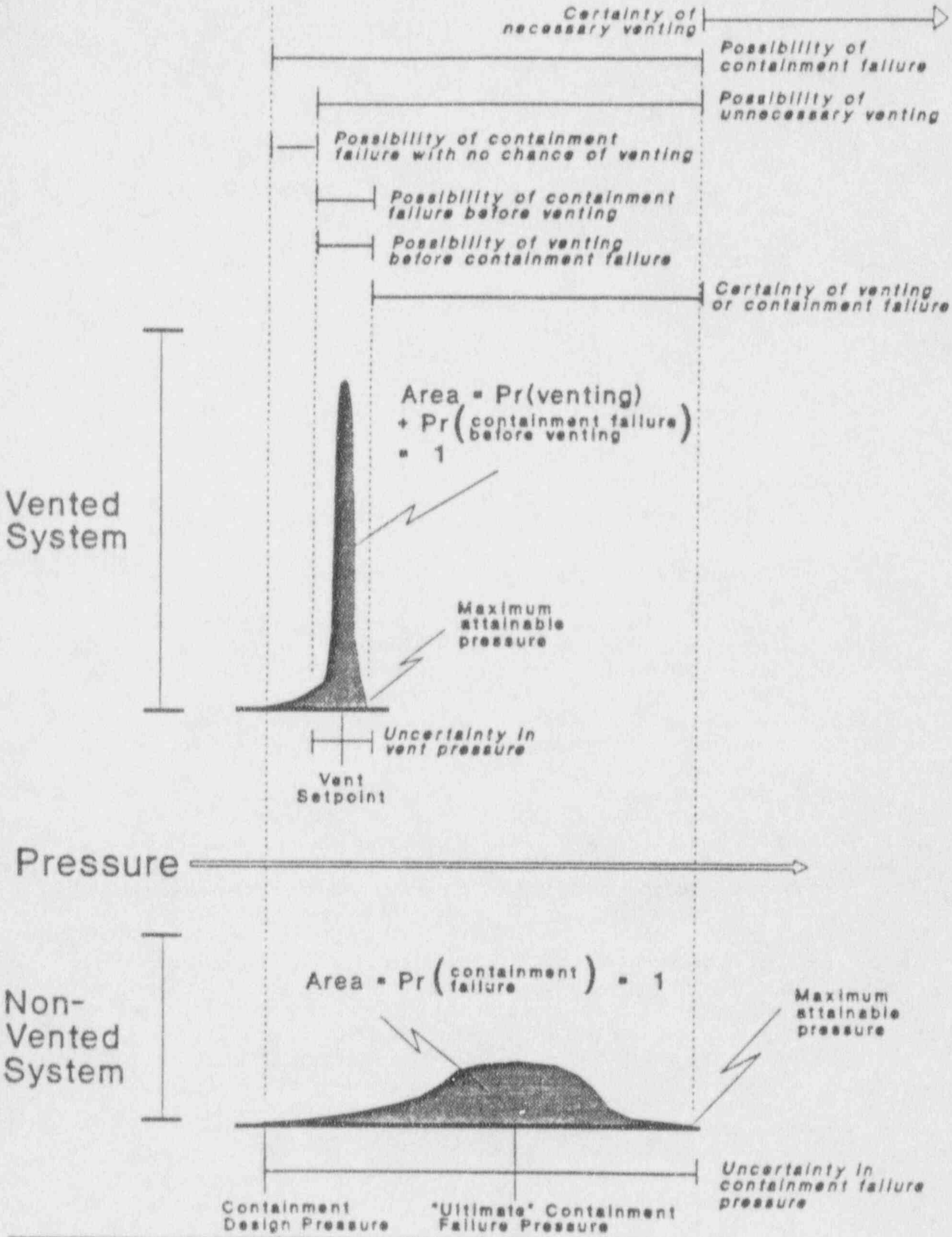

Figure 19.6-5 Hypothetical distributions illustrating the uncertainty in the vent setpoint and the containment failure pressure.

6.4.2.3 Passive Flooder System

The purpose of the passive flooder system is to introduce water into the lower drywell to quench the molten debris and produce a safe and stable state by forming a coolable debris bed. If this debris bed configuration can be achieved, the core debris temperature would be such that the concrete floor and walls would not be affected because nearly all of the decay heat would be used to boil the water. In such a configuration, several challenges to the containment integrity would be eliminated:

- o High temperature gases from extensive core/concrete interaction, which can cause a overpressure and overtemperature failure of the containment.
- o Mechanical and thermal degradation of the pedestal by the core debris, which could allow the vessel to tilt and damage containment penetrations.
- o Mechanical and thermal degradation of the basemat by the core debris (although for the ABWR design, this did not appear to be a serious threat due to the thickness of the basemat - 5 meters).

In earlier amendments of the GE SSAR, the passive flooder system was not a part of the lower drywell design. The ABWR design allowed the core debris to erode the pedestal and, in GE's view, the erosion would be stopped when the core/concrete interaction reached the drywell/wetwell connectors imbedded in the pedestal wall; at that point, the water from the suppression pool would enter the lower drywell and quench the core debris. The passive flooder system was subsequently added to the lower drywell design in Amendment 8 to the GE SSAR (Reference 19.9). The system was designed to rapidly deliver water to the molten debris to form a coolable debris bed and keep the pedestal degradation to a minimum.

GE's scenario for the passive flooder system operation assumes that upon vessel breach, the core debris enters the lower drywell and is uniformly distributed on the lower drywell floor. The decay heat and heat from oxidation reactions melts the fusible plug actuating the passive flooder system. In about 15 minutes, the ten 4-inch diameter flooder ports fill the lower drywell with water from the suppression pool. Minimal core/concrete interaction as predicted by the MAAP code will occur due to rapid quenching of the molten core debris. An ex-vessel fuel/coolant interaction due to a rapid energy transfer from the debris entering water in the lower drywell is precluded by pressure/temperature regimes and the debris particle size (Section 19E.2.3.1 of Reference 19.9). Steaming from the core debris passes to the upper drywell, through the drywell/wetwell connectors and into the suppression pool. The heat delivered to the suppression pool is rejected to the environment through the RHR system. Should the RHR system be unavailable, the suppression pool would become saturated, upon which time the containment would pressurize and eventually actuate the overpressure protection system.

Focusing for the moment on GE's view of the passive flooder operation, where core debris enter the lower drywell and the passive flooder system actuates to cool the debris, the following statements can be made:

1. The core/concrete interaction would be minimal and prevent pedestal failure, thus, avoiding the risks of containment failure induced by the reactor vessel tilting and damaging containment penetrations.

2. Radioactive release from the core debris in the lower drywell have to pass through and be scrubbed by the flooder water overlying the corium and the suppression pool. NOTE: Radioactive releases from core debris ejected out of the lower drywell (i.e. direct containment heating) and releases from debris remaining in the reactor vessel would not be scrubbed by the water from the passive flooder system.

3. Ambient temperatures in the drywell are significantly reduced, thus protecting the operability of equipment and structures.

4. Reliance is placed on the RHR system to reject heat to the environment as a consequence of the steaming from the core debris passing into the suppression pool. GE believes that the three-train RHR system adequately addresses this concern.

5. Reliance is placed on other means of pressure suppression when the RHR system is unavailable. GE relies on its manually operated firewater system to address this concern.

However, the staff's view of the passive flooder system operation differs in some respects from GE's view:

1. MELCOR calculations and other calculations discussed in letter report titled, "Effects of Debris Depth, Debris Composition, and Debris Power on the Limits of Coolability," from E. R. Copus, Sandia National

Laboratories, to C. Tinkler, U.S. Nuclear Regulatory Commission, March 13, 1990 (Reference 19.71), indicate significantly more concrete degradation of the pedestal wall than that predicted by the MAAP code, even after operation of the passive flooder. Continued core concrete interactions would increase the potential for pedestal failure would undermine the system's objective for protecting the lower drywell floor and walls.

2. MELCOR calculations done in support of the staff's review suggest a possibility of the passive flooder system actuating before significant amounts of the core debris enter the lower drywell. This leads to a possibility of a subsequent fuel/coolant interaction and rapid pressurization, potentially causing an early containment failure. The pre-existing water pool, conversely, could increase the likelihood that debris would be quenched upon entry into the pool and form a coolable debris bed. (References 19.62, 19.72, and 19.73).

3. Experiments discussed in a paper by B. W. Berman, et al., "Recent Intermediate Scale Experiments on Fuel-Coolant Interactions in an Open Geometry (EXO-FITS)," dated February 1986 (Reference 19.74), suggest that the possibility of a fuel/coolant interaction and rapid pressurization also exists when water is poured onto core debris in the lower drywell, as during the intended operation of the passive flooder system.

Deleted point about water on debris or vice versa causes SE.

It should be noted that an effective flooder system is necessary to maintain safe temperatures within the lower drywell above the core debris following a severe accident. In this regard, the staff believes that GE should further examine this aspect of the design. Such an examination should consider whether to introduce water into the lower drywell in a controlled or uncontrolled manner, how fast to introduce the water and when to introduce the water. This is an outstanding item.

6.4.2.4 Lower Drywell Composition

The erosion of the pedestal wall by a core/concrete interaction may threaten the integrity of the reactor pedestal. Amendment 8 of GE's SSAR (Reference 19.9) indicated that the lower drywell would be composed of limestone concrete. However, recent information from GE (Reference 19.69) indicates that basaltic concrete will be used. The staff's early analyses showed that the erosion of limestone concrete is more extensive than the erosion of basaltic concrete;

much of the erosion products of limestone concrete are non-condensible gases which would be driven out of the melt and contribute to pressurizing the containment. However, the erosion products of basaltic concrete appear to dilute the melt, without posing an overpressure threat to containment integrity. GE's analysis shows neither type of concrete to be a problem in core/concrete interaction due to quenching provided by the passive flooder. The staff considers the use of basaltic concrete to be preferable to limestone concrete.

Important pt. deleted about the wall not being able to withstand debris. meaning is changed.

6.4.2.5 Containment Structural Integrity

GE did not consider direct containment heating and fuel/coolant interaction as credible phenomena which could lead to the failure of the ABWR containment. Given this view, GE attributes only 0.1 percent of the accident progression frequency to this failure mechanism. Nevertheless, GE chose to increase the ultimate strength of the containment from 100 psig to 134 psig (Reference 19.69).

The staff's view of direct containment heating and fuel/coolant interaction differs from that of GE. The staff believes these rapid pressure pulse phenomena can occur under certain circumstances and may possibly be of sufficient magnitude to threaten the integrity of the containment. The staff's view is based on the Grand Gulf analysis in the NUREG-1150 study (Reference 19.62). Though the ABWR differs from Grand Gulf, the Grand Gulf analysis of the drywell pressurization appears to us to be applicable as a first approximation. In the Grand Gulf analysis, the pressure loading in the drywell from high pressure melt ejection was characterized with a distribution having much uncertainty, being as high as 300 psi, with a median value of about 80 psi. Because of similar drywell volumes, the staff incorporated this distribution into the ABWR risk analysis, assuming that a pressure pulse in the drywell would not be rapidly transmitted to the wetwell.

Figure 19.6-1 (see Internal Events as analyzed by the staff) shows that containment failure from rapid pressure pulses due to direct containment heating accounts for 7 percent of the accident progression frequency; this assumes the modified ABWR having an ultimate containment strength of 134 psig. In the unmodified design having the ultimate strength at 100 psig, the containment failed in sequences contributing about 13 percent of the accident progression frequency. Therefore, this modification of the containment reduced the conditional early containment failure probability by approximately half (from 13 to 7 percent). Incremental increases in strengthening the containment would further decrease this conditional containment failure probability; however, the cost of redesigning the containment may be significant. Other fractions of the accident progression frequency are not appreciably affected by the increased containment strength because either the vent actuates (e.g. the vent setpoint was not increased) or the failure mechanism is by another means (i.e., thermal degradation of moveable penetration seals).

In the seismic events analysis, Figure 19.6-2 shows much less improvement in risk because of the modification. Here, the primary reason for the failure of the containment is the loss of the RHR system. This causes slow pressurization, which led to vent actuation.

The staff believes that the strengthened containment reduces the conditional probability of early containment failure from direct containment heating and fuel/coolant interaction phenomena.

6.5 Conclusions

1. The abbreviated containment event trees (CETs) developed by GE for the ABWR were of concern in the staff's review. GE's contention was that since many of the containment issues identified in previous PRAs could be eliminated from consideration because of specific ABWR design features and an increased knowledge base, the size of GE's CETs adequately portrayed the potential ABWR accident progression. The staff had the additional concern that GE's analysis of the accident progression events in the CETs relied on MAAP code predictions without any uncertainty analysis, when the staff's experiences with the NUREG-1150 study (Reference 19.62) indicate that CETs should reflect uncertainty ranges which adequately reflect the current level of understanding of severe accident progressions. The staff has concluded, therefore, that while some of the CET simplification is justified by specific ABWR design features, such abbreviated trees combined with the sole use of MAAP and the lack of any uncertainty analysis, do not lend confidence that a thorough identification of the issues important to containment performance has been made. Specifically, with regard to important issues such as the identification of the complete spectrum of challenges to containment integrity and the effectiveness of the mitigation systems in contributing to ABWR defense-in-depth capability vis-a-vis the CCFP goal of 0.10, the following differences with GE's analysis were found:

   (a) The staff considered challenges to the drywell integrity from accident progressions involving rapid pressurization from direct containment heating and rapid steam generation. These phenomena, which received considerable attention in the NUREG-1150 study (Reference 19.62), were found to impact the ABWR design's capacity to meet a CCFP of ≤0.10 in this analysis. GE treated these issues to be negligible probability events without sufficient analytical support. The lack of an uncertainty analysis by GE further precluded their identification as potentially important containment challenges.

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(b) The staff was unable to dismiss the potential for molten corium degrading the structural integrity of the concrete reactor pedestal support, as did GE, for molten core debris that potentially could continue to attack the concrete even with an overlying water pool. GE should perform additional analysis to evaluate this potential containment challenge within the range of possible corium-concrete interactions, and the resultant capability of the concrete pedestal to structurally perform its functions under these conditions. An uncertainty analyses should also be conducted to allow a complete assessment of the magnitude of this threat to containment integrity.

(c) The staff has identified a potentially serious threat to the containment integrity as a result of bypass leakage from the drywell to the wetwell airspaces. This containment challenge, which was not addressed by GE in its CETs, could be significant if drywell-to-wetwell bypass approaches the values observed in operating experience with similar containment designs. GE should provide information to evaluate this containment challenge by addressing the allowable drywell-to-wetwell leakage area and the capability of the vacuum breakers to repeatedly perform during the course of a severe accident without introducing additional leakage paths. Uncertainty analyses also are needed before judgments on the magnitude of this containment threat can be made.

(d) The staff has concluded that GE should further analyze the design of the lower drywell passive flooder system in view of; (i) the possibility of the system creating conditions conducive to an ex-vessel fuel/coolant interaction, i.e., lower drywell could be flooded prior to a complete ejection of the corium from the reactor vessel; (ii) the possibility of the system being unable to prevent serious degradation of the concrete pedestal wall; (iii) the possibility of the system providing steaming rates sufficient to overpressurize the containment, sooner than without the system, via bypass leakage between the wetwell and drywell; and (iv) the essential dependence of the system on the availability of the RHR system to cool the suppression pool and prevent steam overpressurization. It should be noted that an effective flooder system is necessary to maintain safe temperatures within the drywell following a severe accident.

However, the design should be further examined to allow consideration of whether to introduce water into the lower drywell in a controlled or uncontrolled manner, how fast to introduce the water and when to introduce the water.

(e) The efficacy of the containment overpressure protection system to perform its mitigative function is uncertain in view of (i) the possibility that the system pressure relief setpoint may not have taken into account the various downside risks, i.e., premature (reduces the possible time for fission product decay prior to release) or unnecessary (reduces the possible time to recovery the RHR system) releases, and (ii) the possibility of the system releasing unscrubbed fission products to the environment significantly earlier as a result of drywell to wetwell bypass flow without an operable or effective wetwell spray. The deterministic analysis of this system is discussed in Chapter 6.2.

Deleted st. about EPRI guidance

2. Based on the staff's CET analysis both with and without the overpressure protection system, the staff believes that the ABWR design could have difficulty meeting the CCFP of 0.10, for both the NRC goal and the GE goal and for both internal and seismic events, even without any consideration of the potential containment threats previously discussed in 1.(b) - integrity of the reactor pedestal, 1.(c) - drywell/wetwell bypass, and 1.(d) - fuel/coolant interactions, integrity of the reactor pedestal, and overpressure in the containment from debris steaming in the lower drywell. These staff conclusions are based primarily on the fact that direct containment heating and rapid ex-vessel steam generation sequences were found to be potential containment failure mechanisms. Furthermore, it should be noted that a CCFP defined as the quotient of the accident progression frequency for sequences affecting containment integrity and the total core damage frequency, needs to be evaluated very carefully when used as a measure to make judgments on the adequacy of the plant's level of defense-in-depth (see Section 19.11).

3. The staff believes that in light of the proposed drywell-to-wetwell bypass flows, GE should give additional attention to the design of the containment spray system. Specifically, the operability and reliability of the spray system, either via the RHR or the firewater system, need to be addressed in terms of safety significance and design basis criteria.

Deleted a conclusion about ATWS

19.7 SOURCE TERM ANALYSIS

7.1 Introduction

added ?

The objective of the source term analysis review is to assess GE's source term methods and estimates, to assess GE's prediction of ABWR design features for reducing source terms, and to assess GE's conclusions for its estimates of the expected releases. This source term analysis is not used to make a deterministic evaluation of the safety of the proposed design. Instead, it is used to assess, in realistic a manner, the safety profile of the proposed design as expressed in terms of the frequency of severe accidents, the consequences of a spectrum of such accidents of varying severities, and the integrated risk to the public. The staff performed a source term calculation with its codes, MELCOR (Reference 19.65) and STCP (Reference 19.66) to compare to one of GE's source term calculations done with the MAAP code (Reference 19.67). Outside of the review of the ABWR, the staff is currently doing a detailed comparison of MELCOR (Reference 19.65) and the MAAP code (Reference 19.67).

Source term calculations represent a logical piecing together, usually in the form of detailed computer codes, of the knowledge of severe accident progressions. There is much uncertainty in that knowledge however and, as a result, more than one source term code exists. The use of different, i.e. MELCOR (Reference 19.65) and STCP (Reference 19.66) that were developed by the NRC or MAAP (Reference 19.67) that was developed by the industry, will result in varying predictions of accident progression. Hence, an understanding of the assumptions and modelling is necessary to interpret the results of the calculations.

7.2 Methods Discussion

7.2.1 GE Analysis

GE modified the MAAP 3.0B code to account for the unique configurations of the ABWR and developed a representative input deck of the reactor and containment. Using this code and input deck, GE performed source term calculations for 12 accident progression groups defined from the CET analysis (releases for normal containment leakage were estimated from design basis leakage).

7.2.2 Staff Review

The staff's review made use of a variety of computer codes to assess source terms. First, in an audit, a STCP and a MELCOR calculation were compared to a GE MAAP calculation of a single sequence for the timing of the accident progression. Second, to determine release fractions, a parametric code similar to that used in the NUREG-1150 study (Reference 19.62) was developed and exercised such that uncertainty estimates could be made. A brief discussion of each of these calculational analyses follows.

Comparison of Accident Progression Timing

A timing comparison was done for a single sequence. The single sequence, i.e. the base case sequence, was chosen from the sequences for which GE did MAAP calculations. The selection was made on the basis of the sequence which included phenomena of relevance to source terms, not on the dominant sequence. The sequence was a loss of core cooling with vessel failure at high pressure. It resulted mostly from the Class I A (high pressure transients with loss of core cooling and failure to depressurize) and Class III A (small to medium LOCA with loss of core cooling and failure to depressurize) accidents. These types of accidents account for about 30 percent of the core damage frequency, and represent phenomena that are of particular interest in the review of the source terms.

Release Fractions and Uncertainty Calculations

The NUREG-1150 study (Reference 19.62) served as the basis for estimating the uncertainty in the source terms. Parametric codes from that study, collectively called XSOR (Reference 19.75), were used to develop a version called ABSOR for the ABWR review by making the necessary modifications to describe the ABWR design. From the experience gained from the NUREG-1150 study (Reference 19.62), fifteen key inputs of the ABSOR code were chosen and assigned distributions while holding four other parameters constant. While the staff's effort was less detailed than the NUREG-1150 study (Reference 19.62), the staff believes that the selection of variables and distribution assignments serves as a first approximation to uncertainty and is adequate for estimating source terms for comparison with GE's calculations.

To perform the calculations, samples of these distributions were taken with a LHS technique, a form of Monte Carlo sampling (Reference 19.67). One hundred samples of the input distributions produced one hundred sets of inputs for one hundred runs of the ABSOR code to produce one hundred sets of outputs (source terms) for each accident progression group (see Table 19.6-1 in Section 19.6.4.1). The one hundred source terms for each accident progression group taken together formed distributions that constituted an estimate of the source term release fractions and uncertainty. From these distributions of source term release fractions, the 5th, 50th and 95th percentile were determined for each accident progression bin stemming from the staff's CET analysis.

7.3 Assessment of Methods

GE's method for estimating source terms is somewhat different from the approach taken in the NUREG-1150 study (Reference 19.62). In that study, strategic groups of pathways were defined for making mechanistic source term calculations with codes, such as MELCOR (Reference 19.65) and the STCP (Reference 19.66), in order to adjust the parametric codes. Then all pathways through the CET had independent source term calculations made with the parametric XSOR codes. In contrast, GE grouped the pathways first, then calculated source terms for each group with its mechanistic code. The staff finds GE's method to be acceptable. However, uncertainty was not addressed, which is a major deficiency because source terms calculations inherently have much uncertainty in them.

The staff's method for calculating source terms was similar to that used in the NUREG-1150 study (Reference 19.62). However, as stated above, it was limited in scope. Nevertheless, it appears to be adequate to determine and assess source terms in each of the accident progression bins.

7.4 Source Term Results

7.4.1 Presentation of Source Term Results from the GE PRA and the Staff's Review

Table 19.7-1 qualitatively shows the magnitude of the source terms for the accident progression groups in Table 19.6-1 and Figures 19.6-1 and 19.6-2. Here, the staff characterized the source terms as negligible, low, moderate, and high and present source terms for both the GE and the staff's analyses.

Table 19.7-1 Description of the Accident Progression Source Terms in Terms of the Core Damage Frequency Fractions of Table 19.6-1 (in Section 19.6.4.1) and Figures 19.6-1 and 19.6-2.

Containment Challenge	Containment Response and Qualitative Magnitude of the Source Terms	Sequence code in Table 19.7-2
(A) **Same as** (H) subject to assumptions about drywell/wetwell bypass and pedestal failure.	If the bypass flow is large, then increased source terms, particularly of noble gases, occur via containment failure or vent actuation. If there is pedestal failure, then potentially high source terms through failed containment penetrations. If there is no bypass or pedestal failure, then the containment remains intact and the source terms are negligible.	NCL
(B) **No pressurization** when core debris is steaming in the reactor vessel with the RHR system.	Because the containment remains intact, the source terms are negligible.	NCL
(C) **Slow pressurization** when core debris is steaming in the reactor vessel without the RHR system.	Whether the containment fails or the vent actuates, low source terms because the suppression pool is in the release path. (Assumes no drywell/wetwell bypass flow.)	LCHPIVDN

Table 19.7-1 (continued)

Containment Challenge	Containment Response and Qualitative Magnitude of the Source Terms	Sequence code in Table 19.7-2
(D) **High temperature on penetration seals and slow pressurization** from core debris ejection into the upper drywell.	Excessive leakage from degraded penetration seals, equivalent to a failed containment, resulting in high source terms.	NSCHPFPH
(E) **Slow pressurization** prior to vessel failure due to loss of the RHR system and subsequent loss of core cooling.	The overpressure pressure protection system fails, allowing the containment to fail before the reactor vessel is breached and fission product release begins. Source terms are high.	LCHPPFEH
(F) **Rapid pressurization** at vessel failure from direct containment heating.	Early structural failure of the containment results in high source terms.	NSRCPFDH
(G) **Slow pressurization** when core debris is steaming in the lower drywell without the RHR system.	Late containment failure results in low to medium source terms when the containment sprays operate and high source terms when the containment sprays are unavailable.	LCLPFSDL LCLPPFDH
(H) **No pressurization** (according to the GE analysis) when core debris is steaming in the lower drywell with the RHR system.	Because the containment remains intact, the source terms are negligible.	NCL

7.4.2 Discussion of the Source Terms

The following discussion will focus on the two principal components of radioactive source terms, namely, release fractions and timing.

7.4.2.1 Release Fractions

Table 19.7-2 compares GE's and the staff's release fraction estimates for major sequence groups. The table compares release fractions for only cesium and iodine. Noble gas release fractions are essentially 1.0 for both GE's and the staff's estimates and simply not included in Table 19.7-2. However, other fission product species are not included in the table because GE predicted the releases of only iodine and cesium (in addition to noble gases) to environment; other fission product species were either retained in the damage fuel, the reactor vessel, or the containment.

During in-vessel core degradation, the more volatile fission product species (cesium, iodine, noble gases, and tellurium) are released from the fuel. Some of these fission products are retained on the surfaces of the reactor vessel, but most are deposited in the suppression pool. The fission products deposited in the reactor vessel can, over time, heat and eventually revolatilize. The MAAP code predicts that a significant fraction of these fission products revolatilize and are released to the environment if the containment fails; tellurium is not predicted to revolatilize and, hence, is not predicted to be released.

The refractory fission products are released from the fuel after the vessel fails if the core debris remains at high temperatures and vigorously attacks the concrete of the lower drywell. GE assumes that the passive flooder system effectively cools the core debris in the lower drywell to prevent extensive core/concrete interaction and fission product release, thus, reducing fission product releases from this source. The overlying water pool would also scrub the fission products that would be released from the core debris.

added

The staff's release fraction estimates also account for the revolatilization of cesium and iodine; therefore, its uncertainty range in Table 19.7-2 encompass the GE results. However, the staff's estimates differ from the GE estimates in that they allow for the possibility of continued core/concrete interaction, even after the passive flooder system actuates. Thus, the staff's release fraction estimates include the release of refractory fission products (not shown in Table 19.7-2). Notwithstanding the above differences in GE's and the staff's approach, the following general statements can be made regarding the release fractions shown in Table 19.7-2:

- o Minimal source terms occur when there is normal containment leakage. GE's estimates of iodine releases are generally bounded by the staff's estimates. GE's estimate of the cesium release is about three orders of magnitude above the staff's predictions for the normal case leakage, but both are very small.

- o Other than normal containment leakage, minimum source terms are seen in sequences where the core melt is arrested in the reactor vessel, Group (2). The GE estimates are slightly less than the staff's median estimate, but within the uncertainty range.

- o The maximum source terms are seen in station blackout and ATWS sequences, Group (5), and sequences where the containment has failed prior to vessel failure, Group (4); in both cases, the containment fails early. GE's estimated predictions are near the upper bounds (95 percent) of the staff's uncertainty bounds. The staff predicts larger iodine releases than cesium release whereas GE predicts roughly equal amounts of these radionuclides.

- o The effect of the firewater system aligned to the containment sprays is evident in Groups (6) and (7) (the passive flooder system actuates in both cases). GE claims a reduction in source terms by a factor of about 50 to 65. The staff predicts, a significantly smaller reduction, a factor of 1 to 3. Nevertheless, GE's estimates are within the range of uncertainty.

- o Table 19.7-2 and Figures 19.6-1 and 19.6-2 show that GE predicts significant fission product releases for some very low frequency accident sequences and failure modes. Some of these low frequency source terms are towards the high end of the staff's uncertainty range for the volatile fission product groups (As discussed above, GE did not calculate release of the refractory fission products).

Table 19.7-2 Cesium and Iodine Release Fractions, as Estimated by the Staff and GE for the Staff's Accident Progression Bins.

		GE Sequence Code	Staff's Estimates 5th	50th	95th	GE's Estimate	Reference to Figures 19.6-1 & 19.6-2
Cesium	(1)	NCL	$2.1x10^{-11}$	$3.2x10^{-9}$	$3.9x10^{-8}$	$5.1x10^{-6}$	A, B, H
	(2)	LCHPIVD	$2.7x10^{-6}$	$2.7x10^{-4}$	$5.3x10^{-2}$	$4.0x10^{-5}$	C
	(3)	NSCHPFP	$8.0x10^{-5}$	$2.1x10^{-2}$	$4.2x10^{-1}$	$1.0x10^{-1}$	D
	(4)	LCHPPFE	0.002	0.06	0.75	0.46	E
	(5)	NSRCPFD	0.002	0.07	0.75	0.50	F
	(6)	LCLPPFD	$8.3x10^{-4}$	$7.8x10^{-3}$	$1.7x10^{-1}$	$1.7x10^{-1}$	G
	(7)	LCLPFSD	$6.6x10^{-6}$	$2.4x10^{-3}$	$1.4x10^{-1}$	$3.2x10^{-3}$	G
Iodine	(1)	NCL	$3.4x10^{-11}$	$2.3x10^{-6}$	$3.8x10^{-4}$	$3.8x10^{-6}$	A, B, H
	(2)	LCHPIVD	$8.5x10^{-6}$	$9.2x10^{-4}$	$6.1x10^{-2}$	$2.8x10^{-5}$	C
	(3)	NSCHPFP	$9.4x10^{-4}$	$9.4x10^{-2}$	$4.0x10^{-1}$	$1.2x10^{-1}$	D
	(4)	LCHPPFE	0.007	0.19	0.69	0.47	E
	(5)	NSRCPFD	0.007	0.19	0.69	0.50	F
	(6)	LCLPPFD	0.002	0.10	0.16	0.18	G
	(7)	LCLPFSD	$4.7x10^{-4}$	$9.7x10^{-2}$	$3.2x10^{-1}$	$2.7x10^{-3}$	G

Code	Description
(1) NCL	Normal containment leakage (No containment failure).
(2) LCHPIVD	Fission product scrubbed by suppression pool before release, included the "venting" sequences.
(3) NSCHPFP	Early containment leakage due to high temperature failure of penetrations, no suppression pool scrubbing.
(4) LCHPPFE	Early containment failure, no suppression pool scrubbing.
(5) NSRCPFD	Early containment failure due to ATWS, no suppression pool scrubbing.
(6) LCLPPFD	late containment failure due to overpressurization, no spray, no suppression pool scrubbing.
(7) LCLPFSD	Late containment failure due to overpressurization, spray available, no suppression pool scrubbing.

7.4.2.2 Accident Progression Timings

In Table 19.7-3, three sets of timings from the MAAP, STCP, and MELCOR codes are shown. The particular sequence analyzed involved a loss of core cooling, vessel failure at high pressure, passive flooder opening to quench the debris in the lower drywell and, as there is no containment cooling, and containment failure (vent not modelled). Although the calculation was done for an earlier version of the ABWR (Reference 19.9, i.e., 100 psig ultimate strength of the containment and lower drywell composed of limestone concrete), the calculations are still relevant to show the degree to which the timings diverge. The staff believes that similar divergence would appear were the calculations to be redone accounting for the design modifications.

Table 19.7-3 Timing of Key Events in an Accident Progression.[a] [b]

Event	Time in Hours MAAP	STCP	MELCOR
Reactor scram	0.0	0.0	0.0
Core uncovery	0.3	0.4	0.5
Fuel begins to melt	(c)	1.3	0.9
Lower plenum dries out	(d)	2.9	4.5
Vessel failure	1.4	4.2	4.7
Passive flooder opens	2.4	4.7	4.7
Vent opens (e)	19.0	11.3	11.9
Containment failure	20.9	13.6	15.3

(a) Calculations are based on the Amendment 8 design, where the ultimate strength of the containment is 100 psig and the lower drywell is composed of limestone concrete.

(b) The sequence is defined as a loss of core cooling, where the vessel fails at high pressure, the passive flooder opens to quench the debris in the lower drywell, and, as there is no containment cooling, the drywell head fails.

(c) Not given by GE.

(d) Modelling in the MAAP code has the lower plenum filled with water when core debris drops into it and fails the vessel.

(e) Would have opened at this time if the containment was vented.

dryout does not occur because the essel failure is modelled as occurring while water is in the lower plenum.

STCP: 75 percent of the core becomes molten before the entire core slumps; the dryout occurs rapidly because the entire core slumps into the lower plenum.

MELCOR: the core slumps portion-by-portion. This can influence on the predicted times to dryout of the lower plenum; the portion by portion slumping delivers hot debris relatively slower than the STCP, hence, the boiling is slower.

Vessel Failure. The modelling of the vessel failure is important because it determines the amount of time that is available for recovery while the core debris is in the vessel. Other aspects of the accident progression aside, the differences in the models that cause differences in the time to vessel breach are as follows:

MAAP: While the water is in the lower plenum, the core debris heats and fails the penetrations.

STCP: When the water in the lower plenum boils away, the core debris heats the lower head until a gross failure occurs.

MELCOR: When the water in the lower plenum boils away, the penetrations rapidly fail.

Taking the core degradation process and vessel breach together, Table 19.7-3 shows that vessel failure is predicted to occur earliest (1.4 hours) from the MAAP code followed significantly later (4.2 hours) by the STCP and slightly after that (4.7 hours) by MELCOR. Vessel breach occurs earliest in MAAP because the water in the lower plenum does not need to boil away prior to penetration failure and a large amount of core debris supplies a large amount of decay heat to do this. The STCP models deliver a large amount of core debris for heating, but both the water and the lower head must be heated. The MELCOR models boil the water but melt only the penetrations; because a small amount of the core debris does this, the longest times to vessel breach are predicted.

Debris dispersal in the lower drywell. The modelling of the debris dispersal is important because it determines the actuation of the passive flooder system and the failure of containment penetrations in the upper drywell.

As expected, differences and similarities in the timings were found among the predictions from the MAAP, STCP, and MELCOR codes.

- o The time to core uncovery is about the same because this is little more than a calculation of boiling off a given amount of water.

- o The STCP and MELCOR predict approximately the same time to core melt because, in a simplistic way, this can be viewed as a calculation of heating a given amount of metallic mass.

- o The MAAP code calculates vessel failure occurring at 1.4 hours while the STCP and MELCOR calculate vessel failure occurring at 4.2 hours and 4.7 hours respectively.

- o Containment failure is predicted at 20.9 hours by MAAP while it is predicted to be much earlier by the STCP (13.6 hours) and MELCOR (15.3 hours).

Models of both the plant and severe accident phenomena give rise to the predictions. The staff believes that the differences in the plant models are small because the STCP and MELCOR input decks were derived from the MAAP input deck developed by GE. If the staff developed it own input decks, the differences in the predictions might be further complicated by differences in plant models. Also, these calculations have a considerable amount of uncertainty, which has not been determined. Such uncertainty would obscure differences in the predictions and the reasons for such differences.

In this review, the reason for the differences in these times is largely due to the modelling of core degradation, vessel failure, debris ejection into the lower drywell, and core/concrete interaction, each of which are discussed below.

Core Degradation. The modelling of core degradation is important because it in part determines the amount of debris available for subsequent phenomena, such as direct containment heating (high pressure sequences) or core/concrete interaction (low pressure sequences). Other aspects of the accident progression aside, the differences in the models that cause differences in the time to vessel breach are as follows:

MAAP: core degradation is modelled as a large amount (30 percent) of the core remaining in the vessel; the

MAAP: The debris falls into the lower drywell, even in a high pressure sequence where it is also dispersed into the upper drywell, along with water from the reactor vessel because the reactor vessel failure occurs at a vessel penetration before the lower plenum in the vessel boils dry. Once in the lower drywell, the water over the debris must be boiled away before the temperature can reach 500 deg. F. and actuate the passive flooder system.

STCP: Hot gases and some debris enter the lower drywell without water (because the water in the lower plenum of the reactor vessel must boil dry before the vessel fails.) The core debris is dispersed into the upper drywell. The hot gases and core debris actuate the passive flooder system soon after vessel failure.

MELCOR: Similar to the STCP, except that it takes slightly longer to actuate the passive flooder system due to modelling differences.

Core/concrete interaction. In MAAP, the core/concrete interaction is minimal because the core debris is modelled as entering the lower drywell where it is quenched with water from the passive flooder system. In STCP and MELCOR calculations, the interaction is more extensive.

The above discussion illustrates some of the modelling differences among the source term codes involved in the PRA and the review. These modeling differences give rise to the differences in the source term predictions. Some of the differences in the models are a consequence of the need to make approximations of these phenomena in order for the computer codes to operate on a reasonable amount of resources. But more important, the modelling differences arise from incomplete understanding of severe accident phenomena, both by the staff and the industry. Though the time to containment failure was calculated for only one sequence, the relatively short times that are predicted, 13 to 15 hours, for this "typical" sequence suggests that other credible sequences may have containment failure times less than 24 hours.

7.4.3 System Effects

In the staff's review, the staff began to assess the impact of some of the ABWR design features on the source terms. Though the staff's results are mostly qualitative, they bring out important points.

Lower Drywell

Source terms can be affected by the material used in the lower drywell structure. Although limestone concrete ablates at a higher temperature than basaltic concrete, much of the ablation products are non-condensible gases that do not dilute the core debris. Basaltic concrete melts at a lower temperature but the heat flux in the core/concrete interaction tends to be lower because the core debris is diluted by the ablation products. The staff has not thoroughly characterized how the source terms would differ for either limestone concrete or basaltic concrete.

Passive Flooder System

Water is introduced onto the debris by the passive flooder system in an attempt to quench the core/concrete interaction. According to GE's calculations, the core/concrete interaction can be quenched, hence, the source terms from this core debris are greatly reduced. According to the staff's calculations and limited experiments, the core/concrete interaction is not necessarily quenched readily - a significant interaction may persist until both oxidation reactions and decay heat subside enough to allow the heat losses to exceed the heat generation. Assuming for this discussion that a fuel/coolant interaction does not occur, the overlying pool will exert three effects - crust formation, scrubbing, and steaming.

- Experiments suggest that a crust may form on the debris bed. However, these experiments may not be prototypical; crust formation is being studied in ongoing research activities. This may reduce source terms by trapping certain fission products in the underlying molten debris. Also, it may increase other source terms by inhibiting debris cooling, thus helping to maintain the debris at a high temperature where fission products are volatilized and driven off of the debris.

- The scrubbing afforded by an overlying pool of water is dependent on many factors including pool subcooling, bubble dynamics and particle size. Pool scrubbing provided a decontamination factor of 1 to 16 in the staff's calculations. This figure gives some indication of the potential source term reduction value of the passive flooder system, as well as the variability inherent in this factor.

- The steaming exerts a complicated effect on the source terms. On the one hand, it creates a moist environment that would tend to reduce aerosols through condensation and agglomeration mechanisms. On the other hand, volatile species such as iodine are released along with the steaming. Also, the steaming pressurizes the containment, reducing the time to containment failure, or if the overpressure protection system is present, the time to venting. If containment failure or the onset of venting occur more rapidly, this will shorten the time available for reducing the source terms through aerosol deposition and radioactive decay.

Containment Spray System

Containment sprays can reduce source terms by the scrubbing action of the sprays. GE takes much credit for this scrubbing - the release fraction of iodine decreases by about two orders of magnitude, for a sequence where there is a loss of core cooling with vessel failure occurring at low pressure (from 0.18 to 2.7 E-3 in Table 19.7-2). The staff's calculations with the parametric source term code indicate a typical decontamination factor for containment sprays of 1 to 3 in Table 19.7-2.

Overpressure Protection System

The overpressure protection system is designed to route releases in the drywell through the suppression pool where scrubbing occurs. There are source term effects arising due to this system that could increase source terms:

- Drywell/wetwell bypass can partially defeat the overpressure protection system by routing a portion of the drywell release around the suppression pool to the wetwell air space. At least a part of the release would then be unscrubbed. Also, because the containment pressurization would be enhanced, the time available for the decay of fission products prior to vent actuation is reduced.

- When venting occurs, the containment pressure will drop causing a portion of the suppression pool to flash. Fission products can be re-entrained through pool swell and flashing, a phenomenon unaccounted for in either the staff's or GE's PRA: GE made additional calculations showing it to be of little concern (Reference 19.69). The staff reviewing the PRA has not seen these additional calculations nor has the staff made calculations to assess the potential effects of flashing. This is a confirmatory item.

7.5 Conclusions

Rewritten

1. Although GE's source term estimates appear to be within the range of uncertainty developed by the staff, based on the staff's limited consideration of source term estimates from GE's MAAP code and the staff's ABSOR/STCP codes, the staff believes that predictions from these codes could show considerable differences because of differences in models

and assumptions. The staff therefore concludes that GE should include a treatment of source term uncertainty, as part of its overall analysis of uncertainties, since large uncertainties are inherent in these calculations.

2. GE did not calculate fission product release during core/concrete interactions because MAAP predicts that the flooder will quench the core debris. The staff is unable to completely dismiss the potential for continued core/concrete interactions (and hence release of fission products from this source) after operation of the passive core flooder. Accordingly, as part of the uncertainty analysis GE should explicitly consider the potential for continued core concrete interaction and attendant source term releases.

added

3. The efficacy of the containment overpressure protection system to perform its mitigative function is uncertain in view of the concerns raised in Conclusion 1.(e) of Section 19.6.5 (pool bypass). The effect of pool bypass on source terms should also be addressed in the GE analysis. GE should also submit their calculations regarding pool flashing.

added

19.8 CONSEQUENCE ANALYSIS

8.1 Introduction

The objective of this section is to provide an assessment of the consequence analysis, the conclusions made from the analysis, and the impact of the ABWR design on predicted consequences. The codes for calculating the consequences were the CRAC2 code (Reference 19.76) used by GE and a more advanced consequence code, MACCS (Reference 19.77), used by the staff. The inputs to the codes and the way in which the calculations were compiled were also reviewed.

8.2 Methods Discussion

The consequence predictions are the third component of the risk calculation. Here, the fission product release predictions are combined with dispersion patterns, meteorological data, and population data, to yield predictions on the radiological impacts on a population. For this discussion, the times of importance to consequence calculations are illustrated in Figure 19.8-1.

8.2.1 GE Analysis

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[illegible]

GE based its consequence calculations on five sites in the U.S. representing meteorological regions, called northeast, northwest, southwest, south, and west. The meteorological and population data were obtained from previously developed information contained in "Technical Guidance for Siting Criteria Development," Sandia National Laboratories, NUREG/CR-2239, dated December 1986 (Reference 19.78). The source terms were determined using the MAAP code (Reference 19.67) for each of the 13 release categories (12 accident sequences and normal containment leakage) as discussed in Section 19.7.2.1. The five calculations from each of the sites, made for each release class, were averaged together. GE did not provide consequence results for each accident progression group (See Table 19.6-1) in its SSAR (Reference 19.9), but separately provided c.mputer output supporting the SSAR.

GE assumed that the elevation of the release is always 37 meters because of the way in which release paths channel through the design of the ABWR. If releases were to go through a vent, the release would be through the stack of the standby gas treatment system, and hence, it would be somewhat higher (about 76 meters).

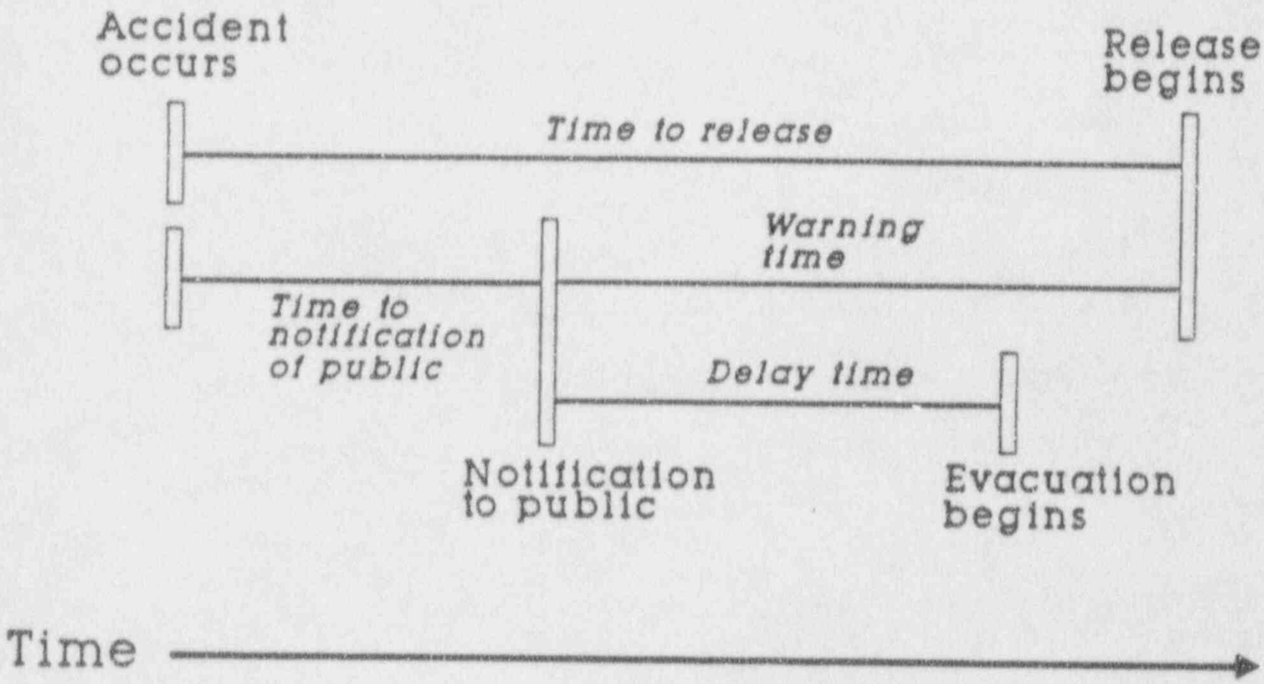


Figure 19.8-1 Important times of consequence calculations.

Some of the assumptions made in the consequence calculations varied among the NRC and the EPRI risk measures appearing in Section 19.9. In calculating the NRC Quantitative Health Objectives and the ALWR requirements, GE assumed 1 hour between accident initiation and public notification, an additional 1.5 hour delay in evacuating, 95 percent evacuation within 3 miles of the plant, at an evacuation speed of 4.47 meters/second (10 miles/hour). The warning time, which is defined at the time between official notification and the release of radioactivity, depends on the time of containment failure, and varies from 1.7 hours to 20.7 hours. In calculating the EPRI ALWR requirements (Reference 19.2) (involving the probability of exceeding 25 rem at 1/2 mile from the plant, i.e. dose definition of containment failure), GE assumed no evacuation and no shielding for 24 hours after plume arrival; in both the internal and seismic events calculations, the assumptions about the evacuation distance, warning time, and speed were the same as those made in calculating the NRC risk measures.

8.2.2 Staff Review

The staff's consequence estimates were calculated using the MACCS code. However, instead of five sites, the staff used two sites representing a low population site (Salem Power Station) and a high population site (Zion Power Station). The consequences were calculated for each of these two sites, then averaged together. The averaging was done because the low population site was thought to under-represent typical consequences and the high population site was thought to over-represent typical consequences. Although the staff averaged the consequences, the averaged results do not necessarily represent an "average" site.

The source terms used for these calculations were derived from calculations using the ABSOR code (Reference 19.75). As discussed in Section 19.7.2.2, the uncertainty was expressed as distributions, from which the 5th, 50th, and 95th percentile source terms were determined for each of the seven release categories calculated with the staff's CETs (See Table 19.7-2). The staff's release height is the same as that of GE. As in GE's analysis, some of the assumptions made in the consequence calculations varied among the NRC and the EPRI risk measures appearing in Section 19.9.

In calculating the NRC Quantitative Health Objectives and the ALWR Requirements, the staff calculated a warning time and assumed 99.5 percent evacuation for internal events or 0 percent evacuation for seismic events, within 10 miles of the plant, at an evacuation speed of 4.47 meters/second (10 miles/hour) for the staff's low population site or 1.1 meter/second for (2.3 miles/hour) for the staff's high population site, with a calculated notification time. The time between accident initiation and the time of public notification, i.e. 1.33 hours, is about the same as GE's assumption. These assumptions are similar to assumptions in the NUREG-1150 study (Reference 19.62).

In calculating the EPRI ALWR requirements (involving the probability of exceeding 25 rem at 1/2 mile from the plant, i.e. dose definition of containment failure), the staff assumed no evacuation and no shielding for 24 hours, as in GE's calculations.

8.3 Staff Assessment

CRAC2 (Reference 19.76), and the MAACS (Reference 19.77) are very similar computer programs, MAACS being the successor of the CRAC2 code. These codes have been shown in previous studies to produce results within a factor of 2 to 3 for similar input assumptions and the staff considers that both provide an acceptable characterization of the consequences of a severe accident. Accordingly, the staff finds GE's use of the CRAC2 to be acceptable.

Reworded and expanded.

The staff has also assessed GE's input assumptions related to warning times, evacuation delay time, and height of release. Although some of the inputs to GE's calculations differ from the inputs to NRC's calculations, the differences in the results are small. GE's warning time is fixed and begins one hour after a reactor trip. NRC's warning time is calculated and begins when the level in the vessel drops to 2 feet below the top of the active fuel; this takes about 1.3 hours for some sequences, based on predictions from the STCP (see Table 19.7-3). The effect of the difference on consequences is small.

Compared to the NUREG 1150 study (Reference 19.62), GE's consequence calculations appear to be similar in terms of the delay time in evacuating and the evacuation speed. Table 19.8-1 shows these times. GE's release height of 37 meters is reasonable for severe accident calculations because of the structure of the ABWR and that the likely failure location in the head of the drywell. The staff concludes that while some of the inputs to GE's calculations differ from the inputs to NRC's calculations, the differences in the results are small. The differences are not explicitly presented here, but are reflected in the integrated risk estimates presented in section 19.9. Based on a review of GE's analysis and supplementary staff analysis, and contingent upon satisfactory resolution of source term issues, the staff finds that GE's consequence analysis would produce results that are in general agreement with the staff's calculations.

8.4 Conclusions

1. The staff believes that GE's consequence calculation method is reasonable and is similar to the staff's method.

added (

2. Contingent upon satisfactory resolution of source term issues (see discussion on Source Term Analysis in section

Table 19.8-1 Comparison of GE's, Staff's, and the NUREG-1150 delay times and evacuation speeds.

	Delay Time	Evacuation Speed
GE (ABWR)	1.5	4.47
Staff (ABWR)	2.3[a.] 1.5[b.]	1.1[a.] 4.47[b.]
NUREG-1150		
Grand Gulf	1.25	3.7
Peach Bottom	1.5	4.8
Surry	2.0	1.8
Sequoyah	2.3	1.8
Zion	2.3	1.1

a. Value used for high population site, i.e. Zion plant.
b. Value used for low population site, i.e. Salem plant.

19.7), the staff believes that GE's consequence methods and assumptions would produce results that are in general agreement with the staff's calculations.

3. GE's consequence calculations are done for an average site. The staff presented results for high and low population sites and also averaged the results. Both procedures produce similar results for the given source terms.

4. EPRI's ALWR requirements document (Reference 19.2) specifies a site which is characterized as at the high end of the site severity range for population density and distribution. GE's consequence calculations do not satisfy to the EPRI requirements (Reference 19.2).

19.9 INTEGRATED RISK ESTIMATES

9.1 Introduction

Risk integration is the final stage of calculating risk. Here, the frequency of the various accident sequences and the consequences are integrated to give the risk results.

9.2 Methods Discussion

9.2.1 GE Analysis

Since GE calculated point estimates, the risk was obtained by simply multiplying the point estimates for the source term group frequency and consequences. The staff presumes that GE considers this a best estimate of risk, though there is no mention of this in GE's documentation.

9.2.2 Staff Review

Uncertainties in the risk estimates were not considered by GE. The staff considered them separately. Based on other studies, such as NUREG-1150 (Ref 19.62), twenty parameters were selected according to what was thought to be the parameters having the highest impact on risk; eleven of the parameters were from the Level 1 analysis and the remaining nine parameters were from the Level 2 and 3 analyses. Distributions were assigned to these parameters, some (such as direct containment heating) from other studies, and others using the staff's judgments. These distributions and others from source terms were sampled, again with the Latin Hypercube Sampling (LHS) process (Reference 19.67) one hundred times, to yield frequencies and consequences. Each accident frequency and consequence value multiplied together yields a risk estimate, and an estimate of the risk distribution.

9.3 Assessment of Methods

GE calculated a point estimate of risk without an estimate of uncertainty. Though not explicitly called a best estimate in the SSAR (Reference 19.9), GE suggests that its estimate is a best estimate (page 65 of Reference 19.70), which does not have a statistical definition. The staff's review indicates the uncertainty to be at least an order of magnitude and often times greater; therefore, the use of a point estimate without an associated measure of uncertainty is misleading.

In the staff's application of the concept of uncertainty, the staff used the LHS process (Reference 19.67) to propagate the uncertainty in selected inputs of the staff's risk analysis through the various mathematical functions to generate an uncertainty distribution in the output, i.e. risk estimate. The staff finds that there are numerous reasonable sets of inputs, each having a degree of credibility. This makes it presumptuous to consider any one set of inputs, and hence, output as "best". Therefore, it is necessary to characterize the risk results in terms of uncertainty.

9.4 Presentation and Discussion of the Risk Results

The risk results are presented in Tables 19.9-1 through 19.9-6 and Figures 19.9-1 and 19.9-2. Both GE's and the staff's risk results are arranged under the NRC Quantitative Health Objectives in Reference 19.80, NRC ALWR Requirements in Reference 19.4, the EPRI ALWR Requirements in Reference 19.2, and GE Goals. The EPRI ALWR Requirements goal and the GE goal do not constitute part of the staff's evaluation; risk estimates are compared to these goals only for information purposes.

To understand the results, several points should be considered:

o Design. Since GE's original submittal, the design was modified so that the ultimate strength of the containment is 134 psig and the lower drywell is composed of a basaltic concrete (Reference 19.75). GE's results are for the unmodified design of the ABWR, where the ultimate strength of the containment is 100 psig and the lower drywell is composed of a limestone concrete. The staff's analysis takes the GE design modifications into account.

added

An additional feature, the containment Overpressure Protection System (OPS), is also now a part of the ABWR design, but only a limited assessment of the effect of this modification has been made due to its late addition to the design. Specifically, no credit for the system has been taken in the staff or GE analyses of internal events, with the exception of analyses reported in Tables 19.5-2 and 19.9-6. However, credit for the OPS was taken in the staff's seismic analysis, since the OPS was part of the system at the time that the review was performed.

o Accounting. GE's analysis accounts for the firewater addition system in the Level 2 analysis only whereas the staff's analysis accounts for this system in both the Level 1 and the Level 2 analyses. As discussed in Section 19.6.4.1, the accounting can affect the results through the various implied assumptions that arise. In the seismic events analysis, GE used a seismic hazard curve developed as the bounding curve some of the nuclear power plant sites east of the Rockies whereas the staff used three sites (i.e., Pilgrim, Seabrook, and Watts Bar) having the highest seismic hazard in the eastern and central United States with two hazard curves (LLNL and EPRI); this is discussed in Section 19.6.4.1.

added

o Arithmetic and Logic Errors. As part of the staff's review of GE's analysis, a number of arithmetic and logic errors were identified. These errors were corrected in the staff's analysis, but not in the GE point estimate values reported in Table 19.9-1.

- Definition of containment failure. Two definitions of containment failure were used. According to the first definition, failure occurs when the containment integrity can no longer be maintained as a pressure boundary; this is known as the pressure boundary definition and was defined by both GE and the staff to occur when the pressure load in the containment exceeds the ultimate strength of the containment. The staff also considered vent actuation relative to containment failure.

  added

  For proper assessment of containment failure when venting is a consideration, the staff looked to SECY-90-016 for guidance. In general, the goal is to maintain containment integrity, to the extent practicable, during the initial 24 hours following the onset of core damage without the need to vent. Beyond the 24 hour guideline, the containment should continue to provide a barrier against the uncontrolled release of fission products.

  The staff has viewed this guidance as a design goal, but does not preclude the consideration of venting during the initial 24 hour period. The most important consideration is the view that venting is a controlled process and does not constitute an uncontrolled release. For the overpressure protection system of the ABWR, a controlled release is established by two isolation valves located downstream of the rupture disk. Once actuated, the operators can close these valves to terminate the release and remotely reopen the valves to reestablish the release if containment conditions warrant such action.

  added

  However, in the presentation of GE's results for internal and external events, and the staff's PRA results for internal events, the effects of controlled venting on the conditional containment failure probability (CCFP) is not specifically addressed. We note that venting in less than 24 hours should not be equated with containment failure. We intend to separate these issues in the FSAR. Until this change is made, the staff's results (for external events only, since this is the only place the OPS was credited) will show containment failure whenever the overpressure protection system is actuated within the first twenty four hours.

- Descriptions of distributions. The distributions of the risk results are described in terms of their width and skewness. The width of the distributions are subjectively described as narrow (range of not more than two orders of magnitude), moderate (range of three to six orders of magnitude), and wide (range of seven or more orders of magnitude); it reflects the reproducibility of the results in terms of their sensitivity to inputs and assumptions. The mean (influenced by extreme values) and the median (not influenced by extreme values) show the central tendencies of the distributions and together indicate the s wness of the distributions.

- o [Uncertainty]. The staff's estimates of uncertainty for risk account for uncertainty in the Level 1 (systems reliability), Level 2 (accident progression as modelled by the CET and source terms), and Level 3 (consequences) analyses. The selection of variables and distribution assignments serves as a first approximation to uncertainty. Additional uncertainty work was done by the staff for seismic events; this is not directly reflected in the distributions. The staff's analysis used the LLNL hazard curves, which generally provide a much higher core damage frequency than either the GE or the EPRI curves, while the latter two curves give rise to core damage frequencies of about the same magnitude. However, the uncertainty ranges of the LLNL curve are large and encompass the other two curves. In performing the uncertainty analysis using the LHS method, the core damage frequency, containment failure probability, and CCFP were ranked separately. Thus, these numbers for the given percentile are not from the same sample, and, consequently, the CCFP is not generally equivalent to the ratio of the containment failure probability divided by the core damage frequency. added

- o Seismic hazard. In regards to seismic events, the major analytical difference between GE and the staff is in the seismic hazard curves (annual frequency of exceeding a specified peak ground acceleration). The staff's analysis was based on three sites having the highest seismic hazard in the Eastern and Central U.S. (Pilgrim, Seabrook, and Watts Bar), and used two seismic hazard curves, one developed by Lawrence Livermore National Laboratory (LLNL) and the other developed by the Electric Power Research institute (EPRI). GE used a hazard curve developed as a bounding curve for many nuclear power plant sites east of the Rockies. Reworded

The risk results are presented in Tables 19.9-1 through 19.9-5. In Table 19.9-1, the seismic events are given for the Pilgrim site using the LLNL hazard curve is provided for comparison. Table 19.9-2 shows the results of the seismic events analyses from other sites along with the Pilgrim site. Figures 19.9-1 and 19.9-2 are plots of the risk results for internal events and seismic events. Reworded

Cut back The risk estimates are interpreted in Table 19.9-5. In accordance with Reference 19.80, the staff determined whether or not the NRC Quantitative Health Objectives and the NRC ALWR Requirements goals were met by comparing its mean risk estimates to the goals. This approach was also used in determining whether or not the EPRI ALWR Requirements goals and the GE definition of the CCFP were met. Table 19.9-5 also describes the staff's uncertainty distributions.

Table 19.9-1 GE's Point Estimates and the Staff's Mean Estimates of the Internal and Seismic Events Risk.[a]

NRC Quantitative Health Safety Goals	Goal	Internal Events		Seismic Events		Internal & Seismic Events	
		Point Estimate GE[b]	Mean Estimate Staff[c]	Point Estimate GE[b]	Mean Estimate Staff[d]	Point Estimate GE[b]	Mean Estimate Staff
Individual risk of early fatality	< $5x10^{-7}$	$2.5x10^{-13}$	$1.6x10^{-12}$	$1.8x10^{-11}$	$3.1x10^{-9}$	$1.8x10^{-11}$	$3.1x10^{-9}$
Individual risk of cancer fatality	< $2x10^{-6}$	$1.9x10^{-12}$	$4.5x10^{-11}$	$1.5x10^{-10}$	$8.4x10^{-9}$	$1.5x10^{-10}$	$8.4x10^{-9}$
Probability of large release (one or more early fatalities)[e]	< $1x10^{-6}$	-	$7.4x10^{-10}$	-	$3.2x10^{-7}$	-	$3.2x10^{-7}$
NRC ALWR Requirements							
Core damage frequency	< $1x10^{-4}$	$1.7x10^{-7}$	$7.5x10^{-7}$	$2.5x10^{-7}$	$2.9x10^{-5}$	$4.3x10^{-7}$	$2.9x10^{-5}$
Probability of containment failure - pressure boundary definition	< $1x10^{-6}$	$8.5x10^{-9}$	$1.1x10^{-7}$	$1.9x10^{-7}$	$2.2x10^{-5}$	$2.1x10^{-7}$	$2.2x10^{-5}$
Conditional containment failure probability - pressure boundary definition	< 0.1	0.05[h] [0.12]	0.18	0.81	0.77	0.51[g,h] [0.53]	0.76

a. Venting not taken into account.
b. Unmodified design - 100 psig containment failure pressure, limestone lower drywell.
c. Modified design - 134 psig containment failure pressure, basaltic lower drywell.
d. Pilgrim site only, based on LLNL hazard curve, provided for comparison.
e. Proposed goal, with final definition of "large" still under staff consideration.

Table 19.9-1 (continued)

f. See "Definition of containment failure" in preceding text for definition.
g. With the overpressure protection system, GE claims a value < 0.10.
h. These numbers are taken from GE submittals and do not reflect staff correction of identified arithmetic and logic errors. Corrected values are in parentheses.

Table 19.9-1 (continued)[a]

EPRI ALWR Requirements	Goal	Internal Events: Point Estimate GE[b]	Internal Events: Mean Estimate Staff[c]	Seismic Events: Point Estimate GE	Seismic Events: Mean Estimate Staff[d]	Internal & Seismic Events: Point Estimate GE	Internal & Seismic Events: Mean Estimate Staff
Core damage frequency	$< 1x10^{-5}$	$1.7x10^{-7}$	$7.5x10^{-7}$	$2.5x10^{-7}$	$2.9x10^{-5}$	$4.3x10^{-7}$	$2.9x10^{-5}$
Probability of a containment failure - dose definition	$< 1x10^{-6}$	$5.0x10^{-10}$	$8.9x10^{-8}$	$3.8x10^{-8}$	$1.6x10^{-5}$	$3.9x10^{-8}$	$1.6x10^{-5}$
GE Goal							
Conditional containment failure probability - dose definition[f]	< 0.1	0.003[h]	0.15	0.15	0.58	0.09	0.53

a. Venting not taken into account.
b. Unmodified design - 100 psig containment failure pressure, limestone lower drywell.
c. Modified design - 134 psig containment failure pressure, basaltic lower drywell.

Table 19.9-1 (continued)

d. Pilgrim site only, based on LLNL hazard curve, provided for comparison.

e. Proposed goal, with final definition of "large" still under staff consideration.

f. See "Definition of containment failure" in preceding text for definition.

g. With the overpressure protection system, GE claims a value < 0.10.

h. These numbers are taken from GE submittals and do not reflect staff correction of identified arithmetic and logic errors. Corrected values are in parentheses.

Table 19.9-2 Staff Point Estimate of Risk from Seismic Events[a]

NRC Quantitative Health Safety Goals	Goal	LLNL Hazard Curve			EPRI Hazard Curve		
		Pilgrim	Seabrook	Watts Bar	Pilgrim	Seabrook	Watts Bar
Individual risk of early fatality	$< 5x10^{-7}$	$1.4x10^{-11}$	$7.4x10^{-12}$	$1.2x10^{-11}$	$4.2x10^{-13}$	$2.9x10^{-13}$	$1.7x10^{-13}$
Individual risk of cancer fatality	$< 2x10^{-6}$	$9.2x10^{-9}$	$4.9x10^{-9}$	$8.1x10^{-9}$	$2.7x10^{-10}$	$1.9x10^{-10}$	$1.1x10^{-10}$
Probability of large release (one or more early fatalities)[b]	$< 1x10^{-6}$	$1.5x10^{-9}$	$8.3x10^{-10}$	$1.4x10^{-9}$	$4.6x10^{-11}$	$3.2x10^{-11}$	$1.9x10^{-11}$
NRC ALWR Requirements							
Core damage frequency	$< 1x10^{-4}$	$7.3x10^{-5}$	$3.9x10^{-5}$	$6.5x10^{-5}$	$1.9x10^{-6}$	$1.4x10^{-6}$	$8.3x10^{-7}$
Probability of containment failure - pressure boundary definition	$< 1x10^{-6}$	$5.7x10^{-5}$	$3.1x10^{-5}$	$5.0x10^{-5}$	$1.5x10^{-6}$	$1.1x10^{-6}$	$6.5x10^{-7}$
Conditional containment failure probability - pressure boundary definition[c]	< 0.1	0.78	0.78	0.78	0.78	0.78	0.78

a. Venting not taken into account.
b. Proposed goal, with final definition of "large" still under staff consideration.
c. See "Definition of containment failure" in preceding text for definition.

Table 19.9-2 (continued)[a]

		LLNL Hazard Curve			EPRI Hazard Curve		
EPRI ALWR Requirements	Goal	Pilgrim	Seabrook	Watts Bar	Pilgrim	Seabrook	Watts Bar
Core damage frequency	$< 1 \times 10^{-5}$	7.3×10^{-5}	3.9×10^{-5}	6.5×10^{-5}	1.9×10^{-6}	1.4×10^{-6}	8.3×10^{-7}
Containment failure probability - dose definition	$< 1 \times 10^{-6}$	4.4×10^{-5}	2.4×10^{-5}	3.9×10^{-5}	1.1×10^{-6}	8.4×10^{-7}	5.0×10^{-7}
GE Goal							
Conditional containment failure probability - dose definition[b]	< 0.1	0.60	0.59	0.60	0.60	0.60	0.60

a. Venting not taken into account.
b. See "Definition of containment failure" in preceding text for definition.

Table 19.9-3 Staff Estimates of Uncertainty in the Risk Measures for Internal Events

NRC Quantitative Health Safety Goals	Goal	95th	50th	Mean	5th
Individual risk of early fatality	$< 5\times10^{-7}$	1.0×10^{-11}	4.2×10^{-14}	1.6×10^{-12}	1.7×10^{-16}
Individual risk of cancer fatality	$< 2\times10^{-6}$	3.0×10^{-10}	1.0×10^{-11}	4.5×10^{-11}	3.2×10^{-12}
Probability of large release (one or more early fatalities)[b]	$< 1\times10^{-6}$	5.3×10^{-9}	4.9×10^{-11}	7.4×10^{-10}	3.4×10^{-15}
NRC ALWR Requirements					
Core damage frequency	$< 1\times10^{-4}$	2.5×10^{-6}	5.7×10^{-7}	7.5×10^{-7}	3.4×10^{-7}
Probability of containment failure - pressure boundary definition	$< 1\times10^{-6}$	4.0×10^{-7}	6.4×10^{-8}	1.1×10^{-7}	3.9×10^{-8}
Conditional containment failure probability - pressure boundary definition[c]	< 0.1	0.70	0.11	0.18	0.05

a. Venting not taken into account.
b. Proposed goal, with final definition of "large" still under staff consideration.
c. See "Definition of containment failure" in preceding text for definition.

Table 19.9-3 (continued)

EPRI ALWR Requirements	Goal	95th	50th	Mean	5th
Core damage frequency	$< 1x10^{-5}$	$2.5x10^{-6}$	$5.7x10^{-7}$	$7.5x10^{-7}$	$3.4x10^{-7}$
Probability of containment failure - dose definition	$< 1x10^{-6}$	$3.4x10^{-7}$	$5.2x10^{-8}$	$8.9x10^{-8}$	$3.1x10^{-8}$
GE Goal					
Conditional containment failure probability - dose definition [b.]	< 0.01	0.56	0.09	0.15	0.04

a. Venting not taken into account.
b. See "Definition of containment failure" in preceding text for definition.

Table 19.9-4 Staff Estimates of the Uncertainty in the Risk Measures for Seismic Events, for the Pilgrim Site, using the LLNL Hazard Curve.[a]

NRC Quantitative Health Safety Goals	Goal	95th	50th	Mean	5th
Individual risk of early fatality	$< 5 \times 10^{-7}$	2.3×10^{-8}	4.0×10^{-11}	3.1×10^{-9}	5.1×10^{-15}
Individual risk of cancer fatality	$< 2 \times 10^{-6}$	5.5×10^{-8}	3.7×10^{-10}	8.4×10^{-9}	1.1×10^{-12}
Probability of large release (one or more early fatalities)[b]	$< 1 \times 10^{-6}$	2.6×10^{-6}	3.6×10^{-9}	3.2×10^{-7}	1.4×10^{-14}
NRC ALWR Requirements					
Core damage frequency	$< 1 \times 10^{-4}$	1.7×10^{-4}	2.5×10^{-6}	2.9×10^{-5}	1.0×10^{-8}
Probability of containment failure - pressure boundary definition	$< 1 \times 10^{-6}$	1.4×10^{-4}	1.9×10^{-6}	2.2×10^{-5}	6.5×10^{-9}
Conditional containment failure probability - pressure boundary definition[c]	< 0.1	0.89	0.77	0.77	0.65

a. Venting not taken into account.
b. Proposed goal, with final definition of "large" still under staff consideration.
c. See "Definition of containment failure" in preceding text for definition.

Table 19.9-4 (continued)[a]

EPRI ALWR Requirements	Goal	95th	50th	Mean	5th
Core damage frequency	$< 1x10^{-5}$	$1.7x10^{-4}$	$2.5x10^{-6}$	$2.9x10^{-5}$	$1.0x10^{-8}$
Probability of containment failure - dose definition	$< 1x10^{-6}$	$8.6x10^{-5}$	$1.3x10^{-6}$	$1.6x10^{-5}$	$5.3x10^{-9}$
GE Goal					
Conditional containment failure probability - dose definition[b]	< 0.1	0.75	0.58	0.58	0.36

a. Venting not taken into account.
b. See "Definition of containment failure" in preceding text for definition.

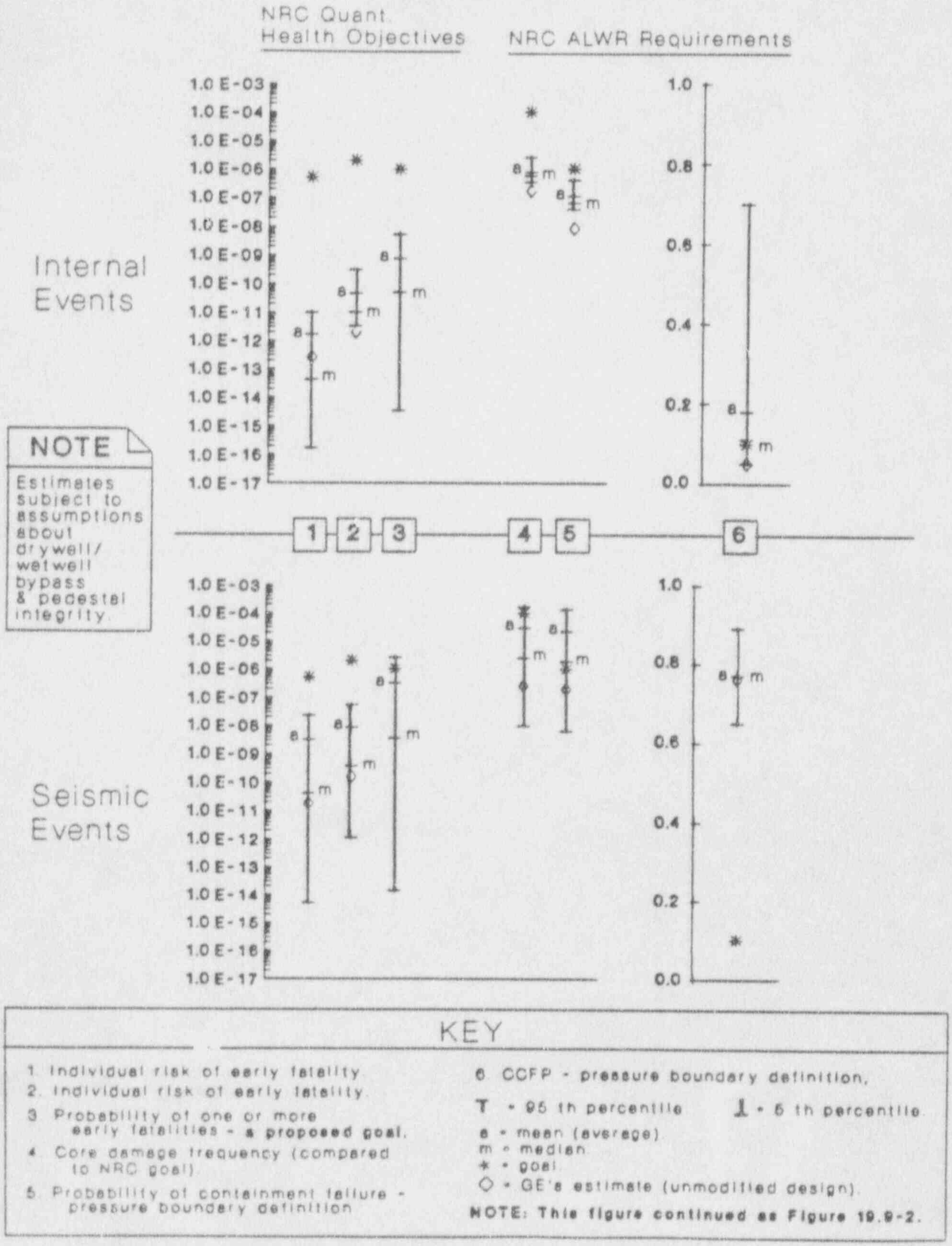
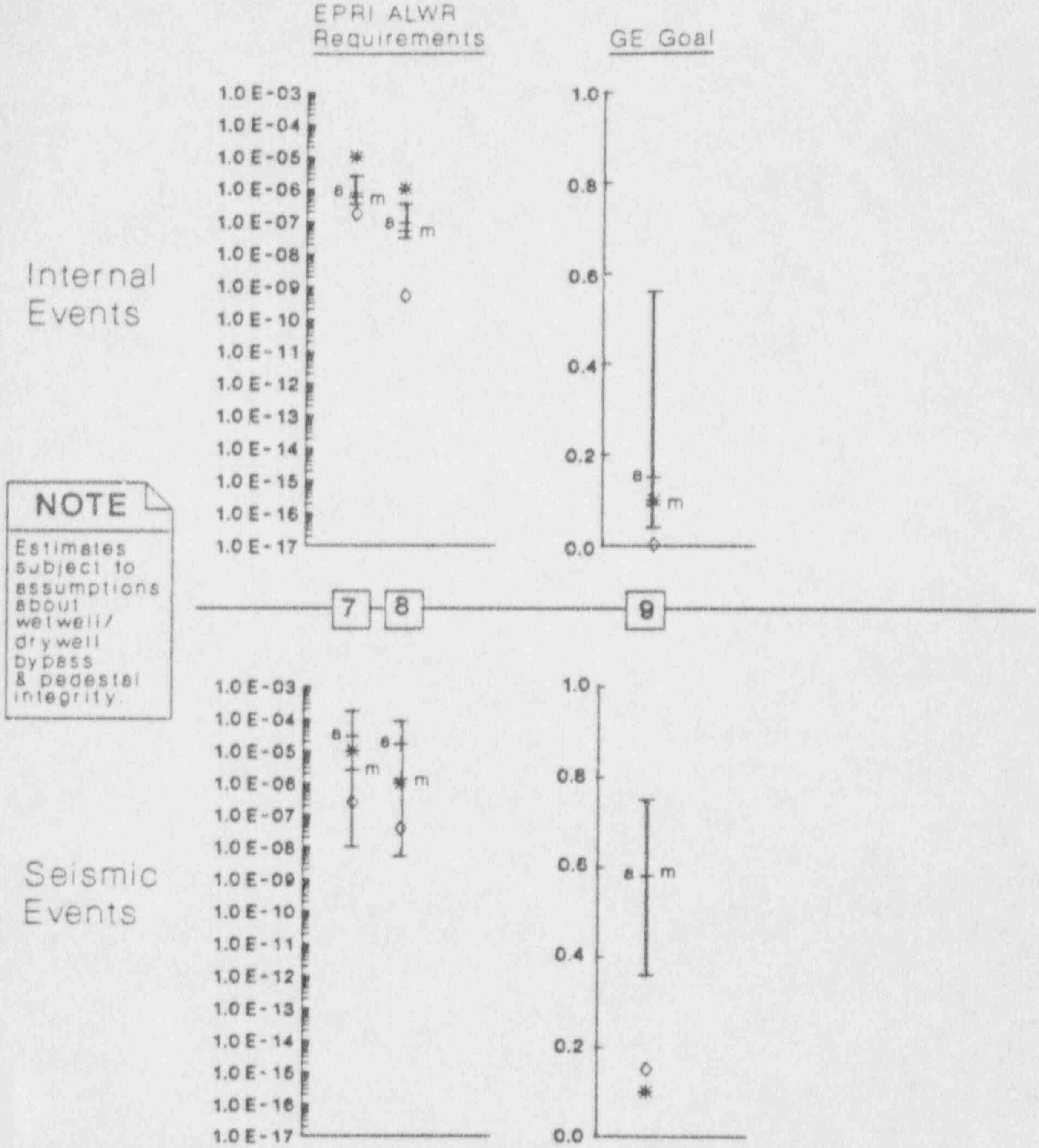


Figure 19.9-1 Plot of internal and seismic events risk results (NRC Quantitative Health Objectives and NRC ALWR Requirements).



KEY

NOTE: This figure continues from Figure 19.9-1.

7. Core damage frequency (compared to EPRI goal).

8. Probability of containment failure - dose definition.

9. CCFP - dose definition.

⊤ = 95 th percentile.

⊥ = 5 th percentile.

a = mean (average).

m = median.

* = goal.

◇ = GE's estimate (unmodified design).

Figure 19.9-2 Plot of internal and seismic events risk results (EPRI ALWR Requirements).

Table 19.9-5 Description of the Risk Results.

NRC Quantitative Health Objectives	Statement
Individual risk of early fatality Goal: $< 5x10^{-7}$	The mean estimate for internal events is below the goal. The mean estimate for seismic events is below the goal. Therefore, the goal is met in both cases. Both uncertainty distributions are well below the goal; for internal events, the upper bound is five orders of magnitude below the goal; for seismic events, the upper bound of the uncertainty range is one order of magnitude below the goal. The width of the distributions is moderate for internal events and wide for seismic events. Both distributions are skewed towards the lower tail (smaller values); hence, the bulk of each distribution is closer to the lower bound, far below the goal.
Individual risk of cancer fatality Goal: $< 2x10^{-6}$	The mean estimate for internal events is below the goal. The mean estimate for seismic events is below the goal. Therefore, the goal is met in both cases. Both uncertainty distributions are well below the goal; for internal events, the upper bound of the uncertainty range is four orders of magnitude below the goal; for seismic events, the upper bound of the uncertainty range is two orders of magnitude below the goal. The width of the internal events distributions is narrow and the width of the seismic events distribution is moderate. Both distributions are skewed towards the lower tail (smaller values); the bulk of each distribution is far below the goal.
Probability of large release (one or more early fatalities)[a] Goal: $< 1x10^{-6}$	The mean estimate for internal events is below the goal. The mean estimate for seismic events is below the goal. Therefore, the goal is met in both cases. The uncertainty distribution for internal events is well below the goal. The uncertainty distribution for seismic events is immediately below the goal; though the upper tail of the seismic events distribution is immediately above the goal, this represents a small fraction of the distribution. The width of the internal events distribution is moderate and the width of the seismic events distribution is wide. Both distributions are skewed towards the lower tail (smaller values). In the seismic events distribution, the position of the mean, median, and upper tail (skewness) relative to the goal suggests that the bulk of the distribution is below the goal with the upper tail touching the goal.

a. Proposed goal, with final definition of "large" still under staff consideration.

Table 19.9-5 (continued)

NRC ALWR Req.	Statement
Core damage frequency Goal: < $1x10^{-4}$	The mean estimate for internal events is below the goal. The mean estimate for seismic events is below the goal. Therefore, the goal is met in both cases. The uncertainty distribution for internal events is well below the goal. The uncertainty distribution for seismic events is immediately below the goal; though the upper tail of the seismic events distribution is immediately above the goal, this represents a small fraction of the distribution. The width of the internal events distribution is narrow and the width of the seismic distribution is moderate. The median and mean indicate that the internal events distribution is symmetrical while the seismic events distribution is skewed towards the lower tail (smaller values). In the seismic events distribution, the position of the mean, median, and upper tail (skewness) relative to the goal suggests that the bulk of the distribution is below the goal with the upper tail touching the goal.
Probability of containment failure - pressure boundary definition Goal: < $1x10^{-6}$	For internal events, the mean is below the goal, saying that the goal is met. For seismic events, the mean is above the goal, saying that the goal is not met. The internal events distribution is below the goal. The seismic events distribution bridges the goal; the upper tail of the distribution is several orders of magnitude above the goal. The width of the internal events distribution is narrow while the width of the seismic events distribution is moderate. The internal events distribution is symmetrical. The seismic events distribution is skewed towards the lower tail (smaller values). In the seismic events distribution, the median being immediately above the goal indicates that about half of the distribution is above the goal and about half is below the goal.
Conditional containment failure probability - pressure boundary definition Goal: < 0.1	The mean estimate for internal events is above the goal. The mean estimate for seismic events is above the goal. Therefore, the goal is not met in both cases. The uncertainty distribution for internal events bridges the goal with most of the distribution being above the goal; only the lower tail is below the goal. The seismic events distribution is far above the goal. The width of the internal events distribution is wide and the width of the seismic events distribution is narrow. The internal events distribution is skewed towards the lower tail (smaller values) and the seismic events distribution is symmetrical. In the internal events distribution, the median being slightly above the goal suggests that the bulk of the distribution is slightly above the goal.

Table 19.9-5
(continued)

EPRI ALWR Req	Statement
Core damage frequency Goal: $< 1x10^{-5}$	For internal events, the mean estimate is below the goal, saying that the goal is met. For seismic events, the mean estimate is above the goal, saying that the goal is not met. The uncertainty distribution for internal events is below the goal. The uncertainty distribution for seismic events bridges the goal; the upper tail is one order of magnitude above the goal. The width of the internal events distribution is narrow and the width of the seismic distribution is moderate. The internal events distribution is symmetrical and the seismic events distribution is skewed towards the lower tail (smaller values). In the seismic events distribution, the median being below the goal and the skewness suggest that although the bulk of the distribution is below the goal, a fair amount is above the goal.
Containment failure probability - dose definition Goal: $< 1x10^{-6}$	For internal events, the mean estimate is below the goal, saying that the goal is met. For seismic events, the mean estimate is above the goal, saying that the goal is not met. The uncertainty distribution for internal events is below the goal. The uncertainty distribution for seismic events bridges the goal; the upper tail is two order of magnitude above the goal. The width of the internal events distribution is narrow and the width of the seismic events distribution is moderate. The internal events distribution is symmetrical. The seismic events distribution is skewed towards the lower tail (smaller values). In the seismic events distribution, the median being immediately above the goal indicates that about half of the distribution is above the goal and about half is below the goal.

Conditional containment failure probability - dose definition Goal: < 0.1	The mean estimate for internal events is above the goal. The mean estimate for seismic events is above the goal. Therefore, the goal is not met in both cases. The uncertainty distribution for internal events bridges the goal with most of the distribution being above the goal; only the lower tail is below the goal. The seismic events distribution is far above the goal. The width of the internal events distribution is wide and the width of the seismic distribution is moderate.	The internal events distribution is skewed towards the lower tail (smaller values). The seismic events distribution is symmetrical. In the internal events distribution, the median being slightly below the goal indicates that about half of the distribution is below the goal; but the mean above the goal and the skewness suggest a large amount of the distribution above the goal.

Table 19.9-6 shows the impact of the overpressure protection system on the containment failure probability and the conditional containment failure probability. As previously stated, venting events have been included as a portion of the conditional containment failure probability. We intend to separate these issues in the FSAR. Only the dose definition of containment failure (i.e., probability of a dose of 25 rem at 1/2 mile from the plant) was used in computing the table. In Table 19.9-6, using the dose definition of containment failure, the reduction in the containment failure probability and the CCFP due to the overpressure protection system was insignificant for internal event and more pronounced for seismic events.

Missing discussion about containment failure and definitions

Missing is a note reminding the reader how to interpret the results

Table 19.9-6 Staff Estimates of Containment Failure Probability and Conditional Containment Failure Probability Showing the Effect of the Overpressure Protection System.

Measure	Goal	Internal Events		Seismic Events	
		No Venting	Venting	No Venting	Venting
Containment failure probability - dose definition	$< 1x10^{-5}$ (NRC) $< 1x10^{-6}$ (EPRI)	$8.0x10^{-8}$	$7.6x10^{-8}$	$4.4x10^{-5}$	$3.2x10^{-5}$
Conditional containment failure probability - dose definition	< 0.1	0.14	0.13	0.60	0.44

In regards to the impact of the evacuation and site assumptions in the seismic events analysis, the staff calculated some of the risk measures again, with other values of these assumptions.

<u>Evacuation assumption</u>. Currently, the staff's calculations are done for an average site (averaged consequence results of the Salem and Zion sites) with 0 percent evacuation. Additional calculations, based on an evacuation assumption of 99.5 percent, were performed for the average site. Individual risk of early fatality for 0 percent evacuation is below the goal (see Table 19.9-4); whereas for the 99.5 percent evacuation, this risk measure estimate decreased about two orders of magnitude further below the goal. Individual risk of cancer fatality is virtually unchanged for either evacuation assumption. The distribution of the probability of one or more early fatalities is slightly overlapping the goal; by assuming 99.5 percent evacuation, this risk measure estimate decreased by about one order of magnitude.

<u>Site assumption</u>. Currently, the staff's calculations are done for an average site (averaged consequence results of the Salem and Zion sites) with no evacuation. Additional calculations were done for a high consequence site, the Zion site, and keeping the evacuation assumption at 0 percent. Individual risk of early fatality is below the goal (See Table 19.9-4); by assuming the high consequence site, this risk measure estimate doubled, but it is still below the goal. Again individual risk of cancer fatality is below the goal and remained unchanged. The distribution of the probability of one or more early fatalities is touching the goal; by assuming the high consequence site, this risk measure estimate doubles, but this is still of no concern.

In regards to the CCFP, a careful examination must be made of the factors going into its estimation. The CCFP is defined as follows:

$$CCFP = \frac{\sum_i \sum_k F_i P_{ik}}{\sum_i F_i}$$

where CCFP = frequency weighted conditional containment failure probability.

F = core damage frequency.

P = conditional containment failure probability.

i = containment failure mode.

k = accident class.

The current low value of CCFP was achieved by eliminating, through the design of the ABWR, threats to the containment, such as hydrogen combustion, core debris directly contacting the shell (known as liner meltthrough), and basemat failure. The remaining threats, such as rapid pressurization from direct containment heating and fuel/coolant interaction, are largely inherent to pressure-suppression type containments. Considering those sequences that still occur but do not cause a failure of the containment, the reductions in the CCFP approach a limit, defined by those sequences that cannot be addressed through design. It is apparent that the CCFP becomes an increasingly difficult measure to interpret as the core damage frequency becomes small.

Discussion about the deterministic ~~aspect~~ measure is missing -

9.5 Conclusions

1. The staff concludes that the NRC's quantitative health safety goals for the individual risk of early fatality and individual risk of cancer fatality along with the goal of $1X10^{-6}$/yr for the probability of one or more early fatalities can be met for both internal and seismic events. The staff further believes that these conclusions would still apply even if the containment threats discussed in Sections 19.6.4.2.1, 19.6.4.2.3, and 19.6.4.2.4, and Conclusions 1.(b) and 1.(c) of Section 19.6 were found to be significant primarily because of the very low core damage frequencies predicted by GE and the staff. Table 19.9-5 has an interpretation of the uncertainty that was calculated in these risk measures.

2. The staff concludes that for internal events, the NRC ALWR goals of $1x10^{-4}$ for the mean core damage frequency and $1.0x10^{-6}$/yr for the containment failure probability (pressure boundary definition) can be met. The staff believes that for seismic events, the mean core damage frequency is met while the containment failure probability (pressure boundary definition) goals is not met. The mean

conditional containment failure criterion of 0.1 would not be met for either internal events or seismic events (see Section 19.6.5, Conclusion 1.(a)). Table 19.9-5 has an interpretation of the uncertainty that was calculated in these risk measures.

3. The staff concludes that for internal events, the EPRI goals of $1x10^{-5}$/yr for the core damage frequency and $1x10^{-6}$/yr for the containment failure probability (dose definition) can be met. The staff believes that for seismic events, the goals are not met. The CCFP of 0.10 is not met. Table 19.9-5 has an interpretation of the uncertainty that was calculated in these risk measures.

4. The staff believes that the CCFP goal ($\leq$ 0.10) in SECY90-016 (Reference 19.4), construed as a measure of the plant's defense-in-depth, would very likely not be met given the uncertainties associated with estimating the goal (see Section 19.9.4).

Conclusion about ATWS missing

Conclusion about limited ST analysis
indicated difficulty meeting 24 hrs

19.10 OPEN ITEMS

"Open items" in the review of the PRA can be divided into four categories: confirmatory items, staff corrections, deficiencies, and interface requirements.

"Confirmatory items" are defined as areas where the staff does not necessarily disagree with GE's submittal, but additional clarification or demonstration is required.

"Staff corrections" are defined as areas where the staff does not agree with a specific numerical value or calculation in GE's submittal, and has substituted its own value or analysis to investigate the sensitivity of the results to this value or calculation.

"Outstanding items" are defined as items where the staff (a) disagrees with the submittal, (b) requires additional supporting documentation, and/or (c) has not yet completed its review and evaluation.

"Interface requirements" are defined as areas which must be confirmed for a specific application for a construction permit.

The open items found in this review are summarized in Tables 19.10-1 through 19.10-4.

Table 19.10-1 Confirmatory Items

Confirmatory items are defined as areas where the staff does not necessarily disagree with GE's submittal, but additional clarification or demonstration is required.

1. The PRA documentation exhibits inconsistency with respect to its functionality for the IORV event, in that the event trees are correct, but the accompanying text is ambiguous. (See discussion in Section 19.3.3, "Success Criteria", page 19-23.)

2. Additional investigation is currently underway to determine the logical minimum injection flow to the vessel needed to avoid core damage following a vessel isolation event coupled with failure to scram and failure to provide poison injection. Preliminary calculations indicate that a flow rate of 800 gpm from a HPCF train alone may not be sufficient to keep the water level above the top of the active fuel for the above scenario. Meanwhile, the staff has used GE's success criteria for the MSIV closure event in its requantification of the ATWS-induced sequence frequencies. However, if the final thermal-hydraulic calculations demonstrate a need for two or more trains of the high pressure injection systems (that is, more than an 800 gpm flow rate) to avoid core damage for the above scenario, then the overall ABWR core damage frequency and risk could increase significantly. (See discussion under Section 19.3.3, "Success Criteria," page 19-24.)

3. GE should provide documentation on the justification regarding the applicability of certain generic common cause/mode failure data to ABWR design-specific components (the diesel generators, the HPCF pumps, the LPCF pumps, and the RHR heat exchangers) involved in the system unavailability modeling. (See discussion under Section 19.3.6.1, "Hardware Reliability Data Analysis," page 19-27.)

4. GE should provide justification regarding the applicability of GESSAR II design information to the ABWR design (on a train basis) for use in test and maintenance data analysis. (See discussion under Section 19.3.6.2, "Test and Maintenance Data Analysis," page 19-28.)

5. The staff has concluded that GE has developed a reasonable plan to use information and insights gained from the HRA to support the system/operational design. The acceptability of any insights realized from the HRA however, must await further design development. (See Section 19.3.7.2.12, page 19-36).

6. GE should provide the calculations it performed to show suppression pool flashing following venting does not lead to significant fission product releases. (See Section 19.7.4.3, page 19-140)

Table 19.10-2 Staff Corrections

Staff corrections are defined as areas where the staff does not agree with a specific numerical value or calculation in GE's submittal, and has substituted its own value or analysis to investigate the sensitivity of the results to this value or calculation.

1. GE has provided neither highlights of the ABWR design improvements in the balance of plant (BOP) systems nor applicable references to such BOP improvements in the ABWR PRA to support the estimate of only one reactor trip per year, which is lower than current experience in the U.S. Due to lack of design details at this stage, the staff has used one event per year for the loss of feedwater frequency and one event per year for the MSIV closure event frequency in its review of the ABWR PRA. Unless GE can provide more justification, the staff finds its own estimates, rather than GE's, to be appropriate at this design stage. (See Paragraph 2 of Section 19.3.2, "Initiating Event Frequency," page 19-21.)

2. GE's estimate for the inadvertent open relief valve (IORV) frequency is about 0.01 per reactor-year. The staff notes that this estimate is substantially lower than the value (0.07 per reactor-year) used for the Limerick plant (Reference 19.23). GE has not provided detailed documentation regarding any design improvements made to the multi-stage relief valves to be installed in the future to support this lower unreliability value. In the absence of evidence to the contrary, the staff has used a higher value (0.1) for the IORV event for its independent assessment. (See Paragraph 3 of Section 19.3.2, "Initiating Event Frequency," page 19-22.)

3. The staff does not agree with the treatment of Class II sequences in the ABWR PRA. More than 86 percent of the Class II sequences are type (i) events, which have already employed fire water as the only available means of low pressure core cooling. For these Class II events, therefore, it is considered wrong to give credit to fire water once again following loss of core cooling in the seismic containment event tree. (See Section 19.4.3.1.3, Page 19-55, Section 19.4.3.5, page 19-69, and Section 19.4.3.6.1, page 19-71.)

4. The staff noted that GE did not explicitly analyze containment venting in the ABWR PRA. When giving credit to containment venting, the staff value for mean annual frequency for Class II events was reduced by an order of magnitude to 5.7 E-07 as compared to the ABWR PRA value of 4.8 E-06. (See Section 19.4.3.6.1, page 19-70.)

5. The seismic capacity of the fuel assemblies was calculated by GE as corresponding to a center deflection of 55 mm, at which scram can be achieved. However, the moment corresponding to this deflection is not the collapse moment as used in the calculations. It is some value between the yield moment and the collapse moment. Therefore the median

ultimate capacity of the fuel assemblies is less than the median value of 1.3g. In a sensitivity study, the staff has used a value of 0.92g median capacity to estimate the accident sequence frequencies. (See Section 19.4.3.3.2, page 19-60.)

6. Although the ABWR PRA report states that the median capacity of large, flat-bottom storage tanks is 2.1g with β_c of 0.45, the only tank used in the seismic system analysis is the fire water tank, with a generic assigned median capacity of 2.8g and a β_c of 0.45. This makes the HCLPF capacity equal to 0.98g. Experience with design and actual performance of these large yard tanks is that this high capacity is not generally achieved. Therefore, use of a median value of 1.43g with a HCLPF capacity of 0.50g was made in a sensitivity analysis. (See Section 19.4.3.3.2, page 19-62.)

7. GE has assigned a median capacity of 2.5g with a β_c of 0.45 to the diesel generators. This means that the HCLPF capacity is about 0.88g. Although diesel generators by themselves have high seismic capacities, the peripheral equipment required for the diesel generators to operate can have low capacities. Therefore, the staff concludes that the diesel generator fragility is rather optimistic. In a later sensitivity study, the staff has assigned a much lower capacity (median of 1.5g and HCLPF of 0.47g) to diesel generators in acknowledgement of lower capacity components in the system. (See Section 19.4.3.3.2, page 19-62.)

8. The seismic capacities assigned to active electrical equipment in the structural failure mode are generally higher than the specific capacities calculated in previous seismic PRAs. For example, the HCLPF capacities of motor control centers, relay switches, and battery and battery racks appear to be too high. (See Section 19.4.3.3.2, page 19-63.)

9. The staff has initially included within the containment failure category those events where the overpressure protection system actuates within 24 hours. In the final SER, we intend to separate the venting issues, since we do not automatically view venting prior to 24 hours as containment failure. This will affect the final values of conditional containment failure probability. (See Section 19.9.4, page 19-157.)

10. LLNL seismic hazard curve will be used in the seismic analysis. The staff's analysis used the LLNL hazard curve, which generally provide a much higher core damage frequency than either the GE or the EPRI curves, while the latter two curves give rise to core damage frequencies of about the same magnitude. However, the uncertainty ranges of the LLNL curve are large and encompass the other two curves. (See Section 19.6.4.1, page 19-105, and Section 19.9.4 page 19-158.)

11. Credit was taken for the firewater system in both the Level 1 and the Level 2 portions of the PRA, instead of just in the Level 2 analysis as did GE. The staff took credit for preventing core damage with the firewater addition system in Level 1. If core damage occurs (Level 2),

then the firewater system could not have been available to <u>prevent</u> core damage, hence, it unlikely to be available to <u>arrest</u> a core melt in the reactor vessel. However, later in the accident progression, the staff took credit for arresting an ex-vessel core melt progression because there is more time available to restore the firewater system than for the in-vessel situation. (See Section 19.6.4.1, page 19-106.)

12. ATWS will be treated as a late containment failure or no containment failure, instead of as an early containment failure (internal events) and late containment failure (seismic events). GE's treatment was apparently done because the frequency of the Class IV accidents is small, less than 2 percent in GE's internal events analysis. Thus, they were conservatively grouped with other sequences which resulted in the largest release of fission products (early containment failure). However, since its frequency in the seismic events was significantly higher, 60 percent, GE performed a more thorough analysis, where most of the sequences resulted in late containment failure. The staff estimated fraction of the ATWS sequences to be higher than GE's estimate: the larger fraction warranted further study of the progression of these accidents, which showed assumptions differing from those of GE in their CET analysis, stemming differing views of accident progressions. The accounting of the ATWS sequences has an effect on the results of the CET analysis. (See Section 19.6.4.1, page 19-106.)

Table 19.10-3 Outstanding Items

Outstanding items are items where the staff (a) disagrees with the submittal, (b) requires additional supporting documentation, and/or (c) has not yet completed its review and evaluation.

1. GE must justify the use of RPS reliability estimates for the Clinton facility in the ABWR PRA. (See Paragraph 1 of Section 19.2.1, "ABWR Safety System Features," page 19-6.)

2. GE must evaluate the impact of the partial and/or total failure of the support systems on plant trips as applicable, including subsequent dependent failures of the mitigating systems needed to provide a vessel coolant makeup function and/or containment heat removal function. (See Paragraph 4 of Section 19.3.2, "Initiating Event Frequency," page 19-22.)

3. GE must provide an accident analyses of postulated interfacing LOCA events as applicable to the ABWR design. (See Paragraph 7 of Section 19.3.2, "Initiating Event Frequency," page 19-23.)

4. GE must provide results of accident analyses of LOCA events outside the containment (in particular, steam line breaks in the RCIC steam piping and the RWCU lines) in combination with failure of the isolation valves. (See Paragraph 8 of Section 19.3.2, "Initiating Event Frequency," page 19-23.)

5. GE must provide justification to support its claim that the train-level rather than component level common mode failure analysis captured the full contribution to common mode failure probability. (See discussion under Section 19.3.5, "System Modeling," page 19-26.)

6. GE must justify the use of reliability data for the mechanical failure probability of the RHR pump and the failure probability of the HPCF pump (See discussion under Section 19.3.6.1, "Hardware Reliability Data Analysis," page 19-27.)

7. GE must provide the HRA-related documentation used to evaluate the approach(es) taken to model human action in the ABWR PRA. The documentation must include the following for each human action modelled: an appropriate task analysis, a description of how the appropriate HRA analysis method(s) were implemented, a discussion of what performance models and performance shaping factors were used, and how human actions were quantified (HEP determined). (See Section 19.3.7.2.1, page 19-31.)

8. GE must describe how each identified HRA method is utilized to develop each individual HEP in the ABWR PRA. (See Section 19.3.7.2.6, page 19-34.)

9. GE must provide a concise discussion of the use of generic data sources for HEP estimation. In addition, GE must justify the use of these generic sources of human error data which are based upon simple manual control tasks for the monitoring and supervisory control tasks of the ABWR operator. (See Section 19.3.7.2.9, page 19-35.)

10. GE must provide the results of HEP uncertainty and/or sensitivity analyses performed in support of the ABWR PRA and provide the criteria that were used for performing such analyses. (See Section 19.3.7.2.11, page 19-36.)

11. GE must perform an uncertainty analysis to address the uncertainty in the relative contributions of the various initiating events to the total core damage frequency. (See Section 19.3.10.1, page 19-38.)

12. GE must perform a severe accident fire analysis. (See Section 19.4.1, page 19-50.)

13. GE must factor the drywell-wetwell bypass into the CETs and provide an analysis of the bypass giving consideration to such matters as (1) the basis to support an allowed leakage area, (2) the basis to support an expected leakage area during the course of a severe accident when the vacuum breakers would be required to perform several times in a severe environment, and (3) uncertainty in the bypass leakage flow rate. (See Section 19.6.4.2.1, page 19-113.)

14. GE must justify the overpressure protection system and, given the system, the pressure relief setpoint, taking into account downside risks. GE must carry out the necessary analysis of the effects of drywell/wetwell bypass before conclusions can be reached that the system has a net benefit from a risk perspective. (See Section 19.6.4.2.2, page 19-118.)

15. GE must perform an analysis of the risk reduction associated with the passive flooder system, taking into account factors such as (but not limited to) the possibility of it failing to quench core debris, the benefits of keeping the area in the lower drywell above the core debris cool, and potential for fuel/coolant interactions. The analysis should consider whether to introduce water into the lower drywell in a controlled or uncontrolled manner, how fast to introduce the water and when to introduce the water. (See Section 19.6.4.2.3, page 19-126 and Section 19.7.4.2.2, page 19-145.)

16. GE must address the threat to containment integrity from a core/concrete interaction in the event that core debris is not quenched by the overlying water pool. The influence on the source terms due to continued concrete attack should also be addressed (See Section 6.4.2.4, page 19-128 and Section 19.7.4.2, page 19-145.)

17. GE must modify the CETs to take into account the following severe accident phenomena and severe accident features: DCH, fuel/coolant

interaction, continued core/concrete interaction, pool bypass, and vacuum breaker effects. GE should perform uncertainty analyses associated with these phenomena. (See Section 19.6.2.1, page 19-99, Section 19.6.2.2, page 19-100, and Section 19.6.3, page 19-103 and Section 19.6.3, page 19-102.)

18. GE must include, as part of its overall analysis, a treatment of uncertainty since large uncertainties are inherent in these calculations (See Section 19.6.2.1, page 19-99, Section 19.9.3, page 19-155, Section 19.7.2.1, page 19-133, Section 19.7.2.2, page 19-133, Section 19.7.4.2.2, page 19-141, and Section 19.9, page 19-178).

19. GE must provide an evaluation of the probability and consequences of containment penetration lines or containment isolation valves failing during a seismic event. (See Section 19.4.3.8.1, page 19-76.)

20. GE must provide a systematic assessment that identifies plant and procedure vulnerabilities when the plant is in modes other than full power. (See Section 19.3.4, page 19-24.)

21. GE must provide (1) a list of systems that were modeled/considered in the ABWR PRA but are not part of the ABWR certified design, (2) a description of any risk significant assumptions for these systems, and (3) the assumed reliabilities for the systems. (See Section 19.3.5, page 19-25.)

22. GE must provide information which describes (1) how PRA insights were used in the ABWR design process, (2) what ABWR design features, if any, were included as a result of PRA insights to reduce risk significant sequences and phenomena, (3) how plant operating experience was factored into the ABWR PRA, and (4) how PRA insights were used to address severe accident phenomena. (See Section 19.11, page 19-190.)

Table 19.10-4 Interface Requirements

Interface requirements are conditions which are specified for those portions of the plant for which GE does not seek certification. These requirements must be addressed by an individual applicant who references the ABWR design in an application for a construction permit\operating license to ensure that the site and design compatible features satisfy the functional performance and safety requirements of ABWR systems.

1. Confirm the estimate of the loss of AC power event, address site-specific parameters (as indicated in the staff's licensing review basis document), such as specific causes (e.g., a severe storm) of the loss of power, and their impact on recovery of AC power in a timely fashion). (See Paragraph 5 of Section 19.3.2, "Initiating Event Frequency," page 19-22.)

2. Provide documentation which describes the material and/or analysis that were used to support the plant-specific HRA. This should include the following areas: detailed function and task analysis (utilizing ABWR staffing goals and staffing philosophy), procedure guidelines, control room design, and work station and display design. (See Section 19.3.7.2.2, page 19-33.)

3. Provide documentation which describes in detail the human-system analyses used by the HRA team to support the plant-specific HRA/PRA. This should include the following areas: detailed task analyses including task requirements on the operating staff, their interfaces with plant systems and components, and any time constraints for critical task accomplishment. Also, GE did not explain how these analyses supported the inclusion of human actions in the PRA event and fault trees. Finally, GE did not describe the use, if any, of a technique such as screening analysis to help identify important human actions. (See Section 19.3.7.2.3, page 19-33.)

4. Provide documentation, including the supporting task analysis, which describes the modelling of human actions related to the advanced technology of the ABWR control and instrumentation. (See in Section 19.3.7.2.5, page 19-34.)

5. Provide documentation which describes how performance shaping factors (PSFs) were utilized to develop each individual HEP in the plant-specific PRA. (See Section 19.3.7.2.7, page 19-34.)

6. Provide additional documentation on the advanced technology aspects of the ABWR person-machine interface, and the human reliability analysis results, criteria, or guidelines that were used as the basis for automating operator functions so as to change, if not practically eliminate, the operator's role in system operation and response to potential abnormal events. (See Section 19.3.7.2.8, page 19-32.)

7. Provide documentation which justifies the use of GESSAR II PRA human error probabilities in the plant-specific ABWR PRA. (See Section 19.3.7.2.10, page 19-35.)

8. Perform a site-specific design verification for the truly "external" events, such as external floods and transportation hazards, for which no analyses can be performed at this stage. (See Section 19.4.1, Page 19-50.)

9. Perform a probabilistic risk analysis for internal floods. (See Section 19.4.1, Page 19-50.)

10. Confirm the assumed seismic capacities for components and structures for site specific applications and incorporate the generic seismic fragility assumptions in the ABWR design specifications. (See Section 19.4.3.3.1, page 19-58, 61, and Section 19.4.3.3.3, pages 19-60 to 19-65.)

11. Perform an evaluation of the potential for seismic-induced soil failures, such as liquefaction, differential settlement, or slope stability for site specific applications. The seismic PRA should be modified accordingly. (See Section 19.4.3.3.2, pages 19-59 and 19-64 to 19-61.)

12. Perform a walkdown of the final constructed plant. The walkdown should include an assessment of potential seismic vulnerabilities, such as marginal anchorage of equipment and gross deviations from the design documents, and spatial systems interactions, such as operators being disabled due to the failure of the control suspended ceiling in a seismic event. (See Section 19.4.3.3.2, page 19-65.) The walkdown should also confirm that the assumed seismic capacities are met or exceeded for site specific applications. (See Section 19.4.3.3.2, page 19-64.)

13. Develop deterministic and probabilistic site specific response spectra for all sites. Demonstrate that the seismic design response spectra for the plant envelope the deterministic site specific response spectra and the probabilistic site spectra used in the ABWR PRA. If the site-specific deterministic or probabilistic response spectra exceed the spectra assumed in the ABWR PRA, perform a plant-specific seismic PRA to confirm that the dominant sequences identified in the ABWR PRA have not been significantly altered. (See Section 19.4.3.3.2, page 19-65.)

14. Confirm the seismic capacities assigned to active electrical equipment such as motor control centers, relay switches, battery and battery racks in the site specific ABWR PRA. (See Section 19.4.3.3.2, page 19-63.)

15. Demonstrate that the applicant has designed each system (i.e., systems modeled/considered in the ABWR PRA, but not part of the ABWR certified design) to meet the system reliability requirements and risk significant assumptions provided by GE. (See Section 19.3.5, page 19-25.)

19..1 CONCLUSIONS

As was stated in the introduction, the review of a PRA is not governed by explicit formal criteria. The PRA and its evaluation are used to assess, in a realistic rather than conservative manner, the safety profile of the proposed design as expressed in terms of the frequency of severe core damage accidents, the consequences of a spectrum of such accidents of varying severities, and the integrated risk to the public, the uncertainty in these parameters, and insights into the safety profile. In addition, a PRA and its evaluation can be used to help make deterministic judgments of the safety of a proposed design.

The staff's review focused significant attention on the quality of the PRA rather than on insights developed from the PRA. The staff believes, however, that knowledge of how PRA insights were employed in the ABWR design underscores the significance of design features which eliminate dominant contributors to the estimated core damage frequency and offsite consequences, and facilitates a balancing of preventive and mitigative design features. The staff therefore requires GE to provide information which describes (1) how PRA insights were used in the ABWR design process, (2) what ABWR design features, if any, were included as a result of PRA insights to reduce risk significant sequences and phenomena, (3) how plant operating experience was factored into the ABWR PRA, and (4) how PRA insights were used to address severe accident phenomena. This is an outstanding item.

The staff also believes that use of PRA insights may be beneficial in resolving open items which have been identified from the review of other SSAR chapters. Therefore, the staff expects GE to employ PRA insights, where feasible, to support issue resolution.

The staff's review of the Level 1 analysis of the core damage frequency due to internally generated events uncovered a number of deficiencies (See Section 19.10 above). Given the existence of these deficiencies, GE's analysis has somewhat underestimated the core damage frequency. Moreover, the identification and ordering of the set of sequences which constitute the principal contributors to the core damage frequency are suspect, since support system failure initiators are missing. It does appear, however, that the internally-generated core damage frequency is quite low. In addition, the analysis did not find any highly dominant accident sequence classes.

The staff review of the Level 1 analysis of the core damage frequency due to external events also uncovered a number of problems. Most notably, GE has not submitted a probabilistic analysis of internal fire initiated accident sequences. Thus, there is no realistic (as opposed to bounding) analysis of internal fires, and no statements as to the relative importance of these sequences in the makeup of the total core damage frequency can be made. It does appear, however, that the ABWR design possesses considerable seismic margin at the 0.3g design basis earthquake level. The actual seismically-induced core damage frequency will have to be calculated for specific applications, since it is highly site dependent.

The staff's review of the Level 2 and 3 analyses has indicated that more effort by GE and the staff is needed before it can be concluded that the severe accident sequences for the ABWR plant have been sufficiently analyzed. Therefore, the adequacy of the level of defense-in-depth provided for in the ABWR containment system design cannot be conclusively judged at this time, based on the information submitted. Major issues are: (a) the absence of an uncertainty analysis present an incomplete picture of the accident progression; (b) the limited size of the containment event trees; (c) system models have not been fully supported, such as the effects of water from the flooder system on core/concrete interaction in regards to the pedestal integrity are uncertain; (d) the assumptions in the analyses treat accident progression events differently, i.e., core/concrete interaction, thus causing differences in risk results and potential accident management strategies.

Notwithstanding the issues described above, the following statements can be made:

(1) According to GE's analysis, the CCFP goal of ≤0.1 which is intended to provide insights into a plant's level of defense-in-depth is met for internal events, whereas the staff's rough uncertainty calculation indicates the mean value to be 0.13, the 95th percentile to be 0.70, and the 5th percentile to be 0.05. Notwithstanding a question as to whether the ABWR meets this specific goal, the staff's analysis indicates that implications drawn based on this CCFP goal need to be carefully considered. It is possible that a plant with a very low total core damage frequency may not meet the 0.10 goal because low frequency threats to the containment are difficult to rule out and could cause the CCFP to exceed the goal of 0.10. Since the ABWR plant's core damage frequency is estimated by GE and us to be very low, the above considerations need to be addressed, especially when conclusions regarding the plant's level of defense-in-depth is being considered in light of this goal alone.

(2) Based on the staff's limited source term analysis, the staff believes that the ABWR design could have difficulty meeting GE's criterion of no containment failure (i.e., 25 rem at the boundary) before 24 hours.

(3) The staff believes that mitigative systems such as the lower drywell passive flooder system and the containment overpressure protection system should be designed to perform their function with a reasonable amount of confidence if they are to be considered viable lines of defense. Specifically:

- o The design of the passive flooder system and the lower drywell should be such that degradation of the pedestal wall from a core/concrete interaction and possible damage from an ex-vessel fuel/coolant interaction does not compromise the integrity of the wall and, hence, pose a threat to the containment.

- o The overpressure protection system, which is designed to provide for a controlled release of scrubbed fission products, should address the bypass flows from the drywell to the wetwell airspace, including the design of the drywell/wetwell spray system.

(4) The individual risk of early fatality and individual risk of cancer fatality calculated for both internal and seismic events by GE and the staff were orders of magnitude below NRC's quantitative health safety goals. Since similar results for internal events were also found in NRC's NUREG-1150 study (Reference 19.62) for the BWR Peach Bottom and Grand Gulf plants, it would appear to indicate that notwithstanding the importance of meeting these goals, conformance with these goals alone does not necessarily indicate an improvement in risk for the ABWR design over that of several generations past.

19.12 REFERENCES

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19.2 Electric Power Research Institute, "Advanced Light Water Reactor Requirements Document," Palo Alto, California, December 1987.

19.3 U.S. Nuclear Regulatory Commission, "Resolution Process for Severe Accident Issues on Evolutionary Light Water Reactors," Commission Paper SECY-89-311, dated December 15, 1989.

19.4 U.S. Nuclear Regulatory Commission, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," Commission Paper SECY-90-016, dated January 12, 1990.

19.5 General Electric, "Amendment 4 to Chapter 19 of the ABWR Safety Analysis Report," Docket 50-605, January 27, 1989.

19.6 Letter, D. C. Scaletti, NRC, to P. W. Marriott, General Electric Company, "Request for Additional Information regarding the General Electric Company Application for Certification of the ABWR Design," dated November 28, 1989.

19.7 Letter, J. Fox, GE, to D. C. Scaletti, NRC, dated January 9, 199[illegible]

19.8 Letter, J. Fox, GE, to D. C. Scaletti, NRC, dated January 11, 1990.

19.9 General Electric, "Amendment 8 to Chapter 19 of the ABWR Safety Analysis Report," Docket 50-605, July 28, 1989.

19.10 Letter, M. Carmel (SNL) to Jae Jo (BNL), November 12, 1990.

19.11 General Electric, "Amendment 10 to Chapter 20 of the ABWR Safety Analysis Report," Docket 50-605, March 28, 1990.

19.12 General Electric, "Amendment 9 to Chapter 19 of the ABWR Safety Analysis Report," Docked 50-605, November 17, 1989.

19.13 Letter, D. C. Scaletti, NRC, to P. W. Marriott, General Electric Company, "Request for Additional Information regarding the General Electric Company Application for Certification of the ABWR Design," dated May 1, 1990.

19.14 General Electric, "Amendment 13 to Chapter 20 of the ABWR Safety Analysis Report," Docket 50-605, July 3, 1990.

19.15 General Electric, "Amendment 14 to Chapter 20 of the ABWR Safety Analysis Report," Docket 50-605, dated October 2, 1990.

19.16 General Electric, "Amendment 13 to Chapter 18 of the ABWR Safety Analysis Report," Docket 50-605, July 3, 1990.

19.17 Letter, D. C. Scaletti, NRC, to P. W. Marriott, General Electric Company, "Request for Additional Information regarding the General Electric Company Application for Certification of the ABWR Design," July 30, 1990.

19.18 General Electric, "Amendment 14 to Chapter 20 of the ABWR Safety Analysis Report," Docket 50-605, August 31, 1990.

19.19 Memorandum, E. S. Beckjord and Thomas Murley to Victor Stello, "Memorandum of Agreement of the RES Role in the Review of the Standard Plant Design-NRR/RES," March 14, 1988.

19.20 A Review of the General Electric ABWR Probabilistic Risk Assessment, NUREG/CR-5676P, to be published.

19.21 "Anticipated Transients without Scram for Light Water Reactors," NUREG-0460, U. S. Nuclear Regulatory Commission, April 1978.

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