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Enclosure 1 contains the subject partial ESBWR Probabilistic Risk Assessment (PRA) document (Revision 1). The complete PRA document (including the sections transmitted herein) is scheduled to be issued in May 2006.

If you have any questions about the information provided here, please let me know.

Sincerely,

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Enclosure:

1. MFN 06-111 – NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment:"
    - Section 8 – Containment Performance
    - Section 9 – Source Terms
    - Section 10 – Consequence Analysis

cc:	WD Beckner	USNRC (w/o enclosures)
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ENCLOSURE 1

MFN 06-111

NEDO-33201, Revision 1, "ESBWR Probabilistic Risk Assessment"

- **Section 8 – Containment Performance**
- **Section 9 – Source Terms**
- **Section 10 – Consequence Analysis**

8 CONTAINMENT PERFORMANCE

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8 CONTAINMENT PERFORMANCE

8.0 INTRODUCTION

A spectrum of potential containment failure modes has been evaluated for the ESBWR. In Section 7, the potential for a break outside of containment was evaluated. In Section 21, potential ex-vessel steam explosion, direct containment heating and basemat penetration challenges were evaluated. In this section, the focus is on the containment challenges associated with potential combustible gas deflagration, overpressurization and bypass. The potential for containment failure due to these challenges is addressed by considering physical characteristics of the containment, notably the inerted condition and containment structural capability, as well as the reliability of passive systems engineered to perform the containment functions of "isolation", "vapor suppression" and "heat removal".

Containment failure due to combustible gas deflagration is shown to be unrealistic considering the inerted containment and time period required to generate enough oxygen to create a combustible gas mixture. The probability of containment failure due to overpressure or bypass requires consideration of the reliability of engineered systems used to isolate the containment and mitigate containment pressurization associated with a severe accident. As will be seen, the containment capability and system reliability are such that the calculated probability of containment bypass and overpressurization can be considered to be negligible.

Consistent with the NRC design certification policy for advanced reactors discussed in Reference 8.0-1, the containment response has been evaluated for a 24-hour period following the onset of core damage. To provide additional insight on the containment performance objective discussed in the reference, containment effectiveness will be quantified to demonstrate that the containment provides a reliable barrier to radionuclide release after a severe accident.

Section 8.1 discusses the potential for combustible gas deflagration. Section 8.2 evaluates the probability of containment overpressurization and bypass. Section 8.3 presents the computer simulation results of containment response to overpressurization challenges. Section 8.4 summarizes key insights from the evaluation. Appendix A.8 quantifies the frequency of all release categories discussed in this section as well as those in Section 21. Appendix B.8 presents the analysis of containment ultimate strength. Appendix C.8 provides the screening analysis to support quantification of the containment isolation system probability.

The results developed in this section, as well as from Section 21, are used to develop conservative source terms in Section 9 for use in the offsite consequence analysis. The offsite consequence analysis is presented in Section 10.

Table 8.0-1 summarizes acronyms and terminology used in this section.

Table 8.0-1

Acronyms and Terminology

General	
ADS	Automatic Depressurization System
BiMAC	Basemat Internal Melt Arrest and Coolability (Device)
CCI	Core Concrete Interaction
CSET	Containment System Event Tree
FAPCS	Fuel and Auxiliary Pools Cooling System
GDCS	Gravity Driven Cooling System
ICS	Isolation Condenser System
PCCS	Passive Containment Cooling System
VB	Vacuum Breaker
Sequence Nomenclature	
MLi	Medium LOCA (GDCS injection line)
T	Transient (e.g., MSIV closure, loss of AC)
T-AT	Transient with failure to insert negative reactivity
nCHR	no Containment Heat Removal
nD	no Deluge
nDP	no Depressurization
nIN	no core Injection
nVB	no Vacuum Breaker (vacuum breaker failure to close)
Containment Release Categories	
BOC	Break Outside of Containment (Connecting RPV to environment)
BYP	Containment Bypass (Connecting containment to environment)
FR	Filtered Release (Through controlled suppression pool venting)
OP	Overpressure (General category)
OPW1	Overpressure due to failure of short-term containment heat removal
OPW2	Overpressure due to failure of long-term containment heat removal
OPVB	Overpressure due to failure of Vacuum Breaker
TSL	Technical Specification Leakage

8.1 POTENTIAL FOR FAILURE DUE TO COMBUSTIBLE GAS DEFLAGRATION

Because the ESBWR containment is inerted, the prevention of a combustible gas deflagration is assured in the short term following a severe accident. In the longer term there would be an increase in the oxygen concentration resulting from the continued radiolytic decomposition of the water in the containment. Because the possibility of a combustible gas condition is oxygen limited for an inerted containment, it is important to evaluate the containment oxygen concentration versus time following a severe accident to assure that there will be sufficient time to implement severe accident management (SAM) actions. It is desirable to have at least a 24-hour period following an accident to allow for SAM implementation. This section discusses the rate at which post-accident oxygen will be generated by radiolysis in the ESBWR containment following a severe accident, and establishes the period of time that would be required for the oxygen concentration in containment to increase to a value that would constitute a combustible gas condition (5% oxygen by volume) in the presence of a large hydrogen release, thus de-inerting the containment in the absence of mitigating SAM actions.

8.1.1 Background

The rate of gas production from radiolysis depends upon the power decay profile and the amount of fission products released to the coolant. Appendix A of Standard Review Plan (SRP) Section 6.2.5 (Reference 8.1-1) provides a methodology for calculation of radiolytic hydrogen and oxygen generation. The analysis results discussed herein were developed in a manner that is consistent with the guidance provided in SRP 6.2.5 and Regulatory Guide 1.7 (Reference 8.1-2).

There are unique design features of the ESBWR that are important with respect to the determination of post-accident radiolytic gas concentrations. In the post-accident period, the ESBWR does not utilize active systems for core cooling and decay heat removal. As indicated earlier, for a design basis loss-of-coolant accident (LOCA), the ADS would depressurize the reactor vessel and the GDCS would provide gravity driven flow into the vessel for emergency core cooling. The core would be subcooled initially and then it would saturate resulting in steam flow out of the vessel and into the containment. The PCCS heat exchangers would remove the energy by condensing the steam. This would be the post-accident mode and the core coolant would be boiling throughout this period.

A similar situation would exist for a severe accident that results in a core melt followed by reactor vessel failure. In this case, the GDCS liquid would be covering the melted core material in the lower drywell, with an initial period of subcooling followed by steaming. The PCCS heat exchangers would be removing the energy in the same manner as described above for a design basis LOCA.

In order to prevent non-condensable related termination of steam condensation, the PCCS heat exchangers are provided with a vent which will transfer any non-condensable gases which accumulate in the heat exchanger tubes to the suppression pool vapor space, driven by the drywell to suppression pool pressure differential. In this way, the majority of the non-condensable gases will be in the suppression pool. The calculation of post-accident

radiolytic oxygen generation accounts for this movement of non-condensable gases to the suppression pool after they are formed in the drywell.

The effect of the core coolant boiling is to strip dissolved gases out of the liquid phase resulting in a higher level of radiolytic decomposition. This effect was accounted for in the analysis.

8.1.2 Analysis Assumptions

The analysis of the radiolytic oxygen concentration in containment was performed consistent with the methodology of Appendix A to SRP 6.2.5 and Regulatory Guide 1.7. Some of the key assumptions are as follows:

Reactor power was 102% of rated

$G(O_2) = 0.25$ molecules/100eV

Initial containment O_2 concentration = 4%

Allowed containment O_2 concentration = 5%

Stripping of drywell non-condensable gases to wetwell vapor space

Fuel clad-coolant reaction up to 100%

Iodine release up 100%

Adequate gas mixing throughout containment

8.1.3 Analysis Results

The analysis results show that the time required for the oxygen concentration to increase to the de-inerting value of 5% is significantly greater than 24 hours for a wide range of fuel clad-coolant interaction and iodine release assumptions up to and including 100%. Thus, the potential for containment failure due to combustible gas deflagration will not be discussed further.

8.2 FREQUENCY OF OVERPRESSURE AND BYPASS RELEASE CATEGORIES

The containment bypass (BYP) failure mode represents the failure to isolate containment before or during a severe accident, thus allowing a radionuclide barrier to be breached. The containment overpressure (OP) failure mode represents the potential for containment pressurization from stored energy and decay heat to exceed the ultimate containment strength. The likelihood of these failure modes was evaluated with "Containment System Event Trees" (CSETs). The end state of a CSET is one of the following potential release categories. The first group depicts containment failure:

- Containment bypass (BYP) represents the condition in which the containment has been bypassed due to failure of the Containment Isolation System. With this failure mode, the containment is assumed to be unavailable as a radionuclide barrier from the start of the severe accident, i.e., the containment isolation function has failed. As a result, there is a direct path from the containment atmosphere to the environment.
- Overpressurization (OPW) represents the condition in which the vapor suppression capability has functioned, but there has been a failure to remove energy from the containment, i.e., the containment heat removal function has failed. Two modes of containment heat removal failure are considered. Short term failure (within 24 hours of accident initiation) is defined as "OPW1" category; long term failure (after 24 hours) is defined as "OPW2".
- Overpressurization due to vacuum breaker failure (OPVB) represents the condition in which a vacuum breaker is open or fails to reclose, which is assumed to defeat the containment's vapor suppression function. In such a situation, containment overpressure occurs earlier than in the OPW failure mode.

Also shown on the CSETs are two end states, which are not considered containment failure because they do not result in the loss of control of the containment boundary:

- Technical Specification Leakage (TSL) represents the condition in which the containment pressure boundary is intact and the only source term is that associated with the allowable leakage rate, as defined by the Technical Specifications.
- Filtered release (FR) is an end state depicting containment venting under operator control. Such a release results in a much lower radionuclide source term than containment failure because the radionuclide pathway is through the suppression pool, which provides filtering of the radionuclides.

The CSETs are discussed in more detail in Section 8.2.1.

8.2.1 Containment System Event Trees

The Level 1 analysis, described in earlier sections, evaluated severe accident sequences with the potential to cause core damage. The core damage frequency associated with each of these sequences is discussed in Appendix A.8. In that appendix, the core damage sequences were grouped according to their similarity and potential containment challenge so that a manageable number of sequences could be evaluated in terms of the containment response.

The class definition and contribution of each accident class to the core damage frequency is summarized as follows:

Accident Class	CDF contribution (per year)	Percentage CDF contribution	Class summary
Class I	2.87 E-8	98.23	Sequences with RPV failure at low pressure
Class II	0.0	0.0	Sequences with containment failure preceding core damage
Class III	3.29 E-10	1.13	Sequences with RPV failure at high pressure
Class IV	1.83E-10	0.63	Sequences involving failure to insert negative reactivity
Class V	4.27E-12	0.01	Sequences involving containment failure due to interfacing systems LOCA (Break outside of containment)

To evaluate the containment response to a severe accident, two types of containment event trees were used to evaluate the complete spectrum of potential challenges to containment integrity. The "Containment Phenomenology Event Trees" (CPETs) were found to be most useful for the phenomenology, as discussed in Section 21. The "Containment System Event Trees" (CSETs) were found to be most useful for evaluating the containment response to bypass and overpressurization events. The CSETs relate the entry event to the containment systems designed to mitigate such an event. The containment is evaluated with a CSET for failure due to overpressurization or bypass if failure by other mechanisms can be ruled out. The low probability of failure from other mechanisms indicates that most of the core damage frequency translates to the entry event of the CSET.

The number of CSETs needed to evaluate the overpressurization and bypass failure modes for the Level I accident classes, was established with the following considerations:

- Class II sequences, by definition, ultimately result in containment failure prior to core damage; thus, an event tree is not required to evaluate the probability of containment failure. Indeed, Class II events do not require evaluation because they do not result in core damage within the mission time, as illustrated in Section 8.3.2.2.
- Class V sequences involve direct communication between the RPV and environment which renders containment systems, and associated event tree modeling, irrelevant.

Thus, containment event trees were required only to evaluate the containment response to Class I, Class III and Class IV events.

The CSETs were developed by establishing the functions and containment systems that were relevant to mitigating the overpressure and bypass challenges. The CSETs were then constructed using appropriate logic to account for mitigating system success or failure by

establishing the logically possible containment responses. Finally, the end states of the CSETs, which are termed "release categories", were defined. The release categories may indicate containment failure or may indicate that the containment has successfully functioned to limit the radionuclide release. These release categories represent meaningfully different outcomes to the containment challenge and are used in the source term evaluation discussed in Section 9.

Review of the CSETs indicates that that there is a common structure to the trees, irrespective of the initiating event. This structure is indicated in Figure 8.2-1. Because of this common tree structure, quantification of different accident classes was unique only because of differing entry event and branch probabilities. Determination of the CSET entry event probabilities is discussed in Section 8.2.1.1. The containment systems evaluated in the CSETs are summarized in Section 8.2.1.2 with the associated top events being discussed in Section 8.2.1.3. The end states of the trees, which become the release categories for the consequence evaluation, are discussed in Section 8.2.1.4. The frequencies associated with the release categories are presented in Section 8.1.1.5. Appendix A.8 provides additional detail on the release category quantification.

8.2.1.1 CSET Entry Events

Quantification of the CPETs indicates that it is very unlikely that containment failure will occur due to the type of containment challenges addressed in the containment phenomenology event trees. Thus, the total probability of all of the CSET entry events is very close to the calculated frequency of a core damage event. The difference lies in the small probabilities of containment failure that were assigned to the core-concrete interaction, ex-vessel steam explosion and direct containment heating events. As illustrated in Appendix A.8, the probability of transferring from the CPET to the CSET becomes the entry, or "initiating" event frequency for each CSET.

The CSET entry event frequencies are summarized in Table 8.2-1. Note that each accident class is divided into two subclasses. The subclasses were necessary to reflect system dependencies on whether or not site power was available. For example, accident Class I was divided into Class IL (RPV failure at low pressure and loss of preferred power) and Class IN (RPV failure at low pressure without loss of preferred power). Class III was similarly subdivided. From the Level 1 analysis presented in Section 7, the probability of a Class IV event resulting in RPV failure at high pressure is negligible. Thus, the subclasses for Class IV are IVL (ATWS with RPV failure at low pressure and loss of preferred power) and IVN (ATWS with RPV failure at low pressure without loss of preferred power).

8.2.1.2 Mitigating Systems

The ESBWR includes systems with the capability to prevent or mitigate containment bypass and overpressurization. The systems considered in the evaluation of containment response are summarized below.

Containment Isolation System

The containment isolation system provides for monitoring and isolation of the containment boundary to prevent unacceptable radiological releases during normal, abnormal and accident conditions.

Isolation Condenser System

The isolation condenser system (ICS) provides the capability to remove decay heat from the RPV. Because the heat exchangers are external to the containment, removal of heat from the RPV also removes energy from the containment. The isolation condensers would be effective primarily when the RPV is at an elevated pressure. The isolation condensers do not condense a significant amount of steam after RPV depressurization and thus, provide little mitigation of a severe accident after RPV depressurization. For conservatism, the ICS was not credited in the severe accident sequence evaluation.

GDCS Deluge and BiMAC

The deluge mode of GDCS operation provides flow through the BiMAC to flood the lower drywell when the temperature in the lower drywell increases enough to be indicative of RPV failure and core debris in the lower drywell. The GDCS deluge system is activated by thermocouples embedded in the lower drywell floor.

By flooding the lower drywell after the introduction of core material, the potential for energetic fuel-coolant interaction at RPV failure is minimized. Covering core debris with water provides scrubbing of fission products released from the debris and cools the corium, thus limiting potential core-concrete interaction. The BiMAC provides additional assurance of debris bed cooling by providing an engineered pathway for water flow through the debris bed.

Containment Heat Removal (PCCS and Suppression Pool Cooling)

Containment heat removal can be provided by either the PCCS or the suppression pool-cooling mode of the FAPCS. For sequences with successful containment heat removal, the analysis assumed that the PCCS was available and that suppression pool cooling was not in operation. This assumption bounds the containment pressure response because the PCCS can only limit pressurization, while suppression pool cooling can limit and reduce containment pressure.

The PCCS receives a steam-gas mixture from the upper drywell atmosphere, condenses the steam using the PCCS pools as a heat sink, and returns the condensate to the GDCS pool. The non-condensable gas is drawn to the suppression pool through a submerged vent line by the pressure differential between the drywell and wetwell. The PCCS is designed to remove decay heat added to the containment after a LOCA, thus maintaining the containment within its pressure limits. Operation of the PCCS requires no support systems and, as illustrated in

Section 8.3, there is adequate inventory in the PCCS pools to provide containment heat removal for more than 24 hours after the onset of core damage.

Drywell Spray

Drywell spray provides the capability to condense steam in the containment atmosphere to limit pressurization and cool a corium debris bed to limit core-concrete interaction. Drywell spray is not credited in this analysis.

Vacuum Breakers

Vapor suppression requires that a pressure differential be maintained between the drywell and the suppression pool. Failure of the vacuum breakers, either due to a preexisting condition or failure to reclose, is assumed to result in loss of the vapor suppression capability. That is, sequences in which vacuum breaker failure occurred were modeled with an open path between the drywell and wetwell airspace.

Suppression Chamber Vent

To prevent overpressurization failure of the containment as a result of long-term core-concrete interaction or failure of containment heat removal, the ESBWR contains a manually controlled vent connecting the suppression chamber gas space to the environment. Opening the vent would greatly decrease the magnitude of a potential release in comparison to containment failure by forcing the radionuclide pathway through the suppression pool. As will be shown in Section 8.3, failure of containment heat removal does not cause the containment to pressurize to the point at which venting is likely to be implemented to prevent containment failure in the 24-hour time frame after onset of core damage.

Reactor Building Effects

Fission product releases to the environment through the paths representing "normal" containment leakage, i.e., leakage up to the amount allowed by the Technical Specifications, could be reduced for some sequences if credit were taken for radionuclide removal by the reactor building HVAC system. However, such a source term reduction was not credited in the severe accident sequence modeling. Therefore, the source terms of sequences with only Technical Specification leakage are conservative in that they represent a direct release from the containment to the environment. Sequences in which the drywell failure is at the drywell head seals are also conservative because credit is not taken for refueling pool scrubbing. Sequences with drywell failure at other locations are not significantly affected because the release path bypasses the reactor building or would overwhelm the capacity of the reactor building ventilation system.

8.2.1.3 Top Events

Section 8.2.1.1 identified the entry events for the containment system event tree. The next step in constructing CSET was to define, as top events, the functions needed to assess the containment response to bypass and overpressurization challenges. These functions are "containment isolation", "vapor suppression", "containment heat removal" and "venting".

Defining top events for the recovery of failed systems and for operator actions was considered, but was judged unnecessary, as indicated in the following sections. As a result, the event tree necessary to model the containment response became simple in form. Further,

because of the passive nature of the containment design, containment systems have no dependencies on the accident initiator that must be reflected in the tree structure itself. That is, the structure of a CSET is the same irrespective of the initiating event being considered. The trees differ only in the quantification as dependencies on the entry event are reflected in the different branch probabilities. The CSET structure is provided as Figure 8.2-1 with corresponding event probabilities provided in Table 8.2-1. A discussion of the treatment of system recovery, operator actions and top events follows. Appendix A.8 provides additional discussion of the top event probabilities.

8.2.1.3.1 Repair of Failed Systems

Recovery of failed systems was not credited in the severe accident analysis.

8.2.1.3.2 Key Operator Actions

Because of the passive nature of the ESBWR containment systems, there are no operator actions required to support the containment response to a severe accident in the 24 hour period after onset of core damage. The containment isolation system, vacuum breakers, and PCCS do not require operator action to initiate or function. Analyses provided in Section 8.3 will show that operator action is not required to maintain containment heat removal through the PCCS for the 24 hour period after onset of core damage and that containment venting will not be required during that period. Thus, operator actions are considered in the containment evaluation only in terms of:

1. Action taken to recover system failures. Such actions are considered in the fault tree analysis. An example would be an operator action to close a redundant valve in the vacuum relief path if an individual vacuum breaker were to fail to properly seat.
2. Action taken as a backup to an automatic action, e.g., to open the connecting valve for PCCS pool makeup if the low-water level signal were to fail.
3. Action taken to initiate a backup system, e.g., to actuate the FAPCS if the PCCS were unavailable.
4. Actions requiring a long time period to initiate. For example, the suppression chamber vent is under operator control. As indicated in Section 8.3, there would be a long time period (more than 24 hours) in virtually all scenarios to initiate venting to prevent containment overpressure due to a loss of containment heat removal.

Because these operator actions are redundant to passive system functioning or are required only after a long time period, such actions do not have a significant effect on the probability of containment failure.

8.2.1.3.3 Top Event CIS

Top event CIS, representing the containment isolation system, assesses the probability that the containment has not been isolated and, as a result, there is a pathway from containment into the reactor building or directly into the environment.

Section 4.18 documents containment isolation from the perspective of analyzing pipe breaks outside of containment for the Level 1 analysis. As indicated in Appendix C.8, a screening evaluation was performed to identify those containment penetrations that could potentially

lead to offsite consequences. The screening analysis found that there were no penetrations that required isolation to prevent significant offsite consequences. Thus, the containment isolation function, as applied to the Level 2 analysis, was modeled considering only the isolation signal common to all penetration paths, as discussed in Appendix A.8. This approach addresses the "failure-to-close" probability of valves that may be periodically opened as well as the potential common mode failure of small penetrations which have not yet undergone detailed design.

If CIS is failed, the event tree path has no additional branching because the containment has been bypassed and operation of the vacuum breakers, containment heat removal or venting functions is irrelevant. The bypass failure is assumed to be present at the onset of core damage and is not recovered for the duration of the sequence.

8.2.1.3.4 Top Event VB

Top event VB models vacuum breaker operation. Successful vacuum breaker function is necessary to maintain the pressure differential between the upper drywell and suppression pool and thus enable the steam condensation by the PCCS and suppression pool. If VB were not successful, i.e., a vacuum breaker fails to reclose or exhibits excessive leakage, the containment would pressurize relatively quickly because the vapor suppression function is ineffective. The failure probability is a conditional probability derived from fault tree modeling as discussed in Section 4.18.

8.2.1.3.5 Top Event W

Top event W models containment heat removal. The event is partitioned into "short-term" and "long-term" heat removal functions, "W1" and "W2", respectively. The passive PCCS system and the active suppression pool cooling mode of the FAPCS are considered in these nodes. As indicated in Section 4.19.2, the PCCS is designed with adequate water in the PCCS pools to mitigate a design basis event for 24 hours after event initiation. Accordingly, event W1 addresses the period from event initiation to 24 hours after event initiation. This is conservative as indicated by Figure 8.2-2, which illustrates that the PCCS pool water level does not drop below the top of the PCCS heat exchangers in the 24-hour period after onset of core damage. The failure probability for W1 is a conditional probability derived from fault tree modeling. There is some dependency on the initiating event because the suppression pool cooling system requires power to operate.

After 24 hours, it is assumed that the PCCS pool must be replenished by opening valves to an additional water pool. Upon connecting the additional pool, there is adequate water to maintain containment heat removal for the longer term, defined as 24 to 72 hours after event initiation. Long-term containment heat removal is modeled as event W2. As with W1, the failure probability for event W2 is a conditional probability derived from fault tree modeling. The W2 event frequency is dependent on the initiating event because DC power is required to open the valves to the additional water source and the suppression pool cooling system requires power to operate.

8.2.1.3.6 Top Event VT

Top event VT models operator action to prevent containment failure by use of a suppression chamber vent path. If Event VT occurs, the release path is controlled and directed through the suppression pool where significant filtering can occur to reduce the potential source term.

As discussed earlier, operator guidance for controlled venting has not yet been defined. However, insight into the ESBWR passive containment capability, and the need for venting, can be gained by evaluating a severe accident scenario in which there is no containment heat removal (i.e., event W1 is failed). From the Level 1 analysis discussed in Section 7, the sequence that dominates the core damage frequency is a transient in which the RPV is successfully depressurized. For such a sequence, Figure 8.2-3 illustrates that, for a dominant Class I contributor to the core damage frequency, the containment pressurizes to less than 1.0 MPa. As will be shown in Section 8.3, similar results were obtained for representative Class III and IV sequences. Thus, it is very unlikely that controlled venting in the 24-hour period after the onset of core damage will be required to prevent containment overpressure failure for the sequences dominating the core damage frequency.

8.2.1.4 Release Categories

Completion of a path through the event tree presented in Figure 8.2-1 provides the necessary information to establish categories for potential radionuclide release to the environment. A release category descriptor for each path is shown in Figure 8.2-1 in the column headed "Rel Cat". The release categories differ in the timing of containment breach and the magnitude of the radionuclide source term. By several orders of magnitude, the most likely path through the CSET results in an intact containment with the source term being associated with containment leakage up to the limit allowed by Technical Specifications. This release category is termed "TSL". The release categories associated with the CSET presented in Figure 8.2-1 are discussed in more detail in the following sections. Drawing on the quantification presented in Appendix A.8, the probability of each CSET release category is summarized in Table 8.2-2.

Containment Bypass (BYP)

The release category "Bypass" represents those sequences in which containment isolation has not occurred due to failure of the containment isolation system. Thus, the BYP failure mode provides for a direct path from the containment to the environment and results in an earlier environmental release than an overpressure event. However, due to the reliability of the containment isolation system, the probability of such a release occurring is approximately four orders of magnitude less than the TSL release category. Because the calculated probability of the BYP release category is so small, containment bypass is not considered a credible containment failure mode for the ESBWR. However, the BYP release category will be considered in Section 9 as a potential source term.

Filtered Release (FR)

The release category "Filtered Release" represents those sequences in which the suppression chamber vent is used to control the containment pressure and potential release point. In such a situation, the containment boundary remains under operator control. As a result, the

magnitude of the release is much less than if the containment were to fail because the release path is through the suppression pool, which provides significant radionuclide filtering.

As indicated earlier, in the 24-hour period after onset of core damage, the ESBWR containment would likely not require venting even in the absence of containment heat removal for the sequences that dominate the core damage frequency. Although venting is not likely to be required in the 24 hour period after onset of core damage, the option is maintained in the containment system event tree. Treating the possibility of FR in this way accounts for uncertainties in the loss of heat removal analysis and provides a conservative estimate of the likelihood of a controlled release. Further, inclusion of venting on the CSET allows for modeling a period longer than 24 hours after the onset of core damage. The probability of the FR release category was calculated as about two orders of magnitude less than the TSL release category.

Overpressurization (OP)

The release category "Overpressurization" represents those sequences in which there has been inadequate post-accident heat removal resulting in the containment pressure exceeding the ultimate containment strength. Two categories of overpressure failure are considered. The category "OPW" applies to severe accident sequences in which the vapor suppression function is successful and only the containment heat removal function has failed. Both early (OPW1) and late (OPW2) failures of containment heat removal are considered. The category "OPVB" applies to sequences in which the vapor suppression function fails; in that situation, the containment heat removal function is also failed. As indicated in Table 8.2-2, the total probability of the overpressure failure mode (OPW1, OPW2 and OPVB) is about three orders of magnitude less likely than the TSL failure mode and is associated predominantly with failure of long term heat removal. Each subcategory is discussed below.

OPVB: The release category "OPVB" applies to sequences in which vacuum breaker failure has occurred. Failure of the vacuum breakers to close, or to be open in a pre-existing condition, results in failure of the containment vapor suppression function. If the vacuum breakers fail to function effectively, the overpressurization occurs fairly early in the severe accident sequence because the vapor suppression function is not effective. Because of the high reliability of the vacuum breakers necessary for the vapor suppression function, the calculated probability of the OPVB release category is more than four orders of magnitude less than the TSL release category. Thus, vapor suppression failure it is not considered a credible containment failure mode for the ESBWR. However, the OPVB release category will be used in Section 9 as part of a conservative evaluation of potential source terms.

OPW1: The release category "OPW1" applies to sequences in which containment heat removal fails within 24 hours after event initiation. In such sequences, vapor suppression has been successful, but the passive PCCS system is unavailable as well as the active FPACS, either of which provides the capability to remove energy from the containment. The 24-hour transition point from W1 to W2 was selected to correspond with the design requirement regarding the amount of water available to the PCCS cubicles without connection to a supplemental pool source. Because of the reliability of the fuel pool cooling mode of the FAPCS and the passive PCCS, the

calculated probability of the OPW1 release category is more than four orders of magnitude less than the TSL release category. Thus, loss of containment heat removal is not considered a credible containment failure mode for the ESBWR. However, the OPW1 release category will be used in Section 9 as part of a conservative evaluation of potential source terms.

OPW2: The release category "OPW2" applies to sequences in which containment heat removal fails between 24 and 72 hours. In such sequences, the passive PCCS system becomes unavailable after PCCS pool dryout due to failure to connect to supplemental water pools; FAPCS availability was also evaluated at this time. Because of the minimum system requirements to provide additional water to the PCCS pools, long term heat removal (>24 hours) is very reliable. The probability of the OPW2 release category is about three orders of magnitude less than the TSL release category. As with other release categories, OPW2 will be represented in the source term evaluation.

Containment failure due to overpressurization is conservatively modeled as a direct path from the drywell to the environment. Thus, potential uncertainty in the location of the failure point is accommodated by the assumption of a direct path to the environment if the containment is overpressurized.

Technical Specification Leakage (TSL)

The release category "Technical Specification Leakage" represents those sequences in which there is neither a release due to containment failure nor a controlled filtered venting. The TSL release category provides a source term that exceeds that associated with normal operation because of the severe accident conditions within the containment. It is assumed that the leakage area corresponds to the Technical Specification allowable containment leakage rate of 0.5% of containment air volume per day at rated pressure.

The leakage path was conservatively assumed to occur between the drywell atmosphere and environment. Thus, no credit is taken for source term reduction if the leakage could be affected by potential refueling pool scrubbing or the reactor building HVAC system.

8.2.1.5 Release Category Frequency and Containment Effectiveness

The frequency of a given release category for each initiator is found by quantifying the CSET path ending with that release category. To provide the total probability of a release category for all initiators, the CSET is evaluated for each entry event and the probabilities are summed. As seen in Table 8.2-2, the most likely release category is that associated with leakage from an intact containment, TSL. Controlled, filtered venting, FR, is the next most likely release category with a release frequency two orders of magnitude lower than TSL. Release categories associated with containment failure, i.e., OP and BYP, are several orders of magnitude less likely than the TSL release category.

The release categories associated with containment failure are so much lower than the TSL category, and their calculated probabilities are so low on an absolute basis, that containment failure due to overpressurization or bypass in the 24-hour period after the onset of core damage is not considered credible. Thus, it is clear that the ESBWR provides a reliable barrier to radionuclide release. This conclusion is reflected in the quantification of

containment effectiveness. The containment effectiveness can be conservatively quantified as the probability of release category TSL (i.e., an intact containment) divided by the probability of all release categories. This method is conservative in that the FR category is included in the denominator although it does not involve loss of containment boundary control. Using the values from Appendix A.8, Table A.8-3, and applying "ε" for probabilities less than 1E-12,

$$\text{Containment effectiveness} \equiv \frac{\text{Probability of TSL release category}}{\text{Probability of all release categories}}$$

$$= \frac{2.84\text{E-}8}{(2.89\text{E-}10+2.9\text{E-}11+\varepsilon+2.52\text{E-}10+\varepsilon+2.84\text{E-}8+2.33\text{E-}10+1.4\text{E-}11+\varepsilon+\varepsilon+1\text{E}12+4\text{E-}12)}.$$

$$= 0.97$$

Table 8.2-1

Summary of CSET Initiating and Branch Failure Probabilities

CSET	Entry Event*	Entry Event Frequency	Top Event CIS	Top Event VB	Top Event W1	Top Event W2	Top Event VT
Class IL (RPV failure at low pressure and loss of preferred power)	IL_CS	1.62E-8	3.50E-5	5.48E-6	3.66E-5	4.12E-7	5.69E-2
Class IN (RPV failure at low pressure without loss of preferred power)	IN_CS	1.20E-8	3.50E-5	6.14E-6	7.97E-6	1.45E-5	5.69E-2
Class IIIL (RPV failure at high pressure and loss of preferred power)	IIIL_CS	3.16E-10	3.50E-5	3.00E-4	1.82E-4	7.60E-1	5.69E-2
Class IIIN (RPV failure at high pressure without loss of preferred power)	IIIN_CS	8.90E-12	3.50E-5	2.92E-4	1.77E-4	6.87E-1	5.69E-2
Class IVL (ATWS with RPV failure at low pressure and loss of preferred power)	IVL_CS	4E-12	3.50E-5	5.32E-6	6.18E-6	8.98E-8	5.69E-2
Class IVN (ATWS with RPV failure at low pressure without loss of preferred power)	IVN_CS	1.75E-10	3.50E-5	5.33E-6	5.76E-6	4.72E-8	5.69E-2

*Nomenclature used in event tree quantification provided in Appendix A.8

Table 8.2-2

CSET Release Category Frequencies

Release category	Frequency (per year)*
TSL	2.84E-8
FR	2.33E-10
BYP	1E-12
OPVB	<1E-12
OPW1	<1E-12
OPW2	1.4E-11

*The frequency is the summed contribution to the release category from all accident classes, as shown in Table A.8-3.

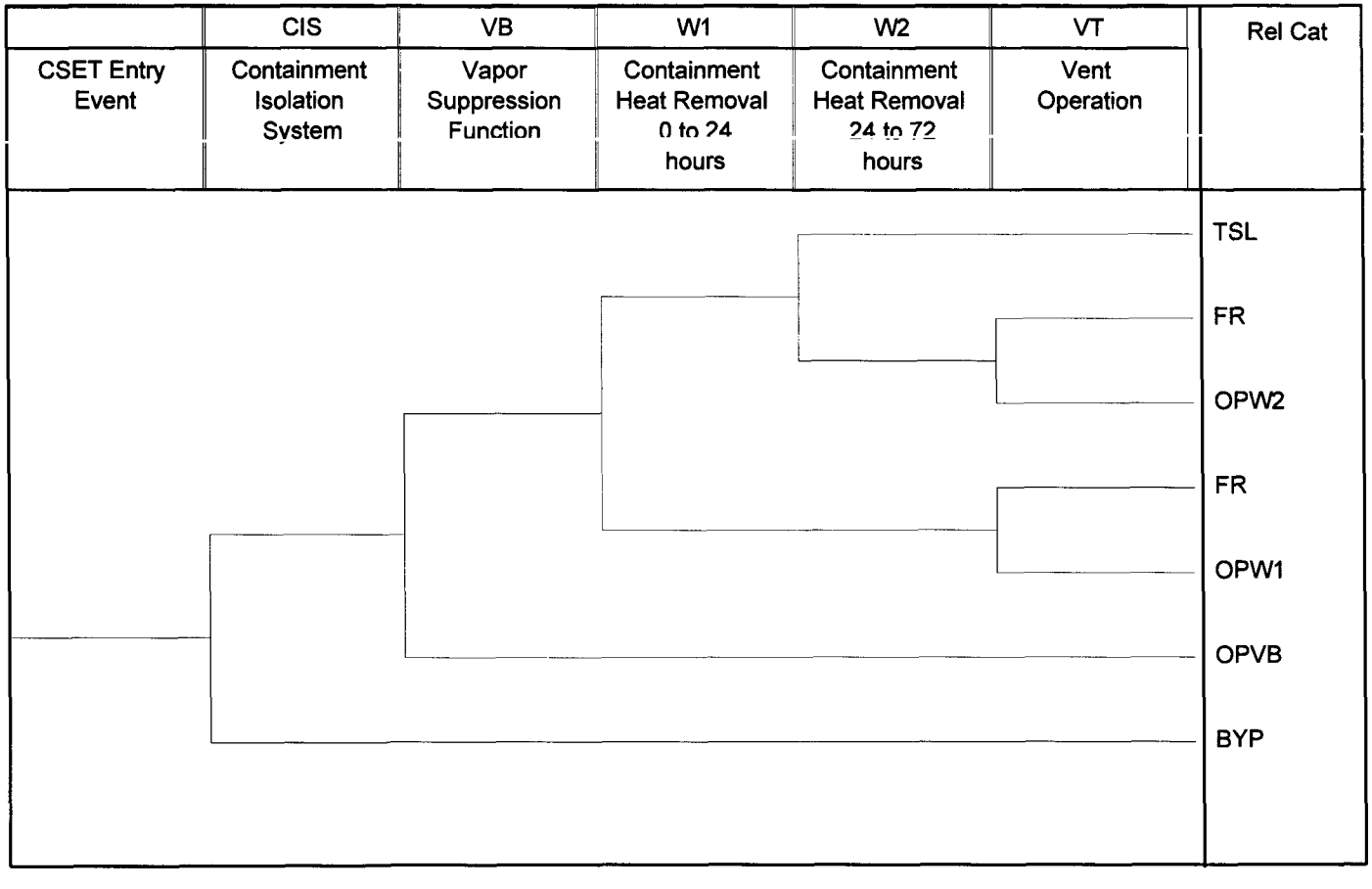

Figure 8.2-1. Containment System Event Tree

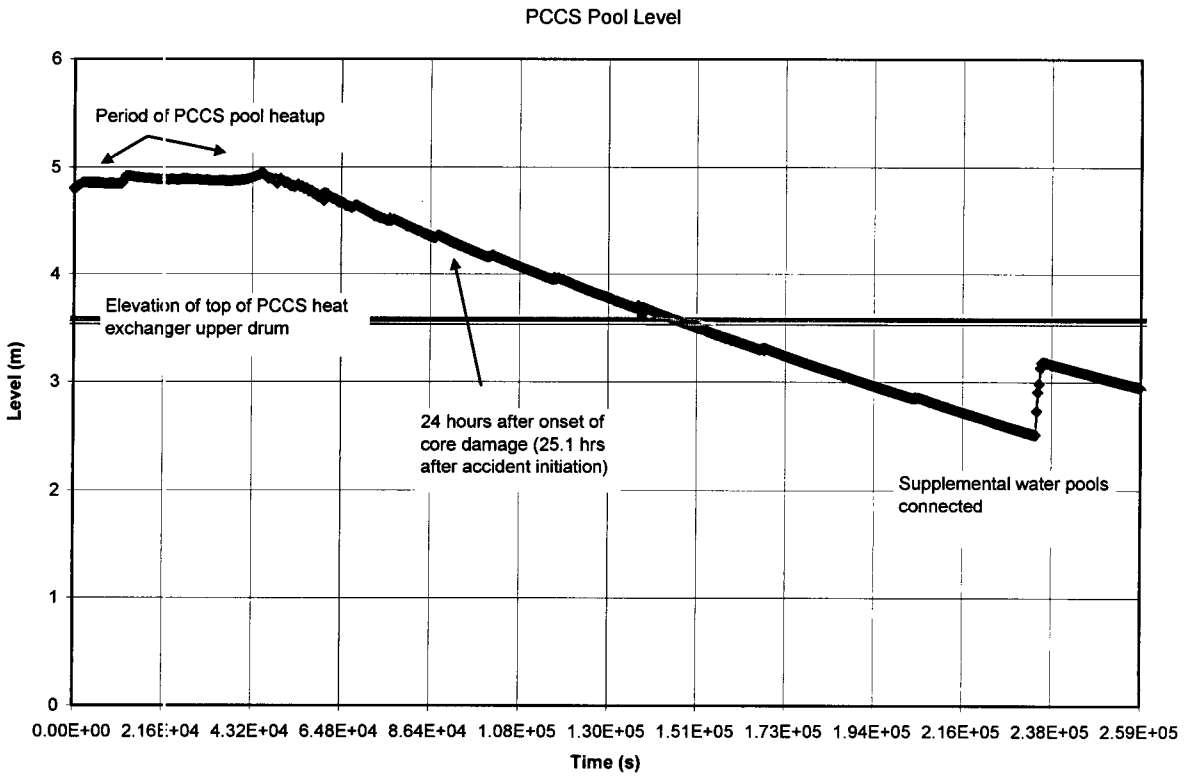

Figure 8.2-2. PCCS heat removal capability for 24 hour period

Example shown is for a dominant Class I sequence, a transient followed by loss of core injection. The PCCS heat exchangers remain covered for more than 24 hours after onset of core damage.

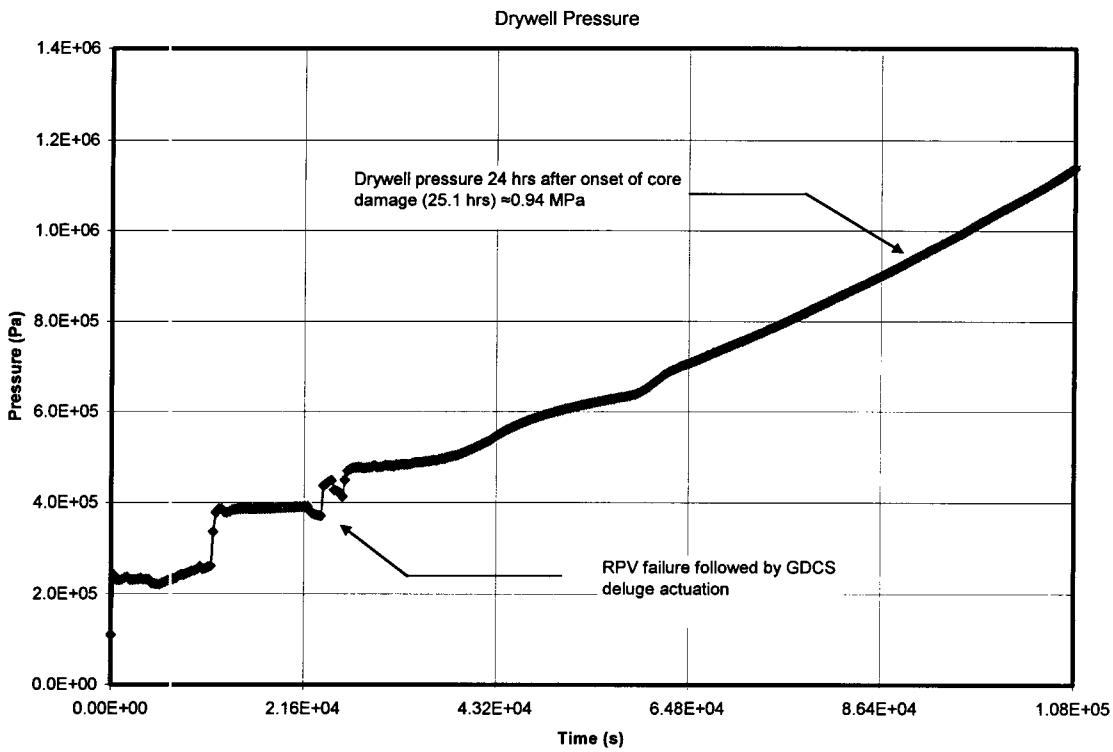

Figure 8.2-3. Containment Pressure with No Containment Heat Removal

Example shown is for a dominant Class I sequence, a transient followed by loss of core injection. With vapor suppression function successful, containment does not pressurize to failure within 24 hours after onset of core damage.

8.3 CONTAINMENT PERFORMANCE AGAINST OVERPRESSURE

To determine the key characteristics of the containment response to a severe accident, an ESBWR simulation model was developed. The model is used to gain insights into the timing of severe accident progression, the containment pressure-temperature response and ultimately the potential source term if the containment were to fail. As demonstrated in the prior section, the reliability of containment systems designed to mitigate a severe accident is such that the calculated probabilities of containment bypass and overpressure failure are so small that they may be considered negligible. Thus, only the TSL and FR release categories are discussed in this section. Hypothetical scenarios in which the containment is bypassed or fails due to overpressurization are considered in the evaluation of potential source terms, as presented in Section 9.

Analysis of the ultimate strength of the containment indicates that the drywell head is the most likely failure location if the containment were to overpressurize. The analysis also illustrates that the containment pressure capability is a function of temperature. This pressure capability profile was used in the simulation modeling.

Section 8.3.1 summarizes the code used for accident simulation. Section 8.3.2 provides the simulation results for a spectrum of potential severe accidents representing each accident class. Appendix B.8 provides the ultimate containment strength analysis.

8.3.1 Simulation Code

The ESBWR was modeled using a computer code capable of modeling the integrated plant response to a severe accident. The code used for this purpose is the Modular Accident Analysis Program code (MAAP), Version 4.0.6, Reference 8.3-1. The code was developed as part of the Industry Degraded Core Rulemaking (IDCOR) program to investigate the physical phenomena that might occur in the event of a severe light water reactor accident leading to core damage, possible RPV failure, and ultimately possible failure of containment integrity and release of fission products to the environment. MAAP development was sponsored by the Atomic Industrial Forum. MAAP includes models for the important phenomena that might occur in a severe light water reactor accident.

MAAP has a long history of use in severe accident analysis, including severe accident analysis of the ABWR as described in Reference 8.3-2, which was based on an earlier version of MAAP. MAAP requires that phenomenological information and plant specific design characteristics be provided in the form of a parameter file. Parameter file inputs related to accident phenomenology were based on the values provided in MAAP sample files, which are maintained for the MAAP Users Group; these values were provided by the code developer. Parametric values related to the ESBWR design were based on review of design documentation information, as it was available in February 2005. In some cases, design information was updated between February and August 2005 when significant design decisions were made.

8.3.2 Sequences Representative of Each Accident Class

As discussed in earlier sections, severe accidents were grouped in five categories in the Level 1 analysis. The Level 1 analysis results were reviewed to identify sequences which were

dominant contributors to the core damage frequency. A single dominant sequence was then selected to represent each of the accident classes for detailed modeling. In this way, the containment response to the complete spectrum of accidents contributing to the core damage frequency could be evaluated.

Table 8.3-1 identifies the sequences that were used to represent each accident class. The "core damage sequence descriptor" used in the table derives from the results of the Level 1 analysis. Table 7.2-3 identified the sequences which were significant contributors to the core damage frequency. The representative sequences shown in Table 8.3-1 are based on the Level 1 results presented in Table 7.2-3. For example, Table 7.2-3 indicates that about 99% of the Class I frequency is associated with loss of preferred power (T-LOPP) or loss of feedwater (T-FDW) sequences. From the perspective of modeling the containment response to a severe accident, both of these sequences can be represented as a transient with loss of injection "T-nIN". A similar approach was used in selecting the representative sequence for the other accident classes. Table 8.3-1 provides a summary description for each representative sequence.

Table 8.3-2 couples the representative core damage sequence with one of the release categories illustrated on the containment system event tree, Figure 8.2-1. The resulting scenario is assigned a "containment response sequence descriptor" to summarize the core damage and containment release information. Recalling that Table 8.2-2 provided the total contribution of all accident classes to each release category frequency, Table 8.3-2 provides additional information by presenting the release category frequency in terms of the contribution from each accident class. As indicated in the table, there is a negligible probability of a core damage sequence resulting in overpressure or bypass failure. However, such hypothetical scenarios are retained for evaluation in Section 9 to assure that a conservative source term is developed.

8.3.2.1 Class I: Sequences with RPV Failure at Low Pressure

Accident Class I involves sequences in which the RPV fails at low pressure; this accident class represents approximately 98% of the core damage frequency. As indicated in Tables 7.2-3 and 7.2-5, the class is dominated by transient sequences in which there is no core injection. Thus, the sequence "T_nIN" described below was used to evaluate the containment response to Class I events.

8.3.2.1.1 Sequence T_nIN_TSL

The initiating event for the T_nIN sequence is a transient initiated by a loss-of-preferred power. No short or long-term coolant injection to the RPV by the feedwater, CRD or GDCS is available. The ADS functions to reduce the RPV pressure. As stated earlier, heat removal by the isolation condensers is not credited. Containment heat removal in the short-term is accomplished by successful PCCS functioning; PCCS pool makeup is successful, thus allowing long-term containment heat removal. The GDCS deluge system and BiMAC are available for debris bed cooling. With successful containment isolation, vapor suppression and containment heat removal the containment remains intact. Technical Specification leakage is the only mode of fission product release.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-1a through e show the system behavior throughout the accident sequence.

In this event, the primary system rapidly depressurizes due to opening of the first ADS-actuated valves at about 47 seconds. The pressure in the containment increases as the drywell is filled with steam and heats up. About thirty minutes into the event, core uncovery occurs which results in fuel rod heatup and melting. Fission products and non-condensable gases are swept into the containment through the DPVs as the core melts. This leads to further heating and pressurization of the drywell air space.

The reactor pressure vessel lower head penetrations fail about 6.4 hours into the event. Corium is deposited on the lower drywell floor, which results in local temperatures that are high enough to cause the GDCS deluge line to open. As a result, the GDCS pool water drains into the lower drywell and covers the debris bed. Because the debris is quenched by the successful GDCS deluge and BiMAC function, significant core-concrete interaction does not occur. Therefore, no significant fission product aerosols or non-condensable gases are generated in the ex-vessel phase of the accident sequence.

Continued heating by debris of the water in the lower drywell leads to the temperature in the overlying water pool reaching saturation. Steam generation in the lower drywell then leads to further increases in the containment pressure until the PCCS heat removal capacity becomes consistent and comparable to the decay heat generated by the core debris. The containment pressure reaches about 0.65 MPa 24 hours after onset of core damage, well below the point at which containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

8.3.2.1.2 Sequence T_nIN_nCHR_FR

Sequence T_nIN_nCHR_FR is the same as the representative Class I sequence T_nIN, except that the containment response differs because containment heat removal has failed. As a result, containment pressurization increases and controlled venting may be implemented to limit the pressure rise and control the radionuclide release point. Specific guidance for the use of the suppression pool vent has not been developed. Indeed, as discussed earlier, venting in the ESBWR does not appear necessary to limit the containment pressure to less than its ultimate strength in the 24-hour period after core damage. The venting scenario is evaluated here to provide insights into vent timing and provide a basis for the FR release category used in the source term evaluation.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-2a through d show the system behavior throughout the accident sequence. The sequence proceeds as discussed in the previous section except that venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength. As indicated, in Figure 8.3-2b, the drywell pressure has reached less than 70% of the ultimate containment strength within 24 hours after onset of core damage; thus venting would not likely be implemented in this time frame. The 90% assumption is met at 32.3 hours, which is about 2.7 hours before containment overpressurization would occur if controlled venting were not implemented.

The sequence demonstrates that venting is not required to prevent containment failure in the 24-hour period after onset of core damage due to a Class I event, even if containment heat removal were unavailable. In such a scenario, there is a long time period after core damage to prepare for venting and take other mitigating actions.

8.3.2.2 Class II: Sequences with Containment Failure Preceding Core Damage

Accident Class II involves sequences in which containment failure, due to a loss of heat removal capability, precedes RPV failure. After containment failure, RPV makeup capability is assumed to be lost due to the gradual boiloff of water in the passive systems; potential damage to piping connections renders active makeup systems unavailable. As a result, core damage and RPV failure occur after containment failure. The sequence MLi_nCHR was selected to represent the containment response to Class II events because the sequence provides containment pressurization due to the break and failure of the containment heat removal function.

As indicated in Section 3.2.4, Class II sequences do not contribute to the core damage frequency because of the long time for sequence development and the potential for operator recovery action. The following discussion illustrates this conclusion.

8.3.2.2.1 Sequence MLi_nCHR

The initiating event for the sequence MLi_nCHR is a medium LOCA, assumed to occur in the GDCS injection line. Failure of containment heat removal is followed by containment pressurization to its ultimate capacity. Core cooling occurs by gravity feed through the GDCS injection and equalizing lines. Eventually, the water used for RPV makeup is boiled off.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-3a through c show the system behavior throughout the accident sequence. The figures illustrate that the containment pressurizes until the ultimate strength is reached at about 31 hours. The ADS depressurizes the RPV allowing GDCS tanks to drain into the RPV, then into the lower drywell through the break. The shroud water level initially rises in response to the GDCS tank injection, then decays as the GDCS inventory is depleted. The shroud level decreases below the elevation of the break at about six hours. Further, shroud level decrease occurs until flow through the equalizing line begins at about 8.3 hours. Flow from the suppression pool maintains RPV level above the top of active fuel for about 71 hours. Shortly thereafter, core heat up begins.

The results of the sequence simulation indicate that the core damage following containment failure due to loss of containment heat removal does not occur within a 24 period after accident initiation. In fact, core temperatures do not reach the point of fuel damage until more than 72 hours after accident initiation. Given the long time for mitigating action to supplement RPV makeup, Class II events are not considered contributors to the core damage frequency and inclusion of such events in the offsite consequence analysis is unnecessary.

Because core damage is not indicated for such a long period from the start of the severe accident sequence, no release category is identified for the representative Class II sequence in Table 8.3-2.

8.3.2.3 Class III: Sequences with RPV Failure at High Pressure

Accident Class III involves sequences in which the RPV fails at high pressure; this accident class approximately 1.4% of the core damage frequency. As indicated in Tables 7.2-3 and 7.2-5, the class is dominated by transient sequences in which there is no core injection. Thus, sequence "T_nDP_nIN" described below was used to evaluate the containment response to Class III events.

8.3.2.3.1 Sequence T_nDP_nIN_TSL

The initiating event for the sequence T_nDP_nIN is a loss-of-offsite power. The sequence differs from T_nIN in that depressurization fails, although the SRVs remain functional in the relief mode. The ICS was not credited. The CRD and Feedwater systems are unavailable. Because depressurization is unsuccessful, the RPV fails at high pressure, i.e., at the pressure controlled by the relief valve setpoint. GDCS deluge and BiMAC function to cool the debris bed in the lower drywell.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-4a and b summarize the system behavior throughout the accident sequence.

The RPV fails about 4.9 hours. Actuation of the GDCS deluge line and successful BiMAC function prevent significant core-concrete interaction from occurring in the lower drywell. Material dispersed to the upper drywell does not result in significant CCI because the large dispersal area allows the material to be cooled. Continued heating of the water by debris in the lower drywell leads to continued steam generation, which increases containment pressure. The PCCS removes heat from the containment, thus preventing overpressurization. The containment pressure reaches about 0.7 MPa 24 hours after onset of core damage, well below the point at which containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

8.3.2.3.2 Sequence T_nDP_nIN_nCHR_FR

Sequence T_nDP_nIN_nCHR is the same as sequence T_nDP_nIN except that containment heat removal has failed. As a result, containment pressurization increases and controlled venting is implemented to limit the pressure rise and control the radionuclide release point. As indicated earlier, specific guidance for the use of the suppression pool vent has not been developed, thus, venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-5a and b show the system behavior throughout the accident sequence. As indicated, in Figure 8.3-5b, the drywell pressure has reached less than 70% of the ultimate containment strength within 24 hours after onset of core damage; thus venting would not likely be implemented in this time frame. The 90% assumption is met at 42.5 hours after accident initiation, which is about 2.9 hours before containment overpressurization would occur.

The sequence demonstrates that venting is not required to prevent containment failure in the 24-hour period after onset of core damage due to a Class III event, even if containment heat

removal were unavailable. In such a scenario, there is a long time period after core damage to prepare for venting and take other mitigating actions.

8.3.2.4 Class IV: Sequences with Failure to Insert Negative Reactivity

Accident Class IV includes sequences that are initiated by an ATWS and followed by failure to initiate negative reactivity. Such sequences represent less than 1% of the core damage frequency. From the Level 1 analysis summarized in Table 7.2-3, the largest Class IV contributor to the core damage frequency is a general transient followed by failure to scram. Thus, the sequence termed "T-AT_nIN", which defines the ATWS initiator with no core injection, was selected to evaluate the containment response to Class IV events.

8.3.2.4.1 Sequence T-AT_nIN_TSL

Sequence T-AT_nIN is a general transient followed by ATWS. The standby liquid control system is ineffective or unavailable. The RPV is not initially depressurized because ADS inhibit is successful. To control the ATWS power level, feedwater runback is successful with operator control assumed at the top of active fuel. The PCCS is available, but no active containment heat removal (FAPCS) is assumed.

The key events of the sequence are summarized in Table 8.3-3. Figures 8.3-6a through c show the system behavior throughout the accident sequence.

In this sequence, feedwater runback is successful. Control of core water level just above the top of active fuel results in a core power level of about 30% full power three minutes after the transient begins. At that time, it is assumed that feedwater is terminated and safety system injection to the RPV does not occur. (System pressure prevents gravity drain from the GDCS and the CRD system is unavailable for forced flow.) Because the ADS inhibit is successful, the RPV is maintained at high pressure, controlled by the SRV setpoint, until the core water level decreases below the point of effective cooling. At that point, manual depressurization is initiated, but injection into the RPV continues to be unsuccessful. RPV failure occurs at about 4.0 hours at low pressure.

Actuation of the GDCS deluge line and successful BiMAC function prevent significant CCI from occurring in the lower drywell. Material dispersed to the upper drywell does not result in significant CCI because the large dispersal area allows the material to be cooled. Continued heating by debris of the water in the lower drywell leads to continued steam generation, which increases containment pressure. The PCCS removes heat from the containment, thus preventing overpressurization. The containment pressure reaches about 0.6 MPa 24 hours after onset of core damage, well below the point at which containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

8.3.2.4.2 Sequence T-AT_nIN_nCHR_FR

Sequence T-AT_nIN_nCHR_FR is the same as sequence T-AT_nIN except that containment heat removal has failed. As a result, containment pressurization increases and controlled venting is implemented to limit the pressure rise and control the radionuclide release point. As indicated earlier, specific guidance for the use of the suppression pool vent has not been

developed, thus, venting is assumed to occur when the containment pressure reaches 90% of the ultimate containment strength.

The key events of the sequence are summarized in Table 8.3-3. Figure 8.3-7a shows the containment response for the accident sequence. As indicated in the figure, the containment pressure 24 hours after the onset of core damage is about 1.0 MPa, within the pressure retaining capability of the containment. The 90% assumption for action to initiate controlled venting is met at about 29 hours after accident initiation.

The sequence demonstrates that venting is not required to prevent containment failure in the 24-hour period after onset of core damage due to a Class IV event, even if containment heat removal were unavailable. In such a scenario, there is a long time period after core damage to prepare for venting and take other mitigating actions.

8.3.2.5 Class V: Sequences with Interfacing LOCA

Because Class V sequences are associated with a direct path from the RPV to the environment the containment response is not relevant to preventing a radionuclide release. The risk of such low probability events is accounted for by defining a release category, "BOC" for break-outside-of-containment, and assigning a frequency in the source term analysis, as discussed in Section 9.0.

Table 8.3-1

Representative Core Damage Sequences

Accident Class	Core Damage Sequence Descriptor	Sequence Summary
I	T_nIN	Transient initiator followed by no short or long-term coolant injection. ADS functions. ICS not credited. PCCS available, but no active containment heat removal (FAPCS). GDCS/BiMAC function successful.
II	MLi_nCHR	Medium liquid line break: GDCS injection line. System is depressurized and injection systems function. Containment heat removal not available.
III	T_nDP_nIN	Transient initiator followed by no short or long-term coolant injection. The RPV is not depressurized; pressure controlled at relief valve setpoint. ICS not credited. PCCS available, but no active containment heat removal (FAPCS). GDCS/BiMAC function successful.
IV	T-AT_nIN	Transient followed by failure to insert negative reactivity. ICS not credited. RPV is not initially depressurized (ADS inhibit successful). SLC is ineffective or unavailable. FW runback is successful. No short or long-term coolant injection. PCCS available, but no active containment heat removal (FAPCS). GDCS/BiMAC function successful.
V	None	No representative sequence assigned for containment evaluation as Class V events involve direct communication between the RPV and environment

Table 8.3-2

Representative Containment Response Sequence

Containment Response Sequence Descriptor	Release Category	Frequency* (per reactor-year)	Containment Response Summary
T_nIN _TSL	TSL	2.81E-08	Release path from drywell through area associated with Technical Specification leakage. All containment systems function effectively.
_nCHR _FR	FR	1E-12	Release path through wetwell vent. Containment heat removal function failed.
_nCHR_W1	OPW1	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails early (<24 hrs); no controlled venting.
_nCHR_W2	OPW2	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails late (>24 hrs); no controlled venting.
_nVB	OPVB	ε	Release path from drywell through area large enough to depressurize containment. Vapor suppression, containment heat removal and controlled venting functions failed.
_BYP	BYP	1E-12	Release path from drywell through open line connecting drywell atmosphere to environment
MLi_nCHR	None	0.0	Release path from drywell through area large enough to depressurize containment. Containment heat removal not available.
T_nDP_nIN_TSL	TSL	7.9E-11	Release path from drywell through area associated with Technical Specification leakage. All containment systems function effectively.
nCHR_FR	FR	2.32E-10	Release path through wetwell vent. Containment heat removal function failed.
_nCHR_W1	OPW1	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails early (<24 hrs); no controlled venting.
_nCHR_W2	OPW2	1.4E-11	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails late (>24 hrs); no controlled venting.
_nVB	OPVB	ε	Release path from drywell through area large enough to depressurize containment. Vapor suppression, containment heat removal and controlled venting functions failed.
_BYP	BYP	ε	Release path from drywell through open line connecting drywell atmosphere to environment

Table 8.3-2

Representative Containment Response Sequence

Containment Response Sequence Descriptor	Release Category	Frequency* (per reactor-year)	Containment Response Summary
T-AT_nIN_TSL	TSL	1.79E-10	Release path from drywell through area associated with Technical Specification leakage. All containment systems function effectively.
nCHR_FR	FR	ε	Release path through wetwell vent. Containment heat removal function failed.
_nCHR_W1	OPW1	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails early (<24 hrs); no controlled venting.
_nCHR_W2	OPW2	ε	Release path from drywell through area large enough to depressurize containment. Containment heat removal fails late (>24 hrs); no controlled venting.
_nVB	OPVB	ε	Release path from drywell through area large enough to depressurize containment. Vapor suppression, containment heat removal and controlled venting functions failed.
_BYP	BYP	ε	Release path from drywell through open line connecting drywell atmosphere to environment

Notes:

"Frequency" indicates contribution from all sequences in accident class, not just the representative sequence. Refer to Table A.8-3 for additional detail regarding release category frequency.

"ε" refers to a calculated frequency of <1.0E-12.

Table 8.3-3

Summary of Results of Severe Accident Sequence Analysis

Sequence Descriptor	RPV Depressurization Initiated (seconds)	Core Uncovered (hours)	Onset of Core Damage (hours)*	RPV Failure (hours)	Deluge Actuated (hours)	Concrete Ablation 24 hrs. after onset of core damage (meters)	Drywell Pressure 24 hrs. after onset of core damage (MPa)	Containment Vent (hours after onset of core damage)
T_nIN _TSL	47	0.47	1.1	6.4	6.5	0.1	0.65	NA
T_nIN_nCHR_FR	47	0.44	1.2	6.6	6.6	0.1	0.9	>24
MLi_nCHR	88	71	>72	>72	NA	NA	NA	NA
T_nDP_nIN _TSL	NA	0.87	1.5	4.9	4.9	<0.1	0.72	NA
T_nDP_nIN_nCHR _FR	NA	0.86	1.5	4.6	4.6	<0.1	0.86	>24
T-AT_nIN_TSL	1015	0.1	0.67	4.0	4.1	0.1	0.61	NA
T-AT_nIN_nCHR_FR	1002	0.1	0.68	4.0	4.0	<0.1	1.0	>24

Key:
MLi: Medium Liquid break (injection line)
T: Transient
T-AT: Transient without negative reactivity insertion
nCHR: No containment heat removal
nDP: No depressurization
nIN: No injection
FR: Filtered release (controlled vent)
TSL: Technical Specification Leakage
NA: Not Applicable
*Time of maximum core temperature > 2499°K

Figs 8.3-1a through e: Sequence T_nIN_TSL

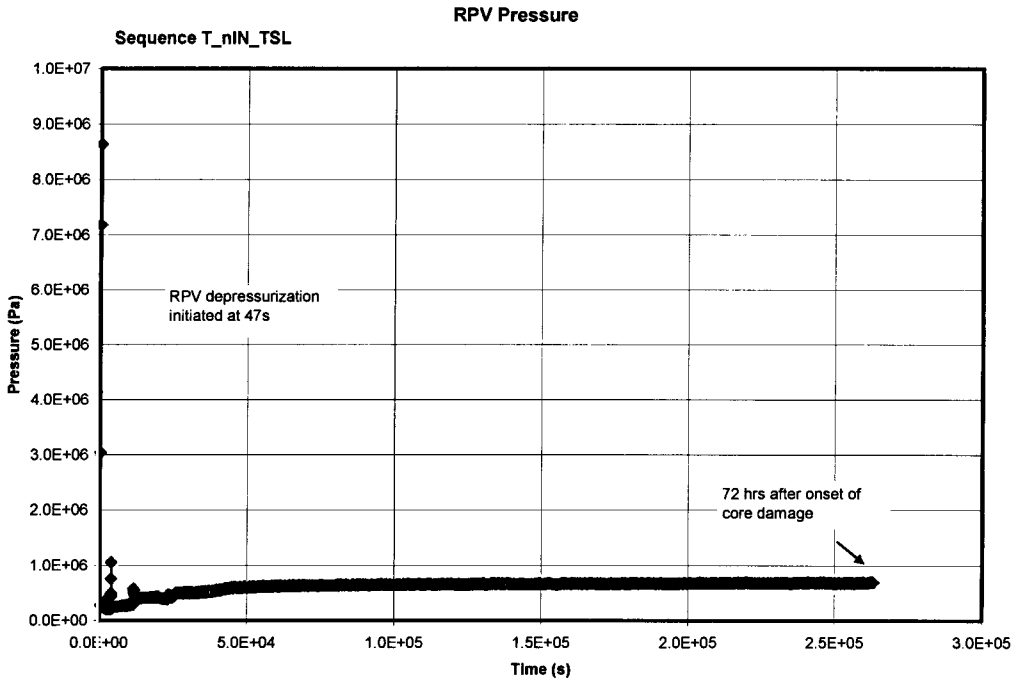

Figure 8.3-1a. T_nIN_TSL: RPV Pressure vs. Time

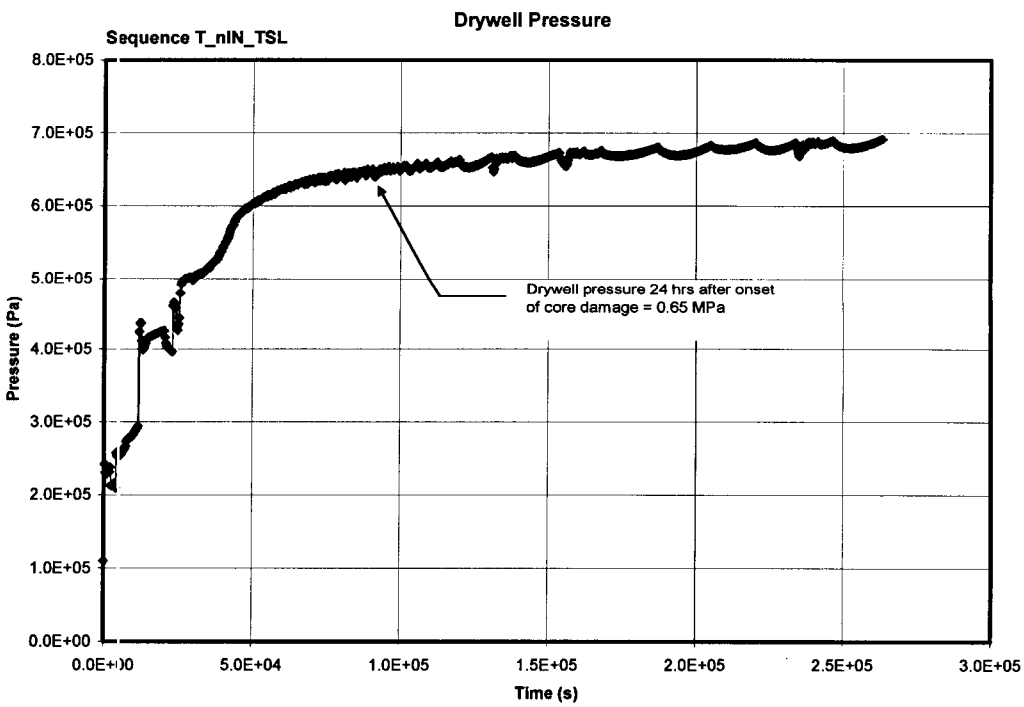

Figure 8.3-1b. T_nIN_TSL: Containment Pressure vs. Time

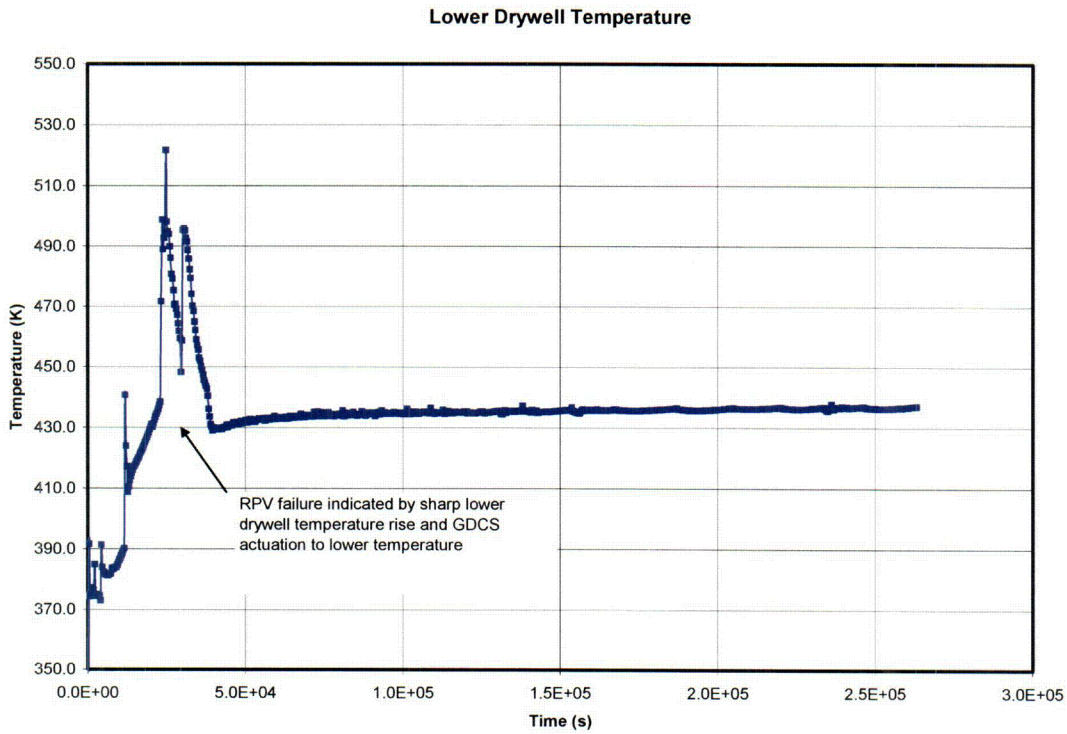

Figure 8.3-1c. T_nIN_TSL: Lower Drywell Temperature vs. Time

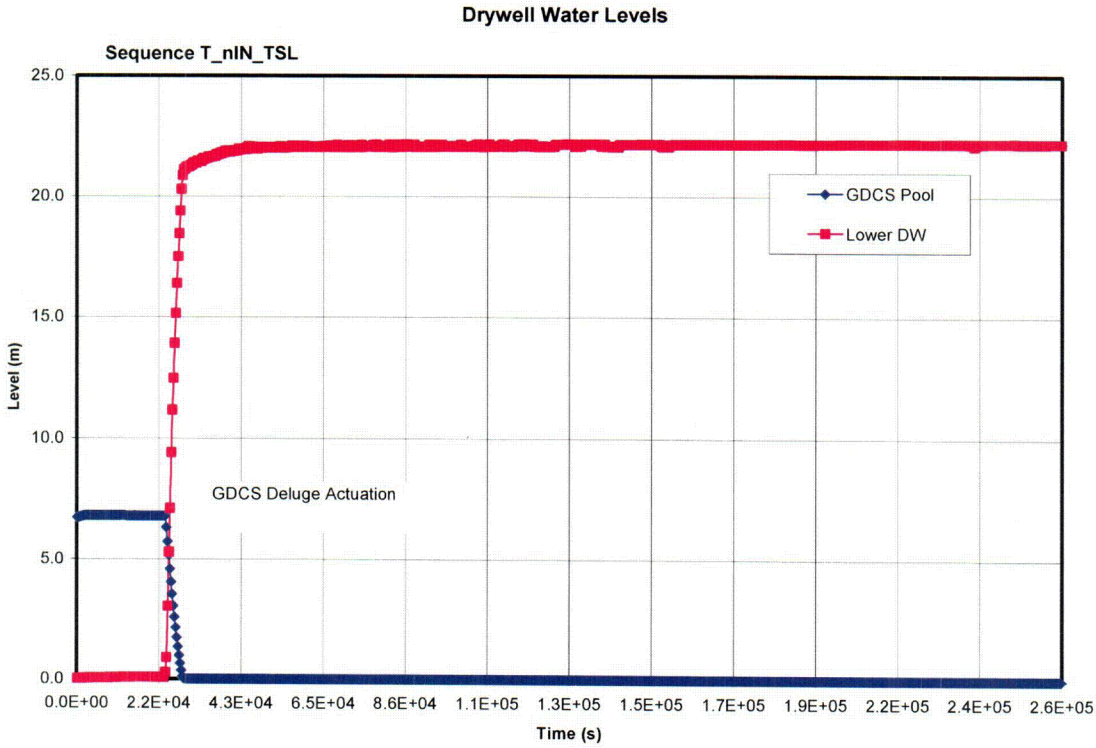

Figure 8.3-1d. T_nIN_TSL: Drywell Water Levels vs. Time

C01

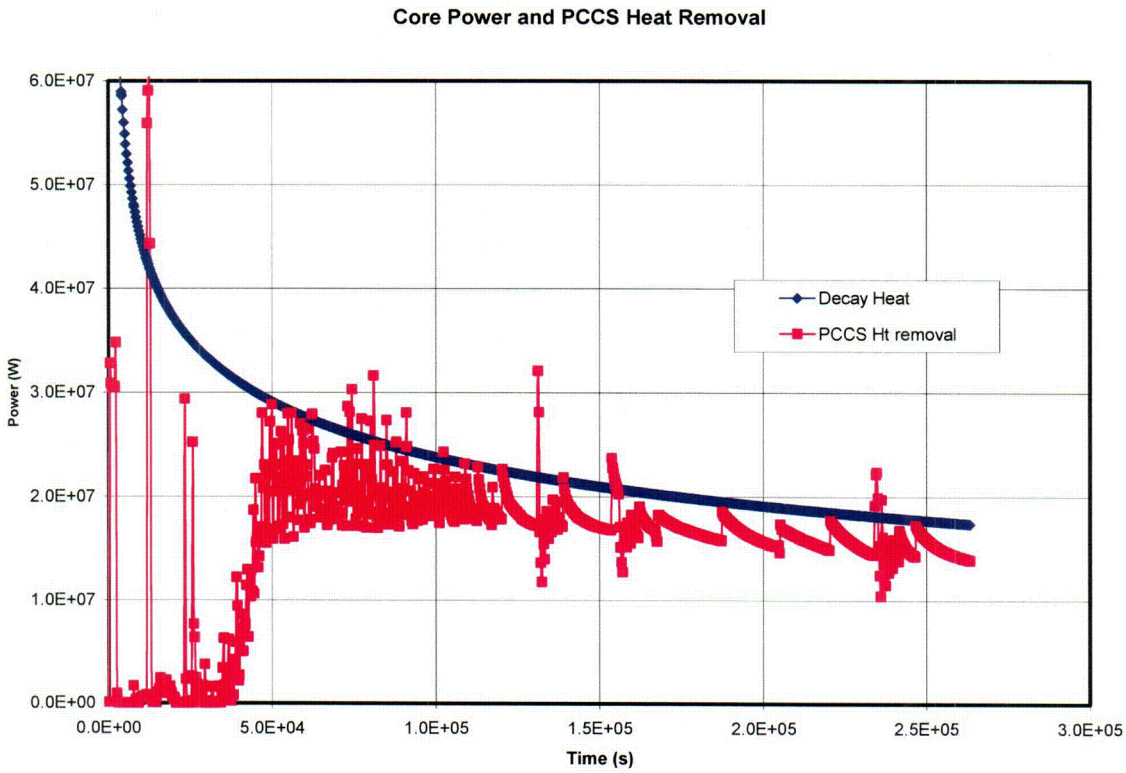

Figure 8.3-1e. T_nIN_TSL: Core Power and PCCS Heat Removal vs. Time

C02

Figs 8.3-2a through d: T_nIN_nCHR_FR

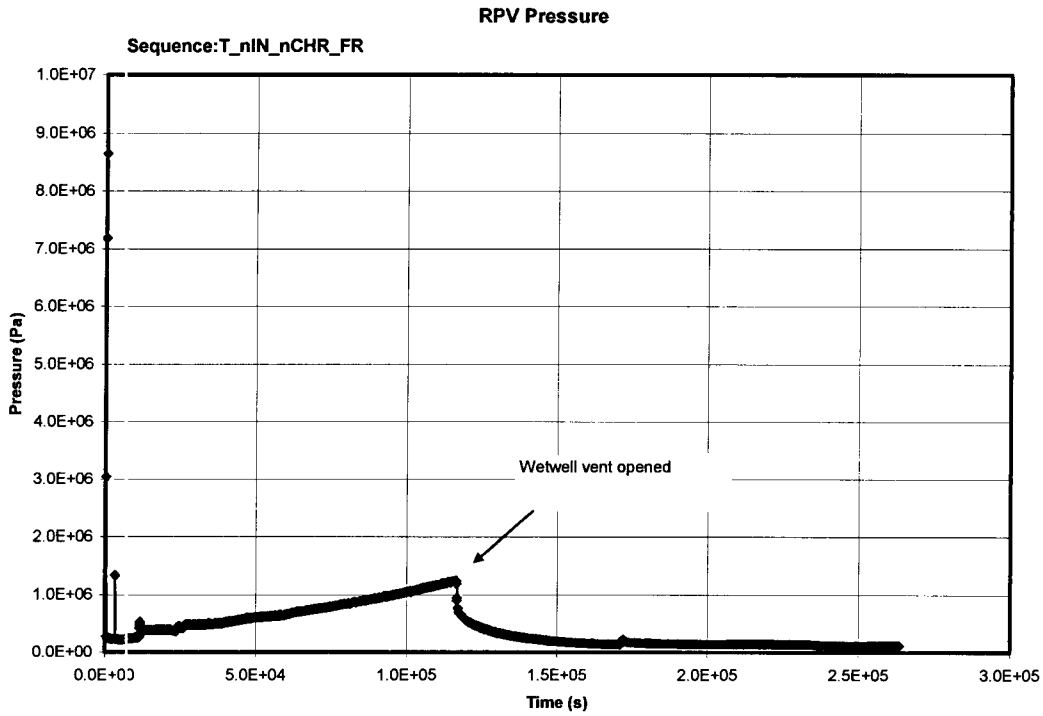

Figure 8.3-2a. T_nIN_nCHR_FR: RPV Pressure vs. Time

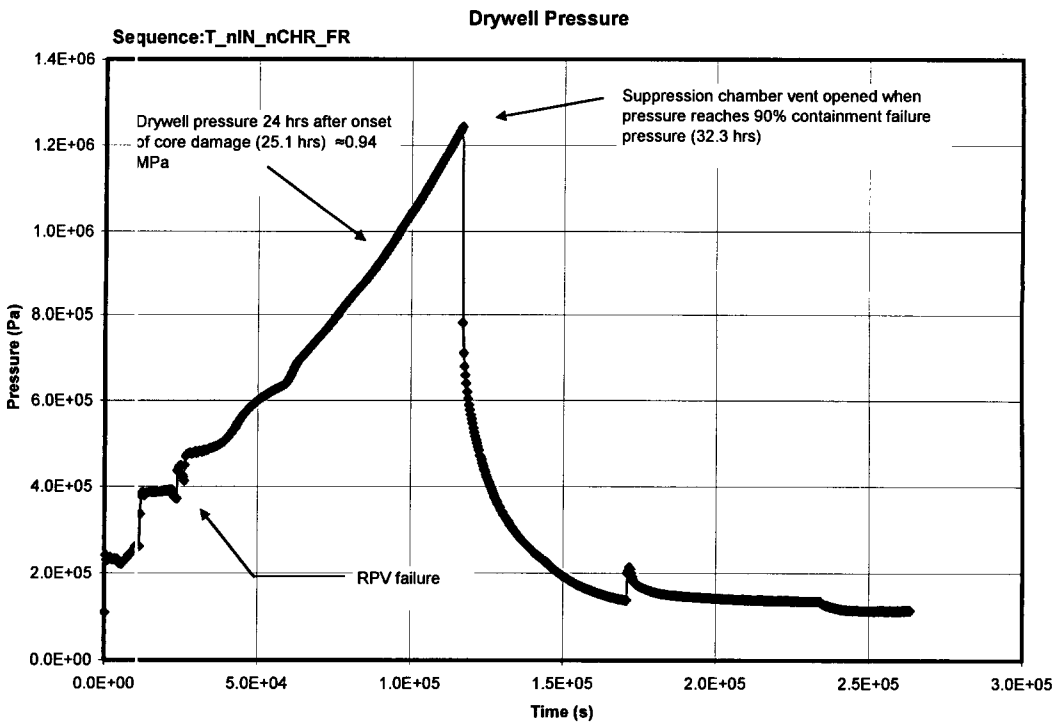

Figure 8.3-2b. T_nIN_nCHR_FR: Containment Pressure vs. Time

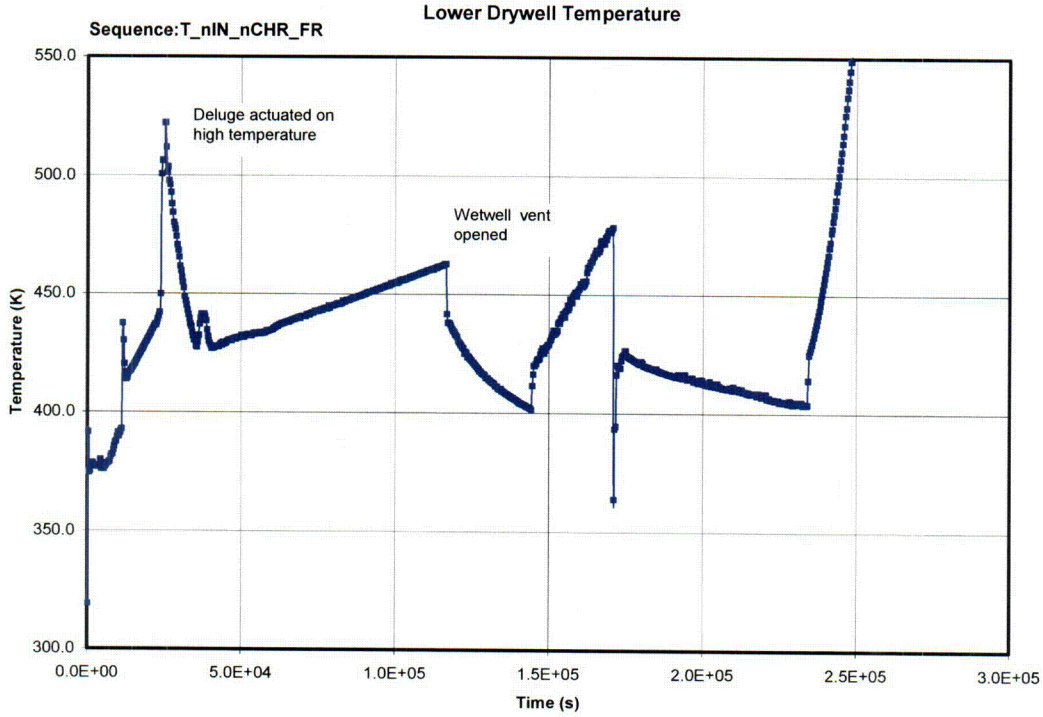

Figure 8.3-2c. T_nIN_nCHR_FR: Lower Drywell Temperature vs. Time

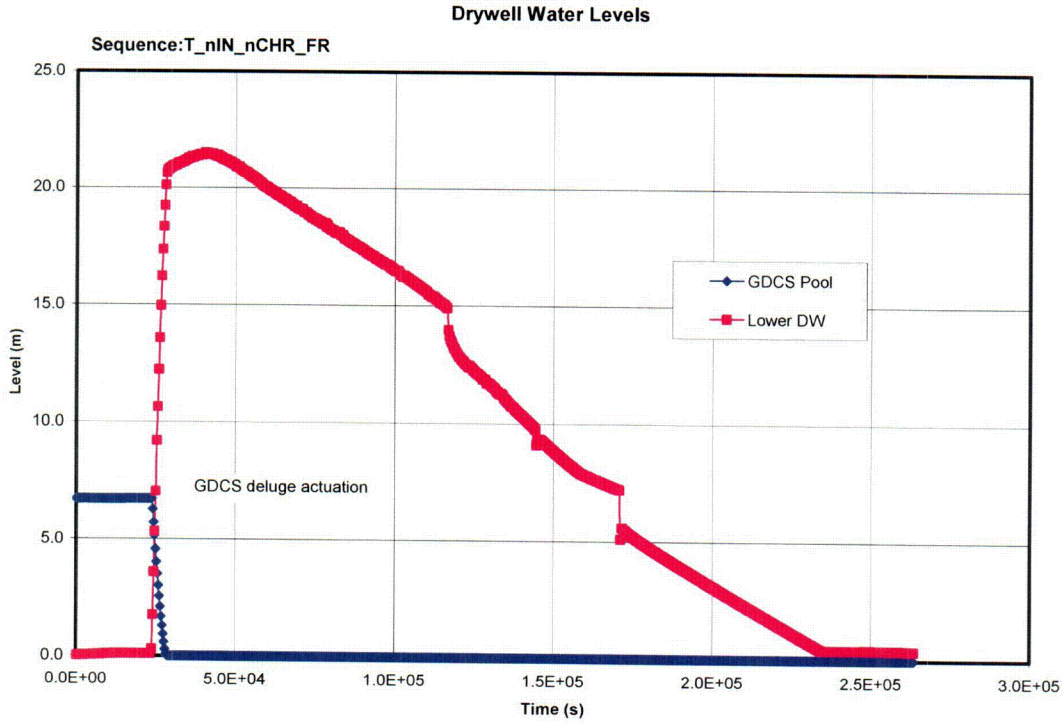

Figure 8.3-2d. T_nIN_nCHR_FR: Drywell Water Levels vs. Time

C03

Figs 8.3-3a through c: MLi_nCHR

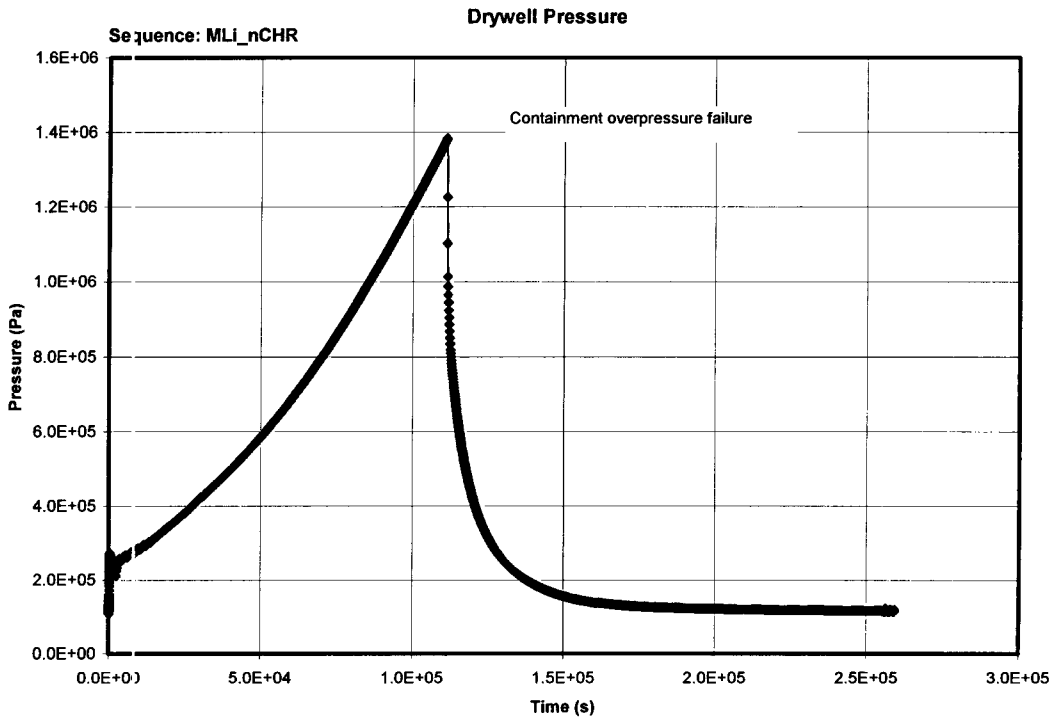

Figure 8.3-3a. MLi_nCHR: Containment Pressure vs. Time

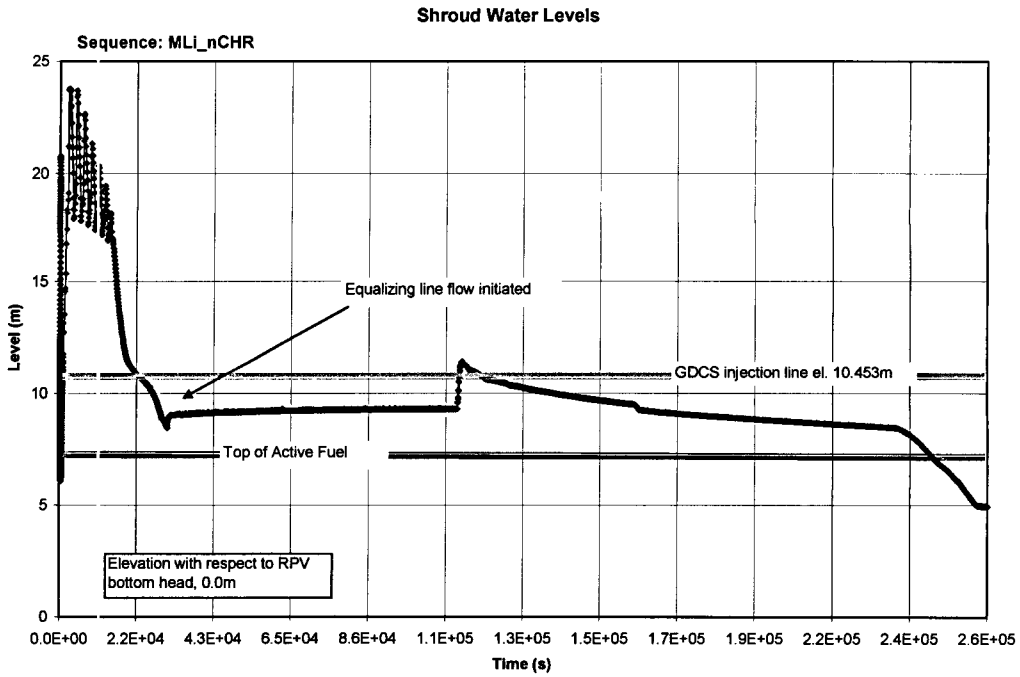

Figure 8.3-3b. MLi_nCHR: Shroud Water Level vs. Time

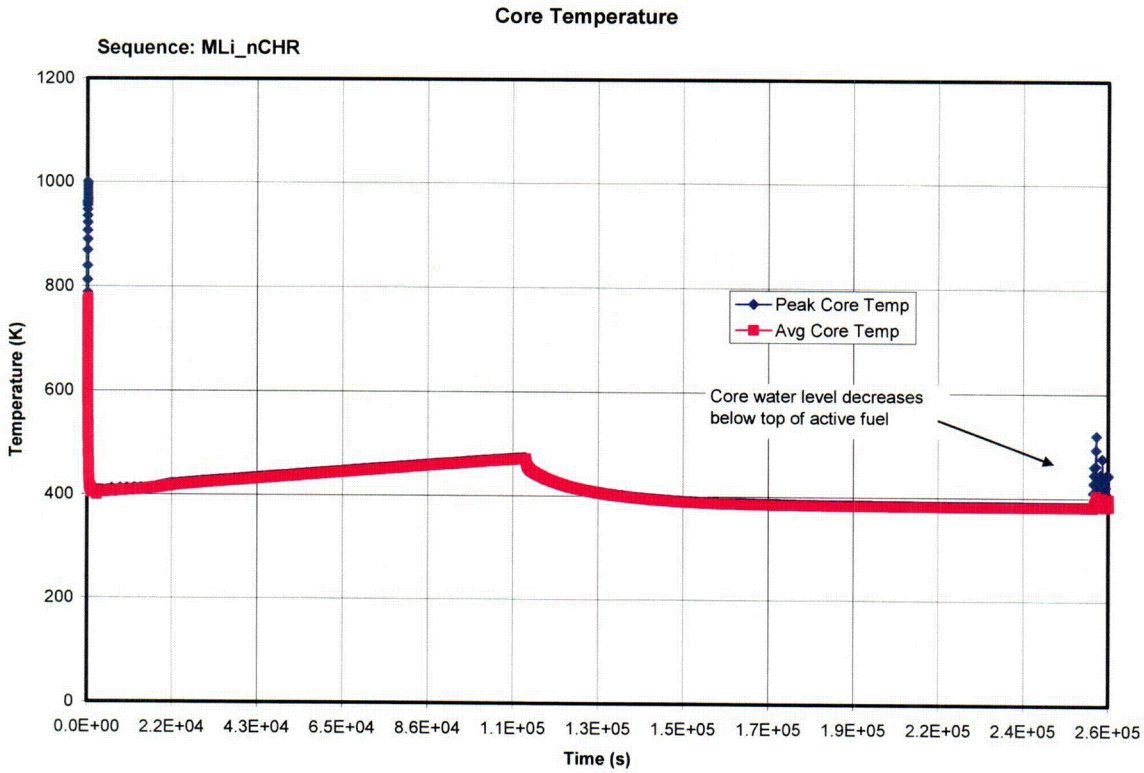

Figure 8.3-3c. MLi_nCHR: Core Heatup vs. Time

C04

Figs 8.3-4a through b: T_nDP_nIN_TSL

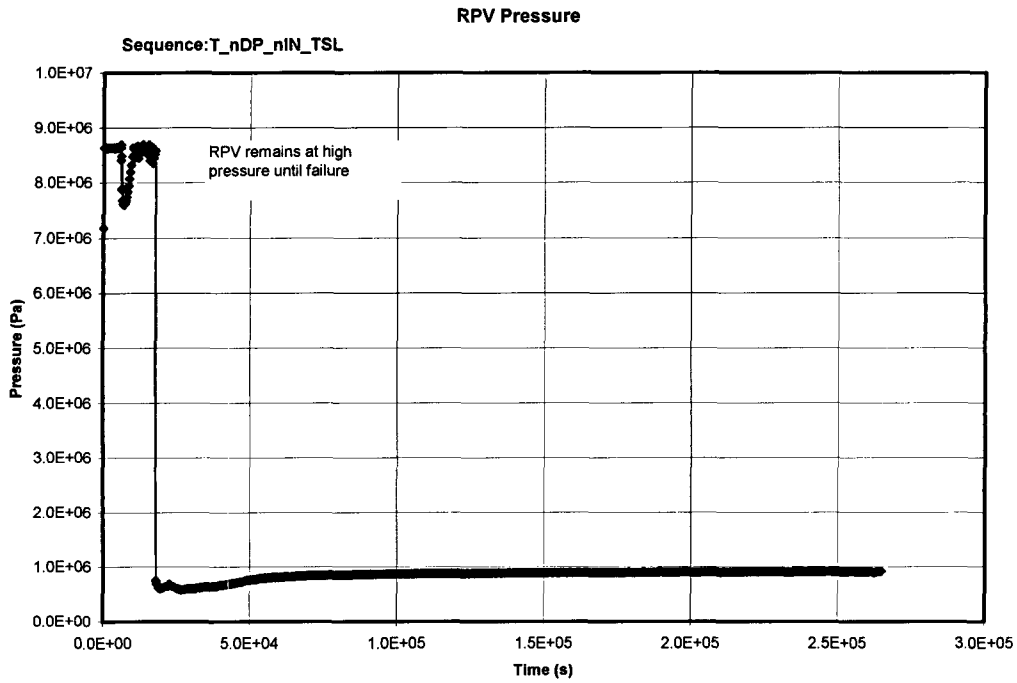

Figure 8.3-4a. T_nDP_nIN_TSL: RPV Pressure vs. Time

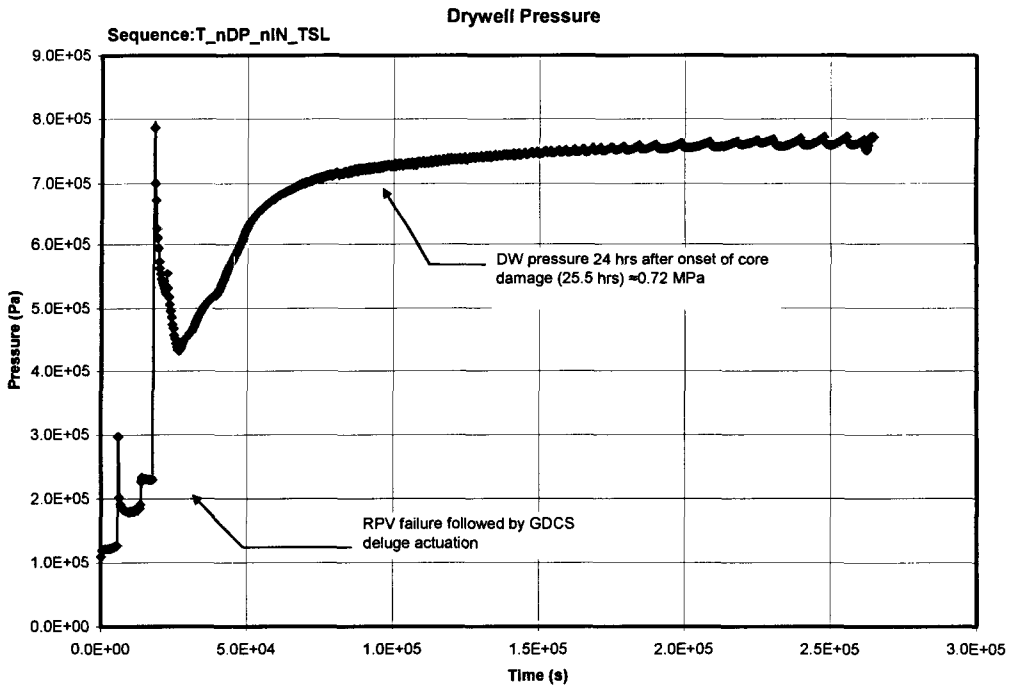

Figure 8.3-4b. T_nDP_nIN_TSL: Containment Pressure vs. Time

Figs 8.3-5a through b: T_nDP_nIN_nCHR_FR

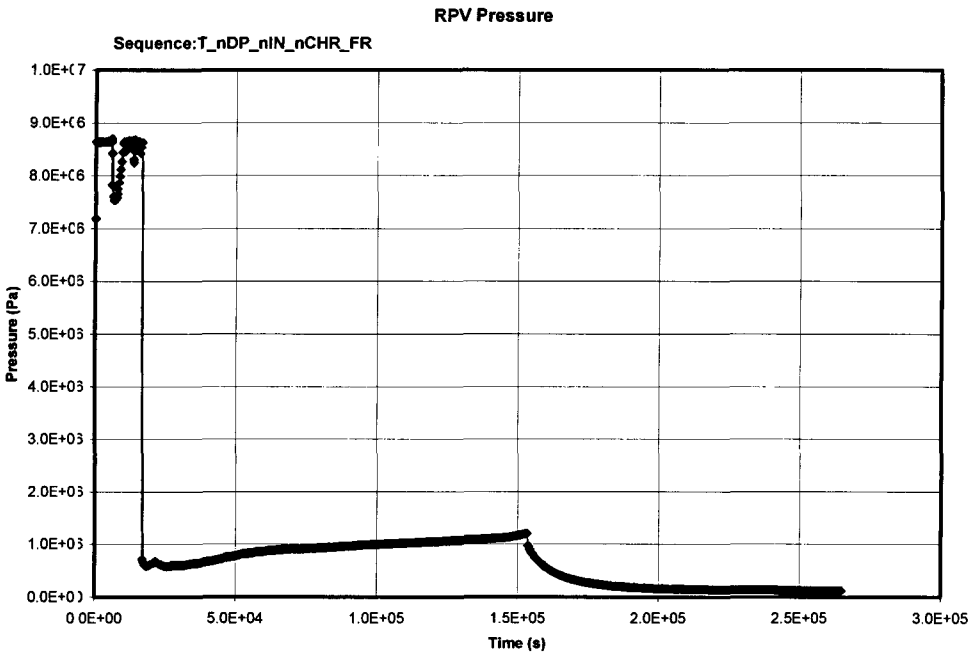

Figure 8.3-5a. T_nDP_nIN_nCHR_FR: RPV Pressure vs. Time

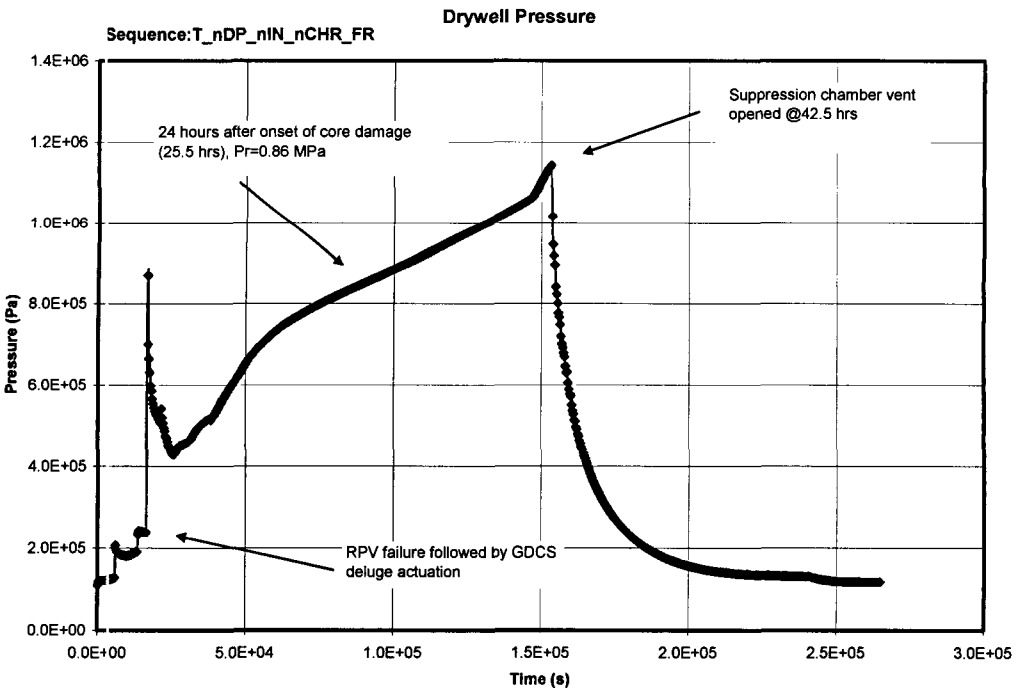

Figure 8.3-5b. T_nDP_nIN_nCHR_FR: Containment Pressure vs. Time

Figs 8.3-6a through c: Sequence T-AT_nIN_TSL

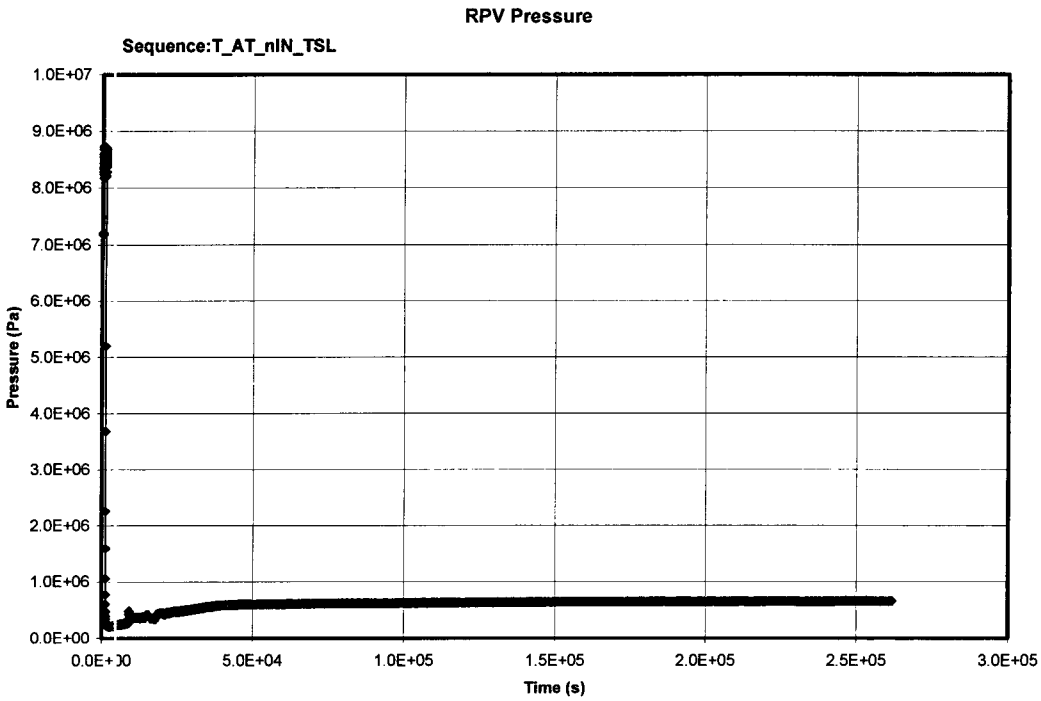

Figure 8.3-6a. T-AT_nIN_TSL: RPV Pressure vs. Time

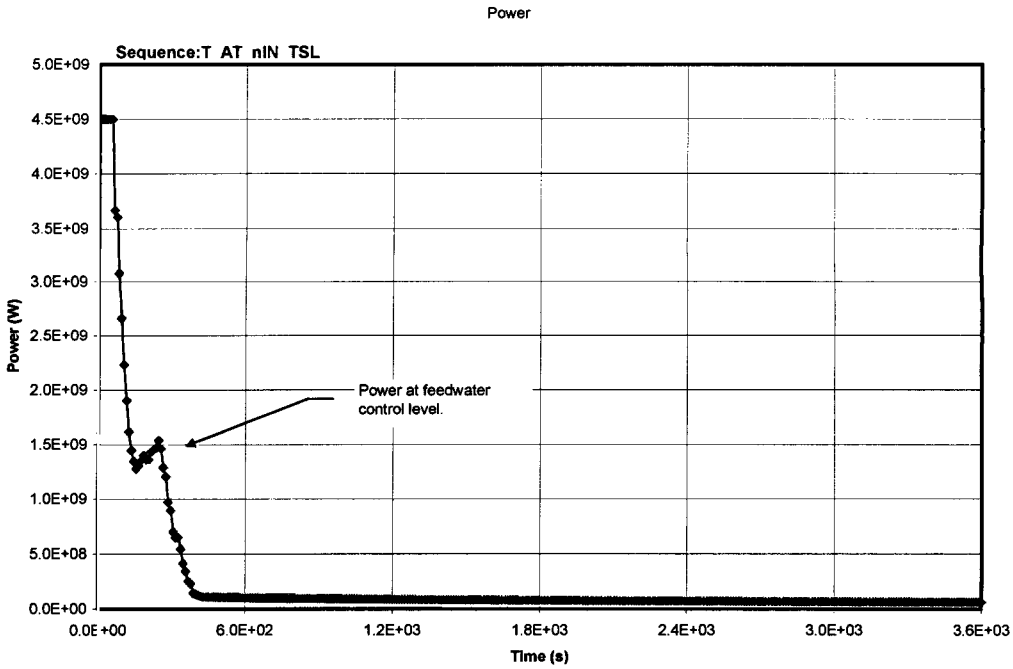

Figure 8.3-6b. T-AT_nIN_TSL: Core Power vs. Time

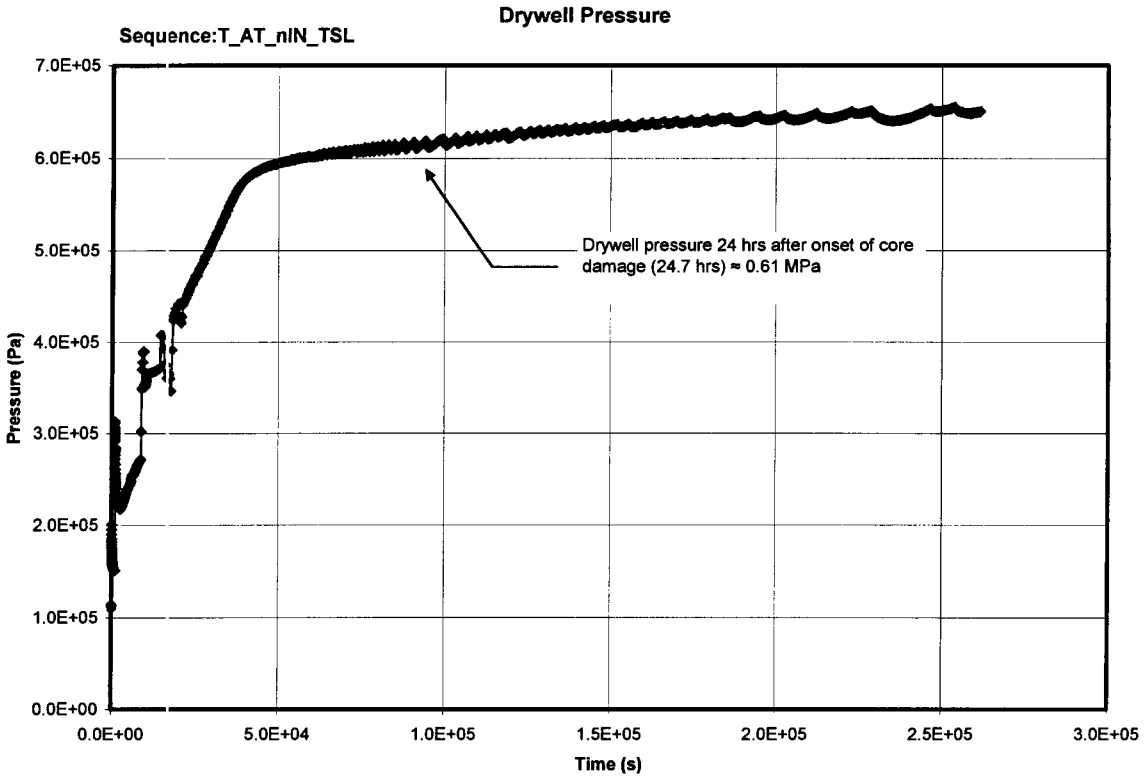

Figure 8.3-6c. T-AT_nIN_TSL: Containment Pressure vs. Time

Fig 8.3-7a: Sequence T-AT_nIN_nCHR_FR

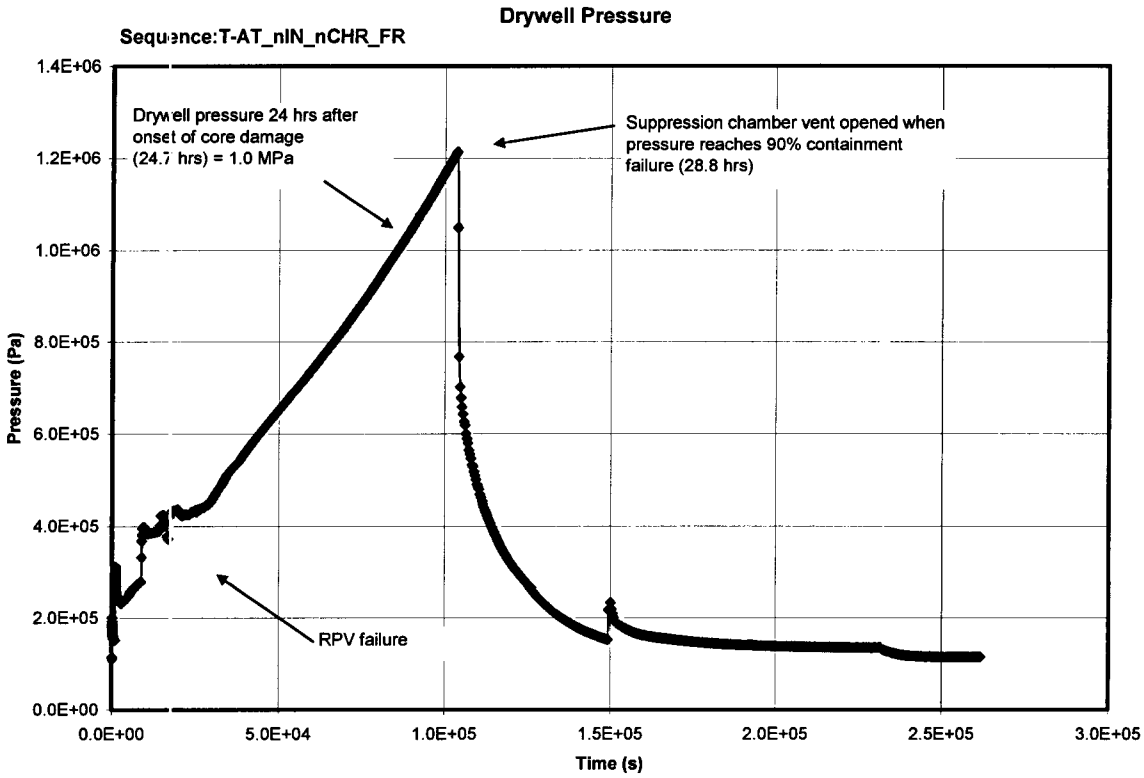

Figure 8.3-7a. T-AT_nIN_nCHR_FR: Containment Pressure vs. Time

8.4 SUMMARY

In this section, the potential for containment failure due to combustible gas generation, containment bypass and overpressurization was evaluated. Because of the ESBWR design and reliability of containment systems, the most likely containment response to a severe accident is associated with successful containment isolation, vapor suppression and containment heat removal. As a result, the containment provides a highly reliable barrier to the release of fission products after a severe accident, with the dominant release category being that defined by Technical Specification leakage (TSL). This conclusion is based on the following insights:

- The combustible gas generation analysis indicated that a combustible gas mixture within containment would not occur within 24 hours after the occurrence of a severe accident. Thus, containment failure by this mechanism is not considered further.
- Containment bypass (BYP) which results in a direct path between the containment atmosphere and environment was evaluated. A containment penetration screening evaluation indicated that there were no penetrations that required isolation to prevent significant offsite consequences. Thus, the probability of the bypass failure mode is dominated by a common isolation signal failure probability, resulting in a calculated frequency of containment bypass about four orders of magnitude lower than the TSL release category.
- Containment overpressurization was evaluated in terms of early and late loss of containment heat removal as well as the loss of the vapor suppression function. Overpressure failure was found to be about three orders of magnitude less likely than the TSL release category after a severe accident, specifically
  - The frequency of loss of containment heat removal in the first 24 hours after accident initiation, release category OPW1, was evaluated to be more than four orders of magnitude lower than the TSL release category.
  - The frequency of loss of containment heat removal in the period between 24 and 72 hours after accident initiation, release category OPW2, was evaluated to be about three orders of magnitude lower than the TSL release category.
  - The frequency of vacuum breaker failure, which would result in the shortest time to containment overpressurization because of the loss of the vapor suppression function, release category OPVB, was evaluated to be more than four orders of magnitude lower than the TSL release category.
- The need for controlled filtered venting, release category FR, in the 24 hour period after onset of core damage was evaluated. The evaluation considered loss of containment heat removal for the spectrum of applicable accident classes. In each representative sequence, operator controlled venting could be implemented to control the containment pressure boundary and potential leak path. However, venting was found not to be necessary to prevent containment failure within 24 hours after onset of core damage for scenarios in which containment heat removal is lost.

Consistent with advanced light water reactor goals established by the NRC, reliability and phenomenological analyses have established that the ESBWR containment maintains its integrity

for a 24-hour period after the onset of core damage in a severe accident. An additional insight regarding the ESBWR containment capability can be gained by calculating the "containment effectiveness". The containment effectiveness was calculated as 0.97, which exceeds guidelines provided in Reference 8.0-1 regarding the "conditional containment failure probability".

The release categories and frequencies discussed above will be retained for use in a conservative evaluation of potential source terms, as discussed in Section 9.

8.5 REFERENCES

8.0-1 SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs", April 2, 1993.

8.1-1 NUREG-0800, "Standard Review Plan", Section 6.2.5, "Combustible Gas Control in Containment".

8.1-2 Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident".

8.3-1 "MAAP4 Modular Accident Analysis Program for LWR Power Plants," Transmittal Document for MAAP4 Code Revision MAAP 4.0.6, Rev. 0, Report Number FAI/05-47, prepared for Electric Power Research Institute, 05/05/05.

8.3-2 "ABWR Design Control Document, Tier 2" Section 19.2.

Contents

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A.8 QUANTIFICATION OF CONTAINMENT EVENT TREES

The purpose of this appendix is to present the quantification of the containment event trees. The results are used to determine the conditional containment failure probability and to calculate the frequencies of the various release categories used in the Level 3 PRA.

The Containment Phenomenological Event Tree (CPET) structure is described in Section 21.1 and the Containment Systems Event Tree (CSET) structure is described in Section 8.2. This structure was appended to the Level 1 full power, internal events cutsets to determine the frequencies of the end states shown on the CPETs and CSETs. In this manner, the total CDF is mapped into appropriate Level 2 end states.

A.8.1 BINNING OF LEVEL 1 RESULTS

In order to determine the values to use on the CPET and CSET branches, the Level 1 cutsets are sorted into subclasses based on accident class and availability of offsite power. These were used to determine the conditional failure probabilities for the systems on the CSETs. The Class I and Class IV subclasses were further divided into bins that specify the water level in the containment at the time of vessel breach. These bins are used to determine the fraction of sequences that are susceptible to steam explosions in the CPET.

Table A.8-1 shows the subclass and bin assignment of the sequences above the Level 1 truncation value.

- Class I

  In these cases, core damage occurs when the RPV is at low pressure. All of these sequences remain at low pressure in the RPV through the time of vessel breach.

  The key information needed for the CPET is the water level in the lower drywell at the time of vessel breach. If the water is above 1.5 m, failure of the pedestal due to steam explosion cannot be excluded, and therefore is conservatively assumed to occur. If the water level is between 0.7 m and 1.5 m, there is the possibility of a steam explosion but failure of the pedestal is physically unreasonable. Water level below 0.7 m does not allow a steam explosion impulse of a magnitude that will challenge the containment structure. The criteria for determining the water level are presented in Section 7.2.5.

  The key information needed for quantification of the CSET is the availability of offsite power. This is needed because the logic flag settings are different in some of the CSET nodes depending on whether the initiating event is a loss of preferred power.

  The following subclasses are defined for quantification:

  - Subclass IN Vessel failure occurs at low pressure (<1 MPa)
    Initiating event does not involve Loss of Preferred Power
  - Subclass IL Vessel failure occurs at low pressure (<1 MPa)
    Initiating event involves Loss of Preferred Power

  The water level is not presented as a separate subclass, but is treated as a split fraction on the CPET.

- Class II

  There are no sequences that are binned to this accident class.

- Class III

  In these cases, core damage occurs when the RPV is at high pressure. Any of these sequences that are initiated by an inadvertent open relief valve (IORV) are grouped with the Class I sequences. The rest of the sequences are assumed to remain with high pressure in the RPV through the time of vessel breach.

  No additional information is needed for the CPET quantification.

  The key information needed for quantification of the CSET is the availability of offsite power. This is needed because the logic flag settings are different in some of the CSET nodes depending on whether the initiating event is a loss of preferred power.

  The following subclasses are defined for quantification:

  - Subclass IIIN Vessel failure occurs at high pressure (>1 MPa)
    Initiating event does not involve Loss of Preferred Power
  - Subclass IIIL Vessel failure occurs at high pressure (>1 MPa)
    Initiating event involves Loss of Preferred Power

- Class IV

  In these cases, core damage sequences are initiated by a failure to reduce reactivity in the core. By the time that the core uncovers, however, the power is essentially shut down due to lack of moderator. The principle difference between these sequences and the Class I or III is the amount of energy transferred to the containment prior to vessel breach. The excess energy is not enough to change the key physics involved in containment failure, so the class can be treated just like the previously defined classes. In the ESBWR Level 1, all Class IV sequences above the truncation limit have depressurization available throughout the sequences, so the core damage is assumed to occur when the RPV is at low pressure. All of these sequences remain at low pressure in the RPV through the time of vessel breach. Therefore, the Class IV sequences use the same quantification model as the Class I.

  Class IV is retained separately because the timing of key events is somewhat faster than the Class I events.

- Class V

  These are cases where core damage occurs with the RPV open to the environment. No containment event trees are needed. All of these sequences are assigned to the release category of BOC.

A.8.2 ASSIGNMENT OF NODE PROBABILITIES FOR CPETS AND CSETS

Each of the CPET and CSET pairs for the subclasses are solved by assigning an initiator value based on the sum of the Level 1 sequences that make up the subclass, followed by nodal probabilities for the various branches. The probabilities assigned to the branches are based on criteria and models that are described later in this section. The event trees are then quantified by

multiplying the node probabilities (or the complement on the success branches) for each sequence.

The following CPET/CSET nodes are independent of the Level 1 sequences:

- BI_FN Debris is Successfully Cooled
- BI_SP GDCS Deluge Supply to BiMAC Successful
- CIS Containment Isolation System
- DCH Containment Intact / Insignificant DCH
- EVE_DAM Pedestal Intact
- RCB_I Reactor Coolant Boundary Intact
- VT Vent Operation

These nodes are assigned the same value in all sub-classes. The nodes themselves are described in either Section 8 or Section 21. The probabilities used for the CPET / CSET nodes are summarized in Table A.8-2.

The event trees in different subclasses may have different probabilities assigned to the other nodes because of dependences with the Level 1 sequences. These nodes are:

- LD_LVL Water Level Prior to RPV Failure
- VB Vapor Suppression Function
- W1 Containment Heat Removal (Short Term: <24 Hours)
- W2 Containment Heat Removal (Long Term: <24 Hours)

Conditional probabilities for the node LD_LVL were calculated by identifying the sequences that contribute to the particular water level bin. The node probability is assigned according to the fraction of the total subclass frequency that results in the given water level condition.

Conditional probabilities for the failure branches of CSET nodes VB, W1, and W2 were calculated by developing fault trees for these nodes, converting the subclass cutsets into fault trees, linking these fault trees using simple event trees, and quantifying these event trees using the cutset methodology. Conditional probabilities were then calculated based on the sequence quantification results of the simple event trees.

The system fault trees were generated by extracting the appropriate gates from the Master CAFTA file of the Level 1 Internal Events PRA. This way, all support systems are accounted for, consistently with the Level 1 model. The top gates of the systems modeled are:

VB	EQU GT10-0001-_1	
W1	AND GT15TOP	GG21-0001-_6
W2	AND GT15-0033-_1	GG21-0001-_6

Where:

GT10-0001-_1: Isolation of Vacuum Breaker Leaks Fails

GT15TOP: 3/6 PCCS Fail

GG21-0001-_6: Both FAPCS Trains Fail SPC Op. Mode Actuation After ADS

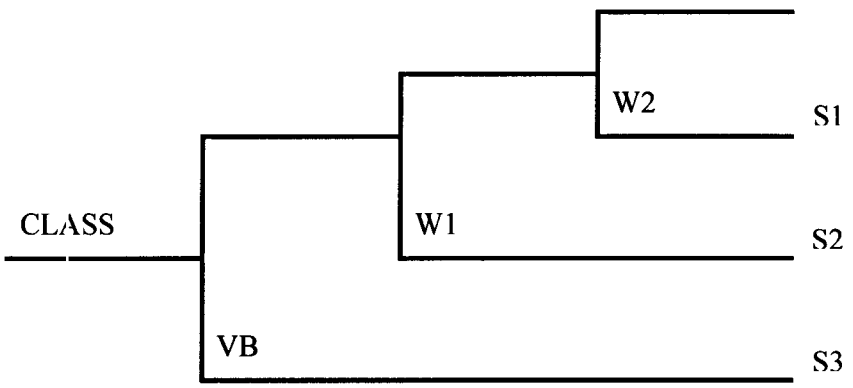
GT15-0033-_1: Loss of Pool Water (72 hours)

These fault trees are shown in Section 4.

The Level 1 accident classes were divided into subclasses (as described in A.8-1) to match the system behavior modeled by the system fault trees. Each accident class was divided in two subclasses, one resulting from the Loss of Preferred Power initiating event, and the other one resulting from any other initiating event. Table A.8-1 shows the mapping of the Level 1 Internal Events accident sequences to the accident subclasses.

Accident sequences that result in containment failure in the in the CPETs are containment bypass sequences. Therefore, dependency of the CSET nodes on these sequences does not need to be considered, and they were not included in the CDF subclasses for Level 2 CSETs. Table A.8-1 identifies which sequences are in this category.

The typical structure of the simplified event trees used for determining conditional probabilities for the system nodes is:



The conditional probabilities for failure branches were calculated as follows:

P(VB) = P(S3) / P(CLASS)

P(W1) = P(S2) / {P(CLASS) * [1 – P(VB)]}

P(W2) = P(S1) / {P(CLASS) * [1 – P(VB)] * [1 – P(W1)]}

where:

P(VB), P(W1), and P(W2) are the conditional probabilities of branches VB, W1, and W2

P(S1), P(S2), AND P(S3) are the sequence probabilities for quantified using the cutset method

P(CLASS) is the probability of the accident subclass obtained by binning the Level 1 accident sequences into subclasses

The probabilities used for the CPET / CSET nodes are summarized in Table A.8-2. The following subsections provide the basis for these probabilities.

A.8.2.1 Debris is Successfully Cooled (BI_FN)

This node is only asked following successful operation of the deluge system. Section 21 identifies the failure of this function as physically unreasonable given successful operation of deluge. This would imply a node probability of less than 10^{-3} for BI_FN, but because the design optimization of the BiMAC has not been completed, the ESBWR PRA assigns a conservative value of 10^{-2} to this node. This value is considered conservative based on the analysis in Section 21.5 and the possibility that the core would be sufficiently spread on the drywell floor to be cooled solely by the overlying pool of water from the deluge system.

A.8.2.2 GDCS Deluge Supply to BiMAC Successful (BI_SP)

Section 21.5 provides the design requirement that the failure rate of the GDCS deluge system (at high confidence) is not to exceed 10^{-3} per demand. It is assumed that the system will be sufficiently independent from any core damage prevention systems to maintain this level of reliability. Therefore the value of 1.0×10^{-3} is assigned to this node for all subclasses.

A.8.2.3 Containment Isolation System (CIS)

Section 8.1 provides a screening of potential containment penetrations that may need to be isolated in a severe accident. No penetrations were identified that would meet this requirement. There are, however, several small (1" diameter or less) pipe penetrations that have not been specified in sufficient detail in the design to completely exclude from the calculation. It is also possible that screened penetrations could be open for a small fraction of the operating time. To account for these, the CIS node is included in the CSET to represent the common mode isolation failures of these lines. A value of 3.5×10^{-5} was selected to represent isolation failure.

A.8.2.4 Containment Intact / Insignificant DCH (DCH_DAM)

Section 21.3 provides the justification for the failure of this function being physically unreasonable. Therefore the conservative value of 10^{-3} is assigned for all subclasses.

A.8.2.5 Pedestal Intact (EVE_DAM)

Section 21.4 provides the justification for the failure of this function being physically unreasonable in cases where the LDW water level is less than 1.5 m but greater than 0.7 m prior to vessel breach. This function is only asked on sequences with this water level, therefore the conservative value of 10^{-3} is assigned for all subclasses.

A.8.2.6 Water Level Prior to RPV Failure (LD_LVL)

The split fractions for this node are assigned using the same method described in Section 7.2.5. In the CPET quantification, the method is refined to consider the split fractions at the subclass level. Table A.8-1 identifies which Class I and IV sequences have high or low water level in the LDW at the time of RPV breach.

Because there were no cutsets that fell into the medium water level, some of the frequency from the low water case was conservatively moved into the medium branch. This would account for

uncertainties associated with truncation level (It is possible that a lower truncation level would allow discovery of a greater fraction of medium LDW water level sequences). It was assumed that 0.1% of the Class I and Class IV sequences would be in the medium LDW water level bin. The split fractions are for LD_LVL shown in Table A.8-2.

A.8.2.7 Reactor Coolant Boundary Intact (RCB_I)

This branch was not used in the CPET quantification. It was identified in Section 21.2 as a "splinter", which means that it is uncertain which path would be followed in any given sequence and a meaningful probability cannot be assigned to the branch. These are treated by solving both paths independently and taking the maximum of the results. In the ESBWR, the path leading to an intact Reactor Coolant Pressure Boundary is more conservative and therefore included in the results.

A.8.2.8 Vapor Suppression Function (VB)

This node was calculated using the method described at the beginning of this section. The values for the various sub-classes are contained in Table A.8-2.

A.8.2.9 Vent Operation (VT)

Vent operation is modeled using the operator action for venting containment: T11-SYS-FF-OPEN. It is assumed that the vent can be operated (manually) independently of any Level 1 mitigation systems.

A.8.2.10 Containment Heat Removal (W1, Short Term: <24 Hours)

This node was calculated using the method described at the beginning of this section. The values for the various sub-classes are contained in Table A.8-2.

A.8.2.11 Containment Heat Removal (W2, Long Term: <24 Hours)

This node was calculated using the method described at the beginning of this section. The values for the various sub-classes are contained in Table A.8-2.

A.8.3 QUANTIFICATION OF CPETS AND CSETS

The node probabilities were assigned to the CSET and CPET branches. If a node was assigned a 0.0 probability, the sequence was truncated for that subclass and not developed further. Figures A.8-1 through A.8-6 show these trees with the assigned node probabilities and the calculated end state values.

For each similar end state, the values from each of the sub-classes were summed to determine the probability of that end state. Table A.8-3 provides these results.

Table A.8-1

Level 1 Sequence Bin Assignments

Sequence	Initiating Event	CDF	Level 1 Class	L2 Subclass	LDW Water Level Bin
T-LOPP044	T-LOPP	1.63E-08	CDI	I L	Low
AT-T-LOPP011	AT-T-LOPP	5.66E-12	CDI	I L	Low
T-FDW044	T-FDW	1.20E-08	CDI	I N	Low
T-IORV029	T-IORV	7.13E-11	CDIII	I N	Low
T-IORV014	T-IORV	2.13E-11	CDI	I N	Low
T-IORV028	T-IORV	1.15E-11	CDI	I N	Low
T-IORV015	T-IORV	1.07E-11	CDIII	I N	Low
AT-T-FDW011	AT-T-FDW	4.94E-12	CDI	I N	Low
AT-T-SW003	AT-T-SW	5.70E-13	CDI	I N	Low
LL-S-014	LL-S	2.16E-13	CDI	I N	Low
ML-L-014	ML-L	2.23E-10	CDI	I N	High
LL-S-FDWB013	LL-S-FDWB	2.36E-11	CDI	I N	High
ML-L-RWCU013	ML-L-RWCU	1.27E-12	CDI	I N	High
LL-S-FDWA013	LL-S-FDWA	1.03E-12	CDI	I N	High
SL-L-RWCU014	SL-L-RWCU	8.07E-13	CDI	I N	High
SL-L-015	SL-L	4.39E-13	CDI	I N	High
SL-L-RWCU027	SL-L-RWCU	2.64E-13	CDI	I N	High
SL-L-029	SL-L	1.06E-13	CDI	I N	High
T-LOPP049	T-LOPP	3.19E-10	CDIII	I I I L	
T-LOPP030	T-LOPP	8.72E-13	CDIII	I I I L	
T-GEN031	T-GEN	4.64E-12	CDIII	I I I N	
T-PCSB030	T-PCSB	2.07E-12	CDIII	I I I N	

Table A.8-1

Level 1 Sequence Bin Assignments

Sequence	Initiating Event	CDF	Level 1 Class	L2 Subclass	LDW Water Level Bin
T-PCS030	T-PCS	1.30E-12	CDIII	I I I N	
SL-S-029	SL-S	5.31E-13	CDIII	I I I N	
T-FDW049	T-FDW	2.26E-13	CDIII	I I I N	
ML-L-015	ML-L	1.21E-13	CDIII	I I I N	
SL-L-RWCU028	SL-L-RWCU	1.11E-13	CDIII	I I I N	
AT-T-LOPP012	AT-T-LOPP	4.07E-12	CDIV	I V L	Low
AT-T-LOPP013	AT-T-LOPP	1.37E-13	CDIV	I V L	Low
AT-T-GEN012	AT-T-GEN	1.19E-10	CDIV	I V N	Low
AT-T-PCS012	AT-T-PCS	3.31E-11	CDIV	I V N	Low
AT-T-FDW012	AT-T-FDW	8.47E-12	CDIV	I V N	Low
AT-T-GEN013	AT-T-GEN	5.49E-12	CDIV	I V N	Low
AT-T-IORV005	AT-T-IORV	4.07E-12	CDIV	I V N	Low
LL-S-016	LL-S	3.33E-12	CDIV	I V N	Low
AT-T-PCS015	AT-T-PCS	1.41E-12	CDIV	I V N	Low
AT-T-PCS013	AT-T-PCS	1.11E-12	CDIV	I V N	Low
AT-T-FDW013	AT-T-FDW	2.75E-13	CDIV	I V N	Low
AT-T-IORV007	AT-T-IORV	1.37E-13	CDIV	I V N	Low
SL-L-RWCU029	SL-L-RWCU	1.71E-12	CDIV	I V N	High
ML-L-017	ML-L	2.51E-13	CDIV	I V N	High
BOC-FDWB046	BOC-FDWB	2.04E-12	CDV	V	
BOC-FDWB041	BOC-FDWB	9.79E-13	CDV	V	
BOC-FDWB045	BOC-FDWB	5.29E-13	CDV	V	
BOC-FDWB042	BOC-FDWB	4.90E-13	CDV	V	

Table A.8-1

Level 1 Sequence Bin Assignments

Sequence	Initiating Event	CDF	Level 1 Class	L2 Subclass	LDW Water Level Bin
BOC-RWCU045	BOC-RWCU	2.33E-13	CDV	V	

Table A.8-2

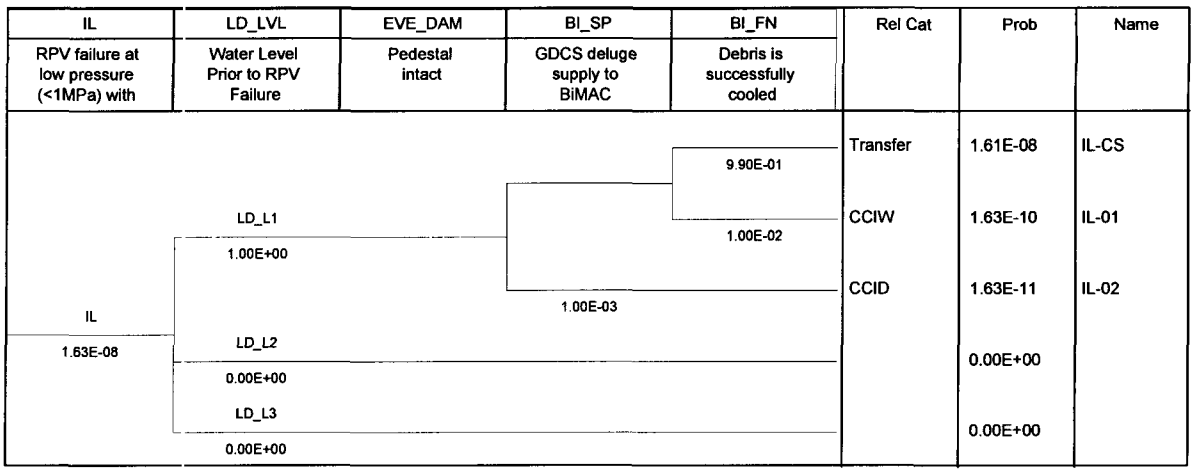
CPET and CSET Node Values

	IL	IN	IIIL	IIIN	IVL	IVN
LD_L1	1.0	9.77E-01			1.0	9.87E-01
LD_L2		2.3E-03				2.3E-03
LD_L3		2.02E-02				1.10E-02
EVE_DAM		1.0E-03				1.0E-03
DCH_DAM			1.0E-03	1.0E-03		
BI_SP	1.0E-03	1.0E-03	1.0E-03	1.0E-03	1.0E-03	1.0E-03
BI_FN	1.0E-02	1.0E-02	1.0E-02	1.0E-02	1.0E-02	1.0E-02
CIS	3.5E-05	3.5E-05	3.5E-05	3.5E-05	3.5E-05	3.5E-05
VB	5.48E-06	6.14E-06	3.00E-04	2.92E-04	5.32E-06	5.33E-06
W1	3.66E-05	7.97E-06	1.82E-04	1.77E-04	6.18E-06	5.76E-06
W2	4.12E-07	1.45E-05	7.60E-01	6.87E-01	8.98E-08	4.72E-08
VT	5.69E-02	5.69E-02	5.69E-02	5.69E-02	5.69E-02	5.69E-02

Table A.8-3

Level 2 End State Frequencies

	IL	IN	IIIL	IIIN	IVL	IVN	V	Totals
CCIW	1.63E-10	1.21E-10	3E-12	ε	ε	2E-12		2.89E-10
CCID	1.6E-11	1.2E-11	ε	ε	ε	ε		2.9E-11
EVE		ε				ε		<1E-12
EVE(CCIW*)		2.50E-10				2E-12		2.52E-10
DCH			ε	ε				<1E-12
TSL	1.62E-08	1.20E-08	7.6E-11	3E-12	4E-12	1.75E-10		2.84E-08
FR	1E-12	ε	2.26E-10	6E-12	ε	ε		2.33E-10
OPW2	ε	ε	1.4E-11	ε	ε	ε		1.4E-11
OPW1	ε	ε	ε	ε	ε	ε		<1E-12
OPVB	ε	ε	ε	ε	ε	ε		<1E-12
BYP	1E-12	ε	ε	ε	ε	ε		1E-12
BOC							4E-12	4E-12



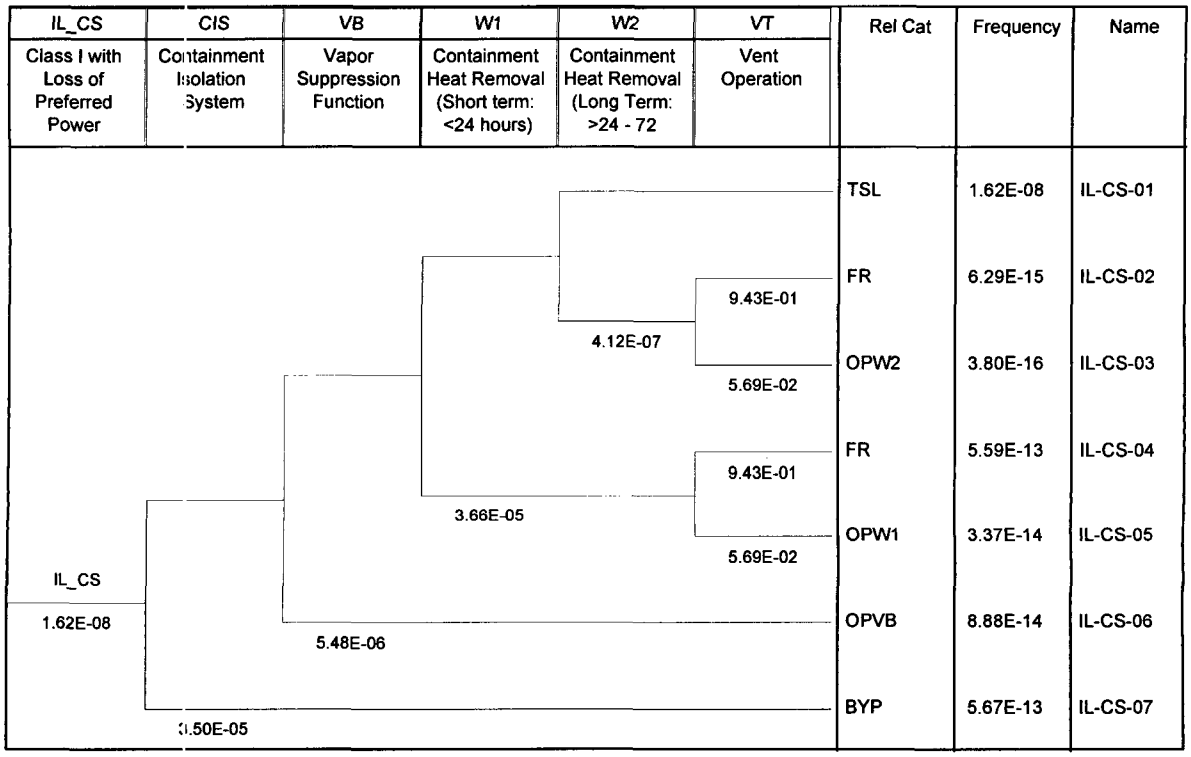
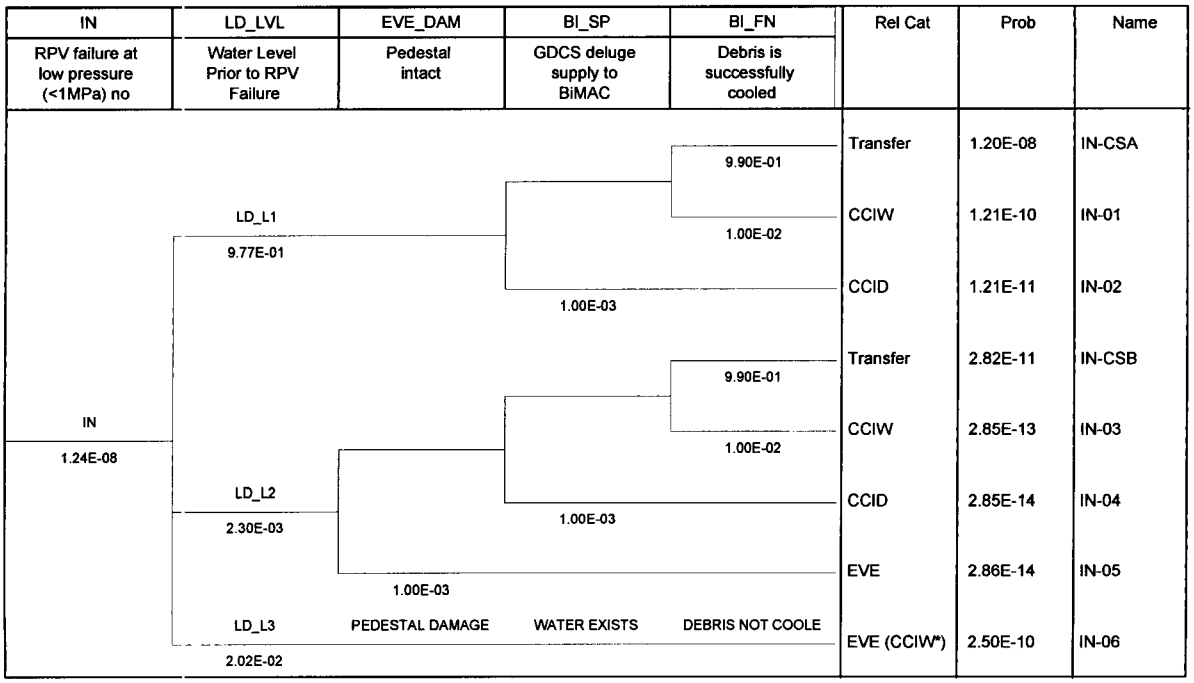


Figure A.8-1. Class I with Loss of Preferred Power CPET / CSET



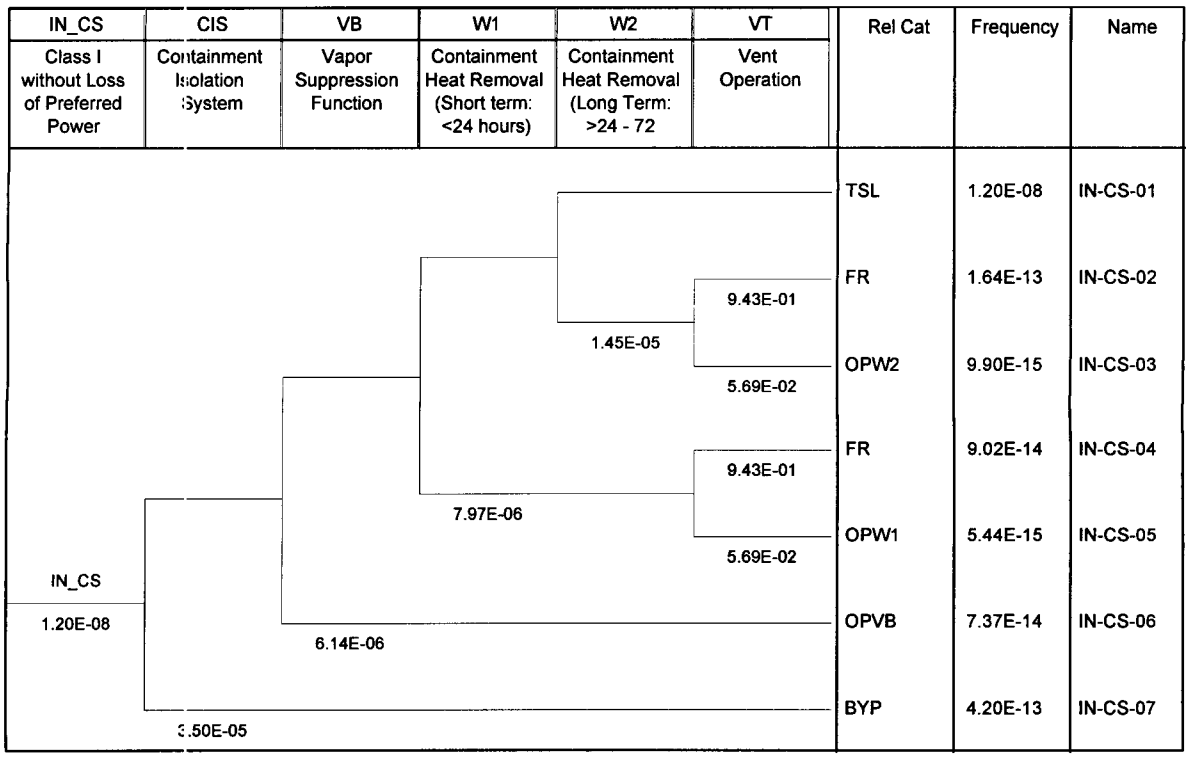
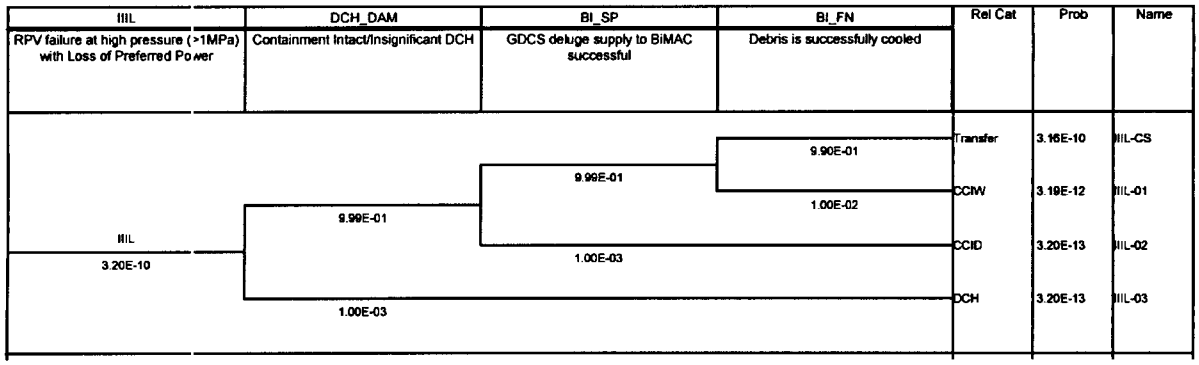


Figure A.8-2. Class I without Loss of Preferred Power CPET / CSET



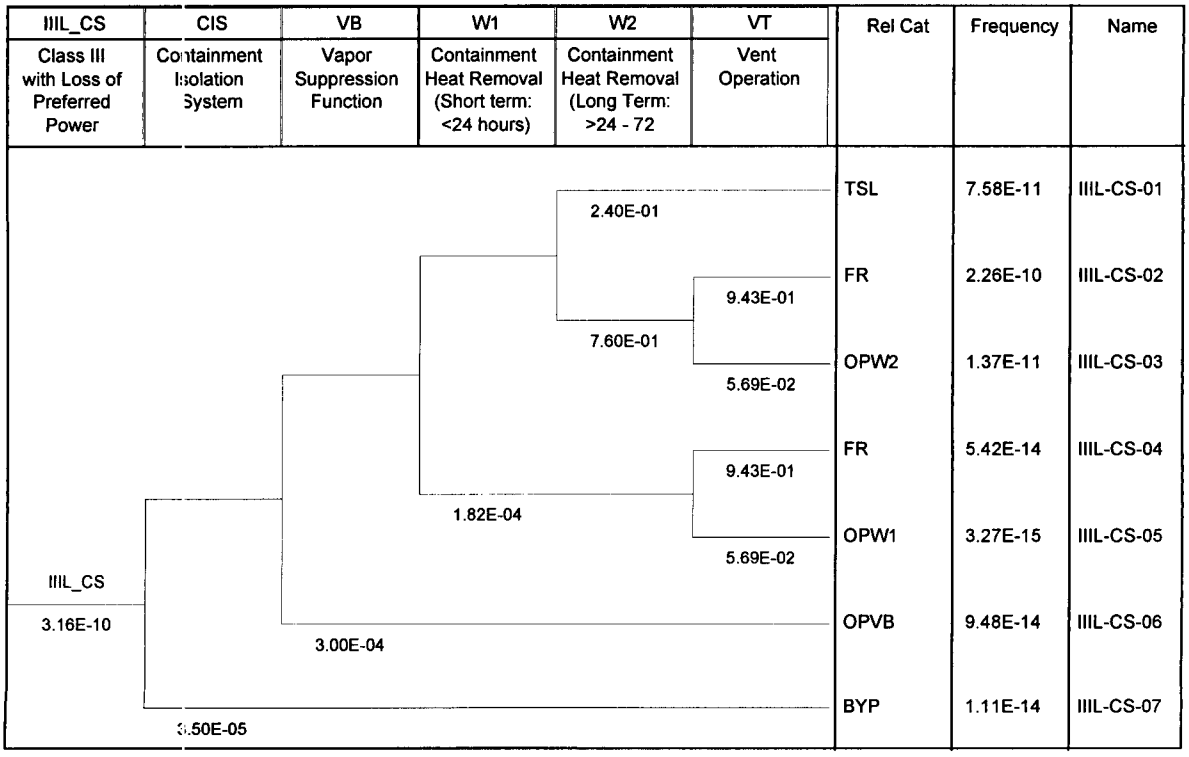
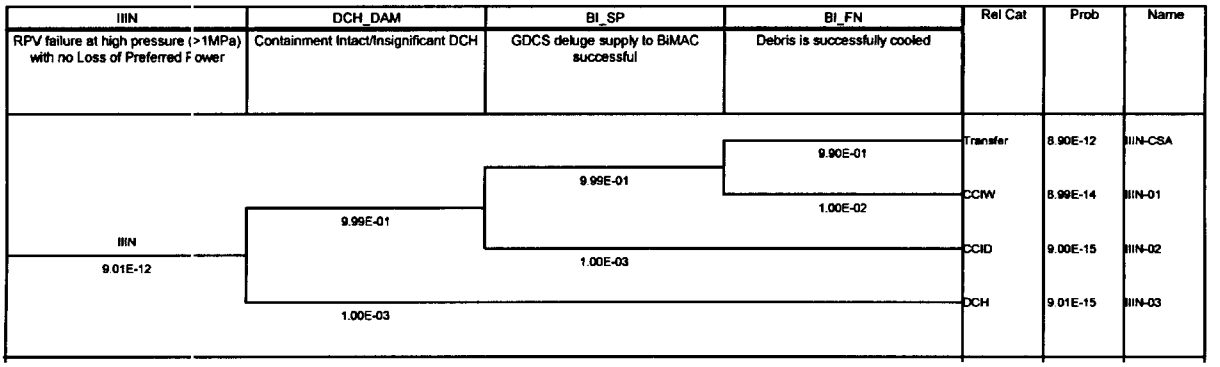


Figure A.8-3. Class III with Loss of Preferred Power CPET / CSET



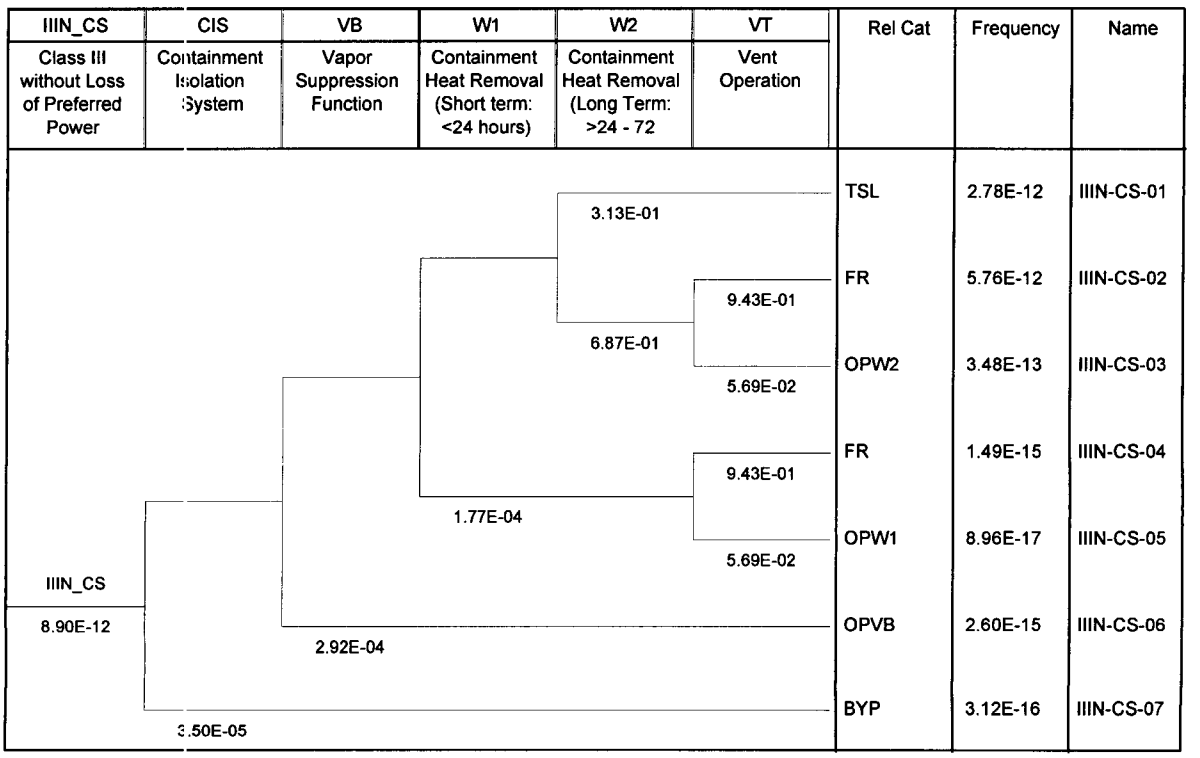
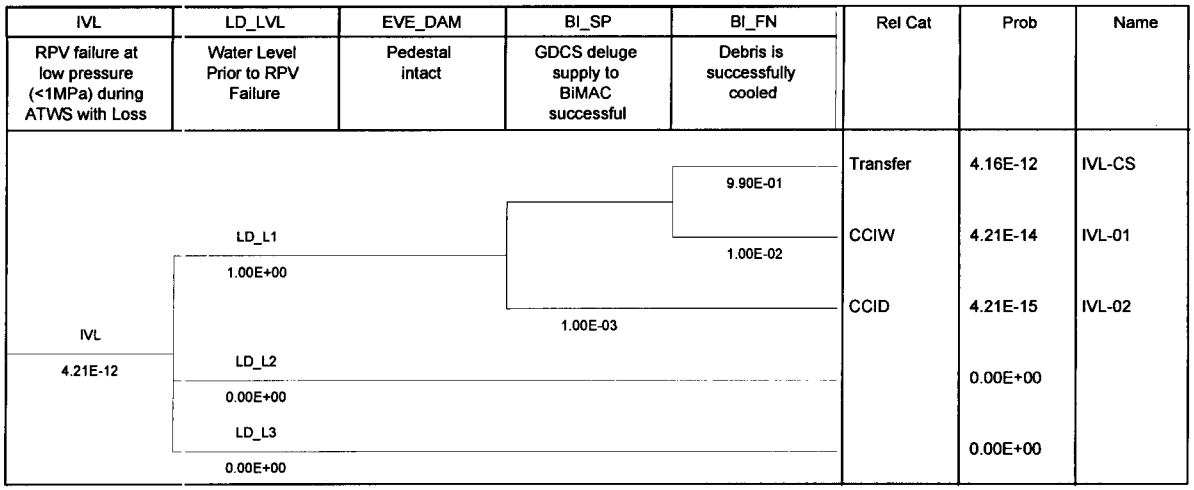


Figure A.8-4. Class III without Loss of Preferred Power CPET / CSET



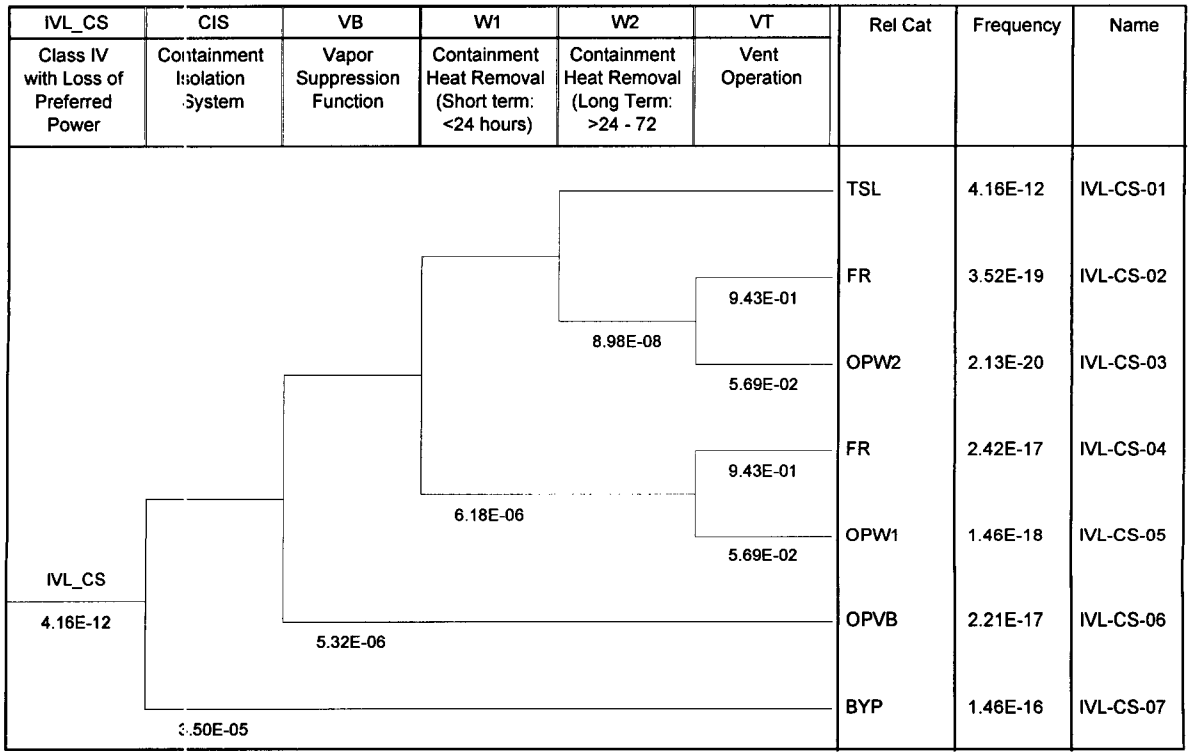
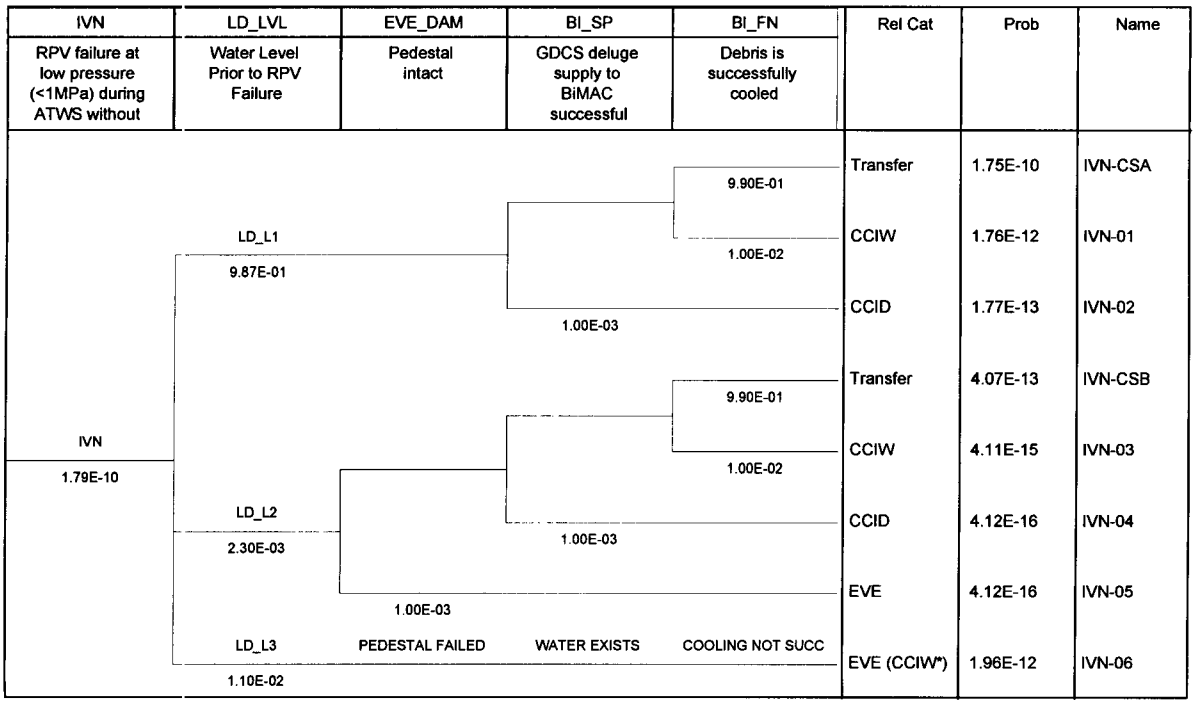


Figure A.8-5. Class IV with Loss of Preferred Power CPET / CSET



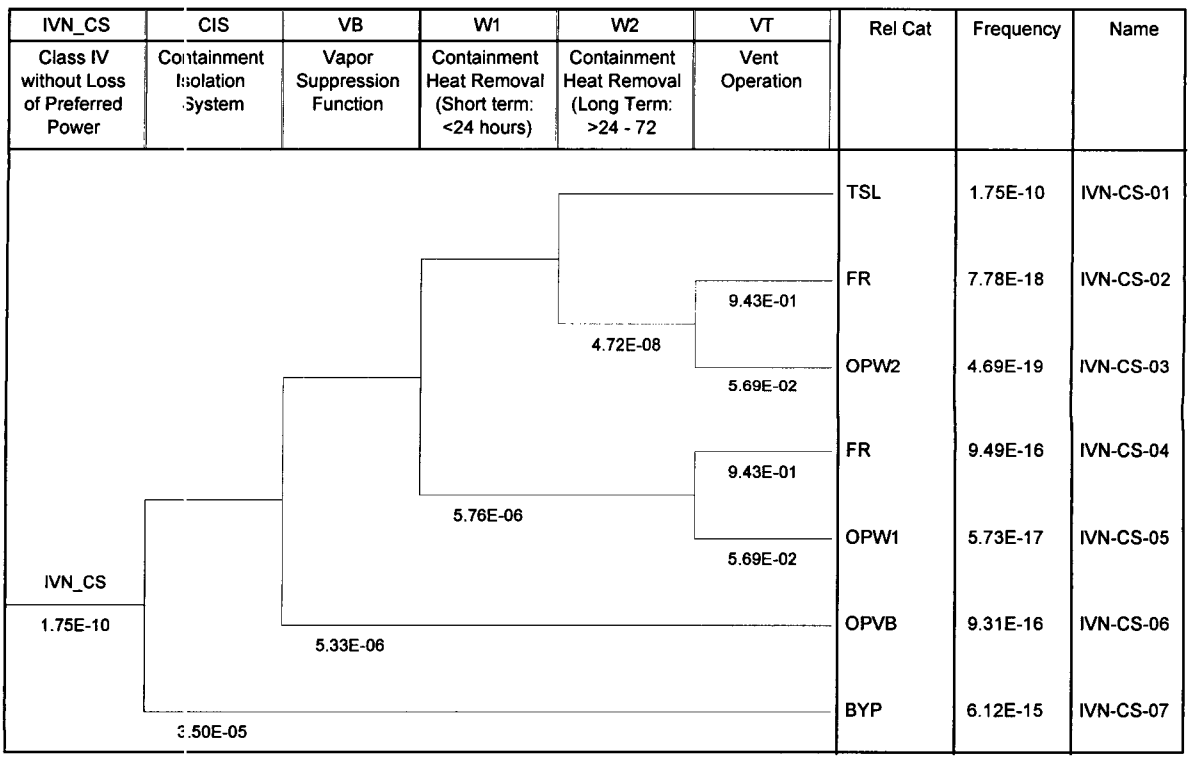


Figure A.8-6. Class IV without Loss of Preferred Power CPET / CSET

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B.8 CONTAINMENT ULTIMATE STRENGTH

This section describes the analysis and evaluation used to estimate the containment internal pressure capability and associated failure mode and location. The ultimate pressure capability of the containment structure is limited by the drywell head whose failure mode is plastic yielding of the torispherical dome. The pressure capability is 1.204 MPa gauge at 533K (500°F). It is a typical temperature for most severe accident sequences. The containment is conservatively assumed to depressurize rapidly when the pressure capability is reached. No significant leakage through penetrations is anticipated before the capability pressure is reached.

The primary function of the containment structure is to serve as the principal barrier to control potential fission product releases. The design basis event for this function is a postulated loss-of-coolant accident (LOCA). Based on this functional requirement, the containment pressure vessel is designed to withstand the maximum pressure and temperature conditions which would occur during a postulated LOCA. The ESBWR containment system employs pressure suppression, which allows a design pressure of 0.310 MPa and a design temperature of 444°K (340°F) for the primary containment pressure vessel. In addition, the suppression pool retains fission products that could be released in the event of an accident. In this section the capability of the containment structural system of the ESBWR standard plant to resist potentially higher internal pressures and temperatures associated with severe accidents is evaluated.

Primary containment, also referred to as "RCCV" for reinforced concrete containment vessel, is a cylindrical structure of steel-lined reinforced concrete. The containment is integrated with the reactor building (RB) walls from the basemat up to the elevation of the containment top slab. The top slab, together with pool girders and building walls, form the IC/PCCS pools and the services pools for storage of Dryer/Separator, fuel handling, new fuel storage and other uses. The elevation view of the reactor building/containment structural system along 0°–180° direction is shown in Figure B.8-1. The containment is divided by the diaphragm floor and the vent wall into a drywell chamber and a suppression chamber or wetwell chamber. The drywell chamber above the diaphragm floor is called the upper drywell (U/D). The drywell chamber enclosed by the RPV support pedestal (a part of RCCV) beneath the RPV is called the lower drywell (L/D). The major penetrations in the containment wall include:

(1) Drywell head

(2) The upper drywell equipment and personnel hatches at azimuth 307° and 52°

(3) The lower drywell personnel and equipment hatches at azimuth 0° and 180°

(4) The wetwell hatch at azimuth 115°

(5) The main steam and feedwater pipe penetrations at the level of the steam tunnel

Additional detail of the containment design is provided in Section 4.0.

The pressure boundary of the containment structure consists of the reinforced concrete containment vessel (RCCV) and the steel drywell head. The structural integrity of the RCCV is investigated for its global strength under internal pressure beyond the design basis using the ANSYS computer program, which is based on the nonlinear finite element method of analysis for 3D reinforced concrete structures. During various severe accident conditions, the ESBWR

containment could also be challenged by high temperatures with a typical temperature of 533°K (500°F) for most accident sequences). At typical accident temperature of 533°K (500°F), the controlling pressure capability is 1.204 MPa associated with the plastic yielding of the drywell head.

In order to evaluate liner response to over-pressurization, liner plates are included in the ANSYS analysis. The analysis results show that the liner strains are much smaller than the ASME code allowable for factory load category when the internal pressure is as high as 1.468 MPa. A separate evaluation further demonstrates that at the governing containment failure pressure of 1.204 MPa at 533°K (500°F), the liner and anchor system will maintain its structural integrity and no liner tearing will occur.

The leakage potential through penetrations is expected to be insignificant.

In conclusion, the ultimate pressure capability is limited by the drywell head. The postulated failure mechanism is the plastic yield of the drywell head. The pressure capability is 1.204 MPa gauge at 533°K (500°F). The pressure capability evaluation described above is based on the deterministic approach. The uncertainties associated with the failure pressure are assessed in Section B.8-3.

B.8.1 RCCV NON-LINEAR ANALYSIS

This subsection describes the non-linear analysis performed for the reinforced concrete containment vessel (RCCV) of the ESBWR Standard Plant. Computer code ANSYS was used for evaluation of the RCCV.

B.8.1.1 Finite Element (FE) Model Description

The containment and the containment internal structures (excluding GDCS pools structures) are axi-symmetric while the RCCV top slab together with the reinforced concrete girders even though not axi-symmetric, are idealized and included in the axi-symmetrical model. Solid elements are used to represent the girders at the top of the RCCV, approximating the stiffness of the actual structure from a detailed model of the walls and slabs in the upper pools.

To represent the restraining effects of the floors outside the containment, horizontal restraining slabs are used with equivalent material properties. The model includes concrete elements, the reinforcing steel, the steel liner plate of the drywell, the drywell head, the wetwell with the vent wall and diaphragm floor structures.

The model consists of 3780 nodal points and 2160 elements. There are 1497 elements representing concrete, whereas 249 elements are isotropic, representing steel plates. The soil below the foundation mat was modeled as 72 spring constants, 342 concentrated mass elements. See Figure B.8-1 for the model.

The ANSYS computer program permits the specification of bi-linear, brittle or ductile material properties. The concrete and soil elements are specified to have properties with no or low tensile capability. The steel plate elements and the rebar elements are specified to have ductile material properties with the same strength in tension and compression. The capability of the ANSYS program to accommodate ductile material behaviors permits both concrete cracking and yielding of steel and rebar. This allows the program to consider redistribution of forces throughout the structure due to the non-linear behavior such as concrete cracking.

B.8.1.2 Analysis

The finite element model was analyzed for internal pressure loading incrementally increased up to 1.486 MPa. The four pressure levels whose results are evaluated and summarized in Table B.8-1 are:

(1) Design pressure, labeled "PD"

(2) Structural Integrity Test 1 (SIT-1) pressure, labeled "IT", with 0.358 MPa pressure in the drywell and wetwell

(3) Severe accident pressures, labeled "SA-1" and "SA-2"

Since ANSYS performs non-linear analysis, it is necessary to apply simultaneously all loads of a loading combination. In addition to internal pressures, only the dead weights are included. The program utilizes a stepwise linear iteration technique. The first cycle results are for elastic analysis. Based upon results of the first cycle, stiffness of all elements is adjusted by the program prior to the next iteration cycle.

B.8.1.3 Results

Table B.8-1 summarizes analytical results for various loading conditions. The results are shown in terms of maximum rebar stresses, concrete stresses, liner strains and structural deformations.

Based on the ANSYS analysis, it can be concluded that the axi-symmetric components of the RCCV, as designed based on ASME Section III Division 2 code requirements, can withstand an internal pressure of 1.486 MPa, i.e., 4.8 times the design pressure, with stresses and strains in the rebar, liner plate and concrete within code allowable limits. The strength is governed by the wetwell wall and the S/P slab junction. The strength of the non-axi-symmetric top slab region is evaluated by extrapolation of the elastic analysis results using a 3D finite element model.

B.8.2 PREDICTION OF CONTAINMENT ULTIMATE STRENGTH

B.8.2.1 Structural Capability

B.8.2.1.1 Concrete Shell

The structural integrity of the RCCV axi-symmetric components has been demonstrated for an internal pressure of 1.486 MPa at ambient temperature from the ANSYS analysis. Based on extrapolation of analysis results, estimate of the ultimate pressure capability is made and discussed in this subsection. The ultimate pressure capability is assumed reached when rebar at both faces of a cross section reaching yield stress or when concrete fails by shear. The estimated pressure capabilities of the various components of the RCCV are shown in Table B.8-2. It should be noted that the extrapolation of results gives only approximate values beyond the analyzed values.

During various severe accident conditions, the ESBWR containment could be challenged by high temperatures with a typical temperature about 533°K (500°F). The effect of elevated temperature on containment pressure capability has been investigated by Argonne National Laboratory (ANL) (Reference B.8-1). The ANL study concluded that for temperatures up to 644°K (700°F), the failure mode and location did not change from the case of internal pressure alone, and the failure pressure was reduced slightly (11% maximum) from that predicted for the

internal pressure alone case. On the basis of the ANL study it is expected that with thermal effects included, the RCCV pressure capability will not be reduced below the drywell head capability for the range of temperatures considered. It is estimated that RCCV pressure capability at 500°F is 90% of the capacity at ambient temperature.

B.8.2.1.2 Drywell Head

This subsection presents an evaluation of the structural capability of the drywell head under internal pressure and temperature loading. The leakage potential of the head closure is discussed in Subsection B.8.2.2.

The drywell head which covers the 10.4 m diameter opening in the upper drywell top slab is a steel torispherical dome assembly. Under internal pressure loading, the most critical location of this type of configuration is the knuckle (or torus) region of the torispherical dome which may fail by plastic yield or buckling.

For torispherical pressure vessel heads, an approximate formula for the limit pressure at which significant plastic deformation occurs was developed by Shield and Drucker (Reference B.8-2) based on the upper and lower bound theorems of limit analysis, and it is

$$P_c = S_Y \left\{ \left(0.33 + 5.5 \cdot \frac{\mathrm{r}}{\mathrm{D}} \right) \frac{t}{L} + 28(1 - 2.2r/D) \left(\frac{t}{L} \right)^2 - 0.0006 \right\} \qquad \text{(B.8-1)}$$

where:

P_c = limit pressure

S_y = yield strength of the material

t = uniform thickness of the head

r = radius of the knuckle shell

D = diameter of the cylindrical shell

L = radius of the spherical cap

Substituting the relevant dimensions into Equation B.8-1 gives

$$P = 0.005156 * Sy \qquad \text{(B.8-2)}$$

The material yield strength depends on temperature. The actual strength of as-built material is generally higher than the specified minimum value used in design. To have a more realistic estimate of the structural strength, the minimum yield strength of material SA-516, Gr. 70 as specified in Appendix I of ASME Section III is increased by 10%. The limit pressure is 1.204 MPa at 533°K (500°F). It is noted that due to the presence of water in the reactor cavity, the outer surface of the drywell head will be at a much lower temperature than the inner surface, which is exposed to the drywell temperature. Consideration of the entire drywell head at 500°F is therefore a conservative assumption.

Buckling is another potential failure mode of the torispherical head under internal pressure since the knuckle is subjected to compressive stress in the hoop direction. Galletly has (Reference B.8-

3) proposed a design equation for preventing buckling in fabricated torispherical shells under internal pressure.

$$P_d = \frac{80\, S_y \left(\frac{r}{D}\right)^{0.825}}{\left(\frac{D}{t}\right)^{1.5} \left(\frac{L}{D}\right)^{1.15}} \qquad \text{(B.8-3)}$$

This equation is based on his previous studies (References B.8-4 and B.8-5) and is formulated for design use with knock-down (capacity reduction) factors included. As compared to all known test results (43 in total), the ratios of the actual buckling pressure to the allowable buckling pressure predicted by this equation were found to range from 1.51 to 4.01. Hence, a minimum factor of safety of 1.5 is ensured by this equation.

The test data presented in Reference B.8-3 (excluding the test performed by Blenkin since no buckling was observed at the maximum test pressure) are summarized graphically in Figure B.8-2, showing the relationship between the test and predicted pressures. The predicted pressures, as can be seen, are at least 1.5 times lower than the test results. In order to gain more insight about the data variability, statistical analyses are performed and the results are given in Figure B.8-3. The PDF (probability density function) of the data shown by solid lines is the histogram of 42 data points expressed in terms of the ratio of test to predicted pressure. It is observed that the data can be reasonably approximated by the lognormal distribution. The medium value of the test to predicted pressure ratios in the data set is 2.27 and the logarithmic standard deviation is 0.293. The resulting lognormal density and cumulative functions are shown in Figure B.8-3. The cumulative probability is 8% for the ratio up to 1.5. It means that the probability of the ratio of actual to predicted pressure being less than 1.5 is 8%. In other words, there is 92% confidence that the margin of safety against buckling is at least 1.5 when Equation B.8-3 is used. The 1.5 factor of safety corresponding to 92nd percentile is deemed sufficient for the assurance of no buckling failure against severe accident loadings of very low probabilities of occurrence.

As mentioned earlier, Equation B.8-3 has a factor of safety of 1.5 as compared to the lower bound of all known test results. From a statistical study of these test results, the medium buckling pressure is estimated to be 2.27 times the value predicted by Equation B.8-3. Multiplying by the median 2.27 value of Equation B.8-3 results in a best estimate buckling failure pressure for the drywell head of 2.667 MPa.

A comparison with the plastic yield limit pressure P_c calculated above indicates that plastic yield will occur before buckling and is the governing failure mode of the drywell head. The capability pressure is 1.204 MPa at 533°K (500°F).

B.8.2.1.3 PCCS Heat Exchangers Ultimate Pressure Capacity

The PCCS heat exchangers are part of containment boundary. Evaluation is performed to determine their ultimate pressure capacity. Analytical calculations are carried out to obtain the

maximum pressure that each heat exchanger component can resist at severe accident temperature, 533°K (500°F).

All of the sections that resist the containment pressure are evaluated in accordance with Service Level D limits of ASME, Section III, Division 1, Subsection NC, Class 2 Components.

The evaluation results reveal that the Level D pressure capacity of the most critical component in the PCCS heat exchangers is 1.77 MPa, which is 1.5 times higher than the pressure capability of the containment structure. The ultimate pressure capability would be even higher; hence the PCCS heat exchangers are not the weak link of the containment pressure boundary.

B.8.2.2 Leakage Potential

The previous subsection has addressed the structural capability of the containment structures under severe accident conditions. However, the containment function can be compromised if excessive leakage occurs before the capability pressure is reached. Leakage above the design allowable could result from failure of the liner plate and penetrations at high pressures and temperatures. The leakage potential of the liner plate and penetrations is evaluated in the following subsections.

B.8.2.2.1 Liner Plate

As discussed earlier, the containment liner plates are included in the ANSYS model. The maximum liner strains are found to be well within the code allowable when the internal pressure is as high as 1.468 MPa.

At the capacity pressure of 1.204 MPa, the maximum liner strain is 0.117% as shown in Table B.8-1 and it is considered as "free-field" strain away from discontinuities such as penetrations. To account for the effects of discontinuities, a strain concentration factor of 33, based on the Sandia containment test results (Reference B.8-6), is conservatively applied to the free-field strain, resulting in 3.96% strain. This strain level is still far lower than the ultimate fracture strain of 21% for the liner plate material. Therefore, it can be inferred that the liner plate will not tear at severe accident pressure of 1.204 MPa.

The most significant effect of thermal loading on the liner performance is a potential buckling failure which may occur if the internal pressure-induced tensile stress is not large enough to overcome the thermal-induced compressive stress. The thermal buckling tests conducted by Construction Technology Laboratories for Electric Power Research Institute (EPRI), Reference B.8-10, showed no buckling for a peak thermal transient exceeding 600°F under a pressure of 65 psi. The representative severe accident temperature for the ESBWR containment is 500°F. Since the increase in internal pressure could be much faster than the heat conduction through the containment wall, it is expected that liner buckling is unlikely to occur under combined pressure and thermal loading associated with severe accidents. As for the thermal effects on linear tearing, an ANATECH study for EPRI (Reference B.8-11) indicated that, for representative reinforced and prestressed concrete containment under WASH-1400 severe accident loading, liner tensile yielding occurred at a higher pressure and the end results near failure were essentially the same as compared to the pressure alone case. On this basis, the liner rupture pressure in excess of 1.204 MPa estimated above for pressure alone is judged to be achievable in combination with temperature. In summary, no liner failure which may lead to leakage can occur before the containment capability pressure of 1.204 MPa at 500°F is reached.

B.8.2.2.2 Penetrations

An ANL study (Reference B.8-8) assigned high priority to the study of large operable penetrations such as the drywell head closure, equipment hatches, and personnel airlocks since they are expected to have high potential for leakage under severe accident conditions. Leakage from fixed penetrations (both electrical and mechanical) appears to be less likely based on the results of experiments conducted to date by Sandia National Laboratories (SNL) and its contractors (Reference B.8-8). In fact, according to the same reference, no leakage was detected from any of the three current electrical penetration assemblies (EPAs) during the severe accident testing (steam environments).

The leakage potential of operable penetrations depends on both the relative position of the sealing surfaces and the performance of the seal material. The position of the sealing surfaces depends on the initial conditions (metal-to-metal contact is maintained under design conditions for most penetrations) and on the deformations induced by accident pressure and temperature. The seal performance depends mainly on temperature as well as the effect of thermal and radiation aging. The recent SNL tests of seals for mechanical penetrations, Reference B.8-8, indicated that

(1) In a steam environment at a constant pressure of 1.069 MPa, the mean degradation temperature was 544°K (520°F) for silicon rubber and 606°K (630°F) for ethylene propylene rubber (EPR), and

(2) In a nitrogen environment at a constant pressure of 1.069 MPa, the mean degradation temperature was 528°K (490°F) for neoprene, and

(3) The degradation temperature was not significantly affected by thermal and radiation aging.

Neoprene is not used for operable penetrations in the ESBWR containment and the seal degradation temperature is conservatively assumed to be 533°K (500°F). The SNL study also showed that even a degraded seal can prevent leakage if the separation of the sealing surfaces is small [less than 0.127 mm (0.005 in.)].

Sandia (Reference B.8-8) has proposed the following equations for "available gasket springback", Sp, for evaluating the leakage potential as a function of the compression set retention and the degradation temperature:

$$S_p = (1-C_B)\, S_q h_i \text{ for } (T<T_d) \qquad \text{(B.8-4)}$$

$$S_p = 0.127 \text{ mm } (0.005 \text{ inch}) \text{ for } (T>T_d) \qquad \text{(B.8-5)}$$

where:

C_B = the compression set retention (a dimensionless measure of the permanent set in the gasket caused by aging),

S_q = the squeeze as illustrated in Figure B.8-4 (a dimensionless measure of the gasket deformation under normal operation conditions),

h_i = the initial seal height, and

T_d = the degradation temperature of the gasket material.

Equation B.8-4 is based on the assumption that significant leakage can be prevented as long as positive compression of the gasket is maintained. Equation B.8-5 is empirical based on test results that even a degraded gasket can effectively prevent leakage if the separation of the sealing surfaces is equal to or less than 0.127 mm (0.005 in).

For the pressure-unseating drywell head closure and equipment hatches, the pressure required to separate the sealing surfaces is a function of the bolt preload, axial stiffness of the bolts and the compression flanges, and the differential thermal expansion between the bolts and the compression flanges. The separation pressure for operable penetrations typically ranges from 1.1 to 1.5 times design pressure (Reference B.8-8). In this study, the separation pressure is assumed to be the average value of 1.3 times design pressure. At and below this pressure, a metal-to-metal contact is maintained and no leakage other than design allowable leak rate is anticipated, even if the seal degradation temperature of about 533°K (500°F) has reached. Additional pressure in excess of the separation pressure is carried entirely by the bolts. The separation displacement between the sealing surface after the separation pressure is reached is:

$$s = \frac{\pi r^2 (p - p_s)}{K_b} \tag{B.8-6}$$

where:

r = the inside radius of the equipment hatch sleeve or drywell head,

p_s = the separation pressure, and

K_b = the total bolt axial stiffness.

The above expression neglects the flexibility due to axial deflection of the compression flanges caused by the Poisson effect which contributes little to the total flexibility of the bolts. This approach for predicting leakage is based on the consideration of structural deformations in terms of separation of connecting flanges of pressure unseating equipment hatches and drywell head. The adequacy of this approach has been recently confirmed by the Sandia hatch leakage tests (Reference B.8-9) in that the predicted leakage onset pressures were in favorable agreement with the test results. The drywell head anchorage to the top slab has a pressure capability higher than the drywell head shell and the leakage path of the drywell head assembly before the failure pressure is reached is through the flanges.

The drywell head is a 10.4-m diameter closure with double seal. One hundred twenty 68-mm diameter bolts hold the head in place. There are 2 drywell equipment hatches and 1 wetwell hatch in the containment wall. All of them have twenty 36-mm minimum diameter bolts with double seal; the diameters are 2.4 m for drywell equipment hatches and 2.0 m for the wetwell hatch. According to Equation B.8-6, the separation displacement at 1.204 MPa capability pressure is calculated to be about 0.146 mm (0.0058 in) for the drywell head and 0.204 mm (0.008 in) for the most flexible hatch. Although they are larger than the springback displacement of 0.127 mm if gaskets are conservatively assumed degraded at 533°K (500°F), the resulting maximum gap of 0.077 mm is deemed small. Hence, no significant leakage is expected before the capability pressure is reached.

For equipment hatches, another potential leakage mechanism is ovalization of the sleeve which causes the sleeve to slide relative to the tensioning ring (or the cover flange). An initiation of

leakage due to sleeve ovalization, however, requires significant deformations of the containment shell around the equipment hatch. The average circumferential membrane strain in the shell that is needed to result in the initiation of leakage from ovalization for equipment hatches identified in the ANL survey (Reference B.8-8) was found to range from 2.5% to 7.3% by SNL (Reference B.8-8). For the equipment hatches under consideration, the ovalization leakage onset strain which is the ratio of the sleeve wall thickness at the sealing surface to the sleeve radius ranges, as a maximum, from about 5.8% to 7.0%. At a pressure of 1.468 MPa, the maximum radial deflection of the wetwell wall was calculated to be 13.02 mm (0.512 in.) from the ANSYS analysis (Table B.8-1). The corresponding hoop membrane strain is 0.072%. It is less than 1.2% and no leakage from sleeve ovalization of the equipment hatches will occur before the capability pressure is reached.

B.8.2.3 Summary

The ultimate pressure capability of the containment structure is limited by the drywell head whose failure mode is plastic yield of the torispherical dome. The pressure capability is 1.204 MPa at 533°K (500°F). No liner leakage will occur before the capability pressure is reached. Leakage through penetrations is expected to be insignificant.

B.8.3 UNCERTAINTY IN THE FAILURE PRESSURE

The uncertainties in the prediction of the failure pressure generally result from uncertainties in the two general areas listed below:

Material Strength (yield strength, tensile strength, modulus of elasticity, etc.)

Modeling (differences between the model and reality, use of simplified models or empirical correlations, uncertainty in dead-loads, etc.)

In a number of the areas listed above very little data may be available to guide the structural analyst in characterizing the uncertainty. Consequently, it is generally necessary to rely to a large extent on engineering judgment and past results to quantify these uncertainties.

As noted above a significant contributor to the uncertainty in the prediction of ultimate capacity derives from uncertainties in the material properties. For most structural materials the lognormal distribution has been shown to be a good model for the variability in material strength. Largely for this reason the lognormal distribution is generally selected to characterize the uncertainty in the prediction of the ultimate pressure capacity for structural components.

The most common form of the lognormal probability density function is:

$$p_f(p) = \frac{1}{p\sqrt{2\pi\beta_c}} \exp\left[-\frac{1}{2}\left[\frac{1}{\beta_c}\ln\left(\frac{P}{P_{med}}\right)\right]^2\right] \qquad \text{(B.8-7)}$$

where:

$p_f(p)$ = the lognormal probability density function for failure pressure,

β_c = logarithmic standard deviation on the pressure capacity p,

p_{med} = the median pressure capacity.

β_c is a combination of the logarithmic standard deviation of material strength uncertainty β_s, and the logarithmic standard deviation of modeling uncertainty β_m. β_c is determined from the standard relationship for combining independent uncertainties.

$$\beta_c = \sqrt{\beta_s^2 + \beta_m^2} \tag{B.8-8}$$

The cumulative distribution function (CDF) is obtained by integrating the PDF

$$P_f(P) = (P \le p) = \int_0^P p_f(p)\, dp \tag{B.8-9}$$

$P_f(p)$ = the probability that the failure pressure is less than pressure p.

The failure pressure of 1.204 MPa [at a drywell temperature of 533 K (500°F)] can be considered to be a lower bound value since a higher failure pressure of 1.632 MPa would be predicted for the drywell head when plastic failure mode is analyzed with Equation 4 of Reference B.8-5 shown below is used.

$$P_{c2} = \frac{20(r/D)^{1.78} S_y}{(D/t)^{1.08} (L/D)^{0.87}} \left[1 + 0.1 \left(\frac{r}{D} \right)^{-1.37} \right] (1 + 0.001 S_y^{1.1}) \tag{B.8-10}$$

The median failure pressure is therefore assumed to be 1.632 MPa. The uncertainties associated with this analysis were estimated using engineering judgment and the results from prior analysis. Typical values for the uncertainties associated with material properties of steel structures range from a β_s of approximately 0.06 to 0.10 and the uncertainties associated with the modeling of simple steel structures range from a (β_m) of approximately 0.10 to 0.16 (References B.8-12 and B.8-13). Using nominal values for β_s of 0.08 and for β_m of 0.14 results in an estimated value for the standard deviation (β_c) of 0.16. The use of 0.14 for β_m is also consistent with Reference B.8-14 in that the variability associated with the modeling error by the use of approximate methods including that for torispherical heads is 0.12. The adequacy of using 0.08 for β_s associated with material property uncertainties at high temperatures is addressed as follows. The ESBWR drywell head material is ASME SA-516, Gr. 70. This material was tested, according to Reference B.8-16, for temperatures up to 477 K (400°F) using the specimens taken from the Sandia's 1/8 scale steel containment model. No actual test data are given but the inferred stress-strain curves (Figures 3.4 and 3.5 of Reference B.8-15) for Gr. 60 of the same material considered in the nonlinear analysis show the same characteristics of nonlinear stress-strain relationships for temperatures up to 477 K (400°F). The same trend is expected to exist for temperatures up to 811 K (1000°F) since it is the upper temperature limit of which the specified minimum material strength is given in the ASME code. Having established this, the variability associated with material strength is expected to be the same regardless of temperatures. The statistical data of 52 tests of A516, Gr. 70 (same as SA-516, Gr. 70) given in Reference B.8-16 show that the average yield strength is 335 MPa and the standard deviation is 24.3 MPa. The coefficient of variation is thus 0.073, which is close to 0.08 used for β_s.

Using the above parameter values results in the containment capacity fragility curve in Figure B.8-5 and Table B.8-3.

The pressure at two standard deviations below the mean is 1.111 MPa. This pressure is 3.58 times the design pressure.

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Table B.8-1

Summary of Stresses and Strains

Loading Case		Maximum rebar Stress/Allow. Stress		Liner Strain		Concrete Compress. Stress/ Allowable stress	Component Rebar Stresses / Allowable Stresses (MPa)										Max. Radial Defl. Wetwell. mm
							MAT		SP/S		Wetwell wall		Upper drywell wall		Top Slab		
Title	P.D. MPa	Merid. MPa	Hoop MPa	Tensile mm/mm	Compress. mm/mm	(MPa)	Mer.	Hoop.	Mer.	Hoop.	Mer.	Hoop.	Mer.	Hoop.	Mer.	Hoop.	
PD	0.310	60.7	11.8	2.40E-04	-1.03E-04	-4.7	23.4	8.0	21.0	10.2	10.0	7.7	20.8	5.7	19.3	7.4	0.68
		248.4	248.4			-20.7	248.4	248.4	248.4	248.4	248.4	248.4	248.4	248.4	248.4	248.4	
IT	0.357	63.9	20.2	2.42E-04	-1.113E-04	-4.8	24.1	5.2	21.4	10.2	9.9	8.7	27.8	6.6	20.6	8.6	0.79
		248.4	248.4			-20.7	248.4	248.4	248.4	248.4	248.4	248.4	248.4	248.4	248.4	248.4	
SA-1	1.210	217.1	262.0	1.17E-03	-5.10E-04	-19.7	217.1	80.4	172.1	70.4	145.6	111.9	188.8	88.7	198.4	41.0	10.19
		414.0	414.0			-34.5	414.0	414.0	414.0	414.0	414.0	414.0	414.0	414.0	414.0	414.0	
SA-2	1.468	284.9	318.7	1.65E-03	-8.59E-04	-31.0	284.9	119.5	221.9	93.3	185.3	142.9	245.4	117.5	267.5	53.7	13.02
		414.0	414.0			-34.5	414.0	414.0	414.0	414.0	414.0	414.0	414.0	414.0	414.0	414.0	

Table B.8-2

Summary of Pressure Capabilities of Various Components of the RCCV and the Drywell Head

Structural Component	Failure Mode[4]	Ultimate Pressure Capability in MPa (gauge)		
		Ambient Temp.[2]	500°F[3]	1000°F[5]
Wetwell	Rebar yielding at DF-Wetwell joint	4.33	3.9	1.94
Upper Drywell	Rebar yielding at DF-Upper Drywell joint	4.8	4.32	1.89
Lower Drywell (Pedestal)	Shear failure at Basemat joint	2.85	2.57	1.16
Suppression Pool Slab	Shear failure at Wetwell joint	1.468	NA[6]	NA[6]
Basemat	Shear failure at Pedestal joint	3.63	3.26	---
Drywell Head	Plastic failure at Knuckle	1.486	1.204[1]	1.13

Notes:

1. Yielding strength based on 10% increase of code-specified values.
2. Extrapolated from Table B.8-1 results for concrete components.
3. Conservatively taken as 90% of the pressure value at ambient temperature.
4. Failure criteria
   - Rebar: 0.01 total strain
   - Concrete: 34.5 MP f'_c compression, 1.95 MPa f_{cT} tension
   - Liner: 0.02 total strain
   - Drywell head: plastic failure
5. Estimated from past study of a similar design.
6. Water stays in suppression pool.

Table B.8-3

Probability of Failure Versus Pressure

(Median Pressure = 1.632 MPa gauge, β_c= .16)

Pressure (MPa) gauge	Ln(p/pmed) β_c	Pf(p)
0..845	-4.114	1.94E-05
0.847	-4.099	2.07E-05
0.849	-4.084	2.21E-05
0.979	-3.184	7.01E-04
1.11	-2.409	7.99E-03
1.197	-1.937	0.026
1.226	-1.788	0.037
1.371	-1.089	0.138
1.458	-0.705	0.241
1.545	-0.342	0.366
1.632	0.0	0.5
1.719	0.325	0.627
1.806	0.633	0.737
1.893	0.927	0.823
1.980	1.208	0.886
2.067	1.477	0.930
2.154	1.735	0.959
2.285	2.103	0.982
2.415	2.449	0.993
2.458	2.56	0.995
2.5	2.666	0.996

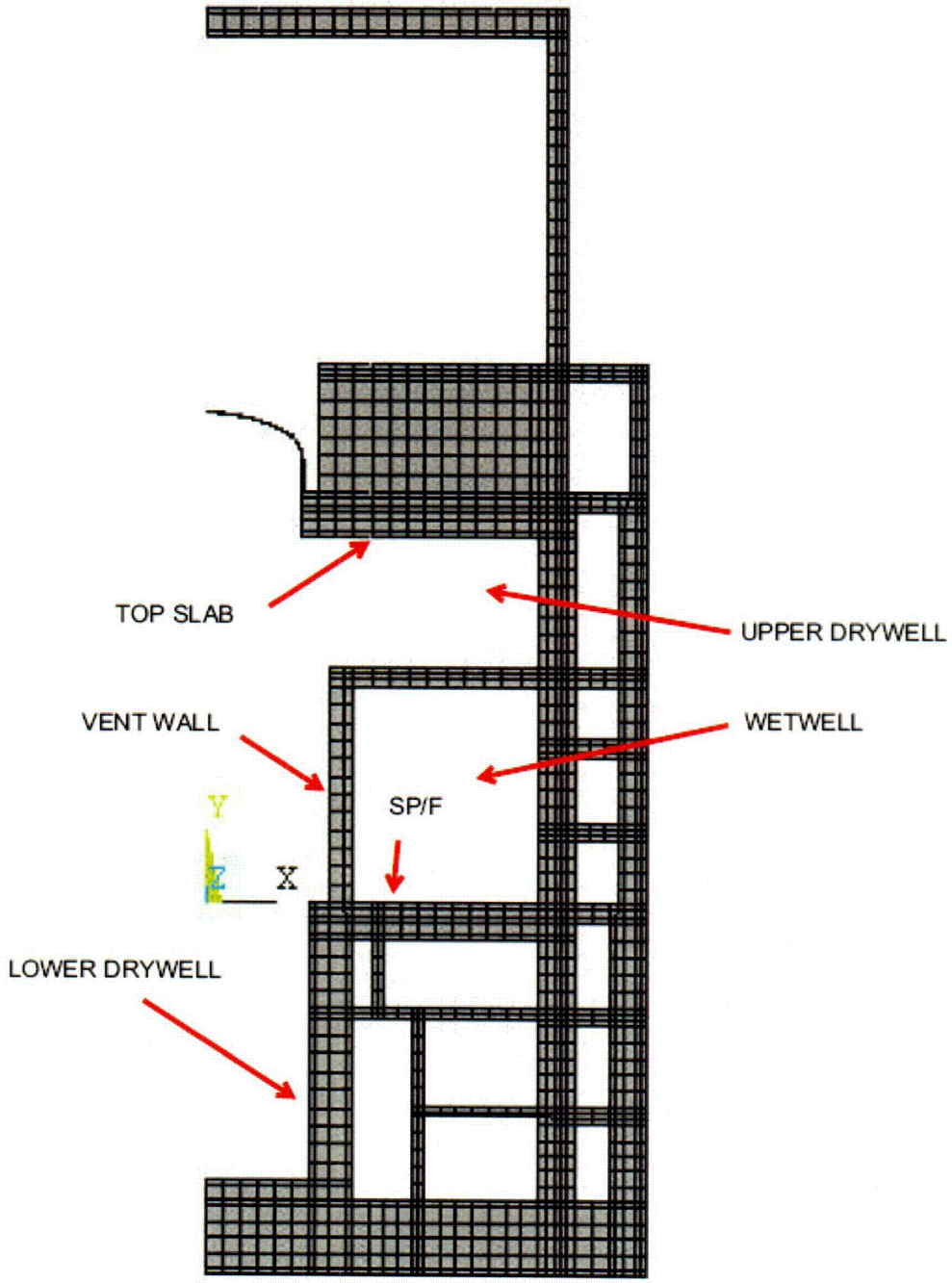


Figure B.8-1. ANSYS model

C05

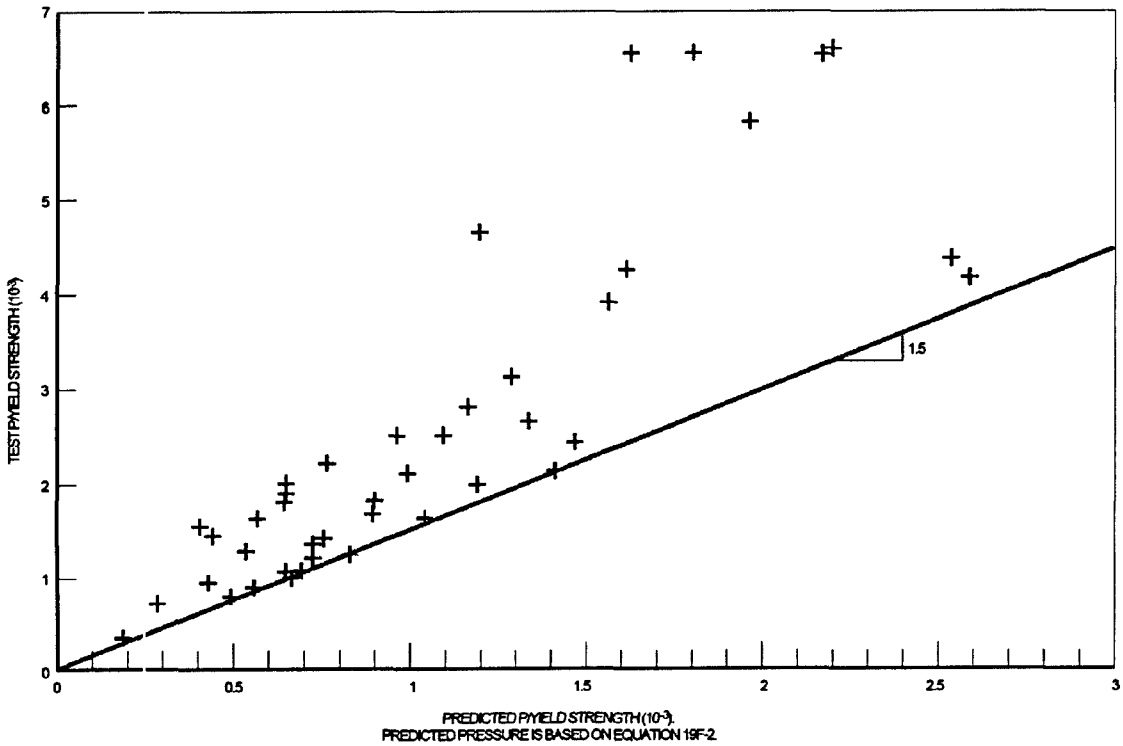


Figure B.8-2. Torispherical Head Buckling Test Data

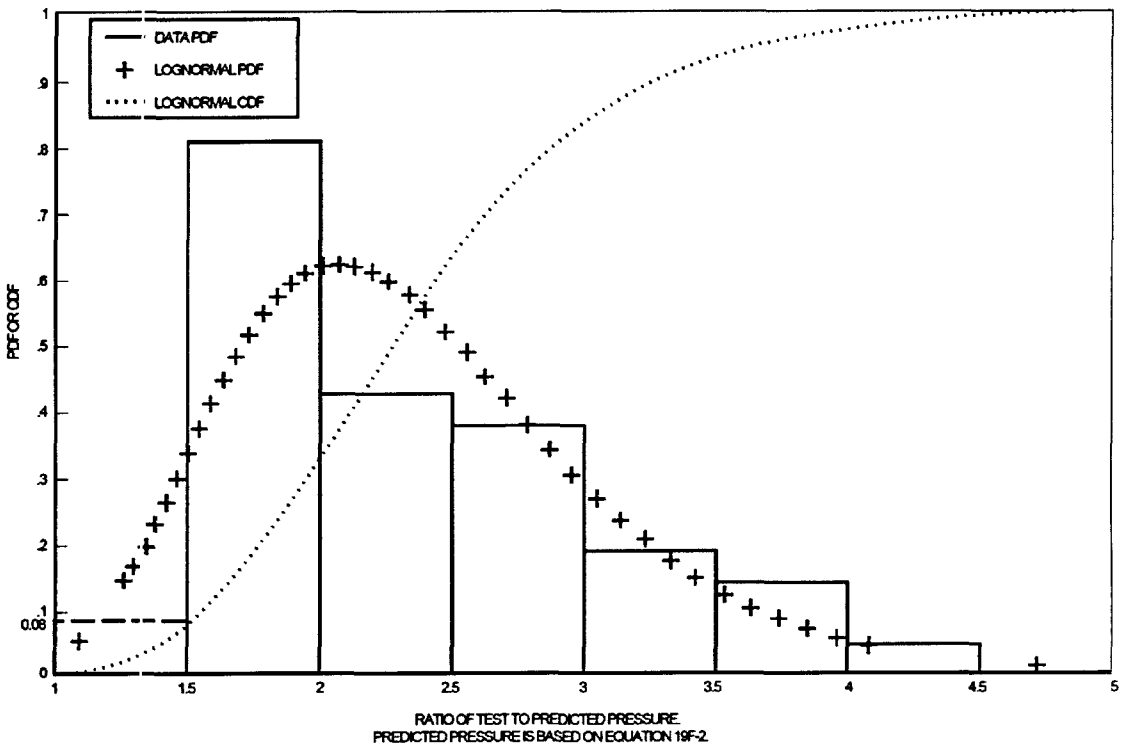
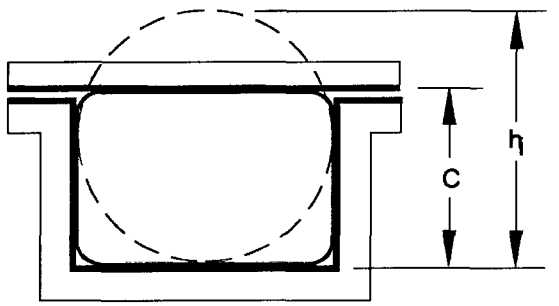


Figure B.8-3. Torispherical Head Buckling Test Data Statistical Distribution



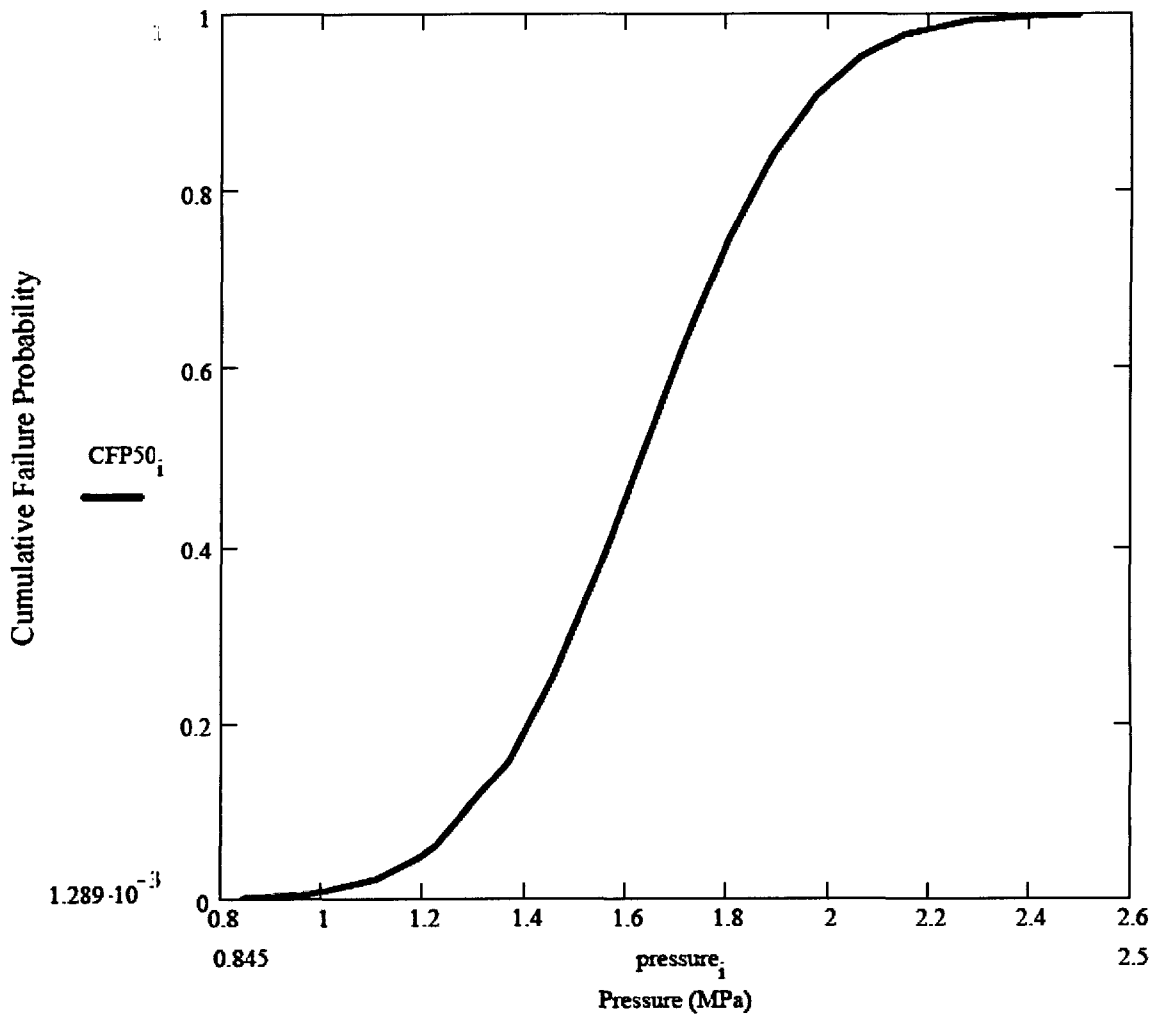
$Sq = (h - C)/h$

WHERE

h = INITIAL SEAL HEIGHT

C = COMPRESSED SEAL HEIGHT IN NORMAL OPERATIONS

Figure B.8-4. Definition of Squeeze for Seals



Cumulative F. P. Pmed=1.632MPa,COV=.16

Figure B.8-5. Cumulative Containment Failure Probability

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C.8 CONTAINMENT PENETRATION SCREENING ANALYSIS

The purpose of this appendix is to present the screening analysis of the containment penetrations identified in DCD Section 6.2.

Section 4.18 documents containment isolation with the perspective of analyzing pipe breaks outside of containment for the Level 1 analysis. In this appendix, a screening evaluation is performed to identify those containment penetrations that could potentially lead to offsite consequences. The screening analysis found that there were no penetrations that required isolation to prevent significant offsite consequences. Thus, the containment isolation function, as applied to the Level 2 analysis, was modeled considering only the isolation signal common to all penetration paths, as discussed in Appendix A.8. The probability of CIS failure is a point estimate that is independent of the Level 1 sequence.

To further clarify the robust containment isolation provided in ESBWR, a screening analysis of the containment penetrations identified in DCD Section 6.2 is performed as follows:

Each of the containment penetrations included in Section 6.2 of the DCD (identified here as Table C.8-1) has been reviewed to select those that potentially could induce:

- a loss of reactor coolant outside the containment,
- a release of radioactive materials outside the containment, in the event of an accident concurrent with a failure of the isolation function.

Next the review identifies those containment penetrations that could potentially lead to off-site consequences. Containment penetrations were eliminated from further analysis if they met one of the following Level 2 screening criteria:

A. Penetrations with normally closed isolation valves.

B. Penetrations connected to a closed system inside the containment.

C. Penetrations having isolation valves plus a closed system outside the containment and with normal operating pressure greater than the expected accident pressure.

D. Penetrations for which isolation failure during an accident event without core damage would not result in an increase of radioactive release compared with normal operation.

Instrument and monitoring lines were not included in this analysis. These RCCV mechanical penetrations are very small (~20 mm, or ¾ in diameter) lines whose failures have negligible contribution to offsite releases.

The containment penetrations screened out and the applicable screening criteria are indicated in Table C.8-1.

Table C.8-1

Containment Penetrations

Penetration ID	Description	Valves				Level 2 Screening Criterion[1]	Level 1 Model
	MAIN STEAM LINE A	F001A	F002A	F016A			Analyzed as BOC-MS
	MAIN STEAM LINE B	F001B	F002B	F016B			Analyzed as BOC-MS
	MAIN STEAM LINE C	F001C	F002C	F016C			Analyzed as BOC-MS
	MAIN STEAM LINE D	F001D	F002D	F016D			Analyzed as BOC-MS
MSL # 3	MAIN STEAM LINE DRAINS	F010	F011			A	
MFWL # 1	FEEDWATER SYSTEM (LINE A)	F102A	F103A				Analyzed as BOC-FDWA
MFWL # 2	FEEDWATER SYSTEM (LINE B)	F102B	F103B				Analyzed as BOC-FDWB
IC # 1A	ISOLATION CONDENSER SYSTEM (LOOP A) STEAM SUPPLY	F001A	F002A			C	Analyzed as BOC-IC
IC # 2A	ISOLATION CONDENSER SYSTEM (LOOP A) CONDENSATE RETURN	F003A	F004A			C	Analyzed as BOC-IC

Table C.8-1

Containment Penetrations

Penetration ID	Description	Valves				Level 2 Screening Criterion[1]	Level 1 Model
IC # 3A	ISOLATION CONDENSER SYSTEM (LOOP A) UPPER HEADER VENT	F007A	F008A			C	
IC # 4A	ISOLATION CONDENSER SYSTEM (LOOP A) LOWER HEADER VENT	F009A	F010A	F011A	F012A	C	
IC # 5A	ISOLATION CONDENSER SYSTEM (LOOP A) PURGE LINE	F014A				C	
IC # 1B	ISOLATION CONDENSER SYSTEM (LOOP B) STEAM SUPPLY	F001B	F002B			C	Analyzed as BOC-IC
IC # 2B	ISOLATION CONDENSER SYSTEM (LOOP B) CONDENSATE RETURN	F003B	F004B			C	Analyzed as BOC-IC
IC # 3B	ISOLATION CONDENSER SYSTEM (LOOP B) UPPER HEADER VENT	F007B	F008B			C	
IC # 4B	ISOLATION CONDENSER SYSTEM (LOOP B) LOWER HEADER VENT	F009B	F010B	F011B	F012B	C	
IC # 5B	ISOLATION CONDENSER SYSTEM (LOOP B) PURGE LINE	F014B				C	

Table C.8-1

Containment Penetrations

Penetration ID	Description	Valves				Level 2 Screening Criterion[1]	Level 1 Model
IC # 1C	ISOLATION CONDENSER SYSTEM (LOOP C) STEAM SUPPLY	F001C	F002C			C	Analyzed as BOC-IC
IC # 2C	ISOLATION CONDENSER SYSTEM (LOOP C) CONDENSATE RETURN	F003C	F004C			C	Analyzed as BOC-IC
IC # 3C	ISOLATION CONDENSER SYSTEM (LOOP C) UPPER HEADER VENT	F007C	F008C			C	
IC # 4C	ISOLATION CONDENSER SYSTEM (LOOP C) LOWER HEADER VENT	F009C	F010C	F011C	F012C	C	
IC # 5C	ISOLATION CONDENSER SYSTEM (LOOP C) PURGE LINE	F014C				C	
IC # 1D	ISOLATION CONDENSER SYSTEM (LOOP D) STEAM SUPPLY	F001D	F002D			C	Analyzed as BOC-IC
IC # 2D	ISOLATION CONDENSER SYSTEM (LOOP D) CONDENSATE RETURN	F003D	F004D			C	Analyzed as BOC-IC

Table C.8-1

Containment Penetrations

Penetration ID	Description	Valves				Level 2 Screening Criterion[1]	Level 1 Model
IC # 3D	ISOLATION CONDENSER SYSTEM (LOOP D) UPPER HEADER VENT	F007D	F008D			C	
IC # 4D	ISOLATION CONDENSER SYSTEM (LOOP D) LOWER HEADER VENT	F009D	F010D	F011D	F012D	C	
IC # 5D	ISOLATION CONDENSER SYSTEM (LOOP D) PURGE LINE	F014D				C	
RWCU # 1	REACTOR WATER CLEANUP/SDC SYSTEM	F002A	F003A			C	Analyzed as BOC-RWCU
RWCU # 2	REACTOR WATER CLEANUP/SDC SYSTEM	F002B	F003B			C	Analyzed as BOC-RWCU
RWCU # 3	REACTOR WATER CLEANUP/SDC SYSTEM	F007A	F008A			C	Analyzed as BOC-RWCU
RWCU # 4	REACTOR WATER CLEANUP/SDC SYSTEM	F007B	F008B			C	Analyzed as BOC-RWCU
RWCU# 5	REACTOR WATER CLEANUP/SDC SYSTEM	F038A	F039A			A	
RWCU # 6	REACTOR WATER CLEANUP/SDC SYSTEM	F038B	F039B			A	
SLC # 1	STANDBY LIQUID CONTROL SYSTEM	F004A	F005A			C	

Table C.8-1

Containment Penetrations

Penetration ID	Description	Valves				Level 2 Screening Criterion[1]	Level 1 Model
SLC # 2	STANDBY LIQUID CONTROL SYSTEM	F004B	F005B			C	
FAPC # 1	FUEL AND AUXILIARY POOLS COOLING SYSTEM	F309	F310			C	
FAPC # 2	FUEL AND AUXILIARY POOLS COOLING SYSTEM	F306	F307			C	
FAPC # 3	FUEL AND AUXILIARY POOLS COOLING SYSTEM	F323	F324			C	
FAPC # 4	FUEL AND AUXILIARY POOLS COOLING SYSTEM	F303	F304			C	
FAPC # 5	FUEL AND AUXILIARY POOLS COOLING SYSTEM	F321				C	
CAC # 1	CONTAINMENT INERTING SYSTEM	F009	F023	F025	F028	C	
CAC # 2	CONTAINMENT INERTING SYSTEM	F007	F008	F023	F024	C	
CAC # 3	CONTAINMENT INERTING SYSTEM	F010	F011	F014	F015	A	
CAC # 4	CONTAINMENT INERTING SYSTEM	F011	F012			A	
CWS # 1 (A+B)[2]	CHILLED COOLING WATER SYSTEM	F001 (A+B)	F002 (A+B)			B	

Table C.8-1

Containment Penetrations

Penetration ID	Description	Valves				Level 2 Screening Criterion(1)	Level 1 Model
CWS # 2 (A+B)[2]	CHILLED COOLING WATER SYSTEM	F003 (A+B)	F004 (A+B)			B	
HPNSS # 1[2]	HIGH PRESSURE NITROGEN SUPPLY SYSTEM	F010A	F011A			C	
HPNSS # 3[2]	HIGH PRESSURE NITROGEN SUPPLY SYSTEM	F003	F004			C	

Notes:

(1) Level 2 Screening Criteria

A. Penetrations with normally closed isolation valves.

B. Penetrations connected to a closed system inside the containment.

C. Penetrations having isolation valves plus a closed system outside the containment and with normal operating pressure greater than the expected accident pressure.

D. Penetrations for which isolation failure during an accident event without core damage would not result in an increase of radioactive release compared with normal operation.

(2) Assumed configuration. Detailed design specified as a COL Applicant responsibility in the DCD

9 SOURCE TERMS

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9 SOURCE TERMS

As discussed in Sections 8 and 21, the containment response to a severe accident is depicted by the end states of containment event trees. These end states become the "release categories" that are used to characterize potential source terms. The source terms will be used in the offsite consequence analysis presented in Section 10.

Table 9-1 summarizes the ESBWR release categories and associated frequencies. As indicated in the table, the release category "TSL", which depicts an intact containment with only leakage providing a source term, is the most likely release category. Other release categories have much lower calculated frequencies. For conservatism, a truncation frequency was used to represent some of these release categories. Specifically, if the calculated probability of the category was less than 10^{-12}, the truncation value of 10^{-12} was carried forward for the consequence evaluation.

The source term evaluation was performed with the MAAP computer code, which produces the distribution of radionuclides released to the environment as a function of time. Each release category is represented by a severe accident sequence that was selected and modeled to represent the group of potential severe accidents that could be associated with that release category. The selection is based on several factors, including the frequency of the various sequences that lead to the end state and the spectrum of response for the various sequences. The selected sequence provides a conservative basis for the source term quantification. The following sections describe the representative sequences and the bases for choosing them. As indicated in the following sections, conservative assumptions were typically made to account for analytical and phenomenological uncertainties. Table 9-1 includes the representative MAAP sequences as well as the time of initial release, and cumulative release fractions of noble gas and CsI at 24 and 72 hours after onset of core damage. Tables 9-2 and 9-3 provide the radionuclide release spectrum for 24 and 72 hours after onset of core damage, respectively.

9.1 BREAK OUTSIDE OF CONTAINMENT (BOC)

The release category "Break Outside-of-Containment" represents sequences in which the RPV communicates directly with the environment due to an unisolated piping break that connects the RPV directly to an area outside of containment. From the Level 1 PRA, two outside-containment break locations contributed to the core damage frequency: breaks in a feedwater line and breaks in a RWCU/SDC line. The RWCU/SDC break event tree includes both a mid-level connection to the RPV and a lower head drain line connection. Although the largest contribution to outside-containment break is associated with the feedwater line, selecting the RWCU/SDC pipe break is conservative because its lower elevation in the RPV results in a more rapid loss of coolant inventory. The mid-level location was selected to represent the BOC release category rather than the drain line because the smaller drain line break produced a lower release fraction.

Therefore, the representative sequence for this category is "BOCsd_nIN". This is an unisolated break outside of containment in the shutdown cooling piping followed by no injection into the RPV. In this scenario, the release begins at the onset of fuel damage and proceeds directly to the environment.

9.2 CONTAINMENT BYPASS (BYP)

The release category "Bypass" represents those sequences in which containment isolation has not occurred due to failure of the CIS function. Thus, there is a direct path from the containment atmosphere to the environment when the severe accident is initiated.

To determine the source term, a large diameter pipe opening was assumed from the time of accident initiation. Sequences in which the RPV is depressurized generally result in an earlier time to core uncovery than those involving failure to depressurize. As a result, the source term is generated earlier and the containment radionuclide concentration is developed earlier because of the path through the DPVs into containment. The low pressure sequences dominate the risk and such a sequence was selected to evaluate the BYP release category. Because of the reliability of the deluge system (i.e., the probability of BYP with failed deluge is below the truncation level), the representative sequence is modeled with deluge success and is termed "T_nIN_BYP".

9.3 CORE-CONCRETE INTERACTION DRY (CCID)

The release category "Core-Concrete Interaction-Dry" applies to sequences in which the containment fails due to core concrete interaction and the lower drywell debris bed is uncovered i.e., the deluge function is unsuccessful.

In these sequences, the core-concrete interaction is not limited by water cooling the debris bed, nor is the radionuclide release limited by the potential scrubbing action of an overlying water pool. Sequences in which the RPV is not depressurized may result in earlier RPV failure, thus initiating earlier CCI. However, sensitivity runs indicate that the potential difference in release fraction between high and low pressure sequences is small in comparison to their relative probabilities. Thus, a low pressure sequence was selected to represent the CCID source term category. The sequence is termed "T_nIN_nD_CCID" to indicate a transient with successful depressurization, but failure of the injection and deluge functions.

9.4 CORE-CONCRETE INTERACTION-WET (CCIW)

The release category "Core-Concrete Interaction-Wet" applies to sequences in which the containment fails due to core concrete interaction even though the lower drywell debris bed is covered with water. In such sequences, the deluge system has functioned to cover the debris bed with water, but the BiMAC is not successful in assuring debris bed cooling. The extent of water penetration into the debris bed, independent of the BiMAC, and thus, the potential for debris bed cooling, is subject to assumption. In the worst-case hypothetical condition, the debris bed is impermeable by the overlying water pool and the extent of CCI could approach that of a dry debris bed. To address this uncertainty associated with the debris bed coolability, the debris bed was modeled as being impermeable, thus maximizing the core-concrete interaction that could occur with an overlying water pool. Unlike the CCID release category, the overlying water pool is present, which provides the potential for scrubbing of the radionuclides evolved from the debris bed.

The representative sequence is termed "T_nIN_CCIW" and differs from the representative CCID sequence only in that the deluge system functions.

9.5 DIRECT CONTAINMENT HEATING (DCH)

The release category "Direct Containment Heating" applies to sequences in which the RPV fails at high pressure and a significant DCH event occurs. From Section 21.3, catastrophic containment failure due to DCH is physically unreasonable but local damage to the liner in the lower drywell cannot be excluded. A conservative approach was used to develop the source term associated with potential DCH damage, specifically assuming that

- Localized DCH effects in the lower drywell could damage wiring, instrumentation or other components such that the deluge system would not function and
- Damage to the lower drywell liner would not result in a direct release path to the environment due to the liner arrangement and backing with structural concrete but would provide a failure area much larger than that associated with normal leakage.

The dominant Class III sequence, a transient with no depressurization and no injection provided the basis for the DCH release category. To address the two points above, the sequence was modeled with deluge failure and a containment failure area that was a factor of 10 larger than that associated with Technical Specification leakage. The representative sequence is termed "T_nDP_nIN_nD_DCH".

9.6 EX-VESSEL STEAM EXPLOSION (EVE)

The release category "Ex-vessel Steam Explosion" applies to sequences in which the RPV fails at low pressure and a significant steam explosion occurs. As indicated in Section 21.4, containment leak tightness and failure of the BiMAC function is physically unreasonable for all but 1% of the sequences contributing to the core damage frequency. A conservative approach was used to develop the source term associated with an EVE, specifically:

- Liner damage was assumed to be significant enough to result in containment depressurization, which occurs at the time of RPV failure,
- No credit was taken for mitigation of the release; i.e., liner damage was assumed to result in direct communication with the environment, and
- Due to uncertainties about potential equipment damage and the distribution of water through containment after the EVE, no credit is taken for a lower drywell water pool that would minimize the source term.

The dominant Class I sequence, a transient with no injection and successful RPV depressurization, provided the basis for this category. To address the preceding points, the sequence was modeled with deluge failure and containment failure occurring at the time of RPV failure. The representative sequence is termed "T_nIN_nD_EVE".

9.6.1 EVE (CCIW*)

The release category applies to the end state depicted in containment event trees shown in Appendix A.8 in which containment failure due to an EVE occurs, but the associated source term would be mitigated by an overlying water pool, i.e. the path in which the initial lower drywell water level exceeded 1.5 meters. For conservatism, no credit is taken for the overlying water pool in the source term evaluation. Thus, the same source term applicable to end state EVE is used to represent this end state and its release category frequency is added to the EVE frequency.

9.7 FILTERED RELEASE (FR)

The ESBWR design includes the potential to manually vent the containment from the suppression chamber air space. This action may be implemented to limit the containment pressure increase if containment heat removal fails or core-concrete interaction generates enough non-condensables to overpressurize the containment. Venting the suppression chamber forces the radionuclides through the suppression pool, which reduces the magnitude of the source term.

To represent the FR category, a sequence with failure to insert negative reactivity was conservatively selected because such a sequence would pressurize containment more quickly than the much more probable non-ATWS sequences. The sequence assumes RPV failure at low pressure, consistent with sequences which dominate the core damage frequency. Operator guidance regarding venting has not been developed, but it is assumed that venting would be delayed until containment integrity is threatened. The analysis assumes that venting does not occur until the containment pressure reaches 90% of the containment ultimate strength. No credit was given in the analysis for closing the vent after reducing the containment pressure. The representative sequence is termed "T-AT_nIN _nCHR_FR".

9.8 OVERPRESSURE-VACUUM BREAKER (OPVB)

The release category "OPVB" applies to sequences in which vacuum breaker failure has occurred. Failure of vacuum breakers to close, or to be open in a pre-existing condition, results in failure of the containment pressure suppression function, which in turn also fails containment heat removal. Thus, such sequences would be expected to result in an earlier release than overpressure sequences with failure of containment heat removal alone.

Review of the containment event trees in Appendix A.8 indicates that vacuum breaker failure is more likely for sequences in which the RPV fails at high pressure. Thus, a Class III sequence was selected as representative of this category. The event trees also illustrate that the OPVB category is logically reached only if deluge/BiMAC function successfully. Thus, the sequence termed "T_nDP_nIN_VB" is used to represent the OPVB release category.

9.9 OVERPRESSURE- EARLY CONTAINMENT HEAT REMOVAL LOSS (OPW1)

The release category "OPW1" applies to sequences in which containment heat removal fails within 24 hours after event initiation. A sequence with RPV failure at high pressure was selected to represent this release category because RPV failure generally occurs earlier than if the vessel were depressurized and the loss of containment heat removal failure probability is higher than for low pressure sequences. Thus, the representative sequence becomes "T_nDP_nIN_nCHR_W1". Containment heat removal is conservatively assumed to be unavailable for the duration of the sequence.

9.10 OVERPRESSURE- LATE CONTAINMENT HEAT REMOVAL LOSS (OPW2)

The release category "OPW2" applies to sequences in which containment heat removal fails after the period covered by OPW1 and up to 72 hours after onset of core damage. In such sequences, the passive PCCS system becomes unavailable after 24 hours due to failure to connect to a supplemental water pool; FAPCS availability is also evaluated at this time. The representative sequence is the same as that used for OPW1 except that containment heat removal is terminated

at 24 hours after event initiation, consistent with the PCCS design basis. The representative sequence is termed "T_nDP_nIN_nCHR_W2".

9.11 TECHNICAL SPECIFICATION LEAKAGE (TSL)

The category "Technical Specification Leakage" applies to sequences in which the containment is intact and the only release is due to the maximum leak rate allowed by Technical Specifications. Sequence T_AT_DP was selected as representative of this category because the core damage time is relatively early for ATWS sequences. For additional conservatism, the area of containment leakage corresponding to the maximum allowable Technical Specification leak rate was doubled to produce the representative source term used for this release category. The representative source term is termed "T-AT_nIN_TSL2x".

9.12 SUMMARY

Potential release categories were defined in Sections 8 and 21. The source terms associated with each release category were developed using MAAP simulations of a representative sequence. Conservative assumptions were used in the selection and simulation of the representative sequence. Table 9-1 summarizes the release category, representative sequence and the cumulative release fractions for noble gases and CsI. Table 9-2 provides source terms for the period 24 hours after onset of core damage. Table 9-3 provides source terms for the period 72 hours after onset of core damage. The source terms and associated release category frequencies are used in the offsite consequence analysis described in Section 10.

Table 9-1

Release Categories

Release Category	Representative Sequence	Frequency (per reactor-year)	Time to Initial Release (hr)	Noble Gas Release Fraction @24 hrs after onset of core damage	CsI Release Fraction @24 hrs after onset of core damage	Noble Gas Release Fraction @72 hrs after onset of core damage	CsI Release Fraction @72 hrs after onset of core damage
BOC	BOCsd_nIN	4E-12	0.6	1.0E+00	8.5E-01	1.0E+00	8.5E-01
BYP	T_nIN_BYP	1E-12	0.5	9.7E-01	4.0E-01	9.8E-01	4.3E-01
CCID	T_nIN_nD_CCID	2.9E-11	14.8	9.1E-01	6.9E-02	9.2E-01	5.3E-01
CCIW	T_nIN_CCIW	2.9E-10	19.3	9.1E-01	8.0E-04	9.2E-01	1.1E-02
DCH	T_nDP_nIN_nD_DCH	<1E-12	4.5	9.0E-01	4.5E-01	9.0E-01	8.0E-01
EVE	T_nIN_nD_EVE	2.5E-10	6.3	8.3E-01	2.5E-02	8.4E-01	2.5E-01
FR	T-AT_nIN _nCHR_FR	2.3E-10	28.5	0.0E+00	0.0E+00	1.0E+00	9.8E-06
OPVB	T_nDP_nIN_VB	<1E-12	18.1	9.1E-01	3.6E-04	9.9E-01	2.8E-01
OPW1	T_nDP_nIN_nCHR_W1	<1E-12	25.4	1.9E-01	1.7E-04	9.9E-01	6.0E-01
OPW2	T_nDP_nIN_nCHR_W2	1.4E-11	40.8	0.0E+00	0.0E+00	9.9E-01	3.8E-02
TSL	T-AT_nIN_TSL2x	2.8E-8	0.3	2.0E-03	1.5E-04	2.0E-03	1.5E-04

Table 9-2

Radionuclide Source Terms (Release Fraction 24 hours after onset of core damage)

Release Category	Xe/Kr	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
BOC	1.0E+00	8.5E-01	7.5E-01	1.5E-02	7.2E-02	3.8E-01	1.5E-02	6.1E-04	3.4E-03	1.5E-01	6.1E-03	2.9E-05
BYP	9.7E-01	4.0E-01	1.9E-01	1.6E-02	1.2E-01	3.5E-01	2.9E-02	5.3E-04	3.3E-03	2.4E-01	4.3E-03	2.4E-05
CCID	9.1E-01	6.9E-02	6.8E-02	2.2E-06	4.7E-06	2.8E-02	1.3E-05	1.3E-07	1.0E-06	1.5E-01	1.4E-02	2.3E-07
CCIW	9.1E-01	8.0E-04	5.9E-05	1.4E-06	4.5E-06	7.6E-04	1.1E-06	1.3E-07	7.6E-07	2.6E-03	3.1E-05	7.9E-09
DCH	9.0E-01	4.5E-01	1.2E-01	3.2E-04	2.7E-04	6.2E-02	3.1E-04	3.1E-04	3.2E-04	8.1E-02	1.1E-04	9.6E-08
EVE	8.3E-01	2.5E-02	6.4E-02	1.3E-02	1.1E-04	7.1E-02	5.7E-03	8.2E-04	6.2E-03	2.1E-01	6.8E-03	5.0E-05
FR	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
OPVB	9.1E-01	3.6E-04	8.2E-04	1.4E-04	1.2E-04	4.9E-03	1.4E-04	1.4E-04	1.4E-04	1.7E-03	2.4E-05	0.0E+00
OPW1	1.9E-01	1.7E-04	2.4E-04	6.0E-07	2.0E-07	5.7E-04	5.8E-07	6.0E-07	6.0E-07	1.3E-02	2.6E-06	0.0E+00
OPW2	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
TSL	2.0E-03	1.5E-04	9.5E-05	2.1E-06	3.9E-05	5.3E-05	7.7E-06	7.3E-08	2.0E-07	9.9E-05	4.6E-08	1.0E-10

Table 9-3

Radionuclide Source Terms (Release Fraction 72 hours after onset of core damage)

Release Category	Xe/Kr	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb	Te_2	UO_2
BOC	1.0E+00	8.5E-01	7.5E-01	1.5E-02	7.2E-02	3.8E-01	1.5E-02	6.1E-04	3.4E-03	1.5E-01	6.1E-03	2.9E-05
BYP	9.8E-01	4.3E-01	2.3E-01	1.6E-02	1.2E-01	3.7E-01	2.9E-02	5.3E-04	3.3E-03	5.0E-01	1.2E-02	2.4E-05
CCID	9.2E-01	5.3E-01	2.3E-01	2.3E-06	5.1E-06	3.2E-01	2.2E-05	1.4E-07	1.1E-06	3.5E-01	1.5E-02	3.0E-07
CCIW	9.2E-01	1.1E-02	2.7E-03	1.4E-06	4.5E-06	2.5E-02	1.1E-06	1.3E-07	7.6E-07	6.8E-03	6.5E-05	7.9E-09
DCH	9.0E-01	8.0E-01	2.2E-01	3.2E-04	2.7E-04	1.5E-01	3.2E-04	3.2E-04	3.2E-04	2.7E-01	1.1E-04	1.6E-07
EVE	8.4E-01	2.5E-01	1.4E-01	1.3E-02	1.1E-04	3.4E-01	5.7E-03	8.2E-04	6.2E-03	5.4E-01	9.8E-03	5.0E-05
FR	1.0E+00	9.8E-06	8.1E-07	1.1E-08	1.9E-07	4.6E-05	3.2E-08	4.1E-10	1.5E-09	2.5E-03	2.9E-05	1.0E-11
OPVB	9.9E-01	2.8E-01	3.9E-02	1.7E-03	1.2E-04	3.4E-02	8.7E-04	1.5E-04	3.1E-04	7.1E-02	3.1E-05	5.2E-07
OPW1	9.9E-01	6.0E-01	2.1E-01	1.6E-03	4.3E-07	1.4E-01	7.5E-04	5.8E-06	1.8E-04	1.6E-01	3.2E-05	5.9E-07
OPW2	9.9E-01	3.8E-02	5.7E-02	1.2E-03	2.0E-07	4.3E-02	5.7E-04	3.9E-06	1.4E-04	1.0E-01	1.5E-05	3.4E-07
TSL	2.0E-03	1.5E-04	9.5E-05	2.1E-06	3.9E-05	5.5E-05	7.7E-06	7.3E-08	2.0E-07	1.2E-04	4.8E-08	1.0E-10

10 CONSEQUENCE ANALYSIS

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10 CONSEQUENCE ANALYSIS

10.1 INTRODUCTION

This section describes the offsite consequence evaluation. Key inputs and assumptions are described. The calculated results are compared to consequence related goals to determine if the goals are satisfied.

The MACCS2 Version 1.12 computer code (Reference 10-1) is used to determine the consequences of potential reactor accidents. The MACCS2 code evaluates offsite dose and consequences such as early fatality risk and latent cancer fatality risk for each source term (i.e., radionuclide release category) over a range of possible weather conditions and evacuation assumptions. The MACCS2 code model is described in Reference 10-1. The rationale for site related input selection is presented in Subsection 10.2. Other more generic input parameters for the MACCS2 analysis are based on "Sample Problem A" of Reference 10-1. ESBWR specific reference data from the plant performance analysis in Section 8.0 and Section 9.0 are used as MACCS2 inputs as presented in Subsection 10.3. The calculated consequence results are compared to the goals in Subsection 10.4.

10.2 SITE ASSUMPTIONS

The evaluation of the offsite consequences of a reactor accident is closely tied to the site parameters (e.g., weather, population, land use). For probabilistic offsite consequence evaluations, site related assumptions are required.

The subsections below describe the rationale for the site meteorology, population, and evacuation. The following tables describe these inputs:

Table	Inputs
10-1	Population Density
10-2	Shielding and Exposure Parameters

10.2.1 Meteorology

In the original WASH-1400 analysis (Reference 10-3), a number of actual site meteorological data were used. However, the original WASH-1400 meteorology data files are not compatible with the MACCS2 code. For this study, the ALWR URD (Reference 10-6) meteorological reference data set is used, which is indicative of meteorological data significantly worse than the average U.S. site. Therefore, the results in this study are not indicative of an average U.S. site as was the original purpose of Reference 10-3, but represents a generally bounding evaluation for most U.S. sites.

10.2.2 Population

For the ESBWR consequence evaluation, the SANDIA Siting Study population density data (Table 3-2 of Reference 10-4) is used to develop a uniform population density corresponding to each spatial interval. The population distribution is developed for distances to 0.5, 1, 2, 3, 4, 5, 10, 20, 30, 40 and 50 miles from the site. The three offsite consequence goals defined for the ESBWR are concerned with consequences within 10 miles of the site; therefore, a bounding 0-10 mile population density is used. The maximum 0-10 mile population distribution value from the "all" sites column of Table 3-2 of Reference 10-4 is used for the ESBWR consequence evaluation and is provided in Table 10-1. As can be seen from Table 10-1, the 0-5 mile population density is larger than the 5-10 mile population density and is used in this bounding analysis as a constant uniform density for the entire 0-50 mile region. This approach provides a more bounding 0-10 mile population density than that provided in the ALWR URD (Reference 10-6).

10.2.3 Evacuation

Many evacuation related characteristics (local roads, population demographics, emergency services) are quite site specific. No guidance is provided by the NRC for generic evacuation evaluations. The evacuation parameters used in this study are conservative assumptions in that no evacuation or relocation in terms of physical movement are assumed and no sheltering is assumed. The public is assumed to continue normal activity during the reactor accident in this bounding analysis. Shielding and

exposure values used for normal activity are the standard MACCS2 assumptions and are provided in Table 10-2.

Table 10-2 provides the following information for people engaged in normal activity:

- Cloudshine Shielding Factor – Fraction of cloudshine dose received from direct external exposure to the plume
- Inhalation Protection Factor – Fraction of inhalation dose received from cloud inhalation
- Breathing Rate – Breathing rate for people in normal activity
- Skin Protection Factor – Fraction of skin dose received from material deposited on skin
- Groundshine Shielding Factor – Fraction of groundshine dose received from material deposited on the ground

10.3 MACCS2 RADIONUCLIDE RELEASE INPUT DATA

10.3.1 MACCS2 Radionuclide Release Input Data

ESBWR specific radionuclide release data is used in this analysis to model the dispersion of a plume of material released to the environment during a reactor accident.

The following tables describe these inputs:

Table	Inputs
10-3	Building Data for Meteorological Modeling of Wake Effects
10-4	Core Inventory Parameters
10-5A	Reactor Accident Release Parameters 24 Hours After the Onset of Core Damage
10-5B	Reactor Accident Release Parameters 72 Hours After the Onset of Core Damage
10-6	Nuclide Release Categories

10.3.2 ESBWR Release Parameters

ESBWR specific parameters are used for wake effect data, core inventory, and reactor thermal power. The width and height of the building wake are used by MACCS2 to model the initial plume dimensions. These parameters for the ESBWR are provided in Table 10-3.

The core inventory and reactor thermal power used in this analysis are ESBWR specific and are provided in Table 10-4. These parameters are used to determine the inventory of each nuclide in the core at accident initiation.

10.3.3 Input to MACCS2 from MAAP

The severe accident sequence analysis (i.e., MAAP) results provide input parameters to the MACCS2 code and are described here and are shown in Table 10-5A and Table 10-5B. Table 10-5A provides the release parameters 24 hours after the onset of core damage, and Table 10-5B provides the release parameters 72 hours after the onset of core damage. The severe accident sequence analysis performed using the MAAP code is further described in Section 8.0. The representative MAAP cases used as MACCS2 inputs are summarized in Section 9.0. Important input release characteristics include the nuclide release time, duration, and fraction. The MAAP cases are used to develop source terms for each release category for the consequence analysis. Tables 10-5A and 10-5B describe the eleven (11) source terms and corresponding radionuclide release categories used for the MACCS2 analysis.

For each source term which represents a release category from Section 8.0, the following data are used (Table 10-5A and Table 10-5B):

- Source Term – Source term developed from the severe accident analysis that characterizes the release category. The source terms are summarized in Section 9.0.
- Release Category – Release category represented by the source term
- MAAP Case – Severe accident sequence analysis results which are used to develop each source term. Section 8.0 provides a summary of the MAAP cases.
- Release Frequency – The frequency per year associated with the radionuclide release category.
- Time of Plume Release – Time (hr) from reactor shutdown (time of accident initiation) until the time of the modeled plume release to the atmosphere. This parameter is based on the severe accident analysis discussed in Section 8.0 results and is approximately the time when the CsI release from containment begins.
- Duration of Release - Duration of release (hr) of radionuclides from the plant is used to determine the dispersion of the release cloud. Each MAAP case for the ESBWR was performed for 72 hours after the onset of core damage. MACCS2 limits the duration of an individual plume to a maximum of 10 hours. Source terms in which the release flattens out after a short time (i.e., less than 10 hours) are characterized by a release duration corresponding to the time the release starts to the time the release flattens out. Each release fraction is reviewed in determining the release duration, with special attention given to the nuclides with the greatest offsite consequence impacts (i.e., iodine and cesium).

- NG – Release fraction of Noble gases from containment to the environment.
- CsI – Release fraction of Iodine from containment to the environment

For this bounding assessment no warning time is assumed. This is the time between official notification of public and release of radioactivity from the plant.

For each source term, the release is modeled to occur at ground level. The thermal content of the plume is assumed to be the same as ambient (i.e., buoyant plume rise is not modeled). These assumptions are conservative for early fatalities based on Reference 10-4.

MAAP provides results for twelve (12) nuclide release fractions from containment to the atmosphere. These nuclide release fractions are related to the MACCS2 release groups as shown in Table 10-6.

The fission product releases for each source term as a fraction of total core inventory are provided in Section 9. MAAP cases that represent each source term are summarized in Section 8.

10.4 COMPARISON OF RESULTS TO GOALS

10.4.1 Goals

Three major offsite consequence-related goals are established in the GE ESBWR Licensing Review Bases based on the NRC Safety Goal Policy Statement. These goals are:

(1) Individual Risk Goal

The risk to an average individual in the "vicinity" of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one tenth of one percent (0.1%) of the sum of "prompt fatality risks" resulting from other accidents to which members of the U.S. Population are generally exposed. As noted in the Safety Goal Policy statement, "vicinity" is defined as the area within 1.61 km (1 mile) of the plant site boundary. "Prompt Fatality Risks" are defined as those risks to which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities. Such risks are the sum of risks that result in fatalities from such activities as driving, household chores, occupational activities, etc. For this evaluation, the sum of prompt fatality risks is taken as the U.S. accidental death risk value of 39.1 deaths per 100,000 people per year based upon Reference 10-7.

(2) Societal Risk Goal

The risk to the population in the area "near" a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one tenth of one percent (0.1%) of the sum of the "cancer fatality risks" resulting from all other causes. As noted in the Safety Goal Policy Statement, "near" is defined as within 16.1 km (10 miles) of the plant. The "cancer fatality risk" is taken as 169 deaths per 100,000 people per year based upon 1983 statistics in Reference 10-8.

(3) Radiation Dose Goal

The probability of exceeding a whole body dose of 0.25 sv at a distance of 805 m (one half mile) from the reactor shall be less than one in a million per reactor year.

The calculated ESBWR consequence results are compared to these goals in the following subsection.

10.4.2 Results

The mean results from the offsite consequence analysis for each source term are shown in Table 10-7A and Table 10-7B. Table 10-7A provides the results 24 hours after the onset of core damage, and Table 10-7B provides the results 72 hours after the onset of core damage. These results are multiplied by the annual release frequency for each source term and then summed to obtain the risk weighted mean consequence results. These results are compared to the consequence goals identified in Section 10.4.1 and

summarized in Table 10-8. A plot of whole body dose at a distance of 805 m (one half mile) against cumulative probability is shown in Figure 10-1. As can be seen, the whole body dose at 805m (0.5 miles) over the entire dose spectrum from 0.1 Sv to >100 Sv is well below the goal of 1E-6/yr exceedance frequency.

Based upon these results, the ESBWR meets the established consequence related goals with substantial margin.

10.5 REFERENCES

10-1 Chanin, D. and Young, M., Code Manual for MACCS2: User's Guide, NUREG/CR-6613, Vol. 1 (SAND97-0594), May1998.

10-2 Spring, J.L. et al, Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, MACCS Input NUREG/CR-4551, December 1990.

10-3 Reactor Safety Study, Appendix 6: Calculation of Reactor Accident Consequences, WASH-1400 (NUREG 75/014), October 1975.

10-4 Aldrich, D.C., et al, Technical Guidance for Siting Criteria Development NUREG/CR-2239, December 1982.

10-5 Criteria for preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654.

10-6 Murley, T.E., Advanced Boiling Water Reactor Licensing Review Bases, Project No. 671, August 7, 1987.

10-7 Accident Facts, 1988, National Safety Council.

10-8 1986 Cancer Facts and Figures, American Cancer Society, 90 Park Ave, New York, NY 10016.

Table 10-1

Population Distribution

Radial Interval	Maximum Population	
	All Sites (people per sq. mi.)	All Sites (people per sq. km.)
0-5 mi (0-8.1 km)	790	305
5-10 mi (8.1-16.1 km)	700	270
10-20 mi (16.1-32.2 km)	730	282
20-30 mi (32.2-48.3 km)	2000	772
30-50 mi (48.3-80.5 km)	2500	965

Data taken from Reference 10-4, Table 3-2.

The 0-5 mile population density (790 people per square mile) is used in the ESBWR bounding analysis as a uniform density for all radial intervals in the 0-50 mile region.

Table 10-2

Shielding and Exposure Data

MACCS2 Parameter	Normal Activity Value
Cloudshine Shielding Factor	0.75
Inhalation Protection Factor	0.41
Breathing Rate (m^3/sec)	2.66E-04
Skin Protection Factor	0.41
Groundshine Shielding Factor	0.33

See Subsection 10.2.3 for additional description of parameters in this table.

All values are based on Reference 10-1

Table 10-3

Site and Reactor Data for Meteorological Modeling

Parameter	Measurement (m)	Measurement (ft)
Reactor Building Length	49.0 m	160 ft.
Reactor Building Width	49.0 m	160 ft.
Reactor Building Height	47.7 m	156 ft.
Fuel Building Length	49.0 m	160 ft.
Fuel Building Height	17.8 m	58 ft.

Table 10-4

ESBWR Core Inventory

ESBWR Core Power is 4500 MWt			
Nuclide	**Bq/MWt**	**Nuclide**	**Bq/MWt**
Co-58	5.10E+12	Te-131m	1.42E+14
Co-60	4.92E+12	Te-132	1.41E+15
Kr-85	1.23E+13	I-131	9.90E+14
Kr-85m	2.73E+14	I-132	1.44E+15
Kr-87	5.27E+14	I-133	2.04E+15
Kr-88	7.42E+14	I-134	2.25E+15
Rb-86	2.35E+12	I-135	1.91E+15
Sr-89	9.93E+14	Xe-133	2.03E+15
Sr-90	9.76E+13	Xe-135	6.72E+14
Sr-91	1.25E+15	Cs-134	1.98E+14
Sr-92	1.34E+15	Cs-136	6.89E+13
Y-90	1.01E+14	Cs-137	1.28E+14
Y-91	1.27E+15	Ba-139	1.84E+15
Y-92	1.34E+15	Ba-140	1.77E+15
Y-93	1.55E+15	La-140	1.82E+15
Zr-95	1.70E+15	La-141	1.68E+15
Zr-97	1.69E+15	La-142	1.62E+15
Nb-95	1.71E+15	Ce-141	1.68E+15
Mo-99	1.89E+15	Ce-143	1.56E+15
Tc-99m	1.68E+15	Ce-144	1.36E+15
Ru-103	1.50E+15	Pr-143	1.53E+15
Ru-105	1.00E+15	Nd-147	6.69E+14
Ru-106	5.21E+14	Np-239	1.93E+16
Rh-105	9.10E+14	Pu-238	3.34E+12
Sb-127	1.03E+14	Pu-239	4.02E+11
Sb-129	3.15E+14	Pu-240	5.21E+11
Te-127	1.05E+14	Pu-241	1.51E+14
Te-127m	1.37E+13	Am-241	1.70E+11
Te-129	3.10E+14	Cm-242	4.01E+13
Te-129m	4.60E+13	Cm-244	1.94E+12

Table 10-5A

Event Release Parameter

24 Hours After the Onset of Core Damage[(1), (2)]

Source Term	Release Category	MAAP CASE	Release Frequency (per year)	Time of Plume Release	Duration of Release[(3)]	NG[(4)] Release Fraction	CsI[(4)] Release Fraction
1	BOC	BOCsd_nIN	4E-12	0.6 hr	2.5 hr	1.0E+0	8.5E-1
2	BYP	T_nIN_BYP	1E-12	0.5 hr	2.2 hr	9.7E-1	4.0E-1
3	CCID	T_nIN_nD_CCID	2.9E-11	14.8 hr	10 hr	9.1E-1	6.9E-2
4	CCIW	T_nIN_CCIW	2.9E-10	19.3 hr	5.8 hr	9.1E-1	8.0E-4
5	DCH	T_nDP_nIN_nD_DCH	<1E-12	4.5 hr	10 hr	9.0E-1	4.5E-1
6	EVE	T_nIN_nD_EVE	2.5E-10	6.3 hr	10 hr	8.3E-1	2.5E-2
7	FR	T-AT_nIN_nCHR_FR	2.3E-10	NA	NA	0.0E+0	0.0E+0
8	OPVB	T_nDP_nIN_VB	<1E-12	18.1 hr	7.1 hr	9.1E-1	3.6E-4
9	OPW1	T_nDP_nIN_nCHR_W1	<1E-12	25.4 hr	0.1 hr	1.9E-1	1.7E-4
10	OPW2	T_nDP_nIN_nCHR_W2	1.4E-11	NA	NA	0.0E+0	0.0E+0
11	TSL	T-AT_nIN_TSL2x	2.8E-8	0.3 hr	10 hr	2.0E-3	1.5E-4

Note:

(1) See Subsection 10.3.3 for definition of parameters in this table.

(2) For this bounding analysis, release height is ground level and release sensible heat is same as ambient.

(3) The release parameters are based on the 24 hours after the onset of core damage value. Each MAAP case for the ESBWR was performed for 72 hours after the onset of core damage. MACCS2 limits the duration of an individual plume to a maximum of 10 hours. Source terms in which the release flattens out after a short time (i.e., less than 10 hours) are characterized by a release duration corresponding to the time the release starts to the time the release flattens out. In general, the nuclides with the greatest offsite consequences (i.e., Iodine and Cesium) are conservatively used.

(4) Noble Gases (NG) and Cesium Iodine (CsI) release fractions are the cumulative release fractions at 24 hours after the onset of core damage.

Table 10-5B
Event Release Parameter
72 Hours After the Onset of Core Damage[(1), (2)]

Source Term	Release Category	MAAP CASE	Release Frequency (per year)	Time of Plume Release	Duration of Release[(3)]	NG[(4)] Release Fraction	CsI[(4)] Release Fraction
1	BOC	BOCsd_nIN	4E-12	0.6 hr	2.5 hr	1.0E+0	8.5E-1
2	BYP	T_nIN_BYP	1E-12	0.5 hr	2.2 hr	9.8E-1	4.3E-1
3	CCID	T_nIN_nD_CCID	2.9E-11	14.8 hr	10 hr	9.2E-1	5.3E-1
4	CCIW	T_nIN_CCIW	2.9E-10	19.3 hr	10 hr	9.2E-1	1.1E-2
5	DCH	T_nDP_nIN_nD_DCH	<1E-12	4.5 hr	10 hr	9.0E-1	8.0E-1
6	EVE	T_nIN_nD_EVE	2.5E-10	6.3 hr	10 hr	8.4E-1	2.5E-1
7	FR	T-AT_nIN_nCHR_FR	2.3E-10	28.5 hr	10 hr	1.0E+0	9.8E-6
8	OPVB	T_nDP_nIN_VB	<1E-12	18.1 hr	10 hr	9.9E-1	2.8E-1
9	OPW1	T_nDP_nIN_nCHR_W1	<1E-12	25.4 hr	10 hr	9.9E-1	6.0E-1
10	OPW2	T_nDP_nIN_nCHR_W2	1.4E-11	40.8 hr	10 hr	9.9E-1	3.8E-2
11	TSL	T-AT_nIN_TSL2x	2.8E-8	0.3 hr	10 hr	2.0E-3	1.5E-4

Note:

(1) See Subsection 10.3.3 for definition of parameters in this table.

(2) For this bounding analysis, release height is ground level and release sensible heat is same as ambient.

(3) Each MAAP case for the ESBWR was performed for 72 hours after the onset of core damage. MACCS2 limits the duration of an individual plume to a maximum of 10 hours. Source terms in which the release flattens out after a short time (i.e., less than 10 hours) are characterized by a release duration corresponding to the time the release starts to the time the release flattens out. In general, the nuclides with the greatest offsite consequences (i.e., Iodine and Cesium) are conservatively used.

(4) Noble Gases (NG) and Cesium Iodine (CsI) release fractions are the cumulative release fractions

at 72 hours after the onset of core damage.

Table 10-6

MACCS2 Release Groups vs. ESBWR Release Groups

MACCS2 Release Groups	MAAP Release Groups	MAAP Output Parameter
1-Xe/Kr	noble gases	FREL (1)
2-I	CsI	FREL (2)
3-Cs	CsOH	FREL (6)
4-Te	TeO2[1] (Sb[1] & Te2[2] fractions are included)	FREL (3), FREL (10) and FREL (11)
5-Sr	SrO	FREL (4)
6-Ru	MoO2 (Mo is in Ru MACCS category)	FREL (5)
7-La	La2O3	FREL (8)
8-Ce	CeO2 (included UO2[2] in this category)	FREL (9) and FREL (12)
9-Ba	BaO	FREL (7)

(1) The larger release fraction of TeO2 and Sb is used as input into MACCS2.

(2) Te2 and UO2 release fractions are negligible.

Table 10-7A

MACCS2 ESBWR Consequence Results by Source Term

24 Hours After the Onset of Core Damage

Source Term	Release Category	Release Frequency (per yr)	Individual Risk (0-1 mile) (1)	Weighted Individual Risk (per year) (2)	Weighted Individual Risk Contribution (%) (3)	Societal Risk (0-10 miles) (4)	Weighted Societal Risk (per year) (5)	Weighted Societal Risk Contribution (%) (6)	Prob.of Dose > .2 Sv (0-0.5 mile) (7)	Weighted Prob of Exceedance (per year) (8)	Weighted Dose Contribution (%) (9)
1	BOC	4.0E-12	8.84E-02	3.54E-13	1.36	1.09E-02	4.36E-14	0.91	1.00E+00	4.00E-12	0.18
2	BYP	1.0E-12	7.62E-02	7.62E-14	0.29	1.50E-02	1.50E-14	0.31	1.00E+00	1.00E-12	0.05
3	CCID	2.9E-11	8.32E-02	2.41E-12	9.31	5.19E-03	1.51E-13	3.14	1.00E+00	2.90E-11	1.34
4	CCIW	2.9E-10	9.33E-04	2.71E-13	1.04	3.42E-04	9.92E-14	2.07	8.30E-01	2.41E-10	11.13
5	DCH	1.0E-12	9.38E-02	9.38E-14	0.36	8.40E-03	8.40E-15	0.18	1.00E+00	1.00E-12	0.05
6	EVE	2.5E-10	9.08E-02	2.27E-11	87.56	1.08E-02	2.70E-12	56.26	1.00E+00	2.50E-10	11.56
7	FR	2.3E-10	0.00E+00	0.00E+00	0.00	0.00E+00	0.00E+00	0.00	0.00E+00	0.00E+00	0.00
8	OPVB	1.0E-12	2.27E-03	2.27E-15	0.01	9.30E-04	9.30E-16	0.02	9.77E-01	9.77E-13	0.05
9	OPW1	1.0E-12	1.44E-02	1.44E-14	0.06	5.85E-04	5.85E-16	0.01	7.76E-01	7.76E-13	0.04
10	OPW2	1.4E-11	0.00E+00	0.00E+00	0.00	0.00E+00	0.00E+00	0.00	0.00E+00	0.00E+00	0.00
11	TSL	2.8E-08	0.00E+00	0.00E+00	0.00	6.36E-05	1.78E-12	37.11	5.84E-02	1.64E-09	75.61
Total	--	2.9E-08	--	2.59E-11	100.00	--	4.80E-12	100.00	--	2.16E-09	100.00

Notes to Table 10-7A

1. The individual risk is calculated as the total number of early fatalities within one mile divided by the total one mile population
2. The weighted individual risk is the individual risk per year and is calculated as the product of the release category release frequency and the release category individual risk
3. The weighted individual risk contribution is the percentage of a release category's weighted individual risk to the total weighted individual risk
4. The societal risk is calculated as the total number of latent fatalities within ten miles divided by the total ten mile population
5. The weighted societal risk is the societal risk per year and is calculated as the product of the release category release frequency and the release category societal risk
6. The weighted societal risk contribution is the percentage of a release category's weighted societal risk to the total weighted societal risk
7. The probability of dose greater than 0.2 Sv is obtained from the MACCS2 output file and is provided in the form of CCDF tables
8. The weighted probability of exceedance is the probability of exceeding a dose greater than 0.2 Sv per year and is calculated as the product of the release category release frequency and the release category MACCS2 probability of dose greater than 0.2 Sv
9. The weighted probability of exceedance contribution is the percentage of a release category's weighted societal risk to the total weighted societal risk

Table 10-7B

MACCS2 ESBWR Consequence Results by Source Term

72 Hours After the Onset of Core Damage

Source Term	Release Category	Release Frequency (per yr)	Individual Risk (0-1 mile) (1)	Weighted Individual Risk (per year) (2)	Weighted Individual Risk Contribution (%) (3)	Societal Risk (0-10 miles) (4)	Weighted Societal Risk (per year) (5)	Weighted Societal Risk Contribution (%) (6)	Prob.of Dose > .2 Sv (0-0.5 mile) (7)	Weighted Prob of Exceedance (per year) (8)	Weighted Dose Contribution (%) (9)
1	BOC	4.0E-12	8.84E-02	3.54E-13	0.96	1.09E-02	4.36E-14	0.73	1.00E+00	4.00E-12	0.13
2	BYP	1.0E-12	8.08E-02	8.08E-14	0.22	1.30E-02	1.30E-14	0.22	1.00E+00	1.00E-12	0.03
3	CCID	2.9E-11	1.00E-01	2.90E-12	7.85	1.05E-02	3.05E-13	5.08	1.00E+00	2.90E-11	0.93
4	CCIW	2.9E-10	2.17E-02	6.29E-12	17.04	1.40E-03	4.06E-13	6.78	1.00E+00	2.90E-10	9.33
5	DCH	1.0E-12	1.02E-01	1.02E-13	0.28	1.12E-02	1.12E-14	0.19	1.00E+00	1.00E-12	0.03
6	EVE	2.5E-10	1.04E-01	2.60E-11	70.41	1.31E-02	3.28E-12	54.65	1.00E+00	2.50E-10	8.04
7	FR	2.3E-10	1.43E-04	3.29E-14	0.09	1.58E-04	3.63E-14	0.61	6.83E-02	1.57E-11	0.51
8	OPVB	1.0E-12	8.00E-02	8.00E-14	0.22	5.82E-03	5.82E-15	0.10	1.00E+00	1.00E-12	0.03
9	OPW1	1.0E-12	9.49E-02	9.49E-14	0.26	8.93E-03	8.93E-15	0.15	1.00E+00	1.00E-12	0.03
10	OPW2	1.4E-11	7.05E-02	9.87E-13	2.67	3.86E-03	5.40E-14	0.90	1.00E+00	1.40E-11	0.45
11	TSL	2.8E-08	0.00E+00	0.00E+00	0.00	6.55E-05	1.83E-12	30.61	8.94E-02	2.50E-09	80.49
Total	--	2.9E-08	--	3.69E-11	100.00	--	5.99E-12	100.00	--	3.11E-09	100.00

Notes to Table 10-7B

1. The individual risk is calculated as the total number of early fatalities within one mile divided by the total one mile population
2. The weighted individual risk is the individual risk per year and is calculated as the product of the release category release frequency and the release category individual risk
3. The weighted individual risk contribution is the percentage of a release category's weighted individual risk to the total weighted individual risk
4. The societal risk is calculated as the total number of latent fatalities within ten miles divided by the total ten mile population
5. The weighted societal risk is the societal risk per year and is calculated as the product of the release category release frequency and the release category societal risk
6. The weighted societal risk contribution is the percentage of a release category's weighted societal risk to the total weighted societal risk
7. The probability of dose greater than 0.2 Sv is obtained from the MACCS2 output file and is provided in the form of CCDF tables
8. The weighted probability of exceedance is the probability of exceeding a dose greater than 0.2 Sv per year and is calculated as the product of the release category release frequency and the release category MACCS2 probability of dose greater than 0.2 Sv
9. The weighted probability of exceedance contribution is the percentage of a release category's weighted societal risk to the total weighted societal risk

Table 10-8

Consequence Goals and Results

Goal	Numerical Goal	ESBWR 24 Hours After the Onset of Core Damage	Safety Goal Achieved 24 Hours After the Onset of Core Damage	ESBWR 72 Hours After the Onset of Core Damage	Safety Goal Achieved 72 Hours After the Onset of Core Damage
Individual Risk (0 – 1 Mile)	$<3.9 \times 10^{-7}$ (0.1%)	2.6E-11	YES	3.7E-11	YES
Societal Risk (0 – 10 Mile)	$<1.7 \times 10^{-6}$ (0.1%)	4.8E-12	YES	6.0E-12	YES
Radiation Dose Probability at 0.25 Sv (0 – 0.5 Mile)	$<10^{-6}$	<2.2E-9	YES	<3.1E-9	YES

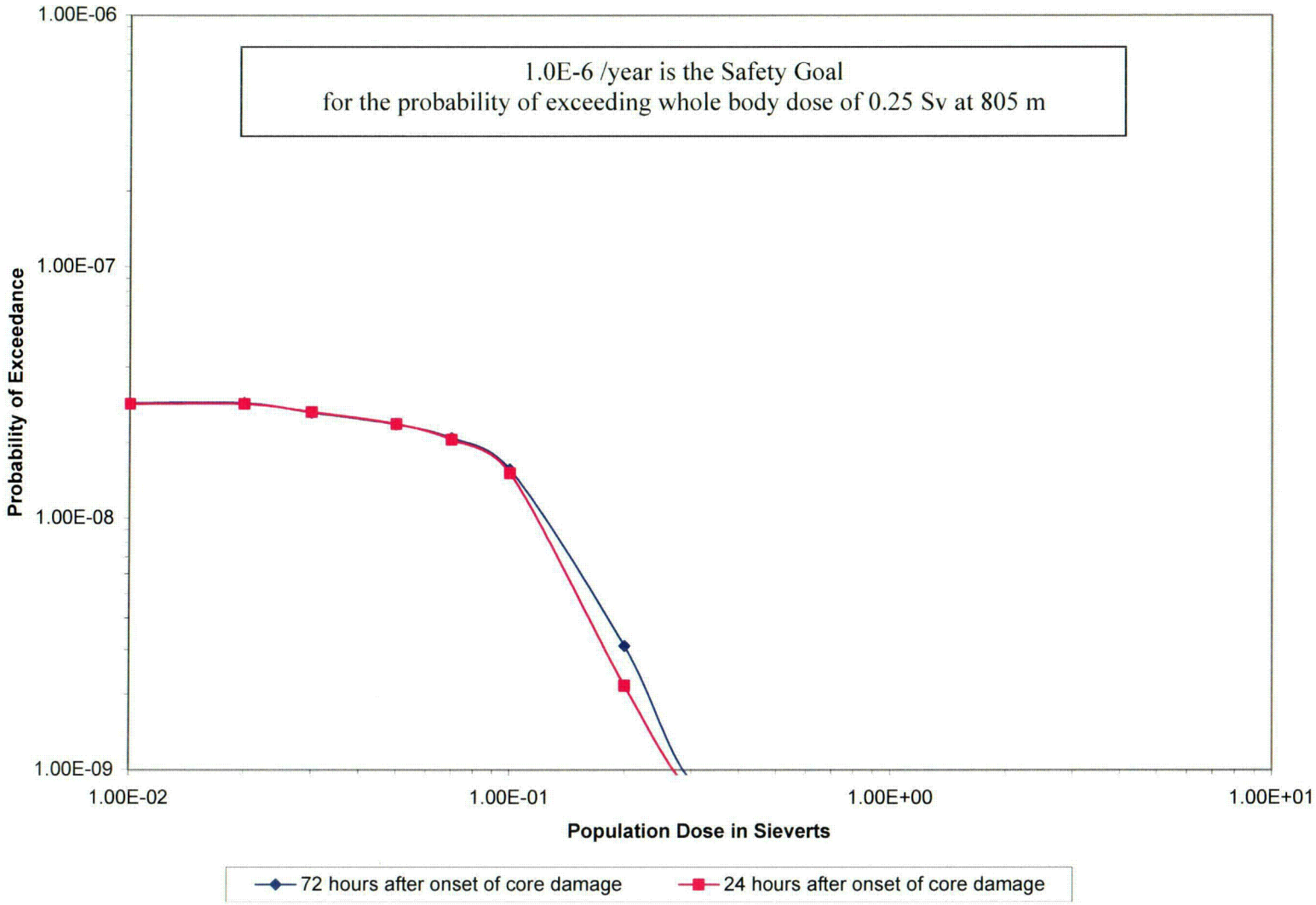

Figure 10-1. Whole Body Dose at 805 m (0.5 Mile) as Probability of Exceedance

*The goal of a maximum probability of 1E-6 is well above the entire dose range at 0.5 mile.

C06