Contents

8 CONTAINMENT PERFORMANCE 8.0-1
8.0 Introduction 8.0-1
8.1 Potential for Failure due to combustible gas deflagration 8.1-1
8.1.1 Background 8.1-1
8.1.2 Analysis Assumptions 8.1-2
8.1.3 Analysis Results 8.1-2
8.1.4 Risk Due to De-Inerted Operation 8.1-2
8.2 Frequency of Overpressure and Bypass release categories 8.2-1
8.2.1 Containment Event Trees 8.2-2
8.2.1.1 CET Entry Events 8.2-4
8.2.1.2 Mitigating Systems 8.2-5
8.2.1.3 Top Events 8.2-6
8.2.1.3.1 Repair of Failed Systems 8.2-7
8.2.1.3.2 Key Operator Actions 8.2-7
8.2.1.3.3 Top Event EVE_DAM 8.2-8
8.2.1.3.4 Top Event DCH_DAM 8.2-8
8.2.1.3.5 Top Event BI_SP 8.2-8
8.2.1.3.6 Top Event BI_FN 8.2-8
8.2.1.3.7 Top Event CIS 8.2-9
8.2.1.3.8 Top Event VB 8.2-9
8.2.1.3.9 Top Event W 8.2-9
8.2.1.3.10 Top Event VT 8.2-10
8.2.1.4 Release Categories 8.2-10
8.2.1.5 Release Category Frequency and Containment Effectiveness 8.2-13
8.3 Containment Performance Against Overpressure 8.3-1
8.3.1 Simulation Code 8.3-1
8.3.2 Sequences Representative of Each Accident Class 8.3-2
8.3.2.1 Class I: Sequences with RPV Failure at Low Pressure 8.3-2
8.3.2.1.1 Sequence T_nIN_TSL 8.3-2
8.3.2.1.2 Sequence T_nIN_nCHR_FR 8.3-3
8.3.2.2 Class II: Sequences with Containment Failure Preceding Core Damage 8.3-4
8.3.2.2.1 Sequence MLi_nCHR 8.3-4
8.3.2.3 Class III: Sequences with RPV Failure at High Pressure 8.3-5
8.3.2.3.1 Sequence T_nDP_nIN_TSL 8.3-5
8.3.2.3.2 Sequence T_nDP_nIN_nCHR_FR 8.3-6
8.3.2.4 Class IV: Sequences with Failure to Insert Negative Reactivity 8.3-6
8.3.2.4.1 Sequence T-AT_nIN_TSL 8.3-6
8.3.2.4.2 Sequence T-AT_nIN_nCHR_FR 8.3-7
8.3.2.5 Class V: Sequences with Interfacing LOCA 8.3-7
8.4 Summary 8.4-1
8.5 References 8.5-1
APPENDIX 8A. QUANTIFICATION OF CONTAINMENT EVENT TREES 8A-1
APPENDIX 8B. REPRESENTATIVE SEQUENCE RESULTS 8B-1

APPENDIX 8C. CONTAINMENT PENETRATION SCREENING ANALYSIS 8C-1

List of Tables

Table 8.0-1 Acronyms and Terminology 8.0-2
Table 8.2-1 Summary of CET Initiating Probabilities 8.2-14
Table 8.2-2 CET Release Category Frequencies 8.2-15
Table 8.3-1 Representative Core Damage Sequences 8.3-8
Table 8.3-2 Representative Containment Response Sequence 8.3-9
Table 8.3-3 Summary of Results of Severe Accident Sequence Analysis 8.3-12

List of Illustrations

Figure 8.2-1. Containment System Event Tree (Deleted) 8.2-16
Figure 8.2-2. PCCS heat removal capability for 24 hour period 8.2-16
Figure 8.2-3. Containment Pressure with No Containment Heat Removal 8.2-17
Figure 8.2-4. Drywell Pressure with No Containment Heat Removal 8.2-18
Figure 8.3-1a. T_nIN_TSL: RPV Pressure vs. Time 8.3-13
Figure 8.3-1b. T_nIN_TSL: Containment Pressure vs. Time 8.3-13
Figure 8.3-1c. T_nIN_TSL: Lower Drywell Temperature vs. Time 8.3-14
Figure 8.3-1d. T_nIN_TSL: Drywell Water Levels vs. Time 8.3-14
Figure 8.3-1e. T_nIN_TSL: Core Power and PCCS Heat Removal vs. Time 8.3-15
Figure 8.3-2a. T_nIN_nCHR_FR: RPV Pressure vs. Time 8.3-16
Figure 8.3-2b. T_nIN_nCHR_FR: Containment Pressure vs. Time 8.3-16
Figure 8.3-2c. T_nIN_nCHR_FR: Lower Drywell Temperature vs. Time 8.3-17
Figure 8.3-2d. T_nIN_nCHR_FR: Drywell Water Levels vs. Time 8.3-17
Figure 8.3-3a. MLi_nCHR: Containment Pressure vs. Time 8.3-18
Figure 8.3-3b. MLi_nCHR: Shroud Water Level vs. Time 8.3-18
Figure 8.3-3c. MLi_nCHR: Core Heatup vs. Time 8.3-19
Figure 8.3-4a. T_nDP_nIN_TSL: RPV Pressure vs. Time 8.3-20
Figure 8.3-4b. T_nDP_nIN_TSL: Containment Pressure vs. Time 8.3-20
Figure 8.3-5a. T_nDP_nIN_nCHR_FR: RPV Pressure vs. Time 8.3-21
Figure 8.3-5b. T_nDP_nIN_nCHR_FR: Containment Pressure vs. Time 8.3-21
Figure 8.3-6a. T-AT_nIN_TSL: RPV Pressure vs. Time 8.3-22
Figure 8.3-6b. T-AT_nIN_TSL: Core Power vs. Time 8.3-22
Figure 8.3-6c. T-AT_nIN_TSL: Containment Pressure vs. Time 8.3-23
Figure 8.3-7a. T-AT_nIN_nCHR_FR: Containment Pressure vs. Time 8.3-24

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 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. Deflagration risk associated with de-inerted operation prior to and following shutdown is considered in Section 8.1.4. 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 is such that the calculated probability of 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 8A quantifies the frequency of all release categories. Appendix 8B displays additional documentation of the representative containment analysis sequences. Appendix 8C provides the screening analysis to support quantification of the containment isolation system probability. Sections 19B and 19C of the DCD document the containment ultimate strength calculation.

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
CET	Containment 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
CCIW	Wet Core-Concrete Interaction (Overpressure failure)
CCID	Dry Core-Concrete Interaction (Overpressure failure)
EVE	Ex-Vessel Steam Explosion
DCH	Direct Containment Heating

8.1 POTENTIAL FOR FAILURE DUE TO COMBUSTIBLE GAS DEFLAGRATION

Because the ESBWR containment is inerted during normal operation, 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. Of particular interest is the amount of time that, in the absence of mitigation SAM actions, is required to generate a combustible containment atmosphere in the presence of a large hydrogen release. A combustible atmosphere is assumed to be 5% oxygen by volume and qualifies as a de-inerted containment.

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 automatic depressurization system (ADS) would depressurize the reactor vessel and the gravity driven cooling system (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 passive containment cooling system (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.

To ensure that non-condensable gases do not degrade the heat transfer efficiency of the PCCS heat exchangers, vents to the suppression pool are provided. The suppression pool vents connect the heat exchanger drums to the suppression pool; non-condensable gas flow is driven by the positive DW to WW pressure differential. The calculation of post-accident

radiolytic oxygen generation accounts for this movement, and assumes that the majority of non-condensable gases move 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, while inerted, 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.

8.1.4 Risk Due to De-Inerted Operation

The ESBWR operates in short de-inerted states prior to and following shutdown. Because combustible gas deflagration cannot be excluded when the containment atmosphere is de-inerted, this analysis conservatively assumes that all core damage events during this de-inerted window will lead to containment failure.. The total time of de-inerted operation per shutdown is limited to 48 hours by Technical Specifications, so 24 hours per year is used to calculate the release contribution. To account for de-inerted operational release, the core damage frequency per day was applied to the “bypass” release category in addition to the separately calculated release frequency.

$$\text{Additional BYP Frequency} = \text{CDF}/365 = 3.34\text{E-}11$$

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 quantified as part of the "Containment Event Trees" (CETs) evaluation. The following potential release categories are the CET end states. The first group depicts containment failure due to containment systems:

- 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.

There are also release categories that describe containment failure due to phenomenological consequences. These failure modes are discussed and evaluated in Section 21, and are summarized below:

- Wet core-concrete interaction (CCIW) represents containment failure after the deluge system is successful but the core is postulated to have been ejected in a non-coolable geometry. Water does cover the core, but significant core-concrete interaction persists and eventually results in containment overpressurization.
- Dry core-concrete interaction (CCID) represents the failure of the deluge system to inject after the core has been deposited on the BiMAC layer. High levels of aerosols and non-condensables are produced and eventually lead to containment overpressurization.
- Ex-vessel explosion (EVE) represents the reactor pedestal, and subsequent containment, failure as a result of a steam explosion. The analysis assumes that an adequate steam explosion occurs every time the core melts through the RPV into a "high" lower drywell water level. The severe accident analysis in Section 21 shows that ex-vessel explosion due to medium water level is physically unreasonable. However, the remote potential for a steam explosion at "medium" water level will be documented in a Section 11 sensitivity with a split fraction of 1E-3.

- Direct containment heating (DCH) represents containment failure due to the direct deposit of the core on the LDW walls during a high-pressure melt ejection (HPME). Similar to medium water level EVE, this failure mode is found to be remote in the Section 21 analysis, and is excluded from the baseline Level 2 analysis. However, the remote potential for an immediate containment failure due to DCH will be captured in a Section 11 sensitivity by assigning a 1E-3 point estimate in the class III event trees.

Also shown on the CETs are two end states that 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 CETs are discussed in more detail in Subsection 8.2.1.

8.2.1 Containment Event Trees

The CETs used for the Level 2 PRA analysis and described in Section 8 and 8C are logically equivalent to the combination of containment phenomenology event trees (CPET) and containment system event trees (CSET) used in the Section 21 discussion. In Section 21, separation of the two trees allowed a more clear distinction between the two analyses; a single, combined CET simplifies the quantification of the Level 2 PRA itself.

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 8A. 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	5.63E-9	46.16	Sequences with RPV failure at low pressure
Class II	4E-11	0.35	Sequences with containment failure preceding core damage
Class III	4.52E-9	37.02	Sequences with RPV failure at high pressure
Class IV	1.87E-9	15.36	Sequences involving failure to insert negative reactivity
Class V	1E-10	1.20	Break outside of containment

Containment event trees were used to evaluate the complete spectrum of potential challenges to containment integrity. Both phenomenological effects and system responses are captured in the CET analyses.

The number of CETs needed to evaluate the containment response to 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.
- 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 CETs were developed by establishing the functions and containment systems that were relevant to mitigating the overpressure and bypass challenges and combining those with the phenomenology discussed in Section 21. The CETs were then constructed using point estimates for phenomenological effects and appropriate logic to account for mitigating system success or failure by establishing the logically possible containment responses. Finally, the end states of the CETs, 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 CETs indicates that that there is a generally common structure to the trees, with only the phenomenology differing between the initiating, or entry events. Determination of the CET entry event probabilities is discussed in Subsection 8.2.1.1. The containment systems evaluated in the CETs are summarized in Subsection 8.2.1.2 with the

associated top events being discussed in Subsection 8.2.1.3. The end states of the trees, which become the release categories for the consequence evaluation, are discussed in Subsection 8.2.1.4. The frequencies associated with the release categories are presented in Subsection 8.2.1.5. Appendix 8A provides additional detail on the release category quantification.

8.2.1.1 CET Entry Events

The CET entry event frequencies are summarized in Table 8.2-1. Note that each accident class may be divided into various subclasses. The subclasses were necessary to reflect the water level in the lower drywell at the time of RPV failure and the associated phenomenological effects that should be considered. For example, accident Class I was divided into Class I_LD (low water level or dry), Class I_M (medium water level), and Class I_H (high water level). Based on Level 1 results, Class III sequences only result in low water level or dry scenarios, so only Class III_LD was considered.

The Class IV (ATWS) sequences experience core damage at high pressure because ADS is inhibited as part of the core damage mitigation effort. However, it is assumed that Emergency Operating Procedures (EOPs) will instruct the operator to depressurize after core damage has occurred in an attempt to preserve containment. It is shown in Appendix 8A that the frequency of ATWS sequences experiencing RPV rupture at high pressure is negligible, so only failures at low pressure were analyzed. Thus, the subclasses for Class IV are IV_LD (ATWS with low water level or dry) IV_M (ATWS with medium water level), and IV_H (ATWS with high water level).

The qualitative binning of Level 1 sequences into water levels is shown in Section 7.

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. In high pressure severe accident sequences the ICS has, by definition, failed already in the Level 1 sequence. Consequentially, 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 a combination of thermocouples embedded in the lower drywell floor and temperature sensors in the lower drywell airspace.

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 thermal-hydraulic 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. Vacuum breakers are required to be leak-tight for effective PCCS operation.

Drywell Spray

In ESBWR, the drywell sprays are designed as a post-event containment clean up and recovery system to limit the exposure to "first-in" personnel after a severe accident. Use of the drywell spray system will not be included as part of the severe accident mitigation guidelines; the drywell sprays are not included in the Level 2 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. The success criteria for vapor suppression is no more than one vacuum breaker may be stuck open, and at least one vacuum breaker must open to break a positive WW to DW pressure differential.

Manual Containment Overpressure Protection (MCOPS)

To prevent overpressurization failure of the containment as a result of containment heat removal failure, the ESBWR contains a manually controlled vent connecting the suppression chamber gas space to the environment through the reactor building ventilation system. 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

Subsection 8.2.1.1 identified the entry events for the containment system event tree. The next step in constructing CET was to define, as top events, the phenomenological events and

system functions needed to assess the containment response to severe accident challenges. The phenomena include ex-vessel explosion, direct containment heating, and dry and wet core concrete interaction. The system 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 trees necessary to model the containment response are simple in form. Further, because the Level 2 PRA model is arranged as an extension of the Level 1 accident sequences, the initiator impact of all sequences is implicitly included. That is, the structure of a CET is the same irrespective of the Level 1 initiating event being considered. The trees differ only in the quantification as dependencies on the entry event are reflected in phenomena that must be considered depending on the entry event bin. For example, the class III CET considers the possibility of containment failure due to DCH because the RPV fails at high pressure. The CET 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 8A 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 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.

(2) Action taken to initiate a backup system, e.g., to actuate the FAPCS if the PCCS were unavailable.

(3) 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 EVE_DAM

Top event EVE, which considers the probability of containment failure due to ex-vessel steam explosion, is only included in CETs in which the entry events include a medium water level in the lower drywell. The phenomenology analysis in Section 21 considers a steam explosion from a medium water level to be physically unreasonable; baseline results only include EVE contribution from high water level cases. The effect of considering medium-level steam explosions is presented in a sensitivity study in NEDO-33201 Section 11.

The ESBWR RPV design features several small lower head penetrations for equipment such as fine motion control rod drive (FMCRD) units and in-vessel instrumentation. The heavily dominant RPV failure mode in a core melt scenario is expected to be one of these lower head penetrations, which would result in a relatively slow flow of corium from the vessel breach, and thus potential premixtures that are much smaller (reduced energetics) than those considered in the analyses presented here. However, since these complex processes are not well known, we assume conservatively that all high water level cases result in ex-vessel explosions that are large enough to damage the pedestal to an extent that could be considered gross containment failure.

8.2.1.3.4 Top Event DCH_DAM

Top event DCH, which considers the probability of containment failure due to direct containment heating, is only included in CETs for which the entry events are Class III core damage sequences. That is, the RPV fails at high pressure. The phenomenology analysis in Section 21 considers containment failure due to DCH to be physically unreasonable. The effect of considering actual containment failure caused by DCH is presented in a sensitivity study in NEDO-33201 Section 11.

8.2.1.3.5 Top Event BI_SP

Top event BI_SP represents the function of the deluge system of the GDCS. The deluge system is a sub-system of GDCS that is not included in the Level 1 analysis, so a new fault tree was created to address its function. The deluge system is comprised of twelve injection lines, each of which includes a squib valve that must be fired for successful operation. Actuation of the system is performed by a deluge-specific, stand-alone actuation system that does not require any support systems. Failure of the deluge system is assumed to result in containment failure due to dry core-concrete interaction.

8.2.1.3.6 Top Event BI_FN

Top event BI_FN estimates the probability that even if the deluge system functions successfully, the water is not successful in cooling the core. As postulated in Section 21, this node is used to quantify both the uncertainty in the BiMAC design effectiveness, and the potential for the core to be ejected in a non-coolable geometry. Failure of this node is assumed to result in containment failure due to wet core-concrete interaction. Node BI_FN is assigned a probability of 1.00E-2.

8.2.1.3.7 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 (BOC) for the Level 1 analysis. As indicated in Appendix 8C, a screening evaluation was performed to identify those containment penetrations that could potentially lead to offsite consequences. The screening analysis found that two systems, Main Steam and Feedwater, require isolation during a severe accident. The same logic is used for these isolations as with the BOC accidents in the Level 1 PRA. This approach addresses both the "failure-to-close" hardware and actuation logic failure modes of the containment isolation system.

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.

8.2.1.3.8 Top Event VB

Top event VB models vacuum breaker operation for vapor suppression. The success criteria for VB are the same as DS-TOPVB in the Level 1 analysis: at least one vacuum breaker must open to relieve a vacuum, and two of three must be leak-tight for vapor suppression. If VB were not successful 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.

8.2.1.3.9 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 Subsection 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 for over 72 hours 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; the initiator impact is addressed in the Level 1 sequences that comprise the CET entry events.

After 24 hours, it is conservatively assumed that the PCCS pool must be replenished by opening valves to the moisture separator storage pools or providing make-up water from the Fire Protection System. Upon connecting the additional pools, 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 AC power to operate.

8.2.1.3.10 Top Event VT

Top event VT models operator action to prevent containment failure by use of a suppression chamber vent path. If Event VT succeeds, 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 within 24 hours. 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. Top event VT contains a dependency on the Level 1 analysis because certain Level 1 sequences include containment venting.

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 at least two orders of magnitude, the most likely path through the CET 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 CET presented in Appendix 8A are discussed in more detail in the following sections. Drawing on the quantification presented in Appendix 8A, the probability of each release category is summarized in Table 8.2-2.

Direct Containment Heating (DCH)

The release category "Direct Containment Heating" is a result of highly energetic phenomenological effects during RPV failure at high pressure. As discussed in Section 21, containment failure due to DCH is considered physically unreasonable. However, a sensitivity study is documented in Section 11 that assigns a probability of 1E-3 to containment failure upon RPV rupture at high pressure.

Ex-Vessel Explosion (EVE)

The release category "Ex-Vessel Explosion" is a high-energy phenomenon that occurs when the RPV fails at low pressure and the core falls into an appropriate depth of water. The

conditions required, and likelihood of, an EVE event are discussed in detail in Section 21. The total frequency of the EVE release category is approximately two orders of magnitude below TSL.

Dry Core-Concrete Interaction (CCID)

Core-concrete interaction is a phenomenological failure mode of the containment discussed in detail in Section 21. The general containment failure mode is overpressure as a result of non-condensable gas generation. The CCID release category occurs when the GDCS deluge system fails its function of water injection to the lower drywell. The frequency of the CCID release category is more than four orders of magnitude below TSL.

Wet Core-Concrete Interaction (CCIW)

The containment failure mode of CCIW is the same as that in CCID. The only difference in the CCIW sequences is that the GDCS deluge system has successfully injected to the lower drywell, but the water is unsuccessful in cooling the ejected core. The split fraction representing this probability is developed in Section 21. The frequency of the CCIW release category is approximately two orders of magnitude below TSL.

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. Due to the reliability of the containment isolation system, the probability of such a release occurring is approximately three orders of magnitude less than the TSL release category. According to the analysis, the only credible failure mode for the containment isolation system is the common cause failure Reactor Protection System (RPS) components to support MSIV isolation.

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 event tree. Treating the possibility of FR in this way accounts for uncertainties in the loss of heat removal analysis and containment venting guidance, and provides a conservative estimate of the likelihood of a controlled release. Further, inclusion of venting on the CET allows for modeling a period longer than 24 hours after the onset of core damage. Although the probability of the FR release category was calculated as more than four of magnitude less than the TSL release category, it will still be considered in Section 9 as a potential source term.

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. Much of the OP release category frequency is derived from Level 1, Class II sequences. This frequency is eventually excluded from the offsite consequences analysis, as discussed in Subsection 8.3.2.2. 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. The high reliability of the vacuum breakers necessary for the vapor suppression function is demonstrated in that the calculated probability of the OPVB release category is more than three orders of magnitude less than the TSL release category. The Level 1 dependency of this node is captured by using the "DS-TOPVB" gate from the Level 1 model.

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 FAPCS in suppression pool cooling (SPC) mode, 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. The Level 1 dependency is captured by using the same gates, "WP-TOPDHR", "DL-TOPVB", and "WS-TOPSPC" that are used in the Level 1 analysis. Because of the reliability of the SPC mode of the FAPCS and the passive PCCS, the calculated probability of the OPW1 release category is three orders of magnitude less than the TSL release category.

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. Although the probability of the OPW2 release category is more than four orders of magnitude less

than the TSL release category, it will conservatively be considered in Section 9 as a potential source term.

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 frequencies of the release categories are calculated by quantifying all of the CET with the single top Level 2 model. The individual release categories can be calculated by multiplying the Fussel-Vessely of that release category's marker (flag) by the overall release frequency, or CDF. As seen in Table 8.2-2, the most likely release category is that associated with leakage from an intact containment, TSL. Release categories associated with containment failure due to phenomenological or system failure events are at least two 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 core damage frequency. Using the values from Appendix 8A, Table 8A-3, and applying "ε" for probabilities less than 1E-15,

$$Containment\_Effectiveness = \frac{TSL\_frequency}{Level\_1\_CDF} = \frac{1.12E-8}{1.22E-8} = 0.921$$

Table 8.2-1

Summary of CET Initiating Probabilities

CET	LDW Water Level	Entry Gate*	Total Entry Gate Frequency
Class I – Low (RPV failure at low pressure)	Low/Dry	I_LD	4.94E-09
Class I - High (RPV failure at low pressure, high water level in LDW)	High	I_H	5.88E-10
Class I – Medium (RPV failure at low pressure, high water level in LDW)	Medium	I_M	9.92E-11
Class III (RPV failure at high pressure)	Low/Dry	III_LD	4.51E-09
Class IV – Low (ATWS with RPV failure at low pressure)	Low/Dry	IV_LD	1.85E-09
Class IV – High (ATWS with RPV failure at low pressure, high water level in LDW)	High	IV_H	2.18E-11
Class IV – Medium (ATWS with RPV failure at low pressure, medium water level in LDW)	Medium	IV_M	7.25E-13

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

Table 8.2-2

CET Release Category Frequencies

Release category	Frequency (per year)*
TSL	1.12E-8
FR	ε
BYP	5.6E-11
OPVB	1.6E-11
OPW1	3.2E-11
OPW2	ε**
CCIW	9.9E-11
CCID	1E-12
EVE	6.10E-10
DCH	0.00
BOC (from Level 1)	1.47E-10

*The frequency is the summed contribution to the release category from all accident classes, as shown in Table 8A-3. BYP is also augmented with frequency from de-inerted operation, as described in 8.1.4.

**Calculated frequencies less than 1E-12 are reported as "ε". The actual calculated number is preserved for input to the offsite consequences analysis in Section 10.

Figure 8.2-1. Containment System Event Tree (Deleted)

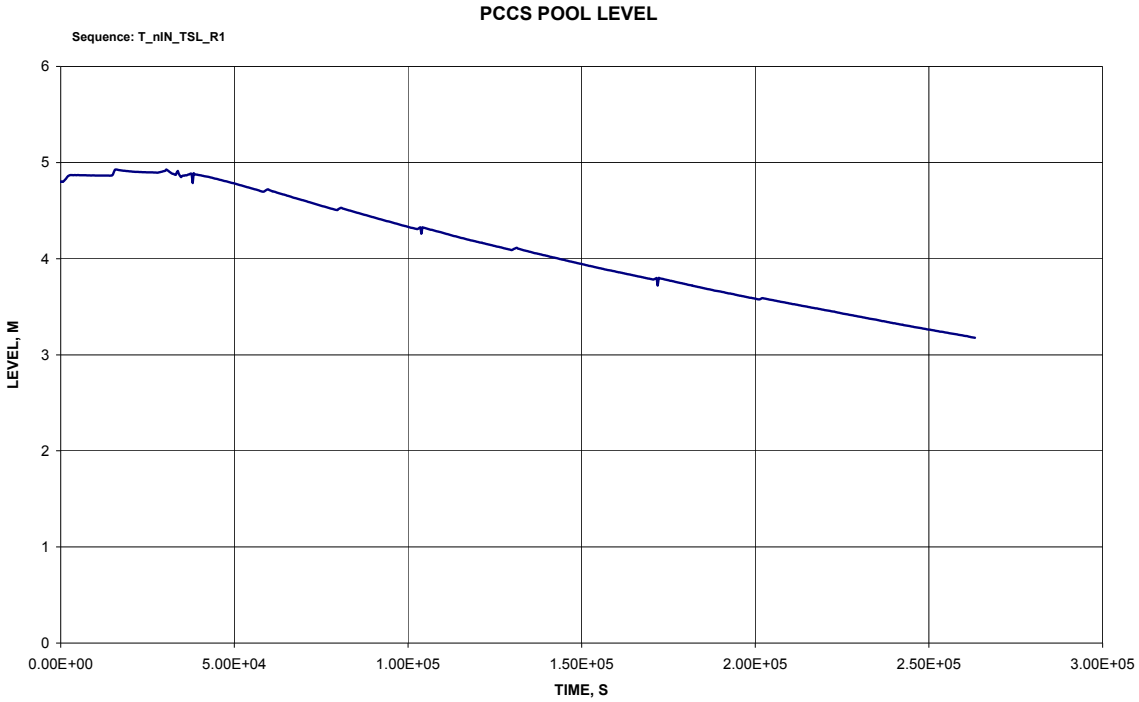

Figure 8.2-2. PCCS heat removal capability for 24 hour period

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.

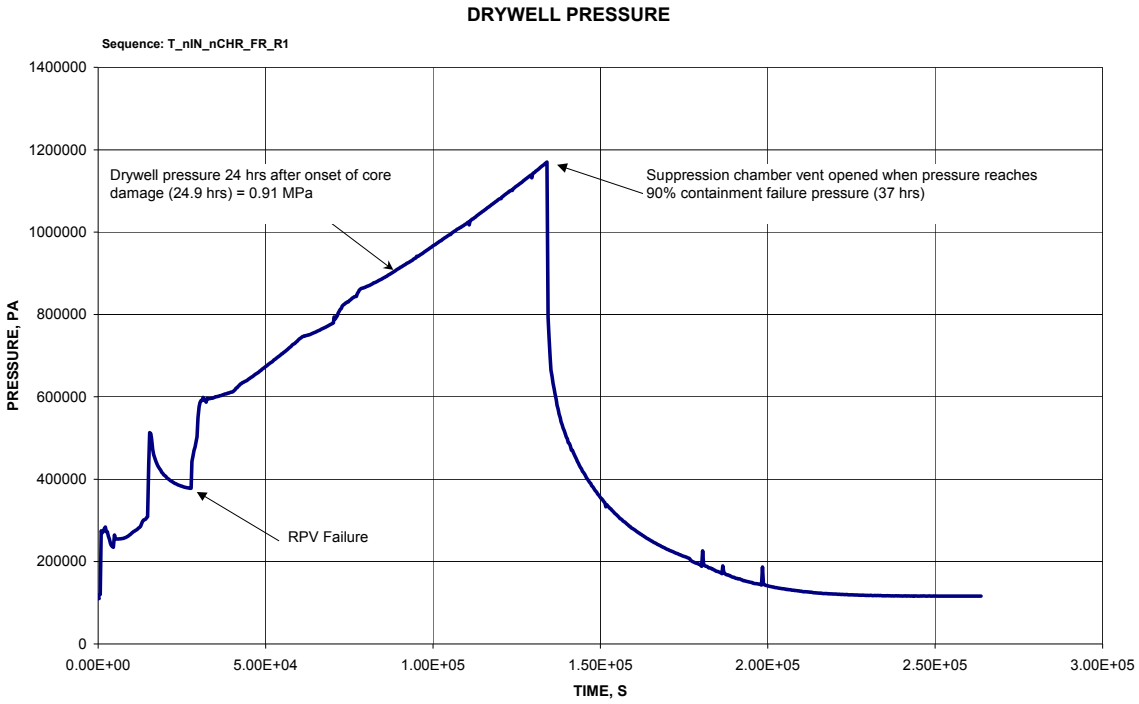

Figure 8.2-3. Containment Pressure with No Containment Heat Removal

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 (8.64E+04 seconds).

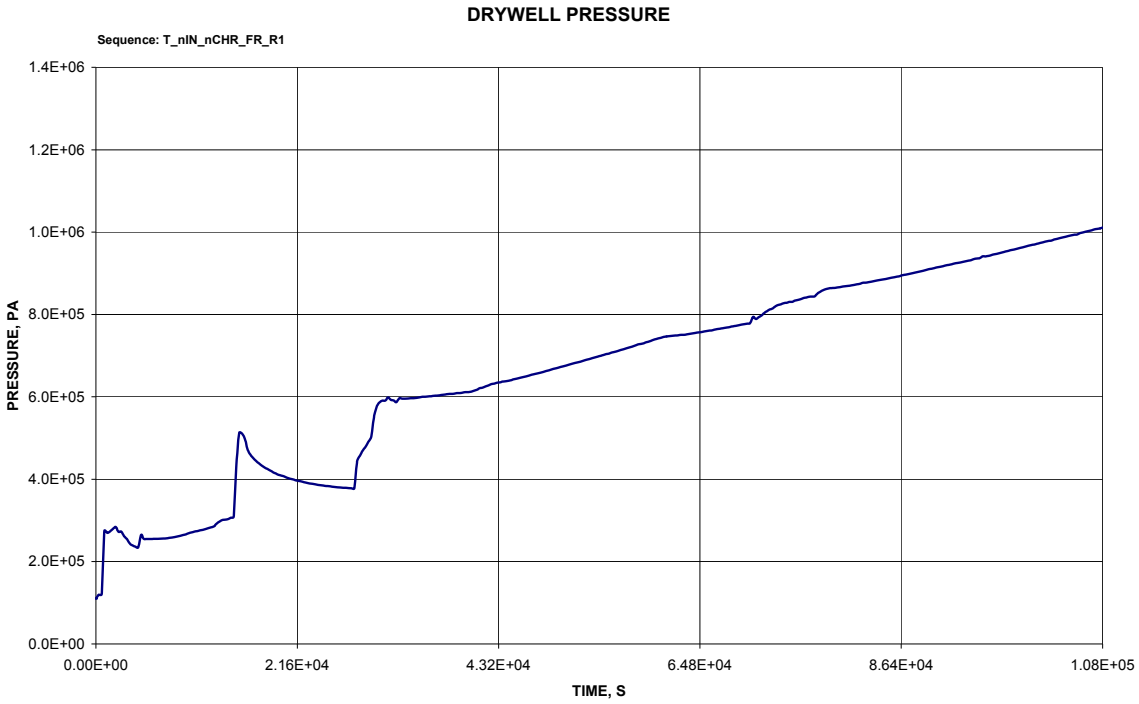

Figure 8.2-4. Drywell Pressure with No Containment Heat Removal

Example shown is for a Class I sequence with a failure of containment heat removal. As shown in the chart, containment does not pressurize to failure within 24 hours after onset of core damage (8.64E+04 seconds).

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 due to system failures 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, overpressurizes, or fails due to phenomenological effects 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.

Subsection 8.3.1 summarizes the code used for accident simulation. Subsection 8.3.2 provides the simulation results for a spectrum of potential severe accidents representing each accident class. The ultimate containment strength analysis is provided in DCD Sections 19B and 19C.

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. With the exception of Class V accidents, in which the containment is completely bypassed, a single dominant sequence was 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 and the definitions of the Level 1 sequence bins. For example, Table 7.2-3 indicates that about 80% of the Class I frequency is associated with a stuck open relief valve (T-IORV), a large feedwater LOCA (LL-S-FDWA/B), or loss of feedwater (T-FDW) sequences. From the perspective of modeling the containment response to a severe accident, all Class Isequences can be represented as a transient with loss of injection T-nIN and successful depressurization. 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.

Graphs of many additional MAAP parameters are shown in Appendix 8B to provide complete documentation of the containment analysis.

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 46.16% 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 because of the low reactor pressure. 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 experiences delayed depressurization due to opening of the first ADS-actuated valves at about 655 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 7.8 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. 58 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 location. 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 32.7 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 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 As shown in representative sequence MLi_nCHR, core damage does not occur during the first 72 hours post-accident. Because core damage is beyond the Level 2 analysis mission time, Class II accident sequences are not included as inputs to the offsite dose consequences in Section 10. The Class II frequency is included in the release frequencies reported in table 8.2-2 so that the entire Level 1 CDF is accounted for..

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.

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-hour 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 offsite consequence analysis However, Class II events are preserved in the release frequency calculation so that the total release is equivalent to CDF.

There are a few Class II accident sequences that do not consider the potential for low-pressure injection during the sequence on the Level 1 event trees. However, if GDCS were considered on these sequences, the failure of GDCS injection resulting in core damage would place the result below the truncation limit. As such, the selected representative sequence is considered to be appropriate.

Class II-a sequences are partitioned into the OPW1, OPW2, and OPVB release categories depending on which system failures resulted in core damage. The Class II-b sequences are binned as release category FR. All Class II frequencies are included in the total frequencies reported in Tables 8.2-2 and 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 represents approximately 37.02% 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 6.2 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 0.62 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 approximately 15.6% 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 5.9 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 0.57 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 1.1 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	9.45E-09	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	6E-12	Release path from drywell through area large enough to depressurize containment. Vapor suppression, containment heat removal and controlled venting functions failed.
_BYP	BYP	2E-12	Release path from drywell through open line connecting drywell atmosphere to environment
_CCIW	CCIW	4.4E-11	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_CCID	CCID	ε	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_EVE	EVE	5.88E-10	Release path from drywell through large area immediately following RPV rupture.
MLi_nCHR	(cdii)	4.2E-11	Release path from drywell through area large enough to depressurize containment. Containment heat removal not available.
T_nDP_nIN_TSL	TSL	4.52E-09	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.

Table 8.3-2

Representative Containment Response Sequence

Containment Response Sequence Descriptor	Release Category	Frequency* (per reactor-year)	Containment Response Summary
_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	4E-12	Release path from drywell through open line connecting drywell atmosphere to environment
_CCIW	CCIW	3.7E-11	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_CCID	CCID	ε	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_DCH	DCH	0.00	Containment failure mode not considered credible.
T-AT_nIN_TSL	TSL	1.84E-09	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	1.7E-11	Release path from drywell through open line connecting drywell atmosphere to environment
_CCIW	CCIW	1.8E-11	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_CCID	CCID	ε	Release path from drywell through area large enough to depressurize containment; no controlled venting.
_EVE	EVE	2.2E-11	Release path from drywell through large area immediately following RPV rupture.

Table 8.3-2

Representative Containment Response Sequence

Containment Response Sequence Descriptor	Release Category	Frequency* (per reactor-year)	Containment Response Summary

Notes:

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

"ε" refers to a calculated frequency of <1.0E-15.

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 (hour)	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	665	0.50	1.1	7.8	7.8	0.05	0.58	NA
T_nIN_nCHR_FR	661	0.48	1.3	7.7	7.7	0.05	0.92	>24
MLi_nCHR	124	>72	>72	>72	NA	NA	NA	NA
T_nDP_nIN _TSL	NA	0.92	1.7	6.2	6.2	<0.1	0.57	NA
T_nDP_nIN_nCHR _FR	NA	0.93	1.7	6.7	6.7	<0.1	1.01	>24
T-AT_nIN_TSL	1163	0.1	0.77	5.9	6.0	0.1	0.57	NA
T-AT_nIN_nCHR_FR	1161	0.1	0.77	5.7	5.7	<0.1	1.1	>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

Figures 8.3-1a through e: Sequence T_nIN_TSL

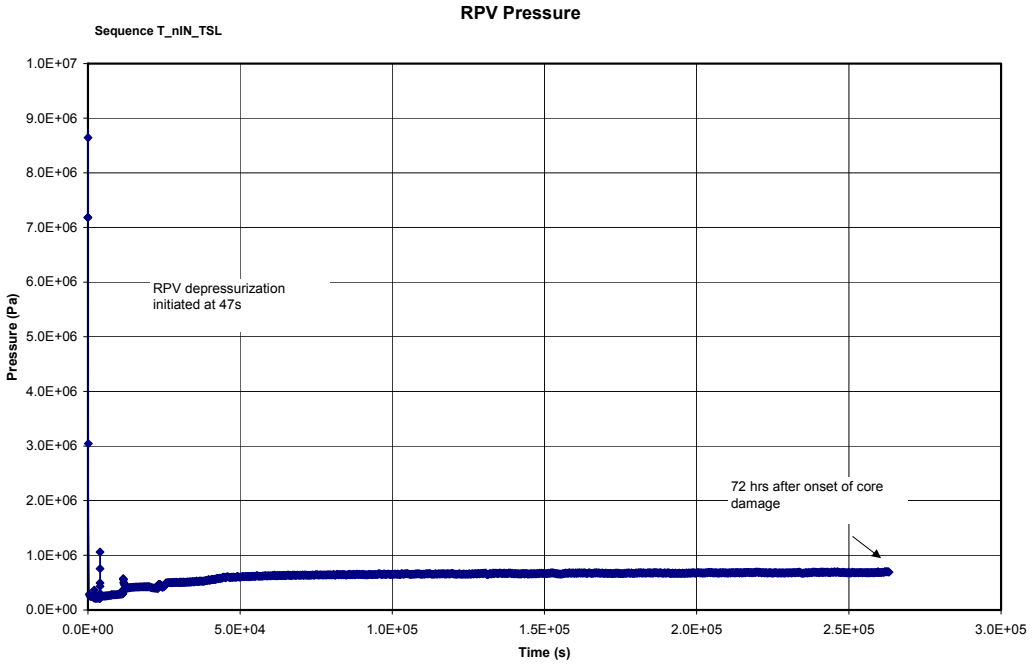

Figure 8.3-1a. T_nIN_TSL: RPV Pressure vs. Time

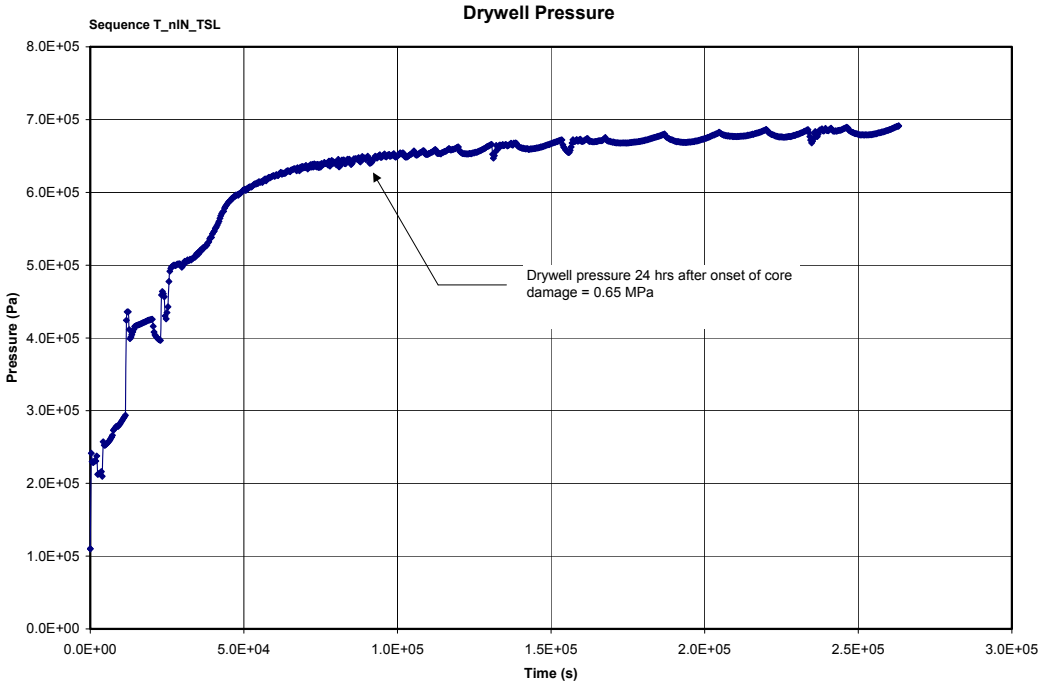

Figure 8.3-1b. T_nIN_TSL: Containment Pressure vs. Time

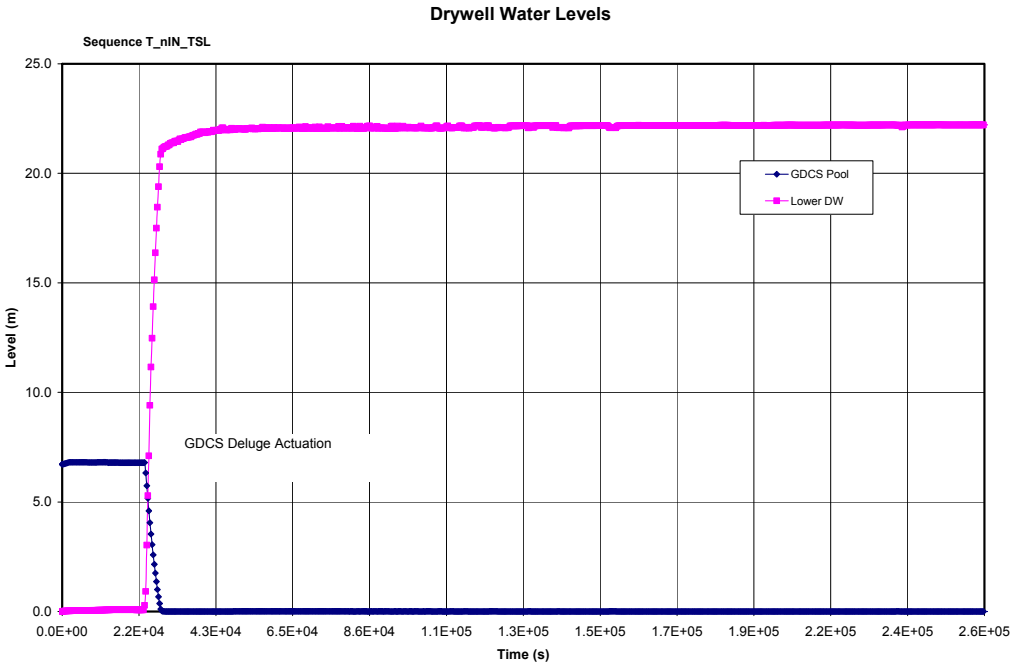

Figure 8.3-1c. T_nIN_TSL: Lower Drywell Temperature vs. Time

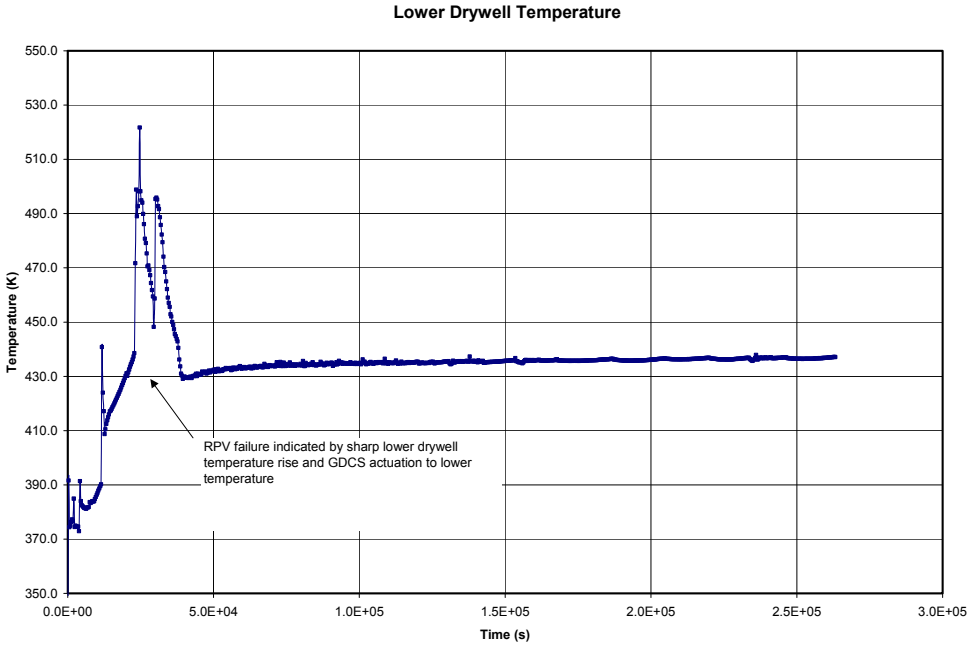

Figure 8.3-1d. T_nIN_TSL: Drywell Water Levels vs. Time

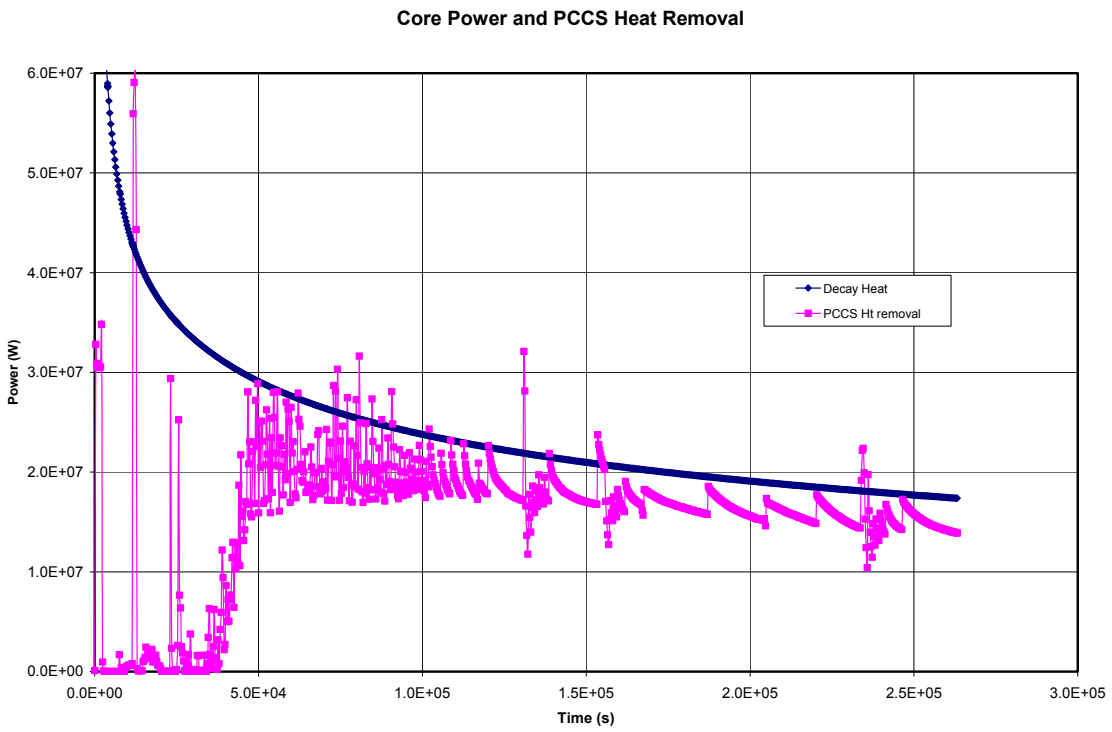

Figure 8.3-1e. T_nIN_TSL: Core Power and PCCS Heat Removal vs. Time

Figures 8.3-2a through d: T_nIN_nCHR_FR

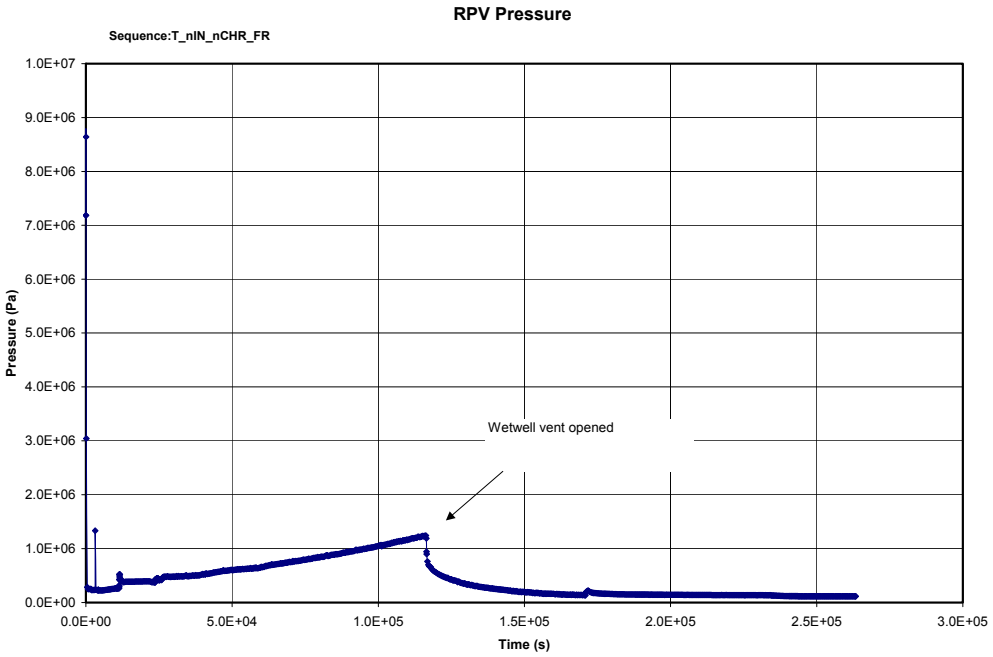

Figure 8.3-2a. T_nIN_nCHR_FR: RPV Pressure vs. Time

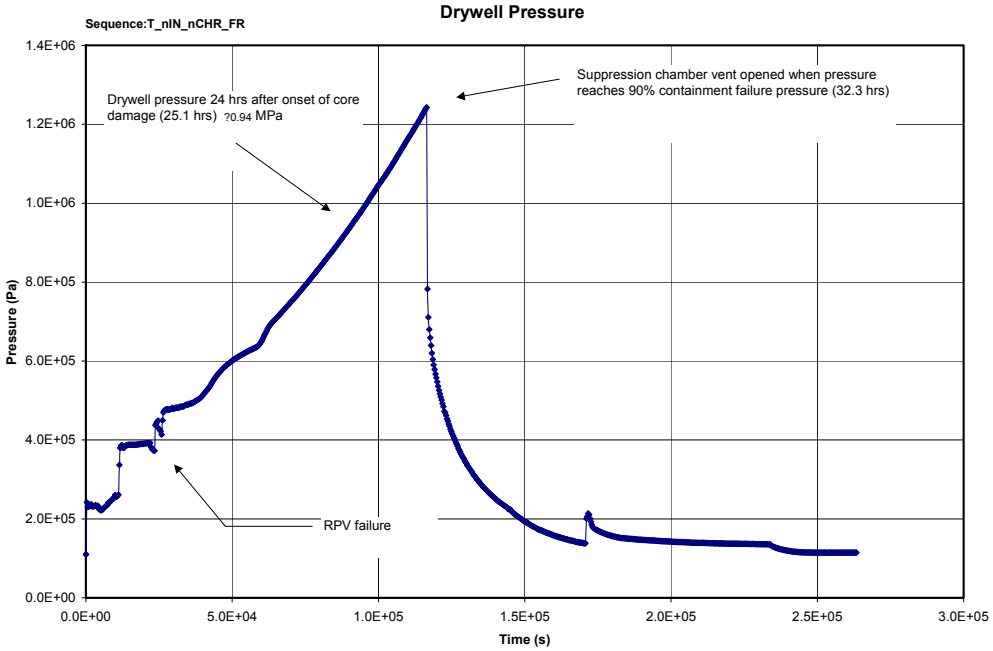

Figure 8.3-2b. T_nIN_nCHR_FR: Containment Pressure vs. Time

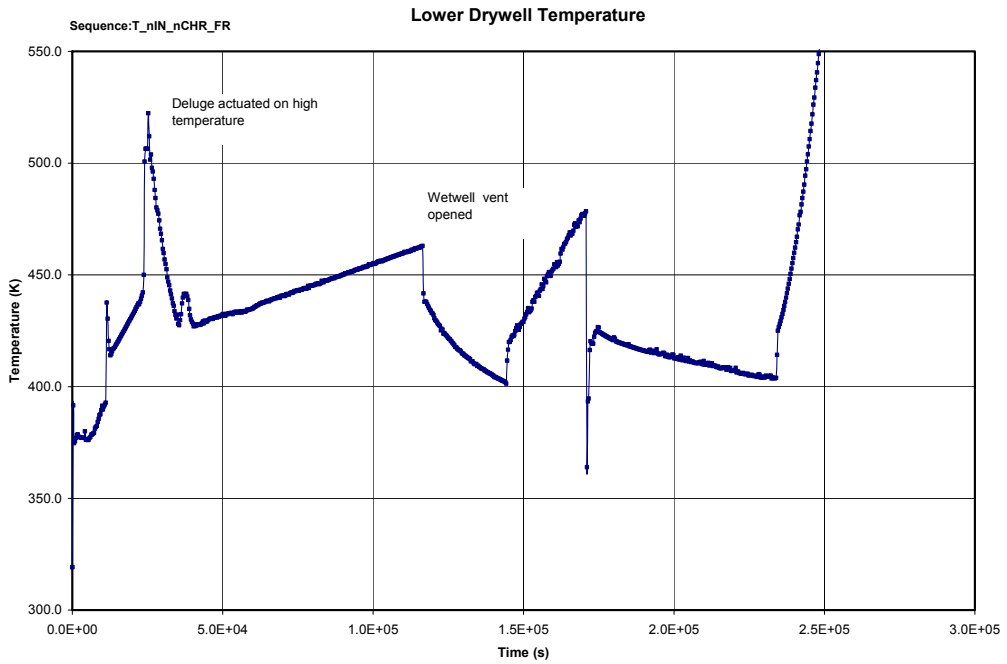

Figure 8.3-2c. T_nIN_nCHR_FR: Lower Drywell Temperature vs. Time

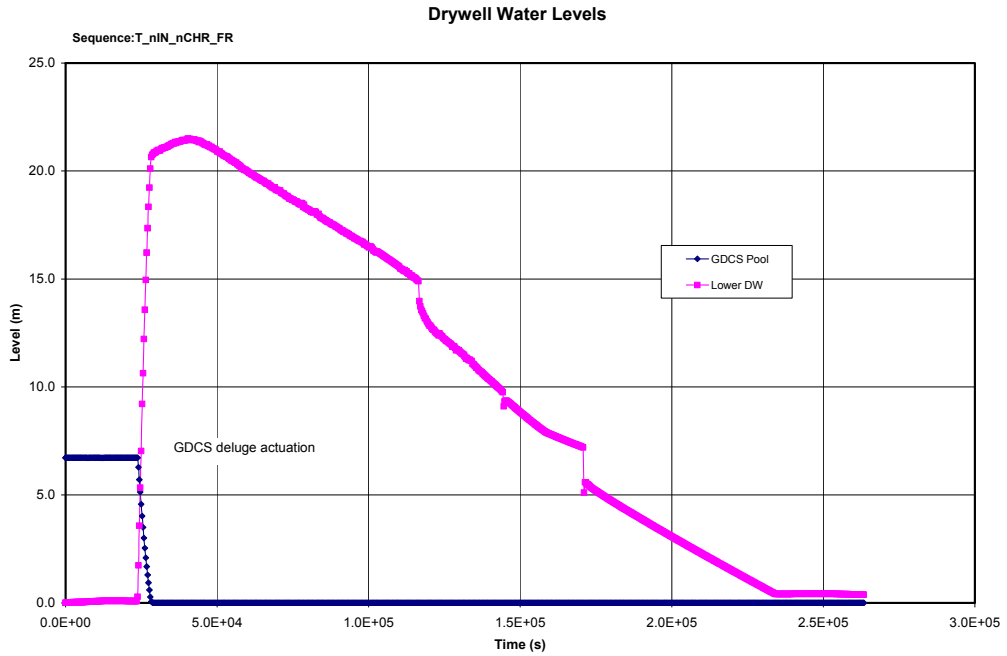

Figure 8.3-2d. T_nIN_nCHR_FR: Drywell Water Levels vs. Time

Figures 8.3-3a through c: MLi_nCHR

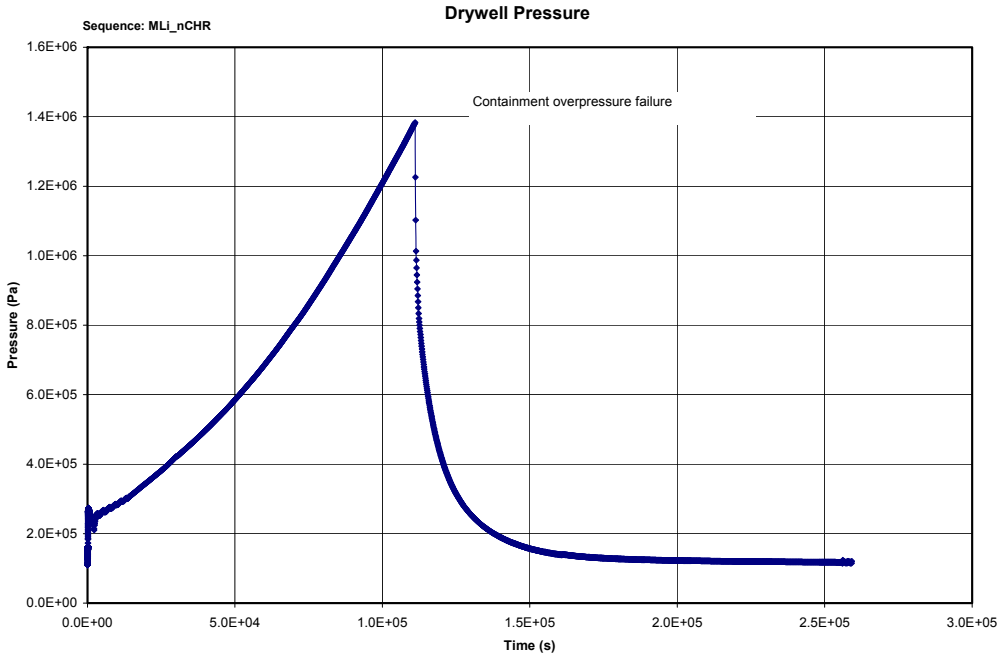

Figure 8.3-3a. MLi_nCHR: Containment Pressure vs. Time

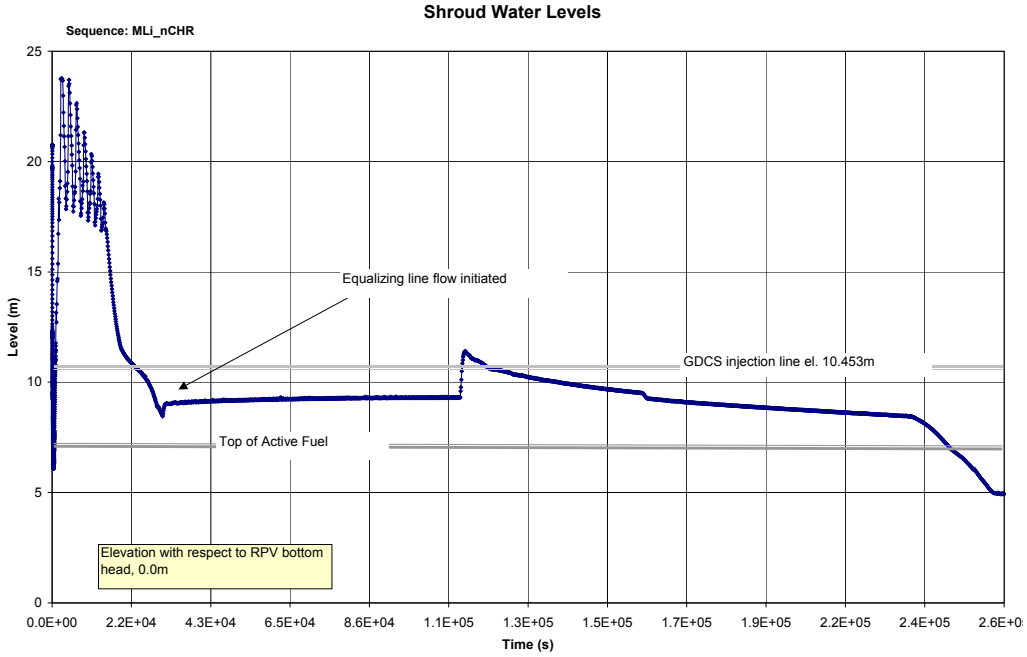

Figure 8.3-3b. MLi_nCHR: Shroud Water Level vs. Time

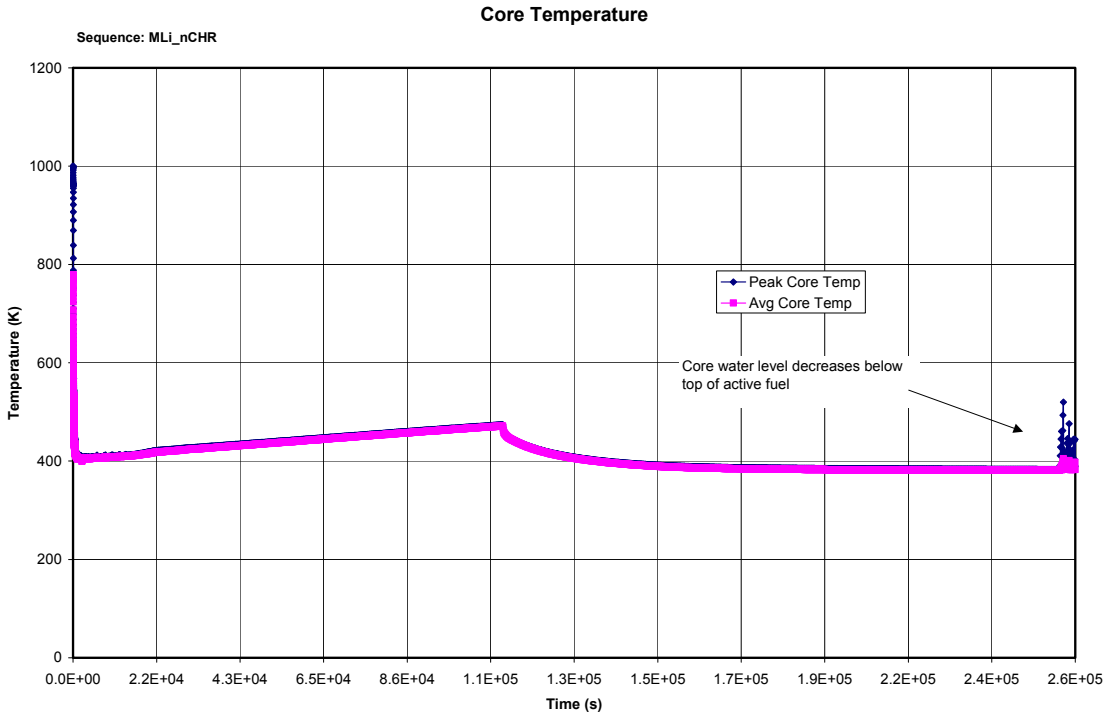

Figure 8.3-3c. MLi_nCHR: Core Heatup vs. Time

Figures 8.3-4a through b: T_nDP_nIN_TSL

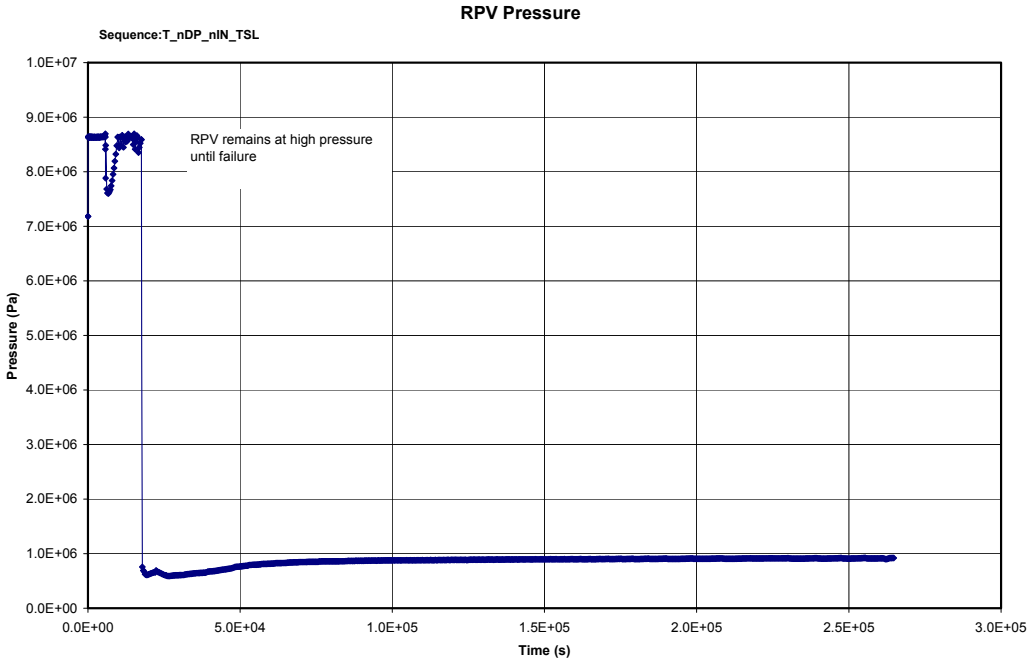

Figure 8.3-4a. T_nDP_nIN_TSL: RPV Pressure vs. Time

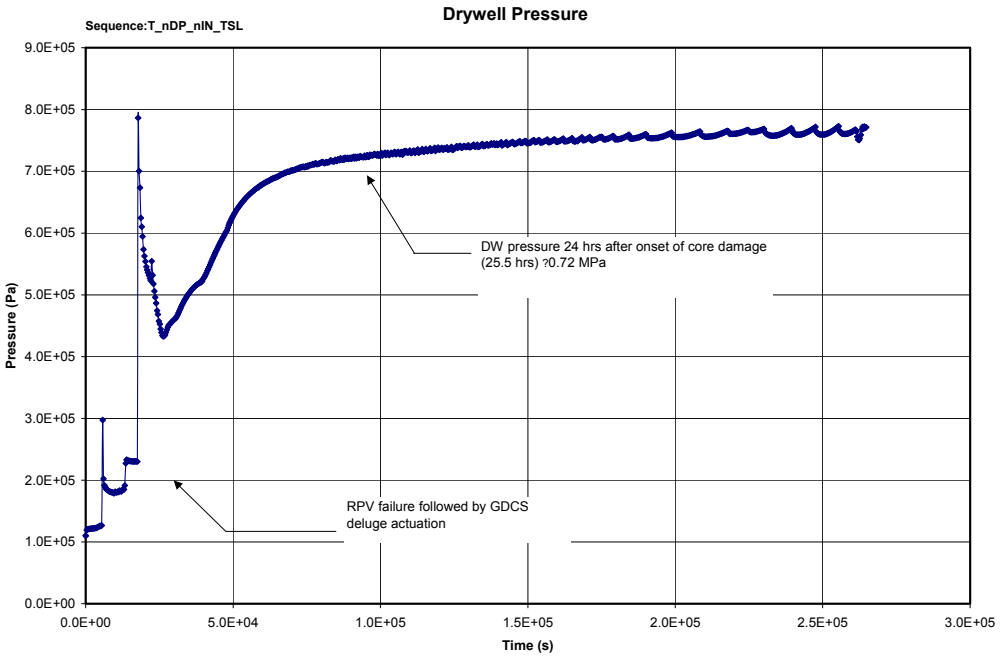

Figure 8.3-4b. T_nDP_nIN_TSL: Containment Pressure vs. Time

Figures 8.3-5a through b: T_nDP_nIN_nCHR_FR

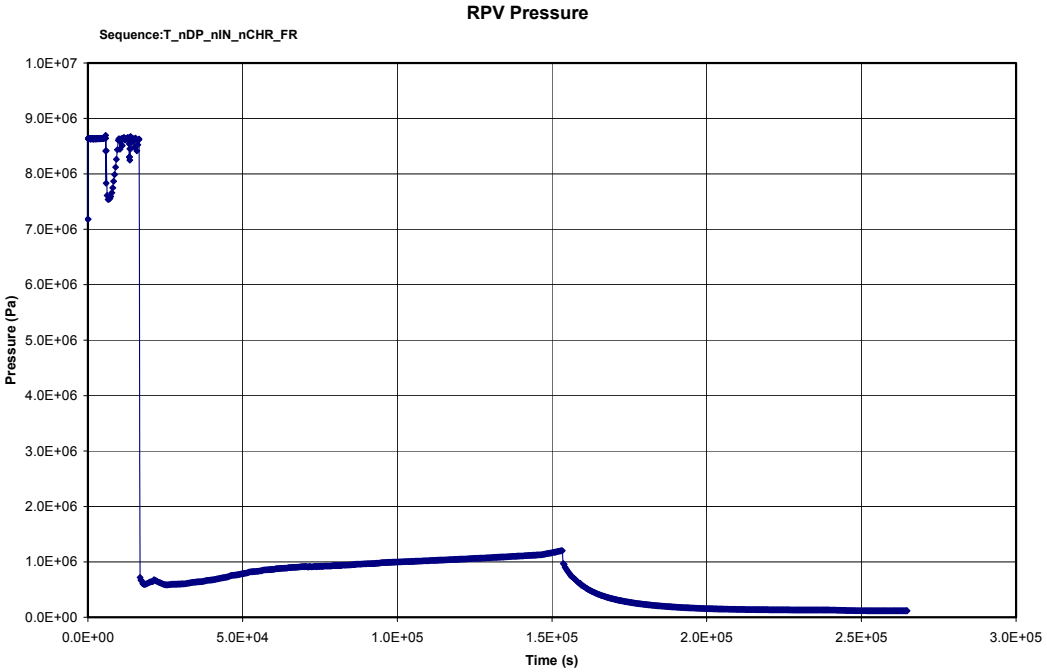

Figure 8.3-5a. T_nDP_nIN_nCHR_FR: RPV Pressure vs. Time

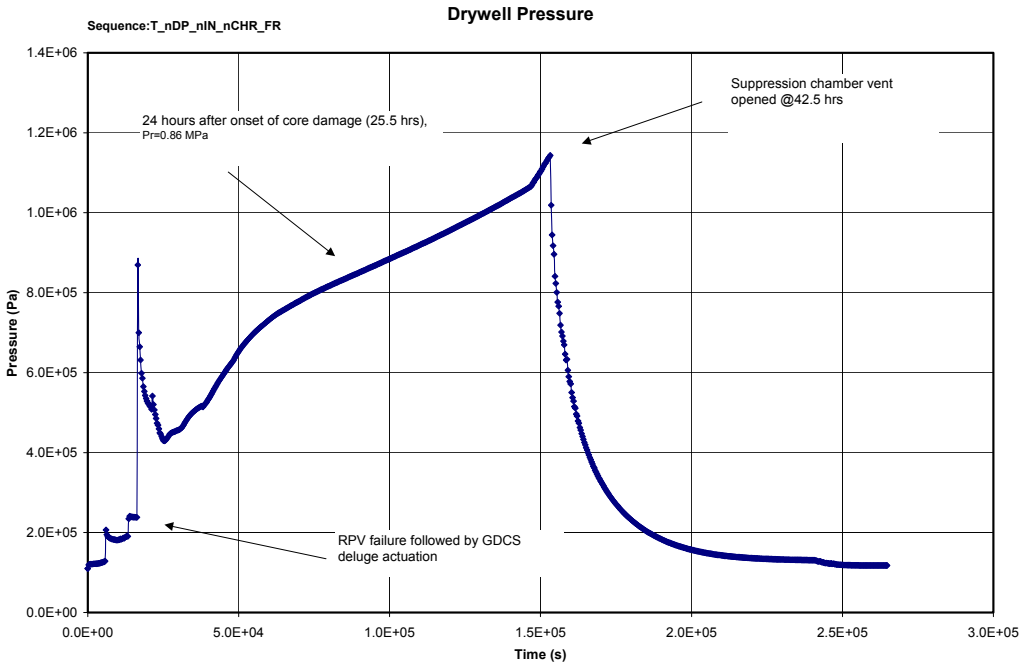

Figure 8.3-5b. T_nDP_nIN_nCHR_FR: Containment Pressure vs. Time

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

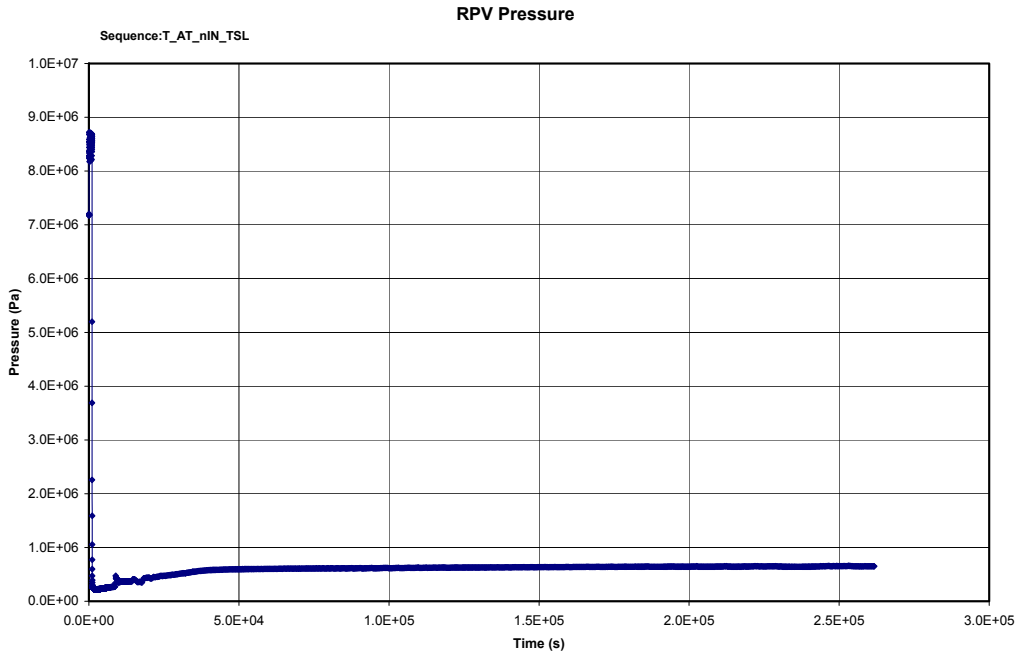

Figure 8.3-6a. T-AT_nIN_TSL: RPV Pressure vs. Time

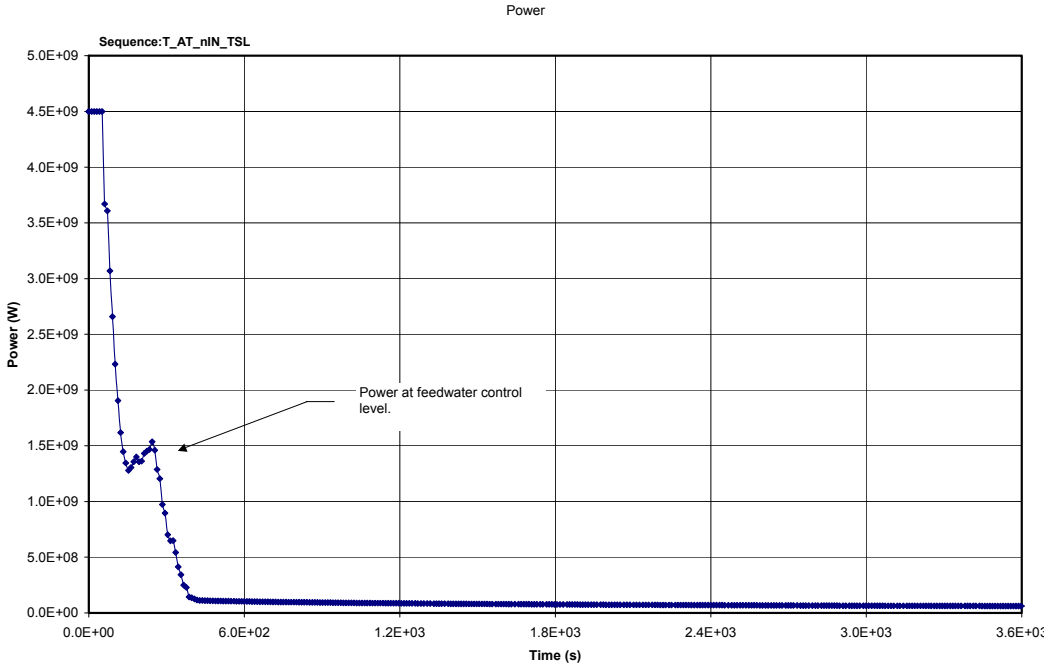

Figure 8.3-6b. T-AT_nIN_TSL: Core Power vs. Time

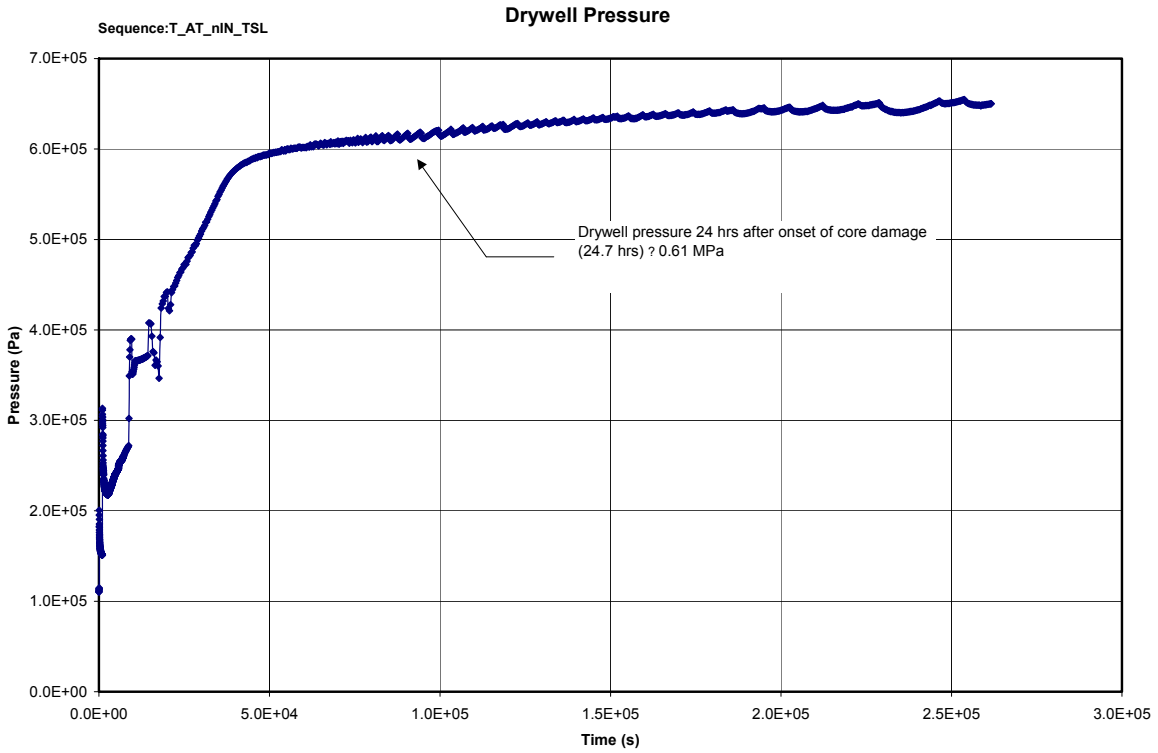

Figure 8.3-6c. T-AT_nIN_TSL: Containment Pressure vs. Time

Figure 8.3-7a: Sequence T-AT_nIN_nCHR_FR

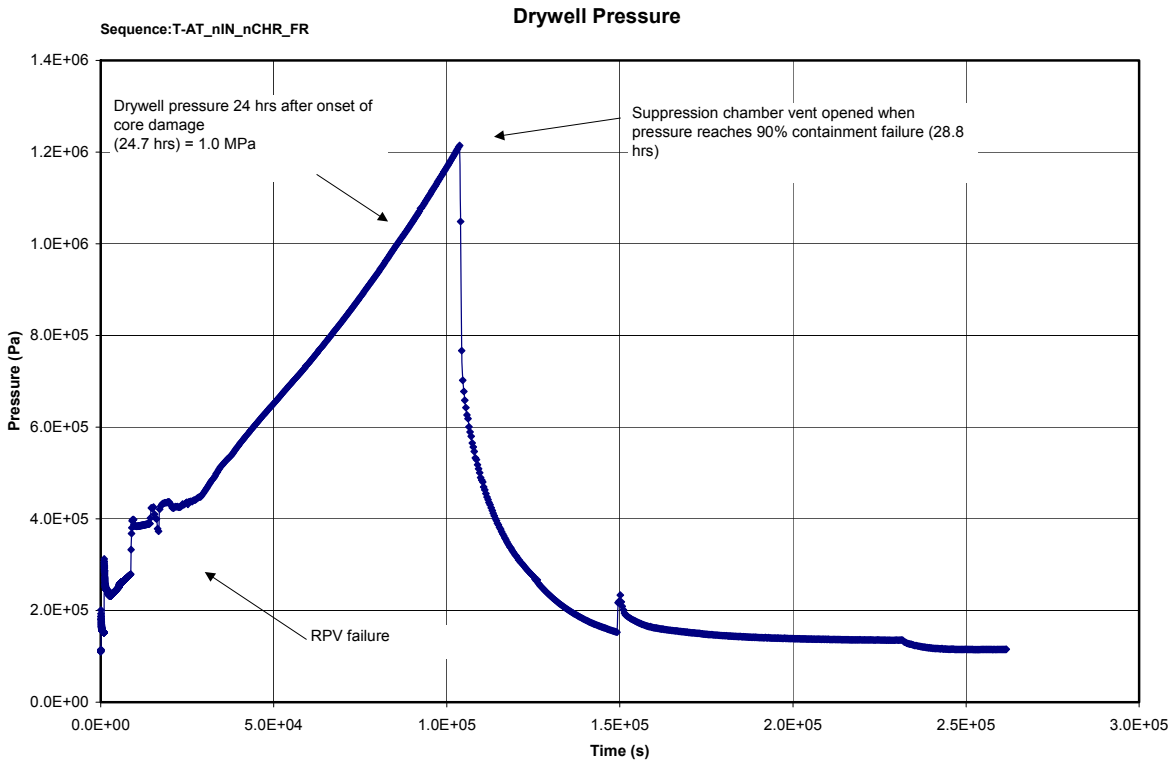

Figure 8.3-7a. T-AT_nIN_nCHR_FR: Containment Pressure vs. Time

8.4 SUMMARY

In this section, the potential for containment failure due to combustible gas generation, containment bypass and overpressurization was evaluated. In addition, the frequency of containment failure events due to the phenomenological events discussed in Section 21 (CCIW, CCID, DCH, EVE) was determined. 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 during inerted operation. Thus, containment failure by this mechanism during inerted operation is not considered further.
- Containment bypass (BYP) that results in a direct path between the containment atmosphere and environment was evaluated. A containment penetration screening evaluation indicated that there are only a few penetrations that required isolation to prevent significant offsite consequences. All potential leakage paths feature multiple containment isolation valves. Thus, the probability of the bypass failure mode is dominated by common cause hardware failures, resulting in a calculated frequency of containment bypass about three 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. Total 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 three 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 more than four 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 three 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. In addition to Level 2 scenarios, core damage Class II-b sequences from the Level 1 analysis are classified as filtered release.

However, for the total of Class I, II-b, III, and IV sequences, release category FR was evaluated to be more than four orders of magnitude below the TSL release category.

- Containment failure due to extensive core-concrete interaction (CCI) is postulated to occur due to containment overpressurization by resultant non-condensable gases. Use of the manual wetwell vent is not considered in the PRA because the suppression pool would not reduce the source term from the non-condensables.
  - Wet CCI (CCIW) events, which feature successful actuation of the GDCS deluge system but unsuccessful core cooling, are discussed in Section 21. The frequency of CCIW was found to be approximately two orders of magnitude below that of the TSL release category.
  - The dry CCI (CCID) containment failure scenarios result from a failure of GDCS deluge actuation. The frequency of CCID events was calculated to be over four orders of magnitude lower than TSL.
- The failure of containment due to direct containment heating was excluded from the baseline results based on the discussion in Section 21. However, DCH failures will be considered in a sensitivity study to be documented in Section 11 by assigning a probability of 1E-3 for all sequences in which RPV failure occurs at high pressure.
- The ex-vessel explosion (EVE) containment failure mode is a result of a high-pressure shock to the containment immediately following a steam explosion in the lower drywell. The conditions necessary for a steam explosion as the core melts through the RPV are discussed in Section 21. The total frequency of the EVE release category was calculated to be approximately two orders of magnitude below that of TSL.

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 0.921, 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.