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19.0 Response to Severe Accident Policy Statement

19.1 Purpose and Summary

19.1.1 Purpose

This chapter documents the Advanced Boiling Water Reactor (ABWR) capability in response to the NRC Policy Statement on Severe Accidents (Reference 19.1-1) and in response to the ABWR Licensing Review Bases (Reference 19.1-2) which would be used for NRC review of the ABWR Standard Plant design. Response to the CP/ML (Construction Permit/Manufacturing License) Rule (Reference 19.1-3) is provided in Appendix 19A. Resolution of applicable unresolved safety issues and generic safety issues is contained in Appendix 19B. For the most part, the ABWR capability is documented by probabilistic risk assessment techniques in Appendix 19D as outlined by Reference 19.1-2. Appendices 19E and 19F support the probabilistic risk assessment and provide the deterministic assessment of the ABWR capability to withstand a severe accident.

Appendices 19H and 19I consider the ABWR response to very large seismic events. Appendix 19K identifies appropriate additional reliability and maintenance actions that are required throughout the life of the plant so that the PRA remains an adequate basis for quantifying plant safety. Shutdown risk is addressed in Appendix 19L and 19Q. A fire protection probabilistic risk assessment is given in Appendix 19M. Detailed information about common-cause failure of multiplex equipment is provided in Appendix 19N. Appendix 19P provides information about the consideration of additional design modifications to reduce the residual risk of severe accidents. Finally, Appendix 19R contains a screening analysis for the potential for flooding to lead to core damage.

19.1.2 Summary

This analysis indicates that ABWR satisfies the severe accident related goals identified in Reference 19.1-2. The individual goals are listed in Section 19.6 where the specific manner in which the goals are satisfied is described. For the purposes of this subsection, this information is further summarized and is organized into three major areas: prevention of core damage, maintenance of containment integrity and minimizing off-site consequences.

Core damage is prevented by three divisions of the Emergency Core Cooling System (ECCS) including the Reactor Core Isolation Cooling System which can function for several hours without AC power. It also includes a reliable and proven reactor depressurization system. Feedwater and condensate pumps also provide protection against core damage. A gas turbine is also available as an alternate supply to key electrical loads. Although an AC-independent Firewater Addition System is incorporated in the design, no credit is taken for it in the calculation of core damage frequency. The calculated core damage frequency is extremely low.

Containment integrity is protected by inerting the containment volume with nitrogen and by providing a three-division heat removal system, many components of which are operated routinely and thus have very high reliability. In addition, the containment design incorporates a containment overpressure protection system. The probability of containment failure resulting from loss of heat removal is extremely small.

19.1.3 References

19.1-1 50FR32138, “Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants”, August 8, 1985.

19.1-2 Thomas E. Murley (NRC) letter to Ricardo Artigas (GE), August 7, 1987, “Advanced Boiling Water Reactor Licensing Review Bases.”

19.1-3 Title 10, Code of Federal Regulations, Part 50, Section 50.34(f).

19.2 Introduction

This section provides background and defines the objective, scope, bases and methodology of the internal events ABWR PRA provided in Appendix 19D, 19E and 19F. It explains how the analysis was conducted and the analytical bases for the methodologies employed.

19.2.1 Definitions

In this study the following definitions are used for the assessment of core damage and risk, subject to the employed methodology:

(1) Probabilistic Risk Assessment

Probabilistic Risk Assessment (PRA) is defined as the systematic identification, analysis and calculation of the probabilities and consequences of occurrence of postulated accident sequences.

(2) Frequency of Core Damage

The probability of core damage during a given year of operation is approximated by the assessed frequency of core damage. The assessed frequency of core damage per reactor year is defined as the product of the expected frequency of initiating events per year and the estimated mean probability of core damage given the initiating events.

(3) Risk

Risk to the public is expressed in terms of the assessed average consequences per reactor year which is defined as the product of the assessed frequency of release categories per reactor year and the estimated average consequences per release category summed over all release categories.

19.2.2 Objective and Scope

The objective of this PRA is to assess the probability of core damage and risk associated with the ABWR as defined in earlier chapters of Tier 2. This is accomplished by evaluating the frequency and consequence of postulated accident sequences.

The PRA analyzes the ABWR at an average site as defined by the site-related assumptions in Subsection 19E.3. Except for the shutdown risk studies, the analysis assumes that the plant is at full power prior to the initiation of an accident. The risk associated with fuel handling, storage and waste disposal accidents are judged to be insignificant and are not evaluated.

19.2.3 PRA Basis

To the extent practical, the analysis has been performed on a realistic basis. Equipment capability, success criteria, and event sequences are modeled realistically to determine, as accurately as possible, the expected course of events and conditions. Wherever possible, major conservatisms were avoided and best estimates were made of the physical effects, phenomena or probability.

19.2.3.1 Key Assumptions and Ground Rules

All of the plant system design detail which is usually required to complete a PRA was not available at the time of the study. This was recognized in Paragraphs 8.5 and 8.8 of the Licensing Review Bases where GE agreed to list key PRA assumptions (Reference 19.2-1). These assumptions are those which relate to systems which are outside the scope of Tier 2 or information about the detailed design which is not yet available. These assumptions form interface requirements or information for the COL applicant. A summary list of these assumptions is shown on Table 19.2-1 which also includes reference to the subsection in which the assumption is discussed in more detail and reference to a “confirming” subsection in which the applicant is advised to confirm the assumption.

Assumptions which were needed to conduct analytical studies are not included in the table, but are discussed in the appropriate section describing the study.

During the later stages of the completion of this PRA, the EPRI ALWR Program developed Appendix A to the Advanced Light Water Requirements document (Reference 19.2-2). This appendix describes PRA Key Assumptions and Ground Rules. For the most part, the PRA follows the assumptions and ground rules in that document. Many of the exceptions to this statement result either because work was done before certain assumptions were identified or because information from the EPRI effort was not available in time to incorporate in the PRA. These exceptions were addressed consistent with the objectives of the ALWR program during the course of the review of this chapter. Most of the remaining exceptions were the result of interactions with the NRC staff.

19.2.3.2 Failure Probability and Field Experience

Realistic component failure probabilities were extracted from domestic operating BWR experience and supplemented by generic component failure probabilities (Subsection 19D.3). The expected loss of offsite power frequency is taken from an EPRI compilation of losses of offsite power at U.S. nuclear power plants for all years through 1986 (Reference 19.2-3). Equipment maintenance or test unavailabilities used in the initial ABWR PRA submittal were taken from the GESSAR II PRA and were based upon BWR experience. In subsequent discussions with NRC regarding applicability of the GESSAR II values to ABWR, it was agreed that ABWR T&M unavailities would be increased over those of GESSAR to provide utility operational flexibility. Consequently, T&M values for RCIC, HPCFB, HPCFC, RHRA, RHRB, and RHRC were each raised to 2% in the PRA model as shown in Table 19D.3-2.

19.2.3.3 Initiating Accident Events

The expected frequency of transient events is based upon operating BWR experience and incorporates the design requirement prescribed in the Advanced Light Water Reactor Requirements Document (Reference 19.2-2) of a maximum of one anticipated transient per year which results in reactor scram. The expected manual shutdown frequency of one per year is based upon a 1985 analysis of operating plant data (Reference 19.2-4). LOCA initiation frequencies are the same as those used in the GESSAR II PRA (Reference 19.2-5) and are based upon the Reactor Safety Study (Reference 19.2-6).

19.2.3.4 System Interactions and Common Cause Failures

Five factors are considered and explicitly incorporated in the analysis of system interactions and common cause failures:

(1) Component commonality at the system level, such as a common initiating signal.

(2) Common divisional services such as common electric power buses or common service water loops.

(3) System dependency, such as ADS dependency on the operability of at least one of the five (two high pressure and three low pressure) emergency core cooling system pumps.

(4) Past experience of losing onsite or offsite power.

(5) Human errors.

19.2.3.5 Human Reliability

The probability of human error is incorporated throughout the analysis by explicit inclusion in the fault trees and event trees of Subsections 19D.6 and 19D.4, respectively. Two types of errors have been considered:

(1) Errors resulting from operator failure to act as directed by normal or emergency procedures.

(2) Errors that contribute to component failure to perform as intended because the component has not been properly calibrated or restored to its operational state as required by plant procedures. Additional discussion regarding human

error prediction and its application in the ABWR PRA is provided in Subsection 19D.7. In general, human errors are expected to be minimized by operator training and symptom-oriented emergency procedures.

Assessment of operator error in this report employs the techniques outlined in the Swain and Guttmann “Handbook of Human Reliability Analysis” (Reference 19.2-7).

19.2.3.6 Reliability Model Definitions

In the event tree analyses, all systems capable of RPV water makeup injection or containment heat removal are modeled as governed by the success criteria (Subsection 19.3.1.3.1). To simplify the analysis, all degraded core sequences are conservatively treated as “binary” core damage sequences, i.e., no partially successful operations of NSSS or BOP systems are considered. Once core damage and fission product release is predicted in an accident sequence, no coolant injection system repair or recovery is considered in the accident event trees. In certain cases, credit for system recovery has been taken in the containment event trees. If adequate RPV water level has been maintained following accident initiation, online repair or recovery of containment heat removal, water injection, and diesel generator systems are modeled.

19.2.3.7 Initial and End Point Conditions

All of the accident sequences in this analysis except those in the shutdown risk assessment are assumed to be initiated with the plant in normal steady-state operation at 100 percent power. This is consistent with the approach taken in the GESSAR II PRA (Reference 19.2-5) and the WASH-1400 Reactor Safety Study (Reference 19.2-6). Consideration has been given to startup and to low power operation in the shutdown risk assessment in Appendix 19L and 19Q.

The conditions of this analysis are the conditions applicable to a mid-life plant with a end-of-cycle core. This provides the widest and best degree of applicability to an operating ABWR. Other conditions of operation are taken as normal with nominal containment and suppression temperatures and pressures and stable external environmental conditions.

Each accident sequence analyzed is terminated in one of two conditions—core damage or safe shutdown. Sequences terminating with a damaged core are then analyzed through the containment event trees. These accident sequences in containment event trees terminate with either successful core melt arrest and therefore no radioactive release, or release to the environment. The criteria for preventing core damage are defined in Subsection 19.3.1.3.1. Recovery or mitigation of core damaging events is investigated and included where appropriate.

For those sequences terminating in safe shutdown, the success criteria as defined in Subsection 19.3.1.3.1 are met. The accident sequence is taken to a point where the reactor is in a condition of hot stable shutdown with the mode switch in shutdown, the reactor subcritical, pressures and temperatures stabilized and within limits, containment and suppression pool cooling being maintained, and vessel water level controlled. The analysis is not carried to cold shutdown due to the potentially long time involved, the low power level and slow progression of events, and the wide variety of test, maintenance, operating, shutdown, or recovery actions that could be involved.

For otherwise successful sequences where suppression pool cooling is not available and the containment overpressure protection system operates to relieve pressure, the time available for recovery actions is extended to the degree that a wide variety of recovery actions are possible. Such accident scenarios are not evaluated further.

19.2.3.8 Source Term and Core Melt Phenomenology

Source term analysis and core damage phenomenology are analyzed with the MAAP-ABWR code as noted in Reference 19.2-2. These analyses cover events and conditions depicted by the accident and containment event trees in Subsections 19D.4 and 19D.5.

19.2.4 Methodology

Methodology used in the ABWR PRA is consistent with the approach and procedures applied in the GESSAR II Probabilistic Risk Assessment, but utilizes current methods for computing the frequency of core damage and radioactive release resulting from postulated accident sequences. A summary description and illustration of the basic procedure followed as well as definitions of the major tasks of the analysis are provided in Subsection 19D.2.1.

19.2.4.1 Outline of the Analysis

As illustrated in Figure 19D.2-1, the basic analysis procedure followed consists of four major sequential tasks:

(1) Assessing the frequency of core damage,

(2) Determining the frequency of fission product release from the core and from the containment,

(3) Calculating the quantity of fission products released, and

(4) Determining the consequences of radioactive release.

Procedures for performing these tasks are diagrammed in Figures 19D.2-2 through 19D.2-5. The first two tasks provide the input necessary to determine the magnitude and consequences of release, and are discussed in Subsection 19D.2. Procedures for assessing the quantities of fission products released are discussed in Subsection 19E.2

and the process for evaluating the consequences of radioactive release are addressed in Subsection 19E.3.

19.2.4.2 Fault Tree–Event Tree Analysis

Given an initiating event, probabilities associated with the accident sequences were evaluated in fault tree and event tree logic models. Approaches taken and methods used to construct and evaluate these models are discussed in Subsection 19D.2.3.

19.2.4.3 Containment Analysis and Key Results

Probabilistic evaluation of containment failure is based on a detailed analysis of the ABWR. The containment ultimate strength under postulated severe accident conditions is evaluated in Appendix 19F. Melt progression analysis is contained in Subsection 19E.2.

The pressure capability of the concrete shell of the prototypical design at ambient temperature is 1.342 MPa for the top slab region as determined by analysis. The pressure capability is estimated to be as high as 1.342 MPa for the cylindrical wall also determined by analysis. The thermal effect of the representative severe accident temperature of 533 K (500°F) on the pressure capability of the concrete shell is expected to be insignificant.

The pressure capability of the drywell head is governed by plastic yield of the torispherical dome. The plastic yield limit pressure is evaluated using an approximate formula developed by Shield and Drucker based on the upper and lower bound theorems of limit analysis. The median limit pressure is 1.025 MPa at 533 K (500°F).

The ultimate pressure capability of the containment structure is therefore limited by the drywell head. When the limit pressure is reached, the containment is conservatively assumed to depressurize rapidly.

Leakage through fixed (mechanical and electrical) penetrations is negligible compared to leakage through large operable penetrations such as the drywell head, equipment hatches, and personnel airlocks. The leakage potential for those operable penetrations was evaluated. Very small [less than 0.0127 cm (0.005 in.)] separation displacements of the sealing surfaces at 0.722 MPa were calculated for the pressure-unseating drywell head closure and equipment hatches. No significant leakage is therefore anticipated before the capability pressure is reached. However, for the purpose of source term calculations, leakage in terms of leak areas is conservatively estimated, assuming no sealing action from degraded seals at temperature above 533 K (500°F), for pressures below the COPS rupture pressure as:

At and below the structural integrity test (SIT) pressure of 0.460 MPa (52 psig), leakage is within the design limit and the equivalent leak area is negligible.

Pressure	Leak Area	
MPa	Cm^2	In^2
0.100	0	0.00
0.412 [design]	0	0.00
0.460 [SIT]	0	0.00
0.515	7.9	1.23
0.584	17.9	2.77
0.653	27.8	4.31
0.722 [COPS setpoint]	37.7	5.85

The evaluation of the accident progression was performed using the MAAP code as described in Subsection 19E.2. MAAP is an integrated code which considers all the important aspects of a postulated severe accident, including both in-vessel and ex-vessel phenomena. In order to accurately model the important phenomena for the ABWR, revision 3.0B of MAAP was modified to create MAAP-ABWR. Additional analyses were performed using separate effects models as described in Subsection 19E.2.

Inputs of the MAAP-ABWR code are the plant parameters and event sequence information. Based on this information, MAAP-ABWR provides information about the pressure and temperature loads on the containment during a postulated severe accident as well as determining the timing and magnitude of any fission product release given the structural containment performance.

19.2.4.4 External Consequence Analysis

Evaluation of external consequences is performed using the CRAC-2 computer code. This evaluation involves:

- Amount and type of fission product release.
- Behavior of the fission products after release from the plant.
- Effects on the population exposed to the fission products.

Input data for the CRAC analysis include containment release data, weather data, demographic data, health physics data, and evacuation assumptions. Details of the CRAC code calculations are provided in Subsection 19E.3.

The calculation of accident consequences starts with the postulated release of fission products to the environment. Following the postulated release, the computer code calculates hourly dispersion, cloud depletion, and ground contamination concurrently with population evacuation. Using the resulting air and ground contamination along with population location with respect to the moving plume and dosimetric models based on the health physics data, individual radiological doses are calculated in terms of early and latent exposure for populations within a 40 km (25 mile) radius of the site. From these exposures, risk is characterized in terms of individual risk of early fatality and injury, societal risk of increased cancer incidence, and probability of dose versus distance.

19.2.4.5 Consequence Analysis Results

Previous PRAs have used a CCDF or Complimentary Cumulative Distribution Function for presentation of results and as a method for comparisons to WASH-1400 and other PRAs. Such a curve provides a highly graphical representation of potential consequences as a function of probability but is extremely dependent upon site characteristics such as evacuation planning and correlation of weather statistics to population demographics which have not been developed for a standard plant site by the NRC. Because of this lack of adequate definition of a standard plant site, the ABWR consequence evaluation has attempted to define consequences in terms of an average site as is explained in Subsection 19E.3. These results in terms of the risks of early fatality for individuals within 1.61 km (1 mile) of the site boundary and increase in risk of cancer fatality for individuals within 16 km (10 miles) of the site can be directly compared to the current NRC Safety Goals.

In addition, a CCDF for probability of dose versus distance for unsheltered stationary detectors could be produced since it is a function of release versus weather conditions only. This evaluation of dose as a function of frequency serves as a comparison to the EPRI ALWR goal of whole body dose less than 0.25 Sv at 0.8 km (1/2 mile) at 1.0E-6 probability.

19.2.5 References

19.2-1 Thomas E. Murley (NRC) letter to Ricardo Artigas (GE), “Advanced Boiling Water Reactor Licensing Review Bases”, August 7, 1987.

19.2-2 “Advanced Light Water Reactor Utility Requirements Document, Volume II, Chapter 1; Appendix A; PRA Key Assumptions and Ground Rules”, Final draft, issued 10/88 by J.C. Devine (EPRI) letter of 10/14/88 to ALWR Utility Steering Committee and ALWR contractors.

19.2-3 “Losses of Off-Site Power at U.S. Nuclear Power Plants, All Years Through 1986”, NSAC-111, Electric Power Research Institute, May 1987.

19.2-4 “Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessment”, NUREG/CR-3862, Idaho National Engineering Laboratory, May 1985.

19.2-5 “GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment”, Appendix A, 22A7007, General Electric Company, March 1982.

19.2-6 “Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants”, WASH-1400, NUREG-75/014, United States Atomic Energy Commission, October 1975.

19.2-7 A.D. Swain and H.E. Guttman, “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications”, Draft Report, April 1980, Final Report, NUREG/CR-1278, August 1983.

Table 19.2-1 Key PRA Assumptions

Summary Assumptions	Reference Subsection	Confirming Subsection
Condensate Storage Pool Volume	19E.2.1.2.1(1)	19.9.9
Battery Loading Profiles for Station Blackout	19D.4.2.9, 19E.2.1.2.1(5)	19.9.9
RCIC Room Temperature Less Than Equipment Design Temperature	19D.4.2.9, 19E.2.1.2.1(6)	19.9.9
Control Room Temperature Less Than Equipment Design Temperature	19E.2.1.2.1(7)	19.9.9
Reactor Service Water System Definition	19D.6.4.2	19.9.21
ECCS Test and Surveillance Intervals	19.3.1.6	19.9.13
Seismic Margins	19H.5	19.9.4

19.3 Internal Event Analysis

19.3.1 Frequency of Core Damage

This subsection describes the approach taken to assess accident event sequences and determine core damage frequency. Human and equipment reliability models and system descriptions provided the bases for constructing system fault trees. Results of these trees and applicable system success criteria were used to construct and evaluate accident event trees to determine the outcomes of accident sequence initiating events. The frequency of core damage was provided directly for Class I and III events by outcomes of the accident event trees. Determination of what fractions of Class II and IV events led to core damage required additional processing through containment event trees, discussed in Subsection 19.3.2, which were used to determine the final outcomes of those sequences involving loss of heat removal or ATWS, Class II and IV events. This approach to assessing core damage frequency and fission product releases is schematically illustrated in Figure 19.3-1.

19.3.1.1 Accident Initiators

This subsection describes the accident sequence initiating events documented in Subsection 19D.3. These initiators are separated into two general groups, transients and loss of coolant accidents (LOCAs). Table 19.3-1 provides a summary of these initiators and their expected frequency of occurrence.

The total frequency of transient initiators used in these evaluations is based upon a 1985 analysis of operating plant data (Reference 19.3-1). The frequency of transients is a design requirement prescribed in the Advanced Light Water Reactor (ALWR) Requirements Document (Reference 19.3-2). Apportioning of the expected transient frequency by initiating event was done on the basis of historical electrical grid and BWR performance data as described in Subsection 19D.3.

LOCA initiation frequencies are based upon the Reactor Safety Study, WASH–1400 (Reference 19.3-4). After reviewing these values and their bases, their use in the ABWR PRA was judged appropriate.

19.3.1.2 Equipment Reliability and Availability

Not part of DCD (Refer to SSAR)

19.3.1.3 Accident Sequence Analysis

19.3.1.3.1 Success Criteria

This subsection provides a discussion of the ABWR success criteria employed in this analysis. These criteria govern the construction of accident event trees which are used to model all accident sequences. The criteria are defined for both non–ATWS events and ATWS events.

(1) Success Criteria for Non–ATWS Events

The success criteria in this subsection are based on best–estimate predictions using the GE licensing approved computer models. Several BWR generic studies have determined that one motor–driven ECCS pump has sufficient reflooding flow to provide adequate core cooling.

(a) Core Cooling

A peak cladding temperature (PCT) of 1478 K (2200°F) was chosen as the criteria in determining the success of a coolant injection system. The resultant ABWR core cooling success criteria to prevent initial core damage for transient and Loss Of Coolant Accident (LOCA) events with scram initiated from the Reactor Protection System (RPS) are given in Table 19.3-2.

The high pressure core flooder (HPCF) pumps, which are the Emergency Core Cooling System (ECCS) high pressure pumps, have a large capacity for making up lost inventory. Following any LOCA or transient event, either one of the HPCF pumps can reestablish the water level and maintain the PCT below 1478 K (2200°F).

The residual heat removal (RHR) pumps, which can be used in the ECCS low pressure core flooding mode, are also large capacity pumps. Following a large LOCA, the Reactor Pressure Vessel (RPV) depressurizes sufficiently and any one of the three RHR pumps can reestablish the water level and maintain the PCT below 1478 K (2200°F). For small or medium LOCA, or transient events, RPV depressurization using at least three (3) depressurization valves is needed to permit timely use of an RHR pump.

The reactor core isolation cooling (RCIC) pump is also part of the ECCS. It is turbine driven and also provides high pressure water makeup to the RPV as long as steam is available at pressures greater than 0.345 MPa differential for the RCIC turbine. The RCIC requires high pressure steam from the vessel to drive the turbine; therefore, its ability to maintain adequate core cooling by itself is limited to the small liquid LOCA or transient (excluding IORV) events.

The capacity of non-safety-related systems, such as the feedwater and condensate pumps, has been estimated based on the ECCS performance analyses. Non-safety-related systems which contribute to a successful conclusion of the event have been included in the success criteria. The Control Rod Drive (CRD) pumps which have limited capacity have not been included in the success criteria.

The condensate pumps are motor-driven pumps and their use depends on the RPV pressure and the availability of makeup water and electrical power. These pumps have higher shut–off heads than the RHR pumps, but still require depressurization before they can be used for core cooling. The source of makeup water for these pumps are the main condenser hotwell and the condensate storage tank. Sufficient makeup water is available to enable these pumps to maintain adequate core cooling for all events except large or medium liquid LOCAs.

A motor driven feedwater pump is combined in series with a condensate pump in order to provide a higher pressure system. Therefore, this option also depends on the availability of makeup water and electrical power. Sufficient makeup water is available to enable this series of pumps to maintain adequate core cooling for the small steam LOCA and transient events.

The fire protection system has two pumps which take suction from the firewater tanks and inject into the RPV through an RHR line. One pump is driven by an electric motor which requires AC power. The other is driven directly by a diesel engine. Once the reactor system has been depressurized, either pump can provide enough makeup water to restore and maintain the RPV water level following any transient (including IORV) event. The analysis to support this conclusion assumes a full ADS blowdown begins within 15 minutes after the vessel water level has reached the level 1 setpoint. The subsequent reactor system depressurization allows injection from the fire protection system about 7 minutes after the start of the blowdown. The ability of the fire protection system to mitigate the consequences of LOCA events is conservatively ignored. For more information about the fire protection system refer to Subsection 5.4.7.

It is conservative to use the 1204°C (2200 °F) PCT licensing limit as an acceptance criteria for the success criteria since tests have been performed which show that the core will remain in a coolable geometry with temperatures as high as 1482°C (2700°F).

A review of Table 19.3-2 shows that, for success, the inventory threatening events require the flow equivalent of only 1 RHR/LPFL or 1 HPCF pump available for large break cases and only 1 HPCF or 1 RHR/LPFL + 3 ADS available for small break cases. The resulting PCTs for the large break cases and transients were between 482°C (900°F) and 593°C (1100°F). For the small break cases with the flow equivalent of only 1 HPCF available the resulting PCTs were less than 538°C and with 1 RHR/LPFL + 3 ADS available the maximum PCT was 982°C (1800°F).

Subsection 6.3.3.7.8 identifies the input parameters that significantly impact the LOCA results. If the above analyses were reanalyzed with these conservative input parameters, it is estimated that only the resulting PCTs for the small break cases with 1 RHR/LPFL + 3 ADS available are above 982°C (1800°F). For these cases the PCT is estimated to be about 1260°C (2300°F). However, even for these conservative LOCA calculations all the PCTs are less than 1482°C (2700°F) which is still acceptable and most LOCA cases and transients are much less than 1482°C (2700°F). Therefore, there is no need to include an uncertainty analysis in the generation of the success criteria.

(b) Containment Heat Removal

Following the success of the core cooling function, heat must be removed from the containment. Containment heat removal is considered a success if the containment pressure is kept below the pressure at which loss of containment integrity is estimated to occur (Appendix 19F). Successful containment heat removal can be achieved by using the RHR System or, depending on the circumstances as defined in Table 19.3-2, the normal heat removal path or the CUW System. The resultant ABWR longterm heat removal success criteria to prevent initial core damage for transient and Loss of Coolant Accident (LOCA) events with RPS scram are given in Table 19.3-2.

The RHR has four major modes of operation and heat is removed from the containment in each of these modes. During the core cooling mode which is initiated automatically, the RHR heat exchanger is in the loop and the heat removal process is established. If core cooling is accomplished without the use of an RHR System, and the suppression pool begins overheating, the suppression pool cooling mode of the RHR will be automatically or manually initiated by the operator. Once initiated, an RHR System will begin removing heat from the containment and eventually terminate the pool heatup.

The normal heat removal path is through the main condenser. This path can be used under transient and accident conditions when the MSIVs or the main steam drainlines are open (or re-opened if closed earlier during the event) and the condensate can be removed. If the RPV is depressurized, the main steam drainline option is not viable since it will not pass enough steam to remove the decay heat energy.

The Reactor Water Cleanup (CUW) System is capable of removing the energy due to decay heat (at greater then 4 hours after scram) at high RPV pressures if the return water bypasses the regenerative heat exchanger. Therefore, its ability to maintain adequate longterm cooling is also limited to the small liquid LOCA or transient events where the reactor system can be maintained at high pressure and temperature producing a large temperature differential across the CUW non–regenerative heat exchanger.

(c) RPV Pressure Relief

A pressure of 150% of the reactor-coolant pressure-boundary design pressure [8.719 MPa], the faulted limit, was chosen as the criterion in determining the success of the system to prevent overpressure failure of the reactor primary system during moderately frequent events. The turbine bypass system and safety/relief valves represent the two success paths in the ABWR overpressure protection success criteria for the events given in Table 19.3-2.

For events resulting in isolation of the primary system, only the safety/relief valves are available to minimize the reactor system pressure rise. The most severe ABWR transient with scram for these events is a closure of all MSIVs. For this case, six of the eighteen safety relief valves are required to open to limit the peak primary system pressure to below 13.029 MPa.

For events which do not result in isolation of the primary system, both the turbine bypass valves and the safety/relief valves are available to minimize the reactor system pressure rise. The most severe ABWR transient with scram for these events is a turbine trip at full power with at least some turbine bypass capability available. For this bounding case, the acceptable combinations of turbine bypass and relief valves available are given in Table 19.3-2.

(2) ATWS Success Criteria

For anticipated transient without scram (ATWS) events, the success criteria are defined in terms of the system mitigation capability to meet the following criteria:

(a) The reactor pressure vessel must be protected from overpressurization.

(b) A coolable core geometry must be maintained.

(c) Long term heat removal must be adequate to preserve containment integrity.

For a postulated system failure, an ATWS event is considered successful (or acceptable) if the resulting ATWS consequences meet these criteria.

An extensive assessment of BWR mitigation of ATWS conducted by General Electric has been reported in NEDE–24222, “Assessment of BWR Mitigation of ATWS”. These success criteria are based upon General Electric’s experience with ATWS analyses, and a consideration of ABWR features.

Three classes of ATWS may be considered. The first class when Reactor Protection System (RPS) scram does not operate but the rods are inserted by alternate means. Electric rod run in is an automatic function which is initiated by the RPS. However, because the rods are inserted more slowly than with scram, the successful operation of this method of reactivity insertion is considered an ATWS. The rods may also be inserted with alternate rod insertion (ARI). High vessel pressure or low water level initiate ARI. Additionally, the operator may manually insert the rods.

The second class of ATWS sequences are those for which there is no rod insertion, but the reactor is brought to a subcritical condition by standby liquid control (SLC) injection. The requirements for this system are discussed below.

The third class of ATWS events has neither rod insertion nor SLC injection. Work performed by General Electric and Idaho National Engineering Laboratory (Reference 19.3-7) has shown that a single high pressure system can maintain adequate core cooling. Maintaining the containment pressure below service level C and the containment overpressure protection (COPS) setpoint is the appropriate success criteria for the containment heat removal system in this highly degraded scenario. If three heat exchangers are available in this case, the containment pressure can be maintained below these levels. In addition, the pressure can be maintained below service level C if the COPS actuates, thus COPS is also an adequate method of containment cooling. However, because the probability of ATWS is very low, no credit is taken for mitigation of this class in the internal events analysis.

The following discussion determines the system and operator requirements necessary to ensure adequate RPV pressure relief, core cooling, and containment cooling during ATWS.

(a) RPV Pressure Relief

As for the non–ATWS transients, a pressure of 150% of the reactor–coolant pressure–boundary design pressure, the faulted limit, was chosen as the criteria in determining the success of the system to prevent over pressure failure of the reactor primary system during moderately frequent events. The turbine bypass system and safety/relief valves represent the two success paths for the ABWR. Overpressure protection success criteria for ATWS events are included in Table 19.3-3.

For events resulting in isolation of the primary system, only the safety/relief valves are available to minimize the reactor system pressure rise. The most severe ABWR ATWS event for overpressure protection is a closure of all main steam isolation valves (MSIVs). For this case fifteen of the eighteen safety relief valves are required to open to limit the peak primary system pressure to below 13.029 MPa.

For events which do not result in isolation of the primary system both the turbine bypass valves and the safety/relief valves are available to minimize the reactor system pressure rise. The most severe ABWR transient for these events is a turbine trip at full power, but in this case some turbine bypass capability is available. For this bounding case, the acceptable combinations of turbine bypass and relief valves are given in Table 19.3-3.

(b) Core Cooling

Adequate core cooling is necessary to prevent fuel failure. Assuming some form of reactivity control is operated, an ATWS event does not substantially differ from its associated transient. Any one high pressure system is adequate to provide core cooling for all initiators except inadvertent opening of a relief valve (IORV). For the case of IORV, the vessel will continually depressurize through the stuck open valve, therefore, at least one High Pressure Core Flooder (HPCF) System must inject to the vessel. These requirements are summarized in Table 19.3-3.

Additionally, for ATWS events where the rods are eventually inserted, the low pressure systems may be used for core cooling. For these events, the core cooling success criteria are identical to those given in Table 19.3-2.

(c) Containment Integrity

Assuming the success of the core cooling function, heat must be removed from the containment. Containment heat removal is considered a success if the containment pressure is kept below the pressure at which loss of the containment integrity is estimated to occur.

Successful containment heat removal can be achieved by using the Residual Heat Removal (RHR) System, or, depending on the circumstances, the normal heat removal path.

For ATWS events, the ability to maintain containment integrity is dependent not only on the availability of the RHR System, but also on the rate at which the reactor is brought to shutdown conditions. Therefore, it is necessary to consider each type of ATWS initiating transient separately in order to determine the energy being passed to the pool as a function of time.

The speed at which poison is injected determines the energy which is passed to the suppression pool. This energy was estimated using GE's realistic best estimate computer code, TRAC, and conservative assumptions for the rate of power reduction with poison injection. Analysis shows that once the reactor has been shut down one RHR loop is capable of maintaining containment integrity.

In order to determine the maximum time available for poison injection, four events are considered below. The required timing for the events shown below was determined assuming that containment cooling is not initiated until the reactor is shut down. Therefore, only one RHR loop must operate. However, for those ATWS initiators for which isolation is not an immediate consequence of the initiator, the main condenser was assumed to be available. Also assumed in the analysis below is that both HPCF Systems and the Reactor Core Isolation Cooling (RCIC) System will automatically initiate if their initiation conditions are reached. This has the effect of predicting the maximum power generation, and therefore, leads to shorter operator action times than a case in which core cooling injection is limited. The results of these considerations is summarized in Table 19.3-3.

(i) MSIV Closure

The sequence of events and anticipated operator actions for an MSIV Closure with failure of automatic rod insertion is as follows. Upon closure of the MSIVs, the reactor pressure will increase sharply and the safety relief valves (SRVs) will open. This will cause 4 of the reactor internal pumps (RIPs) to trip. The water level will rise to level 8, causing a feedwater trip. The operator will observe the failure to insert rods based on the rod position switches, and attempt to insert the rods manually approximately 30 seconds into the transient. If the rods are inserted, then the sequence is successfully terminated.

If the rods are not inserted, then the water level will begin to fall. When the water level reaches level 2, in about 30 seconds, the RCIC will initiate and the remaining 6 RIPs will trip. The water level will continue to decrease, initiating the two HPCF Systems about one minute into the transient. ADS initiation is automatically inhibited due to the ATWS event. Assuming all three water injection systems come on, the power level will drop to approximately 20% of rated. The operator is then instructed to allow the water level to drop by terminating and preventing all injection into the RPV except from the CRD until at least one of the following conditions exist:

- The reactor power drops below the APRM downscale trip.
- The water level in the vessel reaches the top of active fuel.
- All the SRVs are closed and the drywell pressure remains below 0.014 MPaG.

The operator will initiate the SLC System based upon a suppression pool temperature versus power criterion. Assuming that one SLC will operate, approximately 10 minutes are allowed for him to complete this procedure. In order to bring the transient to a complete stop, the RHR System must be initiated in suppression pool cooling or drywell spray mode. A minimum of 30 minutes is allowed for this action.

(ii) Turbine Trip

The sequence of events for the turbine trip is quite different from that for an MSIV closure. When the turbine trips, the turbine bypass valves will open, permitting approximately one third of rated steam flow to pass through them. The remainder of the steam generated will pass through the SRVs into the suppression pool. The initial pressure rise will cause 4 of the RIPs to trip, and the core flow will drop to about 80% of rated. The feedwater pump will attempt to maintain normal water level. The power will equalize at approximately 80% of the rated value. Therefore, about 50% of rated power will be directed to the suppression pool. The operator will be aware of ATWS conditions within 30 seconds. The operator is instructed to attempt to take actions to shut down the reactor. These actions include:

- Placing the mode switch in Shutdown.

- Tripping all RIPs.
- Inserting rods by various means.

If these actions fail to shut down the reactor, the feedwater will run back and prevent RPV injection in preparation for SLC injection. The water level will fall to level 2, and the remaining RIPs will trip if not previously tripped manually. This will cause the power level to drop to about 20%. Both feedwater runback and SLC initiation are assumed to occur within 10 minutes of the initiation of the event. Finally, in order to maintain the pool temperature, the RHR System must be initiated within 30 minutes.

(iii) IORV

This sequence is quite different from the remainder of the sequences considered here. The failure of rod insertion for an IORV event is based on a manual operation taken when the suppression pool temperature limit is reached. It is assumed that one SRV is stuck open. This allows 4.5% of the power to flow to the pool. The turbine and the main condenser are available to remove the remaining energy from the plant. For this sequence, the operator has the capability of shutting down the plant in an orderly fashion.

The RHR will be initiated to attempt to keep the suppression pool cool. The operator will attempt to insert the rods. If that fails, the RIPs will trip, the feedwater will be run back and SLC will be initiated. Even if RHR is not operating, the power being directed to the pool is very low since the bypass valves can be used to divert steam flow to the main condenser, and there will be ample time for SLC initiation.

(iv) Loss of Offsite Power

The loss of offsite power (LOOP) initiated ATWS is similar to the MSIV closure transient. The water level will fall, generating a scram signal. if the water level drops below level 1.5, the MSIVs will close. The feedwater and RIPs will trip. Thus, the power level in the reactor will drop even more quickly than will the MSIV closure case. As for the case of an MSIV closure, the operator will first attempt to insert the rods manually. If this fails, the ADS actuation must be inhibited and the SLC injection must begin within about 10 minutes, and RHR must be started in 30 minutes.

19.3.1.3.2 Accident Sequence Event Trees

This subsection describes construction of event trees used in the analysis to determine accident sequence frequencies. These sequences lead to core damage, safe reactor shutdown, or to intermediate states which require additional treatment in the containment event trees of Subsection 19D.5 to establish final core states. Separate trees have been developed, as shown in Figures 19D.4–1 through 19D.4–15, for each of the initiating events considered. All accident event tree sequences other than those leading to safe reactor shutdown are further treated in the containment event trees of Subsection 19D.5 to determine frequencies of radiation release to the environment.

For purposes of illustration, consider Figure 19.3-2, the event tree for the reactor shutdown initiating event. The initiating event frequency is given as the first branch of the far left column of the tree. The initiating event name and symbol are provided at the top of the column. The tree is developed by identifying the system functions required, in the approximate chronological order of occurrence, for successful reactor shutdown. Success and failure states of each system function are represented by branches in the tree, where the upper branch represents success and the lower branch failure. If a prior system function leads directly to success or failure in the accident sequence, analysis of the remaining system functions is unnecessary.

Information given at the top of the column for each system function consists of an abbreviated definition of success and the symbol for conditional failure probability. The value for each failure probability is shown on the lower branch. Each accident sequence terminates in the column labeled “FREQ” which contains the frequency of occurrence of that sequence. The final column contains the classification of each sequence; either successful termination (OK), core damage, or a sequence which is developed further in another accident tree or transferred to the appropriate containment event tree.

Accident event trees developed in this analysis contain branches which address the primary safety functions of reactivity control, reactor pressure control, core cooling, and containment heat removal. These four functions are considered in all event trees except the reactor shutdown event in which reactivity control is, by definition, provided by event initiation. Success criteria provide the bases for defining minimum combinations of those functions required to bring the plant to a safe stable shutdown condition. Success criteria are presented and discussed in Subsection 19. 3.1.3.1.

19.3.1.3.3 Classification of Accident Classes

Accident sequences identified and evaluated in the event trees were examined and classified on the basis of similarity of timing, potential for fission product release, and containment response. Accident sequence classes used in the analysis are described in Subsection 19D.5.

19.3.1.4 Frequency of Core Damage

Of the ten accident classes defined in Table 19.3-4, eight lead directly to core damage. The remaining two classes can lead initially to loss of the containment heat removal function and subsequently, possible core damage. For these latter sequences, outcomes in event trees documented in Subsection 19D.4 do not necessarily lead to core damage. Detailed analyses of the frequency of core damage following loss of containment integrity is presented in Subsection 19D.5 where it is addressed relative to containment release paths. That analysis is based on the accident event tree outcomes, containment overpressure capability discussed in Subsection 19.3.2, and on the containment event trees of Subsection 19D.5.

Table 19.3-4 summarizes the frequencies of core damage as a function of accident class. As explained above, eight of the ten frequencies (all Class I and III events) are obtained directly from the outcomes of the accident sequence event trees of Subsection 19D.4. Frequencies of the Class II and IV events, where containment heat removal is lost, were determined by processing the loss of heat removal outcomes of the accident event trees through the containment event trees to determine the probability of failure to prevent core damage for these events. The bypass study in Subsection 19E.2.3.3 concluded that the core damage frequency and risk associated with Class V events is negligible. Therefore, the Class V frequencies are not given in the table.

Table 19.3-5 provides a different perspective by showing the breakdown of core damage frequency by initiating event. Expected frequencies are given both in terms of events per year and percent of total. Loss of offsite power and station blackout are the dominant contributors to expected core damage. In the loss of offsite power sequence, dependence on diesel generators alone for electric power results in lower availability and reliability of the emergency core cooling and heat removal systems. This situation is aggravated in the station blackout sequences since all diesel generators are also lost, and adequate core cooling is totally dependent on successful performance of the RCIC System.

19.3.1.5 Results in Perspective

The estimated core damage frequencies are extremely low. It is impossible to calculate such low numbers with a high degree of confidence using the PRA models developed here. For example, a number of potential common cause failures of components such as similar pumps and valves have not been included in the fault tree models, on the expectation that such failures are negligible contributors to overall core damage frequency.

In addition, although the ABWR PRA has addressed those initiating events and event sequences identified as potentially significant contributors to core damage risk, it is impossible to be certain that all initiators and event sequences leading to core damage at such low levels of expected core damage frequency have been identified.

19.3.1.6 Positions and Assumptions Implicit in the Analysis

A number of positions were taken and assumptions made at the outset of the internal events analysis which affect the results obtained and conclusions drawn. Included among these was the decision to apply GESSAR II information to the extent possible to the ABWR PRA. As a result, ABWR ECCS test and surveillance intervals were assumed the same as GESSAR II. In addition, estimated unavailabilities of systems not modeled by fault trees, a number of component failure rates, and certain human error probabilities were taken from GESSAR II and used in the ABWR PRA when judged applicable.

No credit was taken for the Firewater Addition System in the level one analysis for several reasons. First, the core damage frequency is very low as discussed in Subsection 19.3.1.5. Second, the containment event tree analysis is significantly simplified if no credit is taken for the firewater system in the accident event tree. And finally, a relatively short period of time is available for the operator to take the necessary actions.

19.3.2 Frequency of Radioactive Release

19.3.2.1 Overview

Accident event trees developed for each of the accident initiators are described as part of the core damage frequency evaluation. These trees model the event progression for the various accident initiators, and provide the classification and frequency of accident sequences. In these event trees, the sequences which are terminated safely without core damage are designated as “OK”. The event sequences which are not successfully terminated could either directly lead to core damage or in some cases could lead to containment structural failure which in turn could lead to core damage. These event sequences are “binned” into various accident classes depending upon the expected event progression, timing and mode of containment failure and the amount of fission product release to the environment.

There are five basic classes (I through V), and a total of ten classes including subclasses such as IA, IB, IC, etc. A Class IA event, for example, is a transient event with loss of high pressure water makeup systems followed by a failure to depressurize the reactor.

The accident event progression for each of the accident classes was analyzed using the MAAP code (Modular Accident Analysis Program). A detailed description of the analysis is given in Subsection 19E.2. For each accident class, these analyses provide the time of RPV failure, containment pressure and time history, and the time at which radioactivity is released to the environment. Also evaluated are the amount of fission products released to the environment.

The event progressions for each of the ten subclasses of events are modeled in the containment event trees (CETs). The CETs model recovery actions which could prevent core damage or arrest core damage if already initiated. Where recovery actions are unsuccessful, the CETs model core melt leading to reactor vessel rupture, containment structural failure and fission product release to the environment. The CET models are based on core–melt progression analysis discussed in Subsection 19E.2. The mode and location of containment failure is modeled based on a study of the containment capability discussed in Appendix 19F.

There is one CET for each of the ten accident classes. The end states of CETs are either states with insignificant or no release (i.e., core damage prevented or core melt arrested), or states with a release path to the environment resulting from the failure of the containment. Associated with each release path in each of the containment event trees, is a frequency of occurrence and a magnitude of fission product release. The frequencies are calculated by the CETs, and the fission product releases are evaluated using the fission product transport analysis discussed in Subsection 19E.2. The numerous release paths can be consolidated or “binned” into release categories by grouping them based on the expected amount of fission product release to the environment.

The consolidated release categories and the associated frequencies are used as input to the consequence analysis discussed in Subsection 19E.3.

19.3.2.2 Accident Classes

Accident event trees developed for each of the accident initiators are described as part of the core damage frequency evaluation. The end states of these accident event trees are “binned” (grouped) into five basic accident classes based on similarities in the subsequent core melt event progression and the containment response. The key factors that influence the definition of the accident classes are as follows:

- Type of initiating event (transient, LOCA, etc.).
- Relative times of core melt and containment failure.
- Whether suppression pool is bypassed.

The type of initiating event is significant because it determines the speed of the event progression. For instance, when no core cooling is available, core melt occurs faster for the LOCA event than for the transient event because of the faster depletion of the coolant inventory.

The relative times of core melt and containment structural failure are important because if core melt occurs first, the time between core melt and containment structural failure is available for decay and removal of radioactive material released in the accident. This time is also available for enabling the operator to recover failed water makeup systems in order to get water on top of the molten core or to regain suppression pool cooling if it had been lost.

The significance of the suppression pool bypass event is that, following core melt, the fission products are released to the environment without the beneficial effects of passing through the suppression pool.

Five basic accident classes, I through V, have been identified. A brief summary of these five classes is provided below:

Class I:	Transient followed by loss of core cooling.
Class II:	Transient with successful core cooling followed by loss of containment heat removal systems.
Class III:	Loss of coolant accident followed by loss of core cooling.
Class IV:	Anticipated transients without scram (ATWS) events with no mitigation.
Class V:	Events in which suppression pool is bypassed (e.g., LOCA outside the containment).

Some of the accident classes are further divided into subclasses in order to facilitate more accurate modeling of the event progression in the containment event trees (CETs). A brief summary of the Class I and III subclasses is provided below. The remaining classes were not subdivided.

Class IA:	Transient, followed by failure of high pressure core cooling system coupled with failure to depressurize the reactor.
Class IB:	Events are broken into three categories:
Class IB-1:	Station Blackout (SBO) event with RCIC failure, onsite power is recovered in eight hours.
Class IB-2:	SBO event, RCIC is available for operation and keeps the core cooled for eight hours at the end of which RCIC is assumed to fail. The suppression pool continues to heat up during RCIC operation.
Class IB-3:	SBO event similar to Class IB-1 but the onsite power is not recovered in eight hours.
Class IC:	ATWS, followed by failure of boron injection and core cooling.

Class ID: Transients followed by loss of both high and low pressure core cooling systems, reactor at low pressure.

Class IIIA: Similar to Class IA but for a LOCA initiator.

Class IIID: Similar to Class ID but for a LOCA initiator.

19.3.2.3 Accident Event Progression

The accident event progression for the above accident classes were analyzed using the MAAP-ABWR code. A detailed description of the analyses is given in Subsection 19E.2.

A typical core melt sequence may include the following 8 steps:

(1) Core melt in the RPV.

(2) RPV failure.

(3) Discharge of corium (i.e., a mixture of molten metal and core material) and lower plenum water in the lower drywell (LDW) area.

(4) Evaporation of water in LDW producing steam.

(5) Core-concrete interaction producing non-condensible gases.

(6) Drywell heat up causing actuation of a passive flooder system causing suppression pool water to flow to the lower drywell, quenching the corium and terminating the interaction with concrete.

(7) Containment leakage (high containment temperature and pressure discussed in Subsection 19.3.2.4).

(8) Containment overpressure leading to fission product release (see Subsection 19.3.2.4 for limiting pressure).

Each accident sequence is unique with respect to timing of the above events, rate of containment pressurization and pressure rise and the order in which they occur.

The “Passive Mitigation” discussed above in (6) is a unique feature of ABWR containment configuration which allows for core melt arrest without the use of active components.

For purposes of illustration, the timing of a typical Class ID sequence is given below:

RPV failure:	1.8 hour
Water in LDW boils off:	2.7 hours
Passive Mitigation	5.4 hours
Rupture disk opens:	20.2 hours

Class II sequences which involve successful core cooling but no containment heat removal are significantly different. Rupture disk opening occurs in about 20 hours following which core cooling continues for a long time (> 100 hours) before it is necessary to replenish suppression pool water inventory

Associated with each accident sequence is the amount of fission products released to the environment. This depends upon factors such as the amount of release through the suppression pool prior to RPV failure, timing and location of containment structural failure, core decay heat at the time of accident. The fission products released are documented in Subsection 19E.2.

19.3.2.4 Containment Structural Capability

The ABWR containment design pressure is 0.412 MPa. Past stress analyses preformed for other PRAs have shown that the containments are capable of withstanding much higher pressure (typically 2 to 3 times the design pressure). A discussion of the ABWR containment capability is provided in Appendix 19F. The containment structural capability is limited by that of the drywell head. The drywell pressure capability depends upon the containment temperature. At 533 K (500°F), (which is a typical temperature for most accident sequences), the drywell median ultimate strength is evaluated to be 1.025 MPa.

19.3.2.5 Containment Structural Failure Modes and Location

In Appendix 19F it is concluded that when the containment is pressurized, the most likely mode of failure is the plastic yield of the drywell torispherical dome. Containment rupture which impairs the ability of the containment to provide structural support is not judged to be a credible mode of failure. Containment leakage at pressures below the failure pressure is judged to be not significant (i.e., not sufficient to depressurize the containment). However, at high temperatures [i.e., >533 K (500°F)] there is a potential for degradation of seals in the large operable penetrations such as the equipment hatch and personnel air locks. A conservative evaluation shows that leakage is expected to occur only when the containment pressure exceeds 0.460 MPa.

The following failure modes are explicitly modeled in the CETs.

(1) Containment leakage occurs when the temperature exceeds 533 K (500°F) and pressure exceeds 0.460 MPa.

(2) Drywell head failure occurs when the containment pressure exceeds 1.025 MPa and temperature is below 533 K (500°F).

(3) Containment high temperature failure occurs when the containment experiences a very high temperature [>811 K (1000°F)].

There are a number of containment structural failure modes which have been shown to be negligible contributors to plant risk in past PRAs and, therefore, are not included in the ABWR CETs. Examples of such failure modes are steam explosion, basemat penetration, pressure vessel rupture leading to containment failure, etc. Hydrogen detonation is not modeled because the ABWR containment is inerted and hydrogen detonations are, therefore, judged to be negligible contributors to ABWR risk.

19.3.2.6 Suppression Pool Bypass Events

The ABWR suppression pool plays a key role in reducing the fission products released to the environment following a severe accident. Fission products released through the suppression pool benefit from the “scrubbing” action which traps most of the fission products such as cesium iodide. However, if the accident sequence results in bypass of the suppression pool, the magnitude of the associated release could be a factor of 100-10000 more than that for a sequence which discharges through the suppression pool.

It is, therefore, important to study the suppression pool bypass paths and evaluate its impact on the PRA results. There are a number of ways the ABWR suppression pool can be bypassed. Most of them involve some combination of pipe and valve failures, or leakage through closed isolation valves. Examples of suppression pool bypass paths are as follows:

(1) Failure of MSIVs and turbine bypass valves

(2) Failure of MSIVs and main steamline break outside containment

(3) Wetwell-drywell vacuum breaker failure.

A separate study of these suppression pool bypass paths was conducted and it was concluded that the contribution of these paths to ABWR risk was small. With the exception of the wetwell-drywell vacuum breakers not including these paths in the CET, models explicitly will affect the risk results by a small percentage of the total risk. Only the vacuum breakers were modeled in the CETs. However, CETs also model the suppression pool bypass paths resulting from the structural failure of the containment.

For instance, the three containment failure modes discussed in Subsection 19.3.2.5 and modeled in the CETs (leakage, overpressure and high temperature failure) all lead to suppression pool bypass.

The suppression pool bypass study is documented in Subsection 19E.2.3.3.

19.3.2.7 Recovery of Failed Systems

Recovery of failed systems, onsite and offsite power has been modeled in the CETs.

System recovery probabilities are generally calculated using the exponential recovery formula:

$$P_f = \text{Exponential } (-T/MTTR)$$

where:

P_f	=	Probability of failure to recover
T	=	Available repair time
MTTR	=	Mean time to repair

For accident sequences in which core melt had proceeded to the point of RPV failure, it was judged that high radiation might make it difficult to carry out some repair activities. For events involving station blackout, the recovery data was based on historical data.

The time available for repairing or recovering each system was determined by the time within which the system had to be operating to prevent the occurrence of failure (core recovery, containment overpressure, etc.). The available repair times were obtained based on the core melt progression analysis discussed in Subsection 19E.2.

19.3.2.8 Core Melt Arrest Success Criteria

The accident event progression analysis described in Subsection 19.3.2.3 shows that core melt can be arrested by quenching the molten corium. The core melt arrest can take place within the RPV in the early stages of the accident if core cooling can be recovered in time. If this does not occur, then the core melt proceeds to RPV failure and the molten corium is discharged into the LDW. The core melt can be arrested in the containment if the core cooling is recovered before the containment experiences structural failure due to overpressure, leakage or high temperature. In many sequences, the core melt is arrested by passive flooder system operation. In Class IA accident sequences, (i.e., loss of high pressure core cooling system coupled with failure to depressurize reactor), the RPV failure depressurizes the reactor making the low pressure core cooling systems available for arresting core melt in the containment.

In addition to core melt arrest, one containment heat removal system must also be re-established to prevent overpressurizing the containment.

The core melt arrest success criteria is discussed in detail in Subsection 19D.5.8.

19.3.2.9 Containment Overpressure Protection

Sensitivity studies in Subsection 19E.2.8.1.4 were conducted to determine the value of providing a containment overpressure relief feature. The results show a substantial reduction in offsite dose.

19.3.2.10 Containment Release Categories

The amount of radioactive release to the environment depends upon a number of factors such as the timing of containment failure and the location of containment failure. Ideally, there is a specific radioactive release associated with each outcome of the containment event trees. However, evaluating the source terms for each event tree output is very time consuming. Therefore, the releases with similar characteristics are grouped ("binned") together to define release categories as discussed in Subsection 19E.2.2.

19.3.2.11 Containment Event Trees

The results of the accident event trees were grouped into ten accident classes. In general, one CET was developed for each of the accident classes. However, two of the accident classes IC and IV, had negligibly low occurrence frequencies and CETs were not developed for these accident classes. Class IC event frequencies were added to the class IA frequencies and the class IV frequencies were assumed to result directly in core damage and early containment failure.

The CETs model recovery actions and containment failure modes. The end states of CETs are either states with insignificant or no release (i.e., core damage prevented or core melt arrested), or states with a release path to the environment resulting from the failure of the containment. The end states are assigned a source term category grouping which depends on key containment performance criteria as shown in Figure 19D.5-3. These results are then binned into release categories as discussed in 19.2.3.10.

19.3.2.12 Results

The results (discussed in detail in Subsection 19D.5.12) indicate core damage frequency is extremely small for internal events. These results, together with the associated source terms, form the input for the consequence analysis in Subsection 19E.3.

19.3.3 Magnitude and Timing of Radioactive Release

The evaluation of the fission product release was performed using a modified version of MAAP3.0B as discussed in Subsection 19E.2. Representative accident sequences were chosen for study on the basis of the core damage and containment event trees. Each accident sequence was then evaluated for the timing and magnitude of release.

There are three important considerations for the timing of fission product release when considering the consequences of a potential severe accident.

(1) The time available for fission product decay affects the maximum source which could be released. In an extreme case, if all of the fission products were released after an infinite period of time, the offsite dose would be zero because all the fission products would have decayed to stable states. In the ABWR, the COPS ensures that the noble gasses are the only significant release from the containment for most sequences. The potential dose associated with the release of noble gasses drops to less than 10% of its initial value within 7 hours of shutdown. Twelve hours after shutdown, the potential dose has dropped to 5% of its initial value, and it decreases very slowly thereafter. For cases without COPS actuation, the potential dose can be dominated by iodine species. These species decay very slowly retaining two-thirds of their potential dose after 40 hours.

(2) The time between the release of fission products from the core and the time of release from containment (residence time) affects the removal in containment. For releases through the COPS, this term is not important since noble gasses are not retained, and the suppression pool effectively scrubs the remaining fission products as they pass through the pool. This time can be important for the few accidents which have drywell releases. However, for most sequences, a time delay of a few hours after release from the fuel brings the airborne fission product concentration to its equilibrium value. This is primarily the result of the submergence of the debris with water from the Firewater Addition System or the passive flooder.

(3) The time available for offsite evacuation, should it be necessary, is also important. Discussions with several utilities indicate that evacuation of their Emergency Planning Zones (EPZ) can be completed in less than 8 hours, even in the worst weather conditions. Experience has also indicated that ad hoc planning can successfully evacuate a region on about 24 hours (Reference 4 of Appendix 6J).

Based on the forgoing, four time frames were selected in determining the time of fission product release, either via the rupture disk or directly from the drywell. Table 19.3-6 summarizes the results which were obtained by using the probabilities given in

Table 19D.5-3 and assigning them to a time and mode of release based on the accident analysis contained in Subsection 19E.2.2.

19.3.4 Consequence of Radioactive Release

The evaluation for consequences of potential radioactive releases was performed using the CRAC-2 computer code as is detailed in Subsection 19E.3. Based upon the evaluation of plant performance, accident classes were defined in terms of their associated release characteristics and fission product releases. Each accident class was then evaluated by the CRAC-2 code at five sites, one representing each major geographical region of the United States. Each site was chosen as representative of its geographical region based upon meteorological calculations and was further defined as average in terms of population density for that geographical region. The results for the five sites were averaged and compared to three goals, two based upon the NRC safety goal policy of minimizing risk to an individual and the public near a plant, and the third based upon an industry goal of minimizing the dose close to the plant. The results of this study show that the ABWR Standard Plant satisfies these goals.

19.3.5 References

19.3-1 "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessment," NUREG/CR-3862, Idaho National Engineering Laboratory, May 1985.

19.3-2 "Advanced Light Water Reactor Utility Requirements Document, Volume II, Chapter 1; Appendix A: PRA Key Assumptions and Groundrules", Draft, Electric Power Research Institute, August 1988, p. D4.

19.3-3 "GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment," 22A7007, General Electric Company, March 1982.

19.3-4 "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, NUREG-75/014, United States Atomic Energy Commission, October 1975.

19.3-5 "Failure Rate Data Manual for GE BWR Components", NEDE-22056, Rev. 2, Class III, General Electric Company, January 17, 1986.

19.3-6 A.D. Swain and H.E. Guttmann, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications", NUREG/CR-1278, August 1983.

19.3-7 "Analysis of a High Pressure ATWS with Very Low Makeup Flow", DOE/ID-10211, Idaho National Engineering Laboratory, October 1988.

Table 19.3-1 Initiating Event Frequencies

Initiating Event	Frequency Per Reactor Year*
Manual Shutdown	
Isolation/Loss of Feedwater	
MSIV Closure	
Loss of Condenser Vacuum	
Press. Reg./Bypass Valves Closed	
Loss of Feedwater	
Non-Isolation Event (Trip with bypass)	
Inadvertent (Stuck) Open Relief Valve	
Loss of Offsite Power	
Less than 30 minutes	
30 Minutes to 2 Hours	
2 to 8 Hours	
Greater than 8 Hours	
Small LOCA	
[Liquid Break 5.063 cm^2 (0.00545 ft^2) or less] [Steam Break <278.7 cm^2 (<0.3 ft^2)]	
Medium LOCA	
[Liquid Break greater than 5.063 cm^2 (0.00545 ft^2) and less than 278.7 cm^2 (0.3 ft^2)]	
Large LOCA	
[Liquid Break 278.7 cm^2 (0.3 ft^2) or greater] [Steam Break 278.7 cm^2 (0.3 ft^2) or greater]	

* Not part of DCD (refer to SSAR).

Table 19.3-2 Success Criteria to Prevent Initial Core Damage for Transient and LOCA Events With RPS Scram

Event	Success Criteria
CORE COOLING:	
Large Liquid LOCA [>=278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or LPFL(1) – A or B or C
Large Steam LOCA [>=278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or LPFL(1) – A or B or C or 1 Condensate Pump + 1 Condensate Transfer Pump(2)
Medium Liquid LOCA [>=278.7 cm^2 (0.3 ft^2) >5.063 cm^2 (0.00545 ft^2)]	HPCF-B or C or ADS3(3) + LPFL(1) – A or B or C
Small Liquid LOCA [>=5.063 cm^2 (0.00545 ft^2)]	RCIC(4) or HPCF-B or C or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Transfer Pump(2)
All Transients (including IORV)	RCIC(4) or HPCF-B or C or 1 Feedwater Pump + 1 Condensate Pump + 1 Condensate Transfer Pump(2) or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Transfer Pump(2) or ADS8(6) + 1 Firewater Addition System Pump

Table 19.3-2 Success Criteria to Prevent Initial Core Damage for Transient and LOCA Events With RPS Scram (Continued)

Event	Success Criteria
CORE COOLING (Cont.)	
Small Steam LOCA [<278.7 cm^2 (0.3 ft^2)]	HPCF-B or C or 1 Feedwater Pump + 1 Condensate Pump + 1 Condensate Transfer Pump(2) or ADS3(3) + LPFL(1) – A or B or C or ADS3(3,5) + 1 Condensate Pump + 1 Condensate Transfer Pump(2)
LONG-TERM HEAT REMOVAL:	
All Transients or Small Liquid LOCA	RHR-A or B or C(7) or Normal Heat Removal(8) or CUW(9)
All Steam LOCAs or IORV or Liquid LOCA (Large or Medium)	RHR-A or B or C(7) or Normal Heat Removal(8)
PRESSURE RELIEF:	
Isolation Events	6 Safety/Relief Valves
Non-Isolation Events	3 Turbine Bypass Valves or 2 Turbine Bypass Valves + 2 Safety/Relief Valve or 1 Turbine Bypass Valve + 4 Safety/Relief Valves or 6 Safety/Relief Valves

Notes:

(1) The term “LPFL” refers to the low pressure core flooding mode of the Residual Heat Removal (RHR) System.

(2) The condensate pumps take suction from the hotwell which is a limited water source. Therefore, if the MSIVs are not open, a condensate transfer pump is necessary to pump water from the condensate storage tank to the hotwell in order to replenish the water in the hotwell.

(3) The term “ADS3” implies that at least 3 automatic depressurization valves are automatically actuated on low level and high drywell pressure or the same number of SRVs are manually opened when the ADS would have actuated. For transients, the high drywell pressure signal is not present.

(4) The RCIC turbine needs sufficient steam generation at or above the required minimum pressure to drive the pump. For the IORV the RCIC will provide adequate cooling for at least 2 hours.

(5) If none of the motor driven ECCS pumps are running, the ADS will not automatically initiate.

(6) The term "ADS8" implies that the 8 automatic depressurization valves or the same number of SRVs are manually opened within one minute after the reactor vessel water level has decreased to the lower water level 1 setpoint.

(7) If the reactor system is at high pressure, the RHR System would be operated in the pool cooling mode. If the reactor system is at low pressure, the RHR System can be operated in either the pool cooling or the shutdown cooling mode.

(8) The MSIVs or the main steam drainlines must be manually opened, if previously closed, for this system to work. This requires the availability of a nitrogen gas supply. If the RPV is depressurized, the main steam drainline option is not viable since it will not pass enough steam to remove the decay heat energy. Furthermore, the circulating water pumps are required to cool the main condenser. Also, a condensate pump is required to transfer excess water from the hotwell to the suppression pool or the condensate storage tank.

(9) The Reactor Water Cleanup (CUW) System is capable of removing the energy due to decay heat (at greater than 4 hours after scram) at high RPV pressures if the return water bypasses the regenerative heat exchanger. Manual override of CUW isolation signals would be necessary if a level 3 isolation signal is present.

Table 19.3-3 Success Criteria and Required Operator Actions For ATWS Events

Initiator	Success Criteria	Time of Operator Action
PRESSURE RELIEF:		
Isolation Initiators	15 Safety/Relief Valves	
Non-Isolation Initiators	3 Turbine Bypass Valves + 9 Safety/Relief Valves	
	or	
	2 Turbine Bypass Valves + 11 Safety/Relief Valves	
	or	
	1 Turbine Bypass Valve + 13 Safety/Relief Valves	
	or	
	15 Safety/Relief Valves	
CORE COOLING:		
All events with rod insertion	Table 19.3-2	
IORV without rod insertion	1 Feedwater Pump	
	or 2 of RCIC or HPCF-B or HPCF-C	
All other events without rod insertion	1 Feedwater Pump	
	or	
	RCIC	
	or	
	HPCF-B or C	
POWER REDUCTION:		
With Rod Insertion:		
All events	Electric Rod Run In	Automatic
	or	
	ARI	Automatic
	or	
	Manual Insertion	10 minutes
With Boron Insertion		
Isolation Events(1)	RIP Trip	Automatic

Table 19.3-3 Success Criteria and Required Operator Actions For ATWS Events (Continued)

Initiator	Success Criteria	Time of Operator Action
	ADS Inhibit	5 minutes
	Feedwater Runback	10 minutes
	1 SLC	10 minutes
Non–Isolation Events[1]	ADS Inhibit	10 minutes
	Feedwater Runback	10 minutes
	Manual RIP Trip	10 minutes
	1 SLC	10 minutes
LONG-TERM HEAT REMOVAL		
Isolation Events	RHR–A or B or C	30 minutes
Non–Isolation Events	Normal Heat Removal[2] or RHR-A or B or C	30 minutes

Notes:

(1) MSIV Closure and LOSP initiators generate RIP Trip. IORV and Turbine Trip Events require Manual RIP Trip.

(2) Adequate normal heat removal will be provided through the Turbine Bypass valves.

Table 19.3-4 Frequency of Core Damage by Accident Class

Accident Class	Description	Frequency (Events/year)*
IA	Transients followed by failure of high pressure core cooling and failure to depressurize the reactor.	
IB-1	Station blackout events (short term) with RCIC failure.	
IB-2	Station blackout events with RCIC available for core cooling for approximately eight hours.	
IB-3	Station blackout events (long term) with RCIC failure.	
IC	ATWS events without boron injection coupled with loss of core cooling.	
ID	Transients followed by loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling.	
II	Transient, LOCA, and ATWS (with boron injection) events, with successful core cooling but with possible failure of containment.	
IIIA	Small or medium LOCAs with failure of high pressure core cooling followed by failure to depressurize the reactor.	
IIID	LOCAs followed by loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling.	
IV	ATWS events without boron injection but with core cooling available.	
Total		

* Not part of DCD (Refer to SSAR).

Table 19.3-5 Frequency of Core Damage by Initiating Event

Initiating Event	Description	Frequency* (Events/year)	% of Total
TM	Reactor Shutdown		
TT	Non-Isolation (Turbine Trip)		
TIS	Isolation/Loss of Feedwater		
TE2	Loss of Offsite Power for Less Than Two Hours		
TE8	Loss of Offsite Power for Two to Eight Hours		
TE0	Loss of Offsite Power for More Than Eight Hours		
BE2	Station Blackout for Less Than Two Hours		
BE8	Station Blackout for Two to Eight Hours		
BE0	Station Blackout for More Than Eight Hours		
TIO	Inadvertent Open Relief Valve		
S2	Small Break LOCA		
S1	Medium Break LOCA		
S0	Large Break LOCA		
ATWS	Anticipated Transient Without Scram		
Total			

* Not part of DCD (Refer to SSAR).

Table 19.3-6 Frequency Of Fission Product Release

Time of Release	Release Frequency*	
No Release		
	Release via Rupture Disk*	**Release via Drywell***
> 24 hours		
16 to 24 hours		
8 to 16 hours		
< 8 hours		

* Not part of DCD (Refer to SSAR).

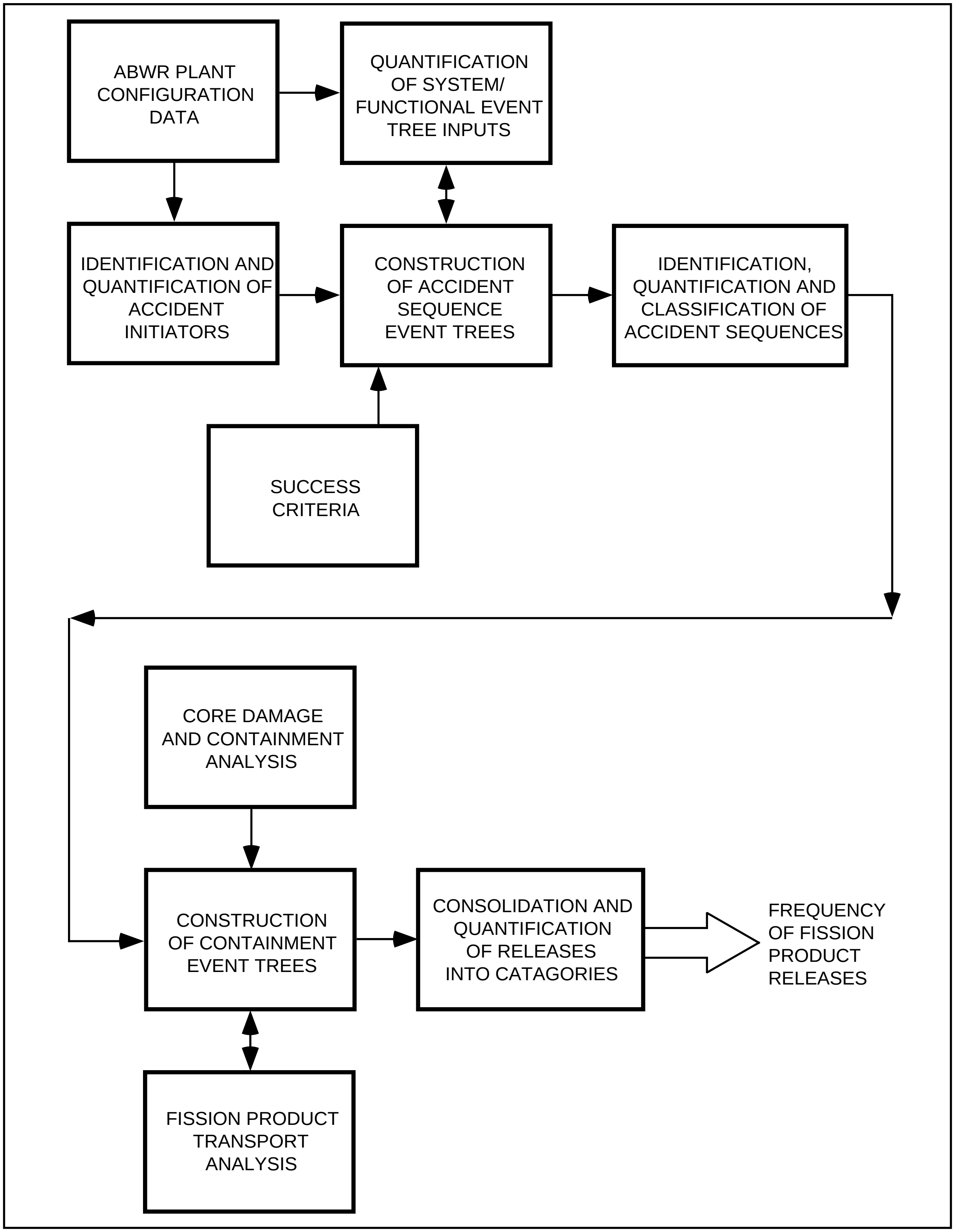

Figure 19.3-1 Overview of Methodology for Assessing Frequency of Core Damage and Fission Product Releases

Figure 19.3-2 Reactor Shutdown Event Tree

Not Part of DCD (Refer to SSAR)

19.4 External Event Analysis and Shutdown Risk Analysis

19.4.1 External Event Review

The Advanced Light Water Reactor (ALWR) Utility Requirements Document (Reference 19.4-1), contains a set of design requirements for the ALWR. The Nuclear Regulatory Commission's Severe Accident Policy (Reference 19.4-2) requires that a probabilistic risk assessment (PRA) be performed for any future nuclear power plant design and that it include consideration of potential external accident initiating events. Therefore, in order to provide a uniform basis for performing these required evaluations, a PRA Key Assumptions and Groundrules Document (Reference 19.4-3) is included as Appendix A of the ALWR Requirements Document. This appendix defines the purpose and scope of the PRA, as well as the types of events to be analyzed and those to be explicitly excluded.

The PRA Key Assumptions and Groundrules (KAG) Document explicitly addresses external initiating events and identifies those events that may be excluded based on qualitative evaluation, as well as those which may require quantitative assessment, for each ALWR. Potential external events identified in the PRA Procedures Guide (Reference 19.4-4) were considered to comprise an exhaustive listing of external events which should be considered for an ALWR PRA.

Potential external initiators identified for exclusion, as well as the accompanying rationale for exclusion, were reviewed. These included all identified events other than tornados and earthquakes. It is assumed that the EPRI assessment that the events listed are considered not to be important contributors to ALWR core damage based on improved design, proper siting, and low probability is an acceptable and sufficient basis for the exclusion of these events from more detailed evaluation in the ABWR PRA.

The above position is supported by the external event evaluations described and the conclusions drawn in Reference 19.4-5. This work was performed by ARSAP in support of the EPRI Requirements Document effort, and is judged applicable to the ABWR design. Assessments were made on the basis of the PRA Procedures Guide, siting requirements contained in the NRC Standard Review Plan (Reference 19.4-6), and EPRI ALWR design criteria.

Only two potential external accident initiators are identified by EPRI in the Key Assumptions and Groundrules Document as events which may require quantitative assessment for each ALWR:

(1) Tornado strikes

(2) Earthquakes

This EPRI assessment is the basis for limiting quantitative external event treatment to these two potential initiators. However, the NRC subsequently required additional analyses for internally initiated fire and flood. Treatment of all four initiators is discussed below.

EPRI qualitative assessment of the ALWR vulnerability to tornado-induced events concludes that most of the vulnerabilities found in past PRAs are not likely to occur in the ABWR design. Rather, the dominant effect of a tornado strike is expected to be prolonged loss of offsite power, and the EPRI position is that a simplified model is sufficient for assessment, provided that it addresses combinations of random failures in combination with loss of offsite power. The Advanced Reactor Severe Accident Program assisted EPRI by developing a method and model to quantitatively evaluate tornado strike impact. This EPRI KAG approach is applied in Subsection 19.4.2 to assess ABWR tornado vulnerability.

EPRI concludes that a seismic analysis is a required part of an ALWR PRA and presents the bases and rationale for its performance in the KAG document. The detailed seismic event analysis is presented in Subsection 19.4.3.

A screening analysis was performed for risk from internally initiated fires. The FIVE methodology was used. The results of the analysis are given in Subsection 19.4.4.

A probabilistic analysis was performed for flooding. All buildings which contain equipment that could be used for safe shutdown were considered. Subsection 19.4.5 contains the results of this analysis.

Shutdown risk was considered in order to evaluate the potential risk for Operational Modes 3, 4 and 5. The results of the analysis are given in Subsection 19.4.6.

19.4.2 Tornado Strike Analysis

As indicated in the preceding subsection, tornado strikes are one of the two classes of external initiating events which, according to the EPRI ALWR Utility Requirements Document, require quantitative assessment for each ALWR. This subsection discusses the basis for the EPRI position, describes the application of this high-level analytic approach to the ABWR, and presents results of the ABWR tornado strike evaluation. This EPRI position and approach are used to estimate ABWR tornado strike core damage frequency.

As part of the support provided EPRI in the development of the ALWR Requirements Document, ARSAP performed an evaluation of ALWR designs (as defined by the EPRI ALWR Requirements Document) to identify vulnerabilities to tornado events, and developed a model and approach to quantitatively estimate expected ALWR core damage frequency due to tornado strikes. Results of this activity are documented in

Reference 19.4-7. The ARSAP qualitative evaluation indicates most vulnerabilities found in past PRAs are not expected to occur in the ABWR and that the dominant effect of a tornado strike is expected to be a prolonged loss of offsite power. Therefore, the need for analysis specifically addressing the consequences of tornado-induced loss of offsite power is indicated.

The ARSAP tornado evaluation developed expected tornado strike frequencies from regional historical data summarized in an EPRI report on tornado missile risk assessment (Reference 19.4-8). Tornados with intensities expected to contribute to core damage events were combined to generate total regional frequencies per square mile per year. Expected tornado strike frequency was then obtained by multiplying the regional values by an assumed plant area of approximately 0.363 square kilometers (0.14 square miles). The resulting regional site strike frequencies were found not to be strongly region dependent, and therefore the maximum assessed regional value was conservatively specified as the basis for evaluation.

Consequently, the loss of offsite power and station blackout accident event trees of Subsection 19D.4 were evaluated using the regional value as the loss of offsite power initiating event frequency in Figure 19D.4-4. In addition, these trees were adjusted to be consistent with the following assumptions resulting from the ARSAP qualitative evaluation of the expected ALWR tornado strike vulnerabilities:

- Condensate storage tank and condenser assumed vulnerable to tornado effects and no credit taken for either.
- Power conversion and feedwater systems assumed unavailable due to loss of offsite power.
- Offsite power recovery not expected within 24 hours following a tornado strike.

Remaining assumptions and conditions for evaluating the loss of offsite power and station blackout event trees for tornado site strike consequences were the same as those documented in Subsection 19D.4.

Evaluation of these event trees on the conservative bases listed above yields an extremely small total core damage frequency due to tornado-initiated events, which is quite small compared to the internal events result and the core damage frequency goal. Since tornado-induced events are predicted to be such small contributors to core damage frequency, this high level evaluation is judged to be sufficient and a more detailed analysis is not warranted.

19.4.3 Seismic Margins Analysis

19.4.3.1 Introduction

A seismic margins analysis (SMA) has been conducted for the ABWR using a modification of the Fragility Analysis method of Reference 19.4-9 to calculate high confidence low probability of failure (HCLPF) accelerations for important accident sequences and accident classes. HCLPF values were calculated for components and structures using the relationship

$$\mathrm{HCLPF} = A_m * \exp(-2.326 * \beta_c)$$

where:

A_m	=	the median peak ground acceleration corresponding to 50% failure probability,
β_c	=	the logarithmic standard deviation of the component or structure fragility.

The resulting HCLPF acceleration corresponds essentially to the 95th percent confidence level that at that acceleration the failure probability of a particular structure or component is less than 0.05 (5%). HCLPFs for accident sequences were evaluated through use of event trees, and seismic system analysis was performed with fault trees to determine HCLPFs of systems.

The seismic margins analysis evaluates the capability of the plant and equipment to withstand a large earthquake (2*SSE).

This subsection discusses the background, objectives, and general approach to the seismic margins analysis. The ground rules and analytical bases for the analysis are also given.

19.4.3.1.1 Background

Seismic event probabilistic analyses have been performed for several PRAs including the WASH-1400 Reactor Safety Study (Reference 19.4-10). The following statement was made in WASH-1400:

> "Although it is difficult to predict with precision the probability of potential accidents due to earthquake damage to a nuclear power plant because of general sparsity of quantitative data on the sizes and effects of earthquakes, it appears possible to make order of magnitude estimates that are useful in the type of risk assessment performed in this study."

Even though there has been a great deal of seismic research and analysis since WASH-1400, the above statement remains largely true today, particularly in regard to uncertainty in establishing an appropriate seismic hazard function. Because of the high degree of uncertainty that presently exists in this regard, a method of analysis has been developed that does not require prediction of an expected seismic hazard function. This methodology, a “seismic margins analysis”, assesses the seismic capacity of the ABWR design in relation to the safe shutdown earthquake (SSE), and in relation to hypothetical seismically induced accident sequences that could lead to damage to the reactor core.

Section 3.2 states the following:

> “ABWR Standard Plant safety-related structures, systems, and components, including their foundations and supports, that are required to perform nuclear safety-related functions during or after a safe shutdown earthquake (SSE) are designated as Seismic Category I.
>
> “The Seismic Category I structures, systems and components are designed to withstand, without loss of function, the appropriate seismic loads (as discussed in Section 3.7) in combination with other appropriate loads.”

Section 3.7 describes the deterministic analyses performed to verify the Standard Plant design relative to seismic events within the design basis envelope. Since the ABWR standard plant is designed for a nominal 0.3g SSE on all soil conditions described in Appendix 3A, considerable margin exists relative to any particular site. It is this design margin that allows the plant to accommodate seismic events far beyond the design basis without significant risk to the public health and safety. The seismic margins analysis discussed in the following subsections and presented in detail in Appendix 19I confirms the low risk for the ABWR standard plant from seismic-initiated events.

19.4.3.1.2 Objectives of the Analysis

The main objectives of the seismic margins analysis are the following:

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

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

(3) To provide an understanding of the most probable sequences of events following a seismic event, and to identify any outstanding vulnerabilities (if any exist) to seismic events.

The seismic margins analysis that has been performed as described in the following subsections achieves the above objectives.

19.4.3.1.3 General Approach to the Analysis

The general approach and methods used in this analysis correspond to guidelines established by the NRC in Reference 19.4-11.

This assessment consists of five primary tasks:

(1) identification of critical structures, systems, and components (SSCs) in regard to potential seismically-initiated accident sequences,

(2) determination of the seismic capacity of critical components and structures,

(3) development of event tree models of potential seismic accident sequences,

(4) development of functional fault tree models of critical systems,

(5) assessment of the seismic margins (HCLPFs) of the ABWR in responding to the seismic accident sequences.

The first step in the analysis is to identify SSCs that are important to safety during a seismic accident and that may be vulnerable (to some extent) to seismic shock. In performing this step, use is made of the internal event analysis (Section 19.3 and Appendix 19D) and a general knowledge of component fragilities. The objective is to limit the size of the analysis by screening-out non-critical SSCs and SSCs that can obviously withstand a severe earthquake without functional damage.

The second step in the analysis is to determine the seismic capacity of the critical structures and components. Seismic capacities of generic components are based on past analyses. Seismic capacities of ABWR unique components are based on analysis or expected capability. The location of components in the plant configuration in relation to structures that may fail is also established. A structural fragility analysis is then conducted for all structures that contain important safety components. The component and structural fragilities are determined in terms of the median value of ground acceleration that would result in failure of the component or structure. Two additional parameters are derived defining the spread of the distribution about the median value.

The next step in the analysis is to construct event trees representing the potential seismically-induced accident sequences. In constructing the event trees and analyzing the event sequences, bases and assumptions of the analyses are established. Important bases of the analysis are listed in the following Subsection (19.4.3.1.4).

The next step in the analysis is to establish the seismic margins of the critical systems by constructing and analyzing functional fault tree models representing the systems.

The final step is to determine the HCLPFs of each of the identified potential accident sequences in the event trees. A computer program is used to perform this step.

19.4.3.1.4 Ground Rules/Analytical Bases

In addition to the PRA bases discussed in Section 19.2, several additional groundrules pertaining to the seismic analysis are given below:

(1) Because of the relatively low seismic capacity of ceramic insulators and offsite transmission lines, the analysis assumes that offsite power will be lost. No credit is given to recovery of offsite power when lost due to the seismic event. This may be somewhat conservative, but is necessary due to the uncertainty of the nature of the failure and actions necessary to recover power.

(2) No credit is given to repair or recovery of mechanical failure of components caused by the seismic event.

(3) Structural failure of a Seismic Category I building containing important equipment results in functional failure of all contained equipment.

(4) Seismic failure of identical redundant components at similar locations are treated as dependent failures i.e., all components fail together. This conservative assumption is used to simplify the analysis. At some future time, it may be desirable to selectively modify this assumption to provide a more accurate model.

19.4.3.2 Seismic Capacity Analysis

In order to determine the capacity of the ABWR plant to resist seismic events, it is necessary to know the seismic margins of plant structures and components. This is accomplished through the development of fragility curves and associated HCLPF capacities. A typical fragility curve is an S-shaped curve which has an increasing probability of failure at higher seismic motion. The mean fragility curve is given in terms of the mean peak ground acceleration (PGA). Fragility curves are generated for those components and structures that have been identified as potentially important to the seismic risk analysis. The resulting HCLPF capacities serve as input to the system analysis following the seismic margins approach.

The development of the seismic capacities for the structures and components of interest is given in Appendix 19H.

19.4.3.2.1 Structural Fragility

Detailed fragility evaluations are presented in Subsection 19H.3 for the following Seismic Category I structures:

- Reactor building shear walls
- Containment
- Reactor pressure vessel pedestal
- Control building.

The radwaste building does not contain safety-related equipment and its failure will not lead to core damage. Consequently, an estimate of the radwaste building fragility is not required.

19.4.3.2.2 Component Fragility

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

(1) ABWR Specific Components—whose fragility evaluation was made according to existing design information.

Detailed seismic fragility evaluations are presented in Subsection 19H.4.2 for the following ABWR specific components:

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

(2) Generic Components—whose fragilities are based on data recommended in Reference 19.4-3 or other data sources as appropriate.

Detailed fragility evaluations for safety-related components other than those specific components presented above cannot be made at this stage of certification due to lack of design details.

The ABWR generic components of interest for this seismic risk analysis are the following:

- Off-site Power (transformers and ceramic insulators)
- Cable trays
- Batteries and battery tracks
- Battery chargers/Inverters
- Electric equipment (chatter failure mode)
- Switchgear/Motor control centers
- Transformers (480 V)
- Diesel generators and support systems
- Turbine-driven pumps
- Motor-driven pumps
- Diesel-driven pumps
- Heat exchangers
- Small tanks (e.g., standby liquid control tank)
- Air-operated valves
- Motor-operated valves
- Safety relief, manual, and check valves
- Hydraulic control units
- Large flat-bottom storage tanks
- Heating, ventilation, and air conditioning ducting
- Air handling units/room air conditioners
- Piping
- Service water pump house

Seismic fragilities and corresponding HCLPF values for these components are summarized in Table 19H-1. These generic seismic capacities are selected from a review of ALWR recommendations (Reference 19.4-3) and other PRA studies. They are considered achievable for the ABWRs with an evolutionary improvement in the seismic capacities of the components designed to a 0.3g SSE.

19.4.3.3 Evaluation of Seismic Margin

The HCLPFs of accident sequences due to seismic events were calculated by constructing and quantifying event trees and fault trees which model the logical relationship of components, systems, functions, and structures that are significant to seismic risk. While structures were not modeled in the internal events analysis of Section 19.3, their inclusion is necessary in the seismic analysis because of the potential for component or function failure due to structural failure. In this analysis, it was assumed that all components housed within a failed structure would fail to function.

Fault trees and event trees are quantified to determine HCLPFs of systems and accident sequences. There are two alternative methods of quantification—"convolution" and "min-max". In the convolution method, accident sequences are evaluated by combining input fragility curves according to the Boolean expression for each sequence. Seismic and random/human failure probabilities are calculated and combined (convolved) for discrete intervals of ground acceleration, and then integrated over the range of interest.

In the min-max method, input fragilities are combined by using the lowest (minimum) HCLPF value of a group of inputs operating in an OR logic, and by using the highest (maximum) HCLPF value of a group of inputs operating in an AND logic. Random/human failure probabilities are reported in combination with HCLPFs for each accident sequence.

Analysis of the effects beyond core damage (Level 2 PRA analysis) was not a part of this seismic margins analysis. However, event trees were constructed to examine the possibility of loss of containment isolation resulting in a large release given the earthquake and a resulting core damaging accident.

Because of the inclusion of a rupture disk in the ABWR design as an ultimate means of containment heat removal, and because an earthquake would not prevent rupture of the disk, failure of containment heat removal is not modeled in the seismic margins analysis. (There are no Class II sequences in the analysis.) There are two valves in line with the rupture disk; however, these valves are left in an open position, and the earthquake would not cause these valves to close.

There are several operator actions included in the seismic margins analysis. These operator actions are discussed in Subsection 19D.7.4.

19.4.3.4 Results of the Analysis

The results of the convolution analysis are shown on the event trees of Subsection 19I.3 and in Table 19I-2 in terms of HCLPF values for the accident sequences, with and without the inclusion of random failures. As seen in the event trees and the table, the HCLPF values for all accident sequences are greater than 0.60g, which is twice the safe shutdown earthquake (SSE = 0.30g). The results of the convolution analysis in terms of accident classes are shown in Table 19I-3.

The HCLPF value of accident sequences obtained from the min-max analysis are printed on the event trees next to the column of accident classes. The combination of HCLPF and random failure probabilities of accident sequences are described in Table 19I-4. As can be seen, no accident sequence has a HCLPF lower than 0.60g.

For most accident sequences, the min-max method of analysis provided lower (more conservative) HCLPF values. However, the use of either method of analysis produced HCLPFs greater than twice the safe shutdown earthquake for all potential accident sequences. The seismic margins analysis has provided confidence that the ABWR design will withstand an earthquake of at least 0.6g intensity—twice the design SSE—and achieve safe shutdown without damage to the reactor core.

19.4.4 Fire Protection Probabilistic Risk Assessment

A fire screening analysis was performed to assess the vulnerability of ABWR to fires within the plant. Mutual agreement was reached earlier with the NRC that a fire screening approach was appropriate and that the Fire Induced Vulnerability Evaluation (FIVE) methodology developed by EPRI provided a proper vehicle for performing this analysis. The methodology is based on conservative assumptions using industrial and plant-specific databases for evaluating fire event sequences while making maximum use of existing plant fire analysis and documentation.

The FIVE methodology provides procedures for identifying fire compartments for evaluation purposes, defining fire ignition frequencies, and performing quantitative screening analysis of fire risk.

Five bounding fire scenarios and corresponding fire ignition frequencies were developed on the basis of the FIVE methodology. The first three of these considered the impact of fires which incapacitate each of the three safety divisions (separated by three-four fire barriers, and each encompassing several fire areas) and, thus, the ECCS equipment which is dependent on each for successful performance. Any fire in a divisional area was assumed to result in the immediate and complete loss of function of the division. The fourth scenario considered the impact of a fire in the control room with the conservative assumption that the only ECCS functions available are those that can be controlled and operated from the remote shutdown panel, and the RCIC system, which can be manually operated outside the control room. The fifth scenario examined the consequences of a fire in the turbine building, based on the assumption that resulting loss of offsite power bounds the possible outcomes of this initiator.

Considering these composite bounding scenarios is an added conservatism to the already conservative FIVE methodology.

Fire ignition frequencies were developed for each of the above scenarios by directly applying the prescriptive steps documented in the FIVE methodology. Bounding core damage frequency estimates were developed by applying these initiating event frequencies to appropriately modified ABWR Level 1 fault and event tree models and reevaluating them.

The final bounding core damage frequency for each of the five scenarios was calculated and determined to be acceptable. These results reflect the inherently conservative nature of the FIVE methodology itself, compounded by its additional conservative application in evaluating fire impact at the divisional fire area, control room complex, and turbine building fire levels. Addressing ABWR fire risk at the fire compartment level, considering ignition sources, fire progression, and suppression in more detail will reduce this value.

19.4.5 ABWR Probabilistic Flooding Analysis

The results of the ABWR Probabilistic Flooding Analysis show that the turbine, control, and reactor buildings are the only structures that required evaluations for potential flooding. The other buildings do not contain any equipment that could be used for safe shutdown or potential flooding would not result in a plant transient.

Flooding in the turbine building could result in a turbine trip due to loss of circulating water or feedwater. Automatic pump trips and valve closure on high water level should terminate the flooding. But if these were to fail, a non-watertight door at grade level in the turbine building should allow water to exit the building. If this door retained water, watertight doors would prevent water entering the control and reactor buildings. The core damage frequency (CDF) for turbine building flooding is extremely small.

The worst case flood in the control building is a break in the reactor service water system (RSW) which is an unlimited source. Floor drains and other openings in the floor would direct all flood water to the first floor where the reactor component cooling water (RCCW) rooms are located. The RCCW rooms contain sump pumps. Water level sensors in the RCCW rooms should actuate alarms in the control room and send signals to trip the RSW pumps and close isolation valves in the RSW system. If these sensors were to fail, watertight doors on each room should limit flood damage to only one of the three RCCW divisions. Breaks in the fire water system could result in interdivisional flooding in the upper floors but floor drains would limit water height to below installed equipment for the first hour. To prevent damage to safety-related equipment after this time requires operator actions to limit the depth of water. The CDF for control building flooding is extremely small.

Reactor building flooding could occur either inside or outside secondary containment. In either case, the flooding sources are finite with the suppression pool and condensate storage tank being the largest sources. Inside secondary containment flooding cannot cause damage to equipment in more than one of the three safety divisions on the first floor because of watertight doors on each safety division room. As was the case in the control building, water from breaks in lines on upper floors will be directed by floor drains to sump pumps on the first floor. The available volume of rooms on the first floor can contain all potential flood sources. Outside secondary containment, floor drains direct all flood water to sump pumps on floor B1F (third floor). If the sump pumps fail or cannot keep up with the flooding rate, an overfill line in the sumps direct water to the corridor of the first floor where it can be contained as discussed above. Interdivisional flooding may occur but floor drains will limit the water elevation such that no damage to safety equipment will occur. The CDF for reactor building flooding is extremely small.

The total CDF for internal flooding is very small.

19.4.6 ABWR Shutdown Risk

The ABWR design has been evaluated for risks associated with shutdown conditions (i.e., Modes 3, 4 and 5). The evaluation included the following shutdown risk categories:

(1) Decay heat removal,

(2) Inventory control,

(3) Containment integrity,

(4) Loss of electrical power,

(5) Reactivity control.

The evaluation also included risk reduction features of the ABWR due to instrumentation, flooding and fire protection, use of freeze seals, and procedure guidelines. ABWR features that are not part of current BWR designs were evaluated to determine if any new vulnerabilities would be introduced. In addition, an evaluation of approximately 200 events at operating BWR plants which were considered precursors to loss of decay heat removal capability showed that ABWR design features could mitigate the effects of all these events.

The results of the ABWR shutdown risk analysis demonstrated that the core damage frequency (CDF) for all shutdown event is very small. The main features that contribute to this low CDF are:

(1) Three physically and electrically independent residual heat removal (RHR) and support systems.

(2) Multiple makeup sources for inventory control (e.g., suppression pool, condensate storage tank, AC independent water addition system).

(3) Two independent off-site sources of electric power and four on-site sources (three emergency diesel generators and a combustion turbine generator).

(4) Reactor protection system (RPS) and standby liquid control system (for boron addition) and interlocks, to prevent accidental reactivity excursions.

19.4.7 References

19.4-1 “Advanced Light Water Reactor Utility Requirements Document, Volume II, Chapter 1: Overall Requirements”, Electric Power Research Institute, June 1986.

19.4-2 “Policy Statement on Severe Accidents”, Federal Register, U.S. Nuclear Regulatory Commission, August 8, 1985, p. 32138.

19.4-3 “Advanced Light Water Reactor Utility Requirements Document, Volume II, Chapter 1; Appendix A: PRA Key Assumptions and Groundrules”, Revision 5, Electric Power Research Institute, December 1992.

19.4-4 “PRA Procedures Guide—A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants”, NUREG/CR-2300, Final Report, U.S. Nuclear Regulatory Commission, January 1983.

19.4-5 Donald Gene Harrison, “Interim External Events Integration for the EPRI ALWR Requirements Document (WBS 4.3.3)”, DOE/ID - 10227, Advanced Reactor Severe Accident Program, U. S. Department of Energy, January 1989.

19.4-6 “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants”, LWR Edition, NUREG-0800, U.S. Nuclear Regulatory Commission, July 1981.

19.4-7 Ricky Lynn Summit, “Estimation of Core Damage Frequency for Advanced Light Water Reactors Due to Tornado Events (Task 4.3.2.1)”, Advanced Reactor Severe Accident Program, U.S. Department of Energy, December 1988.

19.4-8 “Tornado Missile Simulation and Design Methodology”, Volumes 1 and 2, EPRI NP-2005, Electric Power Research Institute, August 1981.

19.4-9 R.P. Kennedy, et al, “Assessment of Seismic Margin Calculation Methods”, NUREG/CR-5270, Lawrence Livermore National Laboratory, March 1989.

19.4-10 “Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants”, WASH-1400, NUREG-75/014, United States Atomic Energy Commission, October 1975.

19.4-11 “Updated Guidance on the PRA-Based Seismic Margins Analysis”, Letter Glenn Kelly, SPSB, DSSA, NRR to Jack Duncan, GE, October 28,1992.

19.4-12 “GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment”, 22A7007, General Electric Company, March 1982.

19.4-13 Campbell, R.D., Ravindra,M.K., and Bahatia A., “Compilation of Fragility Information from Available Probabilistic Risk Assessments”, LLNL, September 1985.

19.4-14 “Handbook of Math Functions”, U.S. Department of Commerce, June 1965, p. 932. Paragraph 26.2.17.

19.5 Source Term Sensitivity Studies

In order to investigate the effect of key assumptions made in the PRA, several sensitivity studies were undertaken. The results of these sensitivity studies are presented in this section.

19.5.1 Core Melt Progression and Hydrogen Generation

Analysis was performed using MAAP to determine the effects of additional hydrogen generation due to oxidation of zirconium (Subsection 19E.2.6.1).

The core melt progression used in MAAP assumes that corium blockages in the channels are formed as the channels melt. This prevents steam from flowing past the fuel in the later stages of the core melt progression. This starves the upper region of steam and thus limits the metal water reaction. Further, MAAP assumes that no metal water reaction can occur once the corium reaches the eutectic temperature of the fuel. For these reasons, MAAP predicts less metal water reaction and, consequently, less hydrogen generation than do other models.

In order to investigate the response of the ABWR to an increase in the amount of hydrogen generated, four sensitivity studies were performed: two with vessel failure at low pressure and two with vessel failure at high pressure. In all four studies, MAAP was run with both the blockage and eutectic cutoff models disabled.

For the low pressure melt sequences, the rate of zirconium oxidation increased from 6.3% of the active cladding to 15.8%. The increase in the metal-water reaction caused the time of vessel failure in both cases to decrease from 1.8 hours to 1.1 hours. For the dominant case with the Firewater System operating, the rupture disk opens at 30.6 hours as compared to 31.1 hours for the base case. The time of rupture disk opening decreased from 20.2 hours to 16.7 hours for the case with passive flooder operation. The change in the magnitude of fission product release for both cases was negligible.

In both high pressure melt sequences, the fraction of active zirconium oxidized increased from 5.1% to 35.9%. The increased hydrogen generation reduced the time to rupture disk opening from 25.0 to 19.7 hours for the high pressure case with passive flooder actuation and RHR spray. The change in fission product release for the case was negligible.

For the less likely case with passive flooder activation only, this resulted in an 11 hour decrease in the elapsed time (from 18.1 to 7.1 hours) to the onset of fission product leakage from the drywell. Additionally, the magnitude of the CsI release fraction at 72 hours increased from 8.7% to 12.5%.

19.5.2 Effect of Overpressure Relief Rupture Disk on Fission Product Release

A wetwell airspace overpressure relief rupture disk is described in detail in Subsection 6.2.5. Such a rupture disk has potential benefits for reducing the fission product release and its associated dose. This subsection examines the magnitude of these potential benefits.

The rupture disk is designed to open when the pressure in the wetwell airspace reaches 0.72 MPa. This assures that any releases caused by overpressurization of the containment will be scrubbed by the suppression pool. A series of MAAP cases were run in order to determine the timing and magnitude of fission product release associated with the various accidents (Subsection 19E.2.2). It was found that all of the sequences with the rupture disk can be accurately represented by two bins (for discussion of bins see Subsection 19E.2.2). The difference between the two is the opening time of rupture disk.

Three types of loss of containment integrity were assumed to be unaffected by the presence of the rupture disk:

(1) Cases with suppression pool drainage following a seismic event result in loss of containment integrity at low pressure (Subsection 19E.2.4.5).

(2) For cases with leakage through the movable penetrations due to high temperature, the leakage through the penetrations occurs before the rupture disk opens. Further, in these cases, the drywell head may be weakened by high temperatures, resulting in drywell head failure before the rupture disk opens. [See Subsection 19E.2.2.2(c) for a more detailed description.]

(3) The third type of sequence for which the rupture disk was assumed to have small impact on the release is early containment structural failure. These failures are hypothesized to occur at the time of vessel failure for sequences in which the vessel fails at high pressure. Since the mode of containment failure is not known, no credit was taken for the rupture disk. (Subsection 19D.5.6.3 describes early containment failure and Subsection 19E.2.4.4 discusses the associated release.)

Subsection 19E.2.8.1.4 examines the benefits and risks associated with the inclusion of the containment overpressure protection system in the design. The analysis indicates a tremendous reduction in the fraction of volatile fission products released from the containment. Typically, the CsI release fraction drops from on the order of 1% to a release fraction of 1E-7. On the other hand, there is a slight decrease in the time of release. However, the effect of the lower fission product release dominates the net impact. This leads to a substantial decrease in risk.

19.5.3 Alternate Definition of Containment Failure

In this PRA, containment failure has been interpreted to mean failure of the containment function. For calculational convenience, this has been taken to be doses greater than 0.25 Sv at 0.8 km (0.5 mile). It has been shown that the ABWR can meet the goal of 0.1 conditional containment failure probability (CCFP) using this definition (Subsection 19.6.8.3).

The NRC staff has proposed an alternate definition of containment failure, one independent of source term:

> "Containment failure occurs when its integrity as a pressure boundary can no longer be controlled."

This definition recognizes the containment function by permitting normal leakage as well as acknowledging credit for suppression pool scrubbing in conjunction with a "last resort" controlled release path, while properly accounting for postulated gross structural failure.

Based on this pressure integrity definition, a new conditional containment failure probability, designated CCFP-PI, can be found. The ABWR meets the containment performance goal regardless of the definition of containment failure.

19.6 Measurement Against Goals

This section summarizes the goals established in the ABWR Licensing Review Bases (Reference 19.6-1) which relate to the prevention or mitigation of severe accidents. In each case the means by which the goal is satisfied is briefly identified, with references to other parts of this chapter which provide additional details.

19.6.1 Goals

The goals summarized in Table 19.6-1 were identified in the Licensing Review Bases. These goals are addressed in Subsections 19.6.2 to 19.6.9.

19.6.2 Prevention of Core Damage

From the internal events analysis (Subsection 19D.5.12.2), the core damage frequency was calculated. For external events, conservative, bounding analyses were performed which conclude that the total core damage frequency is less than the goal of 1.0E-5 and the goal is satisfied.

19.6.3 Prevention of Early Containment Failure For Dominant Accident Sequences

Two modes of early containment failure were identified.

(1) There was judged to be a small chance of drywell failure for sequences in which the core melts with the reactor vessel at high pressure. These are the sequences with EH (for “early” containment failure and “high” release) as the last two characters in the sequence designators. Since depressurization is very reliable and since failure is unlikely even if depressurization fails, the frequency of these failures is calculated to be negligible.

(2) Suppression pool bypass was also identified as being potentially risk significant. If the vacuum breakers fail open, the steam and fission products will not be retained in the suppression pool. Therefore, significant early releases may be possible. The probability of these sequences is insignificant.

A third category of events have been conservatively included in this group. ATWS events with successful core cooling but failure of reactivity control may lead to early containment failure due to high power levels. No containment event trees were developed for these sequences due to the very low accident class frequency. Therefore, an extremely small frequency from the Level 1 analysis is included.

The total frequency is negligible and a very small percentage of the total core damage frequency, so it is concluded that this goal is satisfied.

19.6.4 Hydrogen from 100% of Active Zirconium

A separate effects calculation (Subsection 19E.2.3.2) indicates that the containment can withstand the static pressure of about 0.618 MPa that would be generated were this maximum hydrogen production to occur. This is substantially below the Service Level C Limit of 0.77 MPa for the containment.

19.6.5 Reliable Heat Removal to Reduce Probability of Containment Failure

Containment heat removal capacity is addressed in detail in Subsection 19.3.1.3.1(b), in which success criteria are developed to show the systems necessary to prevent overpressurizing the containment. There are four general means of removing heat from the containment: normal heat removal through the main condenser, reactor water cleanup system, drywell cooling system, and residual heat removal system. No credit is given for operation of the drywell coolers in the PRA.

The heat removal path to the main condenser can be used when the main steamlines are open (or reopened).

The reactor water cleanup system can remove decay heat which is generated four hours after scram if the reactor vessel is at high pressure.

The Residual Heat Removal (RHR) System is described in Subsection 5.4.7. For the worst case event with scram, overpressurization of the containment can be prevented by any one of three RHR divisional subsystems (Loops A, B and C). These subsystems are located in different quadrants of the plant and are protected from common mode failures by divisional separation criteria. Each of these subsystems is automatically initiated by two-out-of-four logic for each initiating parameter (low reactor water level or high drywell pressure). Each of the four input signals to the two-out-of-four logic is provided by separate Class 1E instrument divisions. The heat exchanger is always in the cooling flow path, so that containment cooling to the ultimate heat sink starts as soon as RHR injection flow begins. Under normal operating conditions, there is one Reactor Building Cooling Water (RCW) pump, one Service Water (RSW) pump and two RCW heat exchanger in operation in each of the three loops. When required, the standby pumps and heat exchanger are put in operation. This cooling occurs for all modes of RHR operation. Any one of the following four RHR operating modes will satisfactorily prevent overpressurizing the containment for the dominant sequences:

(1) Low pressure flooder mode

(2) The suppression pool cooling mode

(3) The shutdown cooling mode

(4) The containment spray mode

Key components of the supporting RCW and service water systems (one pump and two heat exchangers in each loop as noted above) are in operation during normal plant operation. At least each month, the standby pumps and heat exchangers are started and the previously running RCW and RSW equipment is placed in a standby mode. This design and operating philosophy results in high reliability since, for example, if power is lost and later regained, the equipment in half of each loop can be thought of as having been tested within the last few minutes or few hours; the other half can be thought of as having been tested within the past week (Summer operation) or month (Winter operation). The RHR and supporting systems are also used to maintain low suppression pool temperatures during normal operation. Depending on the ultimate heat sink temperature, this could occur with frequencies ranging from about once per day to once per week.

The reliability of RHR and its supporting systems is assessed in the fault trees of Subsection 19.D.6.

Finally, as noted in Subsection 19.6.3, there is a substantial amount of time available during which the above heat removal systems might be repaired if they had initially failed. This time effectively increases the reliability of these systems.

The calculated frequency of containment structural failure resulting from loss of heat removal for internal events is extremely small (Subsection 19.D.5.12). Only an extremely small percentage of these events result in core damage due to the high degree of diversity and redundancy in the core cooling systems.

Thus, very reliable heat removal is provided which makes the probability of containment failure very small and the goal is satisfied.

19.6.6 Prevention of Hydrogen Deflagration and Detonation

The inerting system is described in Subsection 6.2.5. The primary containment vessel is inerted with nitrogen gas to below 3.5% oxygen by volume. This provides some margin against instrumentation errors or an (unexplained) increase in oxygen concentration. Any increase in actual oxygen concentration is highly improbable since measures are taken to eliminate any source of oxygen in the containment. This includes substituting nitrogen for air in all pneumatic systems and seals, and maintaining the containment at a slightly positive pressure during reactor power operation to prevent in-leakage of air (oxygen). The containment oxygen concentration is expected (and has been observed in operating plants) to slowly decrease during prolonged power operation as nitrogen makeup is periodically added to compensate for the slight leakage from the containment at positive pressure. Therefore, the margin against entering a potentially flammable regime is normally more than 1%.

The ABWR follows standard industry inerting design, establishing a nitrogen storage and delivery system sufficient to inert the containment in less than four hours. In addition, deinerting (to at least 18% oxygen) is possible within four hours.

In the ABWR, the drywell cooler flow rate is very high such that the residence time (drywell volume divided by the inerting or deinerting flow rate) is on the order of a few minutes. Therefore, good mixing is assured. In addition to conserving nitrogen, this good mixing assures that “pockets” of uninerted atmosphere are swept away and the containment is truly inert. Pocketing in the wetwell is much less of a concern since it is a relatively open space.

In conclusion, the inerted containment atmosphere provides passive protection against hydrogen deflagration and detonation. This protection is not vulnerable to loss of power and is available during all accident sequences. Therefore, the goal is satisfied.

19.6.7 Offsite Dose/Large Release

As shown in Figure 19E.3-1 and Table 19E.3-7, the probability of a 0.25 Sv whole body dose at 0.8 km (1/2 mile) from the reactor is extremely small per year, less than the 10^{-6} goal. This goal is satisfied. No attempt was made to define the term “large release” but the 0.25 Sv dose is considered to be “much less than large”, so the large release goal is satisfied.

19.6.8 Containment Conditional Failure Probability

A conditional containment failure probability was determined as outlined below.

19.6.8.1 Potential Mechanisms of Containment Failure

There are several potential mechanisms which could cause significant fission product release and thus might be considered to be “containment failure”:

(1) Energetic steam explosions.

(2) Hydrogen deflagration/detonation.

(3) Suppression pool bypass.

(4) High pressure/temperature combinations.

Energetic explosions are reviewed in Subsection 19E.2.3.1 and it is concluded that there is no potential for steam explosions of sufficient magnitude to overpressurize the containment. Without such overpressurization, there is no potential for significant fission product release. Therefore, for purposes of measuring against the goal, the probability of containment failure resulting from steam explosions is taken as zero.

Hydrogen deflagration / detonation is precluded by inerting the containment as discussed in Subsection 19.6.6. For purposes of measuring against this goal, the probability of containment failure resulting from hydrogen deflagration and detonation is also taken as zero.

Suppression pool bypass which results from certain random equipment failures before or during the accident (as opposed to bypass which results because of increasing temperature or pressure) are examined in Subsection 19E.2.3.3. For internal events, this evaluation showed that the conditional probability of full bypass is extremely small and that the contribution of this bypass is a small percentage of the total plant risk. The potential for suppression pool bypass was also considered from the standpoint of external events and no significant additional mechanisms were identified. Since the conditional probability of full bypass is much less than the 0.1 goal, this potential containment failure mechanism is not considered further for the purposes of measurement against this goal.

High pressure / temperature combinations within the containment under certain conditions can cause containment failure. These potential failures are treated in Subsection 19.6.8.2.

19.6.8.2 Definition of ”Containment Failure“

Containment failure is defined here in a manner which provides an indication of failure of the containment function: Containment failure is considered to have occurred for any sequence which gives an offsite dose at 0.8 km (1/2 mile) of 0.25 Sv or more. In general, this occurs as a result of increased pressure and / or temperature as noted below.

Increased leakage from the containment could occur through penetrations as a result of increasing pressure and temperature. Analysis in Appendix 19F indicates that this could occur if the containment pressure exceeds 0.46 MPa and the temperature exceeds 533 K (500°F).

Drywell head failure is most likely to occur as a result of reduced drywell head load carrying capability at increased temperatures as noted in Appendix 19F. Failure pressure is estimated at 1.025 MPa at an upper drywell temperature of 533 K (500°F).

19.6.8.3 Measurement Against the Goal

From the offsite dose frequency plot (Figure 19E.3-1), the frequency of exceeding 0.25 Sv at 0.8 km (1/2 mile) is extremely small. Dividing by the core damage frequency, gives a conditional containment failure probability less than the goal of 0.1 and the goal is satisfied.

Measurement against an alternate definition of containment failure based on maintenance of containment integrity is discussed in Subsection 19.5.3.

19.6.9 Safety Goal Policy Statement

As noted in Table 19E.3-7, the calculated individual risk is insignificant and the societal risk is negligible. The calculated risks are many decades below the numerical goals and the goals are satisfied.

19.6.10 Not Used

19.6.11 Conclusion

As noted in the discussions in Subsections 19.6.1 through 19.6.9, the Licensing Review Bases goals are satisfied.

19.6.12 References

19.6-1 Thomas E. Murley (NRC) letter to Ricardo Artigas (GE), August 7, 1987, "Advanced Boiling Water Reactor Licensing Review Bases."

Table 19.6-1 Summary of Goals in Licensing Review Bases

Licensing Review Bases Para. No.	Subsection In Which Goal Is Addressed	Summary Statement of Goal
7.5.1	19.6.2	**Prevention of Core Damage**—Mean core damage frequency from internal and external events less than 10^{-5} per reactor year.
7.5.2		**Mitigation of Core Damage**
7.5.2a	19.6.3	Measures to reduce the probability of early containment failure for dominant accident sequences.
7.5.2b	19.6.4	Measures to accommodate hydrogen generated from the reaction of 100% of the zirconium in the active fuel cladding.
7.5.2c	19.6.5	Highly reliable heat removal systems to reduce the probability of containment failure by loss of heat removal.
7.5.2d	19.6.6	Reliable means to prevent hydrogen deflagration and detonation.
7.5.3		**Offsite Consequences**
7.5.3(1)	19.6.7	Mean frequency of offsite doses in excess of 0.25 Sv beyond one-half mile radius (typical United States site boundary) from the reactor less than 10^{-6} per reactor year, considering both internal and external events.
7.5.3(2)	19.6.8	Containment design is to assure that the containment conditional failure probability is less than 0.1 when weighted over credible core damage sequences.
8.10		**Safety Goal Policy Statement**—Comply with eventual requirements. Since the eventual requirements are not known, the current policy is addressed here. The current policy is:
	19.6.9	Risk to average individual in the vicinity of the plant less than 0.1% of sum of prompt fatality risk from other accidents.
	19.6.9	Risk to population within 16.1 km of the plant less than 0.1% of sum of cancer fatality risk from other causes.
	19.6.7	In addition, although not stated in the Licensing Review Bases, GE intends to satisfy the goal (Federal Register, page 28047) of overall mean frequency of large release less than 10^{-6} per reactor year. This goal is addressed along with the above Paragraph 7.5.3(1) goal relating to offsite doses.

19.7 PRA as a Design Tool

In addition to its use as a measurement tool to assess the degree to which PRA-related goals were satisfied as summarized in Section 19.6, the PRA was used to substantially influence the design. During the course of the review of this PRA, the NRC requested that the way in which operating experience was factored into the design and the ways in which the PRA influenced the design be described. This description is provided here.

19.7.1 ABWR Design and Operating Experience

The design of the ABWR covered a period of about 12 years, from 1978 to 1990. The world wide experience of several companies including ABB-Atom, Hitachi, Toshiba, ANM, and GE was used to establish the original design. During the design process, methods were employed to ensure that operating experience was factored into the design. These are summarized in Subsection 1.8.3, particularly Table 1.8-22.

In addition to the general design process noted above, three specific design improvements compared to earlier designs were introduced which provide benefits from a PRA perspective:

(1) The plant is designed for a safe shutdown earthquake (SSE) of 0.3g. Most operating BWRs have an SSE of 0.2g or less. Thus, the ability to withstand earthquakes is improved. Very large margins are expected at low seismic sites.

(2) The elimination of recirculation piping has substantially reduced the potential for LOCAs, particularly large LOCAs.

(3) The use of three separated ECCS divisions, provides the benefits shown in the internal events analysis. In addition, this separation reduces ABWR vulnerability to fires and floods.

19.7.2 Early PRA Studies

PRAs were used extensively in the early design effort for making design decisions. This has resulted in millions of dollars of cost savings without compromising the plant safety. Several key studies are summarized here.

(1) Core Cooling Systems

A core cooling system optimization study was performed. This study enabled the core cooling and heat removal functions to be combined and the total number of ECCS divisions to be reduced from 4 to 3, resulting in significant cost savings.

A RCIC reliability study was performed. This study enabled the elimination of one high pressure core cooling system by upgrading the RCIC System's reliability.

A risk comparison study was performed. This compared the core damage frequency for BWR/4, 5, and, 6 plants with the ABWR and identified the importance of modifying the ADS logic to initiate on low water level. This change improved the ABWR safety significantly for transient event sequences.

(2) Reactivity Control

Studies of ABWR scram system reliability and scram system unavailability with alternate rod insertion enabled the incorporation of a less expensive ATWS mitigation system in place of an alternate system proposed for an earlier design. This change also results in significant cost savings.

(3) Instrumentation Studies

An ABWR instrument reduction study and reliability assessment enabled the elimination of 60% of the sensor instrumentation in the reactor safety systems without impacting plant safety. Other studies performed have identified significant cost reductions in the ABWR multiplexing systems and other instrumentation systems.

(4) Control Rod Drive Improvements

The early ABWR ATWS design was based on utilizing the capabilities of the new fine motion control rod drives (FMCRD) to meet the intent of USNRC ATWS Rule 10CFR50.62 for improvement of hydraulic scram reliability. Adoption of the FMCRDs provided improved scram reliability by elimination of the scram discharge volume, which is a potential common mode failure point for current BWRs using the locking piston-type CRDs. The scram reliability goals were met without use of the Alternate Rod Insertion (ARI) valves specified in 10CFR50.62. However, subsequent PRA studies showed that adoption of the ARI valves in the design would provide a further substantial reduction in the probability of ATWS. Since the cost of adding the ARI valves to the design at that time was minor, it was decided that their incorporation into the design was appropriate.

The FMCRD brake mechanism is provided to prevent a rod ejection in the event of a break of the scram insert line. As a result of PRA studies, the design was changed from the centrifugal-type brake used in the early design to the current electro-mechanical-type break. The PRA studies indicated that the brake design had to be fully testable on an annual basis to meet the goals for rod ejection frequency. It was determined that the electro-mechanical brake design was easier to test, and would not have any impact on the plant outage critical path.

(5) RIP Trip Study

The reliability of RIP power supply was evaluated. The probability of simultaneous trip of all RIPs was calculated. The objective of this study was to assure that the probability of an all RIP trip event is low enough to classify such an event as an accident. The study confirmed the 4-bus configuration for the RIP power supply. In addition, motor generator sets were adopted to prevent an all RIP trip event from occurring following a loss of AC power.

19.7.3 PRA Studies During the Certification Effort

As part of the ABWR certification effort, the PRA was further used to improve the design. This effort was first reported in the 1991 Probabilistic Safety Assessment and Management Conference. An AC-independent Water Addition System and a combustion turbine generator were added to reduce the probability of core damage. A lower drywell flooder and a containment over pressure protection system were added to mitigate the effects of core damage in the unlikely event that such damage should occur. The studies which lead to these and other improvements are summarized here.

(1) Initial Probabilistic Risk Assessment

The initial PRA effort for ABWR Certification indicated that the ABWR had abundant means of preventing severe accidents and mitigating their consequences and that the goals (Section 19.6) could be satisfied. However, key insights gained from this effort led to the selection of additional features as described in the following paragraphs.

The core damage frequency from internal events was determined to be extremely small. Although this result was very favorable, the core damage frequency was dominated by station blackout. A simple, “AC-independent water addition system” was added to the design. The cost impact is quite small since only a few small lines and manually operated valves are added. A combustion turbine generator, required by the Electric Power Research Institute Advanced Light Water Reactor Requirements Program was also added to the design. These features significantly decreased the frequency of core damage due to station blackout. As compared to current plants the frequency of these events is extremely low.

In other evaluations, it was determined that if molten core material were present in the lower drywell, it would ablate the reactor vessel pedestal in the region of the wetwell / drywell vents, allowing suppression pool water to enter the lower drywell. This would quench the corium and terminate core-concrete interaction, non-condensable gas generation, and drywell atmosphere heatup; all favorable effects which lessen the potential to fail the containment function. However, it did not seem prudent to take favorable credit for a rather uncertain process. Earlier conceptual studies had identified the concept of a “passive drywell flooder” which could be relied on with much greater certainty to produce the desired favorable effects.

The drywell head was found to be the most probable failure location should the containment be pressurized to a point well above the design pressure. If such an unlikely failure were to occur, fission products could be released without the benefit of suppression pool scrubbing. Fission product retention in BWR suppression pools has been found to be very beneficial in reducing the amount of fission products released from the containment. Even before specific numerical calculations had been performed, the potential benefits of a device that would relieve containment pressure through the suppression pool were apparent. Therefore, a containment overpressure relief feature was added to the design to accomplish this function.

Examination of dominant severe accident sequences indicated several areas in which the Emergency Procedure Guidelines or plant operating procedures could be improved for the ABWR. Prevention of accidents can be improved in seismic initiated loss of offsite power events by instructing the operator to manually operate heat removal system valves if transformer loss has made power operation of those valves impossible. Accident mitigation can be improved for the ABWR accident sequences in which corium has penetrated the reactor vessel by filling the drywell with water to the level of the bottom of the reactor vessel, rather than to the top of the active fuel as done for earlier BWRs.

(2) Feature Descriptions and Resulting Benefits

As a result of the studies summarized above, four new features were added to the design to enhance the plant’s performance under severe accident conditions. The added features are described in the following paragraphs.

(a) AC-Independent Water Addition

Two fire protection system pumps are provided on the ABWR: one pump is powered by AC power, the other is driven directly by a diesel engine. A fire truck can provide a backup water source. One of the fire protection standpipes is cross-connected to the RHR injection line to the reactor vessel through normally closed, manually operated valves. From this line, fire protection water can be directed to the reactor vessel after the reactor vessel has been depressurized. Fire protection water can also be directed to the drywell spray header to reduce upper drywell pressure and temperature. Should drywell head failure occur (an extremely unlikely event, especially given the containment overpressure protection feature discussed below), use of drywell spray also reduces the release of volatile fission products from the containment.

(b) Combustion Turbine Generator

A combustion turbine generator (CTG) starts automatically. It is automatically loaded with selected investment protection loads. Safety-grade loads can be added manually. This provides diverse power if none of the three safety-grade diesel generators are available.

The CTG is a standby non-safety power source to feed plant investment protection loads during loss-of-offsite power events. It is not seismically qualified. The unit also provides an alternate AC power source in case of a station blackout event.

The CTG is designed to supply standby power to the three turbine building (non-Class 1E) 6.9 kV buses which carry the plant investment protection loads. The CTG automatically starts on detection of a 30% voltage drop on the 6.9kV bus. The 6.9kV bus is tripped and the CTG sequentially assumes the loads.

CTG failure will not affect safe shutdown of the plant. The unit is not required for safety but is provided to assist in mitigating the consequences of a station blackout event.

The CTG can supply power to nuclear safety-related equipment if there is complete failure of the emergency diesel generators and all offsite power. Under this condition, the CTG can provide emergency backup power through manually-actuated Class-1E breakers in the same manner as the offsite power sources. This provides a diverse source of onsite AC power.

(c) Lower Drywell Flooder

The lower drywell flooder allows water from the suppression pool to enter the lower drywell during severe accidents where core melting and subsequent vessel failure occur. Several pipes run from the vertical pedestal vents into the lower drywell. Each pipe contains a fusible plug valve connected by a flange to the end of the pipe that extends into the lower drywell. In the unlikely event that molten corium flows to the lower drywell floor and is not covered with water, the lower drywell atmosphere will rapidly heat up. The fusible plug valves open when the drywell atmosphere (and the fusible plug valve) temperature reaches 260°C (500 F). The fusible plug valve is mounted in the vertical position, with the fusible metal facing downward, to facilitate the opening of the valve when the fusible metal melting temperature is reached. When the fusible plug valves open, suppression pool water will be supplied through pipes to the lower drywell to quench the corium, cover the corium, and remove corium decay heat. The result will be a reduced interaction between corium and drywell floor concrete which, in turn, will reduce drywell temperature and pressure from non-condensable gas generation. There will be less chance of overpressurizing the containment and causing radionuclide leakage to the atmosphere. The lower drywell flooder is a passive injection system. No operator action is required.

(d) Containment Overpressure Protection System

If an accident occurs which increases containment pressure to a point where containment integrity is threatened, the pressure will be relieved to the atmosphere by a line connecting the wetwell to the plant stack. Providing a relief path from the wetwell vaporspace precludes an uncontrolled containment failure. Directing the flow to the stack provides a monitored, elevated release. The relief line, designed for 1.136 MPa, contains a rupture disk which opens at a pressure above the design pressure but below the Service Level C capability of the containment. The relief line also includes a second rupture disk, set at a very low value. This allows the discharge path to be inerted. If overpressure occurs, the rupture disks will open and pressure is relieved in a manner that forces escaping fission products to pass through the suppression pool. Relieving pressure from the wetwell, as opposed to the drywell, takes advantage of the fission product scrubbing provided by the suppression pool. After the containment pressure has been reduced and normal containment heat removal capability has been regained, the operator can close two normally open air-operated valves in the relief path to reestablish containment integrity. Initiation of the pressure relief system is totally passive. No power is required for initiation or operation of the pressure relief function.

(e) Seismic Capability of Added Features

After the above added design features were further developed, additional PRA studies were performed focusing on seismically-initiated events. The combustion turbine generator is not seismically qualified so no credit was taken for its operation in the analysis. The other three features have relatively high seismic capacities. The AC-independent water addition system including the direct diesel-driven pump and the associated piping and manual valves have a seismic HCLPF of 0.5g. The lower drywell flooder is virtually invulnerable to a seismically induced failure (pipes and valves whose likely failure mode would probably introduce water to the lower drywell). The overpressure protection system is Seismic Category I, and a seismically-induced failure is not likely to prevent the relief function provided by the rupture disks.

(3) Emergency Procedure Guideline Improvements

Emergency Procedure Guidelines (EPGs) were improved in several areas. Two examples are described here.

(a) Accident Prevention

In a high fraction of seismically initiated station blackout sequences, diesel generators are available to supply power to pumps in the heat removal system but lower voltage power necessary for operation of MOVs may not be available because of transformer failure. The transformer seismic capacity is less than that of the EDGs. However, the necessary valves can be operated manually and this capability will be reflected in the detailed procedures to be developed to supplement the EPGs.

(b) Accident Mitigation

EPGs developed for earlier BWRs call for the operator to fill the containment to the level of the top of the active fuel if the reactor vessel water level cannot be determined or cannot be maintained above the top of the active fuel. For an ABWR plant which has undergone a severe accident, this strategy can be improved. Filling the containment to a lower level than the TAF is appropriate for two reasons. First, noncondensable gases in the containment are compressed to a lesser degree and containment pressure is reduced compared to the earlier strategy. Second, filling the containment to a lower level avoids flooding the containment overpressure protection system and the potential for subsequent damage to system piping if the rupture disk setpoint pressure is reached. Therefore, the operator is directed to fill the containment to the level of the bottom of the reactor vessel. In the very long term, for post accident recovery and cleanup operations, it would probably be necessary to increase containment water level to an elevation above the top of the active fuel.

In the process of preparing the PRA, human actions were summarized and sensitivity studies were preformed. An overview of this process is provided in Section 19.11.

(4) Further Improvements

Subsequent to the above described improvements, several other improvements were identified and incorporated into the design.

The pressure capability of the drywell head was enhanced to increase the containment pressure capability. Basaltic concrete was added to the lower drywell cavity floor to reduce the potential for non-condensable gas generation which could result if core damage occurs.

As a result of the fire PRA studies (Appendix 19M), the capability of controlling automatic depressurization of the RPV from the remote shutdown panel was improved.

Based on studies of the potential effects of failures in Safety System Logic and Control, surveillance testing of microprocessor-based controllers was increased in frequency to quarterly to improve the ability to detect failures which are not detected by the continuous self-test feature.

As a result of the internal flood PRA studies, several improvements or additional design details were developed to reduce the potential for internal flooding to pose a significant threat. These additional features, which are shown in Table 19R-7, include the following:

- condenser bay water level sensors to terminate serious flooding in the turbine building;
- control building floor water level sensors to terminate major potential flooding sources;
- a limitation on the reactor service water (RSW) pipe length to the first RSW isolation valve to limit the water volume which could be drained into the control building following isolation of a RSW break, addition of anti-siphon capability to the RSW System to prevent siphoning the ultimate heat sink into the control building;
- and floor drains, sump overfill lines, and water tight doors in the reactor building to prevent floods from having significant impact.

Based on a detailed PRA evaluation of a reactor water clean up (CUW) system pipe break outside primary containment, a remote manual shutoff valve was added to the system. This valve is located upstream of the inside containment CUW isolation valve and is intended to terminate outside containment CUW line breaks if the two automatic isolation valves fail to close following a line break. The operator can close the valve from the control room. This valve reduces the probability that the operator would have to control RPV water level lower than the normal range for postulated CUW line breaks.

Consideration of severe accident phenomena indicated the ability to cool the core debris in the lower drywell could be compromised if a significant debris mass were to enter the containment sumps. If the core debris were not quenched, continued core concrete interaction with its resultant noncondensible gas generation could lead to containment pressurization even with successful containment heat removal. To prevent this possibility, a protective barrier around the sumps was added to the design. This barrier prevents the intrusion of molten debris into the containment sumps in the event of a severe accident while allowing water to enter the sumps during normal operation.

Several key safety functions, previously performed manually, were automated.

(5) Summary

Probabilistic Risk Assessment studies conducted for the Advanced Boiling Water Reactor during the certification effort provided valuable insights to plant performance under transient and accident conditions. Although the studies indicated that the established safety goals could be satisfied, an AC-independent water addition system and a combustion turbine generator were added to the design to substantially reduce the probability of a sequence of events which lead to core damage. To reduce the potential consequences of a core damage event, should one occur, a passive means of flooding the drywell with water and a passive containment over pressure relief system were added to the design. EPGs were also improved to further enhance the capability to prevent accidents from occurring and to mitigate subsequent consequences.

The studies discussed above were conducted by examining the plant design and operation from many different perspectives and thus are judged to constitute a thorough search for design and procedure “vulnerabilities.” No prescriptive attempt was made to define the term vulnerabilities in this context. It was judged the better approach to give engineers experienced in many disciplines a wide latitude in identifying potential weaknesses and then dealing with each issue as it was raised case by case.

19.7.4 Conduct of the PRA Evaluations

The PRA was conducted in accordance with the Key Assumptions and Groundrules developed under the Advanced Light Water Reactor Program. This document was developed with input from many individuals experienced in PRA.

PRA models consisted of fault trees and event trees as described in the "PRA Procedures Guide", NUREG/CR-2300. Detailed plant models included plant systems and equipment and dependencies arising from common cause failure, human error and support system failure, thus enabling potential vulnerabilities to be identified.

19.7.5 Evaluation of Potential Design Improvements

PRA techniques were used in the evaluation of whether there are additional potential design modifications which would be cost-beneficial to implement (Appendix 19P) and in the technical support of the evaluation of Severe Accident Mitigation Design Alternatives (SAMDA) for compliance with the National Environmental Protection Act (NEPA). Evaluations used the PRA event trees as a guide for estimating conservative benefits from a variety of potential modifications.

19.8 Important Features Identified by the ABWR PRA

Introduction

The ABWR PRA has been reviewed to identify important design features, i.e., those features and actions that contribute significantly to the mitigation or prevention of a particular accident sequence or event scenario. These may be important contributions relating to

- System capability
- Structures, systems, and components denoted by importance measures such as Fussell-Vesely
- Bypass sequences (containment and suppression pool)
- Features identified in SECY 93-087
- How the design meets containment performance goals
- External events
- Shutdown events
- Important core damage sequences
- What keeps core damage frequency (CDF) low
- What has large uncertainty and in the extreme could become a significant contributor to CDF

This section describes the logical process used to identify the important design features and provides the basis for the importance of the feature. These design features are listed in Tables 19.8-1 through 19.8-7. These tables are annotated to show where the features are addressed by ITAAC in Tier 1. These ITAAC verify that the as-built facility correctly incorporates the annotated feature. Other items which have not been identified as important design features have have also been identified in those tables with appropriate reference to Subsection 19.9 and Appendix 19K.

Logical Process Used to Select Important Design Features

Although each design feature that can prevent or mitigate core damage is important to some degree and should be correctly and fully implemented, there are features that provide a greater degree of protection than others and can be considered more "important." For each initiating event (e.g., flood, fire, LOCA), there are components or features that are more important than others for the prevention or mitigation of the event being evaluated. Where contributions to CDF have been determined by the calculation of Fussell-Vesely or Risk Achievement factors, these parameters can be used to identify the most important features. If the analysis does not result in the calculation of importance measures, other bases are used. For example, a single feature that can fully mitigate or prevent an event by completing its function is more important than features that only contribute to the prevention or mitigation of an event or only partially control that event. Also, components whose degradation can result in an increase in severity of an event are more important than those components with larger design margins. The specific bases for the selection of features that are considered important within each analysis category is provided with the features selected.

As a final check to ensure that important features were not overlooked, the processes in each area were reviewed by PRA engineers who performed reviews in the other areas and by senior engineering managers with broad system knowledge. This additional review resulted in the addition of a few features and the deletion of others.

It should be recognized that in identifying important features from a PRA perspective, those identified will generally be more important relative to the specific event (i.e., flood, fire, etc.) than to overall core damage. That is, a feature important for flood mitigation will have a lower overall significance than features for mitigating events with a higher contribution to CDF.

19.8.1 Important Features from Level 1 Internal Events Analyses

19.8.1.1 Summary of Analysis Results

The ABWR internal events probabilistic risk assessment (PRA) was performed to assess plant vulnerability to potential internal accident sequence initiators. The ABWR Level 1 internal events PRA is based upon detailed fault tree models of the various plant systems as well as event trees which define possible progressions and outcomes of each potential accident initiator. These fault trees and sequences of events are used to estimate core damage frequency due to each potential accident sequence. The sum of the sequence outcomes is the estimate of total internal event core damage frequency. The estimated total CDF for all internal events analyzed is very small.

19.8.1.2 Logical Process Used to Select Important Design Features

Following completion of the Level 1 internal events PRA, it was systematically reviewed to identify important features. The internal events PRA allows compilations of minimal cutsets leading to core damage as well as importance measures of those components and systems represented as basic events in the models. These results provided one basis for a systematic review to identify important features and capabilities. In the majority of cases, cutsets and importance measures identify “features” at the component level. By reviewing the accident sequences and cutsets resulting from their detailed evaluation, it was possible to identify those systems, features and capabilities which are most important in assuring that the ABWR core damage frequency will be very low. Further insight was gained regarding risk by examining the Fussell-Vesely and Risk Achievement

Worth importance measures of the basic components contributing to the performance of each system or feature.

As an example, the first 20 cutsets contribute approximately three quarters of the total core damage frequency. Two-thirds of this amount is due to station blackout events, all of which involve failure or unavailability of the Reactor Core Isolation Cooling (RCIC) system. In addition, eight of the twenty basic events of greatest Fussell-Vesely importance belong to RCIC. If the RCIC were not present in the design, the calculated CDF would be approximately one decade higher. These observations highlight RCIC and its capability to operate without AC power for several hours as important features of the ABWR. They also identify the importance of station battery capability to provide RCIC control power for several hours.

As an additional example, failure of the combustion turbine generator (CTG) is included in each of the station blackout failure sequences and cutsets. It is also among the top twenty in Fussell-Vesely importance. These insights identify the diverse source of emergency power provided by the CTG as an important feature of the ABWR design.

Other systems and features which provide diversity in addition to fulfilling redundant functions were identified and their importance assessed. Following these evaluations important ABWR features and capabilities were identified.

19.8.1.3 Features Selected

The specific capabilities and features identified as being important to safety are listed in Table 19.8-1. The basis for the selection of each feature or capability is also provided in the table.

RCIC

In the unlikely event that offsite AC power is lost and the three Emergency Diesel Generators and the CTG are not available, the RCIC system can provide core cooling from a diverse power source (reactor steam) for an extended amount of time. RCIC operation for an extended period of time requires that makeup water supply be switched from the CST to the suppression pool. In addition, the station battery capability must be adequate to provide RCIC control and motive power for approximately eight hours. The capability of the RCIC to provide core cooling from a power source diverse from AC provides approximately decade reduction in the calculation of the estimated CDF. Sensitivity studies have shown that RCIC operation for two hours provides most of the benefit.

Combustion Turbine Generator

In the unlikely event that offsite AC power is lost and all three EDGs are unavailable, the CTG provides a diverse source of AC power. It is connectable to any of the three safety divisions and is capable of powering one complete set of normal safe shutdown loads.

No plant support systems are needed to start or run the CTG. The CTG starts automatically (this feature is not "important" in the context of this analysis) and safety-grade loads are to be added manually. Although the probability of losing offsite power and all three EDGs at the same time is very small, the consequences of such an event is potentially very significant. The capability to provide AC power from a diverse source substantially reduces the risk of a loss of offsite power resulting in a station blackout.

High Pressure Core Flooder (HPCF) Logic and Control

The operation of the HPCF is controlled by the digital safety system logic and control (SSLC) system. As identified in SECY 93-087, the common cause failure of digital instrumentation and control logic may result in the failure of redundant equipment. A postulated common cause failure of the SSLC would disable the HPCF without a diverse means to initiate at least one loop of the HPCF. One division of the HPCF has been provided with capability for initiation and operation through an independent and diverse "hard wired" circuit. Although the probability of a common cause failure of the SSLC is very low, an independent and diverse means of HPCF operation further reduces the risk associated with system operation through the multiplexed digital SSLC.

AC-Independent Water Addition (ACIWA) System

The ACIWA provides diverse capability to provide water to the reactor in the event that AC power or the ABWR engineered safety systems are not available. The system has a diesel driven pump with an independent water supply and all needed valves can be accessed and operated manually. In addition, support systems normally required for emergency core cooling systems are not required for ACIWA operation. Even though the ACIWA is not a first line prevention or mitigation system with respect to core damage, it is important in preventing and mitigating severe accidents in the unlikely event all other systems are unavailable.

Reactor Building Cooling Water (RCW) / Reactor Service Water (RSW)

The RCW system and the RSW system are each designed with two parallel loops in each division. Each loop (i.e., 50% of the capacity of each division) is capable of removing all of the component heat loads associated with operation of the ECCS pumps. Together, the two loops in each division are capable of removing heat from the suppression pool through the RHR heat exchangers during LOCA. The parallel loops of RSW and RCW within each division substantially reduce the calculated CDF.

Prevention of Intersystem LOCA

In SECY 90-016 and 93-087 it has been recommended that designers should reduce the possibility of a loss of coolant accident outside containment by designing (to the extent practical) all systems and subsystems connected to the Reactor Coolant System (RCS) to withstand full RCS pressure. All piping systems, major systems components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) which extend outside the primary containment boundary are designed to the extent practicable to an ultimate rupture strength (URS) at least equal to full RCPB pressure. The design provisions provided reduce the possibility of an intersystem loss of coolant accident (ISLOCA) and consequently the probability of a loss of coolant accident outside the containment being an initiating event that could lead to core damage.

Reactor Protection System (RPS) / Control Rod Drive (CRD) System

The ABWR has a highly reliable and diverse CRD scram system incorporating both hydraulic insert and electric run-in capabilities. The control rod drive system utilizes hydraulic pressure as the principal scram mechanism with electric run in capabilities for backup to the hydraulic scram capabilities. The hydraulic scram system also includes additional backup scram valves to relieve scram air header pressure thereby causing the control rods to insert. Redundant and diverse scram signals are provided from the RPS and Alternate Rod Insertion (ARI) System to the hydraulic scram mechanisms and the electric run-in capability. The RPS is a four division system based on a two-out-of-four initiation logic. The ARI System is two-out-of-three initiation logic based on output signals from the Recirculation Flow Control System. This redundant, and diverse scram capability significantly reduces the probability of an ATWS.

Automatic Standby Liquid Control System (SLCS) and Recirculation Pump Trip

The ABWR has a highly reliable and diverse scram system incorporating both hydraulic and electric run-in capabilities to reduce the probability of an ATWS. In the unlikely event of an ATWS, the standby liquid control system and recirculation pump trip provide backup reactor shutdown capability. Automatic initiation of the SLCS avoids the potential for operator error associated with manual SLCS initiation and further reduces the already low probability of an ATWS leading to core damage.

Three Divisions of Engineered Safety Features (ESF)

There are three independent and separated divisions of ESF, each containing both high and low pressure emergency core injection and decay heat removal systems. Providing three complete divisions of ESF substantially reduces the calculated CDF for events that require ESF. The integrity of divisions is important. The high pressure or high temperature piping lines should not penetrate walls or floors separating two different safety divisions. Piping penetrations should be qualified to the same differential pressure requirements as the walls or floors they penetrate.

Automatic Depressurization System (ADS)

The Automatic Depressurization System provides a highly reliable means of depressurizing the reactor in the event of failure of the high pressure injection systems. This permits core cooling with low pressure systems, avoids high pressure core melt sequences, and substantially reduces the calculated CDF.

Three Emergency Diesel Generators (EDG)

There are three independent and separated EDGs, one dedicated to each of the three ESF divisions and each capable of powering the complete set of normal safe shutdown loads in its division. This configuration provides redundant sources of emergency AC power as added defense against loss of offsite power events. Three EDGs, each capable of powering a complete set of normal safe shutdown loads, substantially reduces the calculated CDF for events that require emergency AC power.

Four Divisions of Safety System Logic and Control (SSLC)

There are four divisions of self-tested safety system logic and control (SSLC) instrumentation designed on the basis of two-out-of-four actuation logic. This configuration provides highly reliable initiation of ESF core cooling and heat removal systems as well as actuating the CRD scram system for defense against ATWS events. A four division two-out-of-four SSLC provides protection against inadvertent actuation in addition to assuring the highly reliable actuation capability. This redundancy in the SSLC substantially reduces the calculated CDF for events that require SSLC signals as well as the reduction in unwanted system actuation resulting from inadvertent signals due to spurious inputs, surveillance and maintenance errors, and other causes of single signals.

Each microprocessor-based logic processing unit within the Essential Multiplexing System (EMS) and SSLC undergoes continuous self-test, with a reasonable certainty of fault detection. Undetected faults are identified during periodic (quarterly) surveillance testing, using the operator initiated, offline self-test feature available within each processing unit. This self-test function exercises all programmed logic and also causes outputs to toggle between untripped and tripped states. Faults are logged in each unit's self-test memory and are reported to the operator and process computer. The offline tests are expected to identify any faults not detected by the continuous self-test feature because more logic paths and trip states can be checked with reduced risk of spurious system actuation. This offline testing was judged to be important in the PRA analysis.

The administrative actions to avoid common-cause failures which are noted in Subsection 19.9.8 were also judged important.

HPCF Pumping Capability

In the discussion of Class II events where core cooling is successful but containment cooling is not (Subsection 19D.5.14), the ability of HPCF to pump 171° C (340° F) water is important.

Many features that are included in the level 1 model were determined to be less important than others in the context of these analyses. Several of these are identified in the following paragraphs.

The capability of the Reactor Water Cleanup (CUW) System to provide an additional means of decay heat removal with the reactor at high pressure was judged to be less important than the features selected as "important features." The additional redundancy provided by this capability does not significantly reduce the calculated ABWR CDF. This is due to the high reliability of other means of decay heat removal such as the various modes of operation of the three RHR loops and the containment overpressure protection system which result in a very small contribution of Class II sequences to total CDF without the CUW capability.

The degree of redundancy in SRVs to perform the ADS function was also judged to be less important than other features. Only three SRVs are required to open to depressurize the reactor so that low pressure pumps can provide the necessary cooling. The eight ADS SRVs plus the remaining ten SRVs that can be manually actuated far exceed redundancy requirements for depressurization. ADS failure is dominated by common cause failure of the ADS valves.

Another feature judged to be less important than other features is the automatic initiation of RHR on suppression pool high temperature. Many hours are available to initiate RHR to remove heat from the suppression pool following transients that dump heat to the suppression pool. The reliability of operators to manually initiate this function when required is judged to be very high, therefore this automatic initiation feature does not significantly reduce the calculated CDF.

The capability to manually initiate scram was judged to be less important than the selected features. The ability to manually initiate scram is not an important feature from the standpoint of CDF due to the highly reliable, redundant, and diverse features of the reactivity control systems.

The capability to use the CRD hydraulic system to provide additional water injection into the core was judged to be less important than the selected features. This is the primary reason that numerical credit for this function was not taken. The valve in some sequences for the coolant injection capability of the CRD pumps is of a lesser importance since adequate core cooling is available from other sources to assure a very low core damage frequency.

It was also judged that the high drywell pressure signal for ADS was less important than the selected features. With the incorporation of the drywell high pressure signal bypass timer, the high drywell pressure signal for ADS is less important.

19.8.2 Important Features from Seismic Analyses

19.8.2.1 Summary of Analysis Results

A seismic margins analysis has been performed for the ABWR to calculate a high confidence low probability of failure (HCLPF) acceleration for important accident sequences and classes of accidents. The results of the analysis indicate that all hypothesized accident sequences and all accident classes had HCLPFs equal to or greater than 1.67 times SSE.

Two implicit assumptions in the seismic margins analysis are that a seismic event will result in the unavailability of offsite power and the combustion turbine generator (CTG). The ceramic insulators in the switchyard are not tolerant of high seismic loads and therefore are assumed to fail. Also, the CTG is not qualified for seismic loads and is assumed to be unavailable in a seismic event. Therefore, all of the seismic analyses assume that only emergency AC power and DC power are potentially available.

19.8.2.2 Logical Process Used to Select Important Design Features

The seismic margins analysis did not include the calculation of minimal cutsets which contribute to CDF. Therefore, there was no calculation of importance parameters such as Fussell-Vesely or Risk Achievement. Since importance parameters were not available, two alternate bases were used to select the important features. The first basis used was the identification of the functions and equipment whose failure would result in the shortest path to core damage in terms of the number of failures required and the relative seismic capacities of the components involved. The second basis used was the identification of the most sensitive functions and equipment in terms of the effect on accident sequence and accident class HCLPFs due to potential variations of component seismic capacities. Using these two bases, the seismic margins analysis was systematically reviewed to identify the “important” features.

19.8.2.3 Features Selected

Table 19.8-2 lists the features selected and the rationale for selection. These features met the criteria of either the shortest path to core damage or the most sensitive components.

Shortest Paths to Core Damage

It is assumed that the failure of any Category I structure leads directly to core damage. The structures with lowest HCLPFs are the containment and the reactor building. It is important that HCLPFs for Category 1 structures not be compromised by future modifications or additions that could affect safety equipment.

Seismic failure of DC power also is assumed to lead directly to core damage. Without DC power, all instrument and equipment control power is lost and the reactor cannot be controlled or depressurized. In the seismic margins analysis it is assumed that this results in a high pressure core melt. The limiting components for DC power are the batteries and the cable trays.

It is possible that a large seismic event could impair the ability to scram due to deformation of the channels that enclose each fuel bundle. In the event that the scram function is impaired, the only means of reactivity control would be the Standby Liquid Control (SLC) System. Seismic failure of the SLC system to insert borated solution into the reactor is controlled by the seismic capacity of the SLC pump and the SLC system boron solution tank.

Emergency AC power and plant service water were both treated as having the same effects in the seismic margins analysis. Failure of either system would require only one additional failure to result in core damage. The limiting components for seismic failure of emergency AC power are the diesel generators, transformers, motor control centers, and circuit breakers. The limiting components for seismic failure of plant service water are the service water pumps, room air conditioners, and the service water pump house.

Most Sensitive Components

The HCLPFs of the accident sequences with the lowest HCLPFs could be increased by increasing the individual HCLPFs of the AICWA pumps, the fuel channels, or the RHR heat exchangers. The HCLPFs of the appropriate accident sequences would be increased by an amount equal to the increase in the HCLPF of any of these components.

The only single item that could, by itself, decrease the HCLPF of any accident sequence below the acceptance criteria is a Category I structure having a HCLPF below 1.67 times SSE. This would also decrease the HCLPF of accident class IE; ATWS with high pressure melt due to loss of inventory.

The only system that could, by itself, result in lowering an accident sequence HCLPF below the acceptance criteria is DC power. DC power has two components that could fail the sequence—the batteries and the cable trays.

AC-Independent Water Addition (ACIWA)

The ACIWA provides a diverse capability to provide water to the reactor in the event that AC power is not available and is important in preventing and mitigating severe accidents. The system has a diesel driven pump with an independent water supply and all needed valves can be accessed and operated manually. In addition, support systems normally required for ECCS operation are not required to function for ACIWA operation. The ACIWA can provide either vessel injection or drywell spray in the event all AC power is unavailable. Although the system pumps are housed in an external building (shed), the collapse of the building would not prevent the diesel driven pump from starting and running. The fire truck provides a backup to these pumps.

Seismic Walkdown

In addition to the above identified features, it was judged important that the seismic walkdown noted in Subsection 19.9.5 be conducted to seek seismic vulnerabilities.

19.8.3 Important Features from Fire Analyses

19.8.3.1 Summary of Analysis Results

An ABWR fire risk screening analysis based on the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology was performed to assess vulnerability to fires within the plant. Each scenario evaluated was calculated to have an acceptably low core damage frequency.

19.8.3.2 Logical Process Used to Select Important Design Features

The screening criterion for EPRI's FIVE methodology provided the primary basis for systematically evaluating important design features. The FIVE methodology provides procedures for identifying fire compartments for evaluation purposes, defining fire ignition frequencies, and performing quantitative screening analyses. The criterion for screening acceptability and dismissal from any more detailed consideration is that the frequency of core damage from any postulated fire be acceptably low.

Five bounding fire scenarios and corresponding ignition frequencies were developed on the basis of the FIVE methodology. Each scenario was calculated to have a core damage frequency less than the agreed upon criteria and hence screened from further consideration. Validity of these outcomes is contingent upon specific assumptions regarding the design features and performance capabilities of structures and equipment.

Consequently, the study was systematically reviewed to identify those procedures, assumptions, and features which are necessary in the fire risk assessment analysis to achieve core damage frequencies less than the screening criteria and thus pass the FIVE methodology screen.

19.8.3.3 Features Selected

Table 19.8-3 lists the features selected and the basis for each feature being considered important. These features are those necessary to maintain fire initiated core damage frequencies below the screening criterion. The proper functioning of these features assures the capability to mitigate the postulated fires. Features identified as a result of the review of the Level 1 internal events analysis are also important in the fire analysis but they are not included here unless they have some fire unique significance.

Fire Detection and Suppression

The principal function of the Fire Protection System (FPS) is fire detection and suppression. It must be demonstrated that safe shutdown of the ABWR can be achieved, assuming that all equipment in any one fire area has been rendered inoperable by fire and that reentry to the fire area for repairs and for operator action is not possible. Divisional separation is provided by three hour fire barriers to contain the fire within the division. Fire detection systems include infrared sensors as well as product-of-combustion type smoke detectors. Automatic fire suppression systems include foam and sprinklers. Manual fire fighting methods use hand held fire extinguishers and water hoses. Fire detection and suppression systems are provided throughout the plant and FPS actuation is alarmed in the control room. Since the primary Containment is inerted during normal plant operation, no FPS system functions are provided in this area.

Remote Shutdown Panel and RCIC and SRV Operation from Outside the Control Room

The dominant contributor to core damage was found to be the potential for a control room fire leading to abandonment of the area and requiring control of the plant from outside the control room. This finding identified the Remote Shutdown Panel as an important feature. Core damage frequency, as initially evaluated by the FIVE methodology, for control room fires was over two orders of magnitude greater than that predicted for a divisional electrical fire, and did not pass the FIVE methodology screening criterion. The Remote Shutdown Panel provides capability to shut down the reactor that is physically and electrically independent from the control room. However, initially the Remote Shutdown Panel had the capability for operating only one loop of high pressure injection (HPCFB) and only three safety relief valves for depressurization. Either the HPCFB or the successful operation of all three SRVs was required to prevent core damage in the event of a fire which led to abandonment of the control room.

Potential courses of action to reduce the risk from control room fires included providing redundancy for depressurization by providing control for a fourth SRV at the remote shutdown panel and redundancy and diversity for high pressure injection by providing the capability to operate the RCIC system from outside the control room. The CDF impact of each of these two options was evaluated by the FIVE methodology. Neither option by itself provided sufficient reduction in the CDF to meet the 1E-6 risk screening criterion. In combination, however, the fire risk screening criterion was met. With the incorporation of both options into the capability of the Remote Shutdown Panel, a CDF of less than the screening criteria was demonstrated for the ABWR.

Divisional Separation of ESF and Support Systems

ABWR fire core damage frequency less than the screening criterion was demonstrated based upon the design feature that safety divisions, including necessary support systems, are isolated from each other by three hour rated fire barriers. The divisional separation requirement extends to and includes the intake structure. This includes fire barriers formed by concrete fire barrier floors, ceilings, and walls; partitions; rated fire doors; penetration seals for process pipes and cable trays; special assemblies and constructions; and fire dampers. In addition, the fire analysis assumes the routing of piping or cable trays during the detailed design phase will conform with the fire area divisional assignment documented in the fire hazard analysis. This design feature assures that the routing of piping or cable trays will not invalidate the requirement that all safety divisions are separated by three hour fire barriers. Subsection 9A.5.5 under “Special Cases—Fire Separation for Divisional Electrical Systems” lists the only areas of the plant where there is equipment from more than one safety division in a fire area. These should be the only areas where multiple divisions share the same fire area.

Smoke Control System

The EPRI FIVE methodology does not directly address the migration of smoke, and its impact is not explicitly estimated in the fire assessment. However, it is implicit in the analysis that the smoke control system will limit the spread of smoke and hot gasses, and fire suppressant between safety divisions to the extent that damage is limited to equipment in the division in which the fire started. SECY 93-087 and SECY 90-016 identify as important the prevention of the spread of smoke, hot gasses, and fire suppressant from migrating from one division to another to the extent that they cannot adversely affect safe shutdown capabilities, including operator actions. It is assumed in the fire analysis that the smoke control system is capable of preventing the migration of smoke or hot gasses between divisions with an open door between the division experiencing the fire and another division to the extent that they cannot adversely affect safe shutdown capabilities, including operator actions. Since this is an implicit assumption in the FIVE analysis and has been identified as NRC guidance in SECY 90-016 and SECY 93-087 as elements to resolve fire protection concerns, the control of smoke, hot gasses, and fire suppressant is considered an important design feature for fire protection.

If there is a fire in the secondary containment that results in the loss of the HVAC system due to one of the valves at the common HVAC supply or exhaust failing to close, hot gasses will migrate upward in the building through pipe chases and HVAC ducts. Safety-related equipment will continue to operate since they are at the lower levels of the secondary containment and the smoke will migrate away from the lower levels and room coolers will maintain temperature in the subcompartments within acceptable limits. Entrances to the secondary containment are at or near grade, therefore, fire fighting personnel can enter at this level to fight a fire and take any other actions necessary even if one of the common HVAC valves fail to close.

19.8.4 Important Features from Suppression Pool Bypass and Ex-Containment LOCA Analyses

19.8.4.1 Summary of Analysis Results

Suppression pool bypass pathways, potential pathways for the release of radioactive material which do not receive the benefits of suppression pool scrubbing, were evaluated. The evaluation included an analysis of the probability of individual bypass pathways existing at the time of a core damage event and the consequence of each path as estimated by the amount of flow accommodated by the pathway. These factors were multiplied to obtain a “bypass fraction” which is a measure of risk.

Ex-containment LOCAs that bypass the suppression pool were evaluated based on simplified event trees. The total calculated CDF for these LOCAs is extremely small.

19.8.4.2 Logical Process Used to Select Important Design Features

The bypass fraction was used to verify that bypass paths contribute a small percentage of the total offsite risk from internal event sequences and therefore do not present an undue offsite risk. A numerical goal was established based on comparison of offsite exposures with and without a full suppression pool bypass. The features that contribute to the prevention or mitigation of containment bypass were systematically reviewed to evaluate their specific contribution to containment bypass. The selection basis used to determine the important features that prevent or mitigate containment bypass was to consider features which, if they were not included in the design, could increase the total bypass fraction above the numerical goal.

The core cooling features that could prevent or mitigate containment bypass were systematically reviewed to determine their contribution to total CDF. Those features that would increase the calculated CDF by more than a factor of 2 if they failed or were not included in the design were identified as important features.

19.8.4.3 Features Selected

Table 19.8-4 lists the features that were identified as important to prevent or mitigate suppression pool bypass events and ex-containment LOCAs. The basis for the selection of the feature is noted in the table. For some cases, the change in the bypass fraction if the feature were to fail is discussed below.

DW-WW Vacuum Breakers

Assuming an event leads to pressurization of the wetwell to the extent that the containment rupture disk opens, the vacuum breakers would open and then close thereby isolating the drywell from the wetwell. Failure of a DW-WW vacuum breaker to close following the assumed event would provide a significant bypass from the drywell into the wetwell airspace. If the rupture disk is open and one of the vacuum breakers has not closed there would be a direct pathway from the drywell to the wetwell and to the environment. The consequence of a vacuum breaker failing to close was evaluated in the PRA. The total bypass fraction was calculated if a vacuum breaker failed to close.

Redundant MSIVs

There are four main steamlines (MSL), each with two in-series automatic isolation valves. The MSIVs are a pneumatic operated, spring close, fail-closed design actuated by redundant solenoids through two-out-of-four logic. If both MSIVs in any one MSL fail to close there will be a large bypass pathway from the RPV to the Turbine Building. The potential bypass pathway is large compared to other potential bypass pathways. Therefore, the failure of two MSIVs to close in any one steamline would result in a higher consequence from a given postulated event. Although it is extremely unlikely, it is possible that two MSIVs in the same steamline could fail to close and, depending on the event, the failure could result in a substantial offsite dose consequence.

Design and Fabrication of the SRV Discharge Lines

The discharge of the SRVs are piped through the drywell and the wetwell airspace to the suppression pool which is inside the wetwell. To ensure the integrity of the SRV discharge lines, especially in the wetwell region, these lines are designed and fabricated to Quality Group C requirements and the welds in the wetwell region above the surface of the suppression pool are non-destructively examined to the requirements of ASME Section III, Class 2. During an SRV discharge, a break in one of these lines in the wetwell airspace could result in the pressurization of the wetwell and possibly result in the opening of the rupture disk. Although it is extremely unlikely, the failure of the SRV discharge line during operation of the SRV and the subsequent opening of the rupture disk would result in a pathway directly from the RPV to the environment. Depending on the event, the consequence of this postulated sequence could be a substantial increase in the offsite dose consequence.

Normally Closed Sample Lines and Drywell Purge Lines

The sample lines and drywell purge lines are normally closed during plant power operation. Although the valves in the sample and drywell purge lines are normally closed in order to limit the risk of bypass, if one or more of these lines are open when an event initiates a potential bypass path can exist. Depending on the event and the size and number of lines open, a substantial fission product release could result in a significant increase in the consequences of a given event.

Cleanup System Isolation Valves

In the event of a break in the CUW system, it is important that the break be isolated. The probability of not isolating a CUW line break outside containment is very low due to inclusion of a remote manual shutoff valve (in addition to the two automatic isolation valves). Even though the exposed structures and ECCS equipment are designed for the loads and environment which could follow from an unisolated break, there is some potential for failure. Further, there is some potential for the operator not properly controlling reactor vessel water level during the recovery phase (Subsection 19.9.1).

Blowout Panels in the RCIC and CUW Divisional Areas

Blowout panels are provided in the RCIC and CUW divisional areas to prevent overpressurization. Failure of the blowout panels during an ex-containment LOCA due to a break in a RCIC or CUW line could result in the pressurization of a divisional area that could impact equipment in an adjacent area and result in a second electrical division being unavailable. This impacts the core damage frequency for ex-containment LOCAs. If a break in one of these areas caused such an impact, the core damage frequency for bypass events could be increased.

Several plant features treated in the analysis were judged much less important than those discussed above. These are noted in the following paragraphs.

Piping dimensions are judged to be less important to suppression pool bypass evaluations than other features. The flow split fraction is determined by design dimensions of the plant such as piping size and length. While important in the evaluation of suppression pool bypass, the evaluation was based on conservatively low estimates of bypass path resistance. Consequently these features were not considered important within the context of the final system design. Only much larger piping sizes in identified pathways would be of concern, but significant variations are not considered likely.

The level of water in the suppression pool is considered less important than other features. Higher suppression pool level tends to increase the amount of flow which passes through a bypass pathway because of the increased resistance within the suppression pool path. This characteristic is less important to the results because the flow split fraction varies as the square root of the differential pressure and thus the suppression pool level. Since the suppression pool water level is limited by the return line elevation to 1.6m above the normal level of the suppression pool, the maximum effect on the bypass fraction is small.

The closing of the turbine bypass valve is considered less important than other features. If the MSIVs fail to close in one of the steamlines, the turbine bypass valve would normally be expected to close in response to the Turbine Pressure Control System after RPV pressure has reduced below normal operating pressure. Failure of this valve to close is one component of the definition of the main steamline bypass pathway. The feature is considered relatively unimportant in comparison with the reliability of MSIV closure.

The instrument check valves are also less important than other features. All instruments which sense RPV or containment parameters contain inline excess flow check valves to limit the release in the event of an instrument line break. However due to their small line size, even if the check valves fail to prevent excess flow, the total bypass fraction from instruments would only contribute a very small percentage of the total bypass fraction. Therefore this feature is considered of lesser importance to the results of the bypass evaluation.

Reliable seating of redundant Feedwater and SLC check valves and ECCS discharge check valves is considered to be of lesser importance than other features that prevent or mitigate suppression pool bypass. Because of the relatively large line size, failure of the redundant feedwater check valves can lead to a bypass if a break occurs in the feedwater, CUW or LPFL A return lines, both check valves fail to prevent full reverse flow and core damage occurs. Redundant SLC lines also result in a bypass path if the check valves fail to prevent reverse flow and a piping failure occurs. Failure of LPFL B or C discharge check valves could be significant if a break were to occur in the pump discharge. If all check valves failed to seat, the total bypass fraction could increase but still be below the goal. Therefore it can be concluded that the check valves are not important to the bypass evaluation as other features.

19.8.5 Important Features from Flooding Analyses

19.8.5.1 Summary of Analysis Results

The ABWR flooding analysis evaluated all potential flood sources and through the use of simplified event trees determined the CDF for each building of interest. The three buildings determined to have the potential for flooding to affect safety-related equipment are the Turbine, Control, and Reactor Buildings. The other buildings do not contain safety-related equipment and are not connected to buildings that do. Tunnels from each of these buildings which are routed to the radwaste building are sealed to prevent interbuilding flooding. Therefore, the interbuilding flooding probability through these tunnels was evaluated to be several orders of magnitude lower than direct flooding due to pipe breaks in each building and was not included in the event trees. The adequacy of the tunnel seals should be confirmed by the COL applicant. The CDF for events initiated by flooding in the Turbine Building is extremely small for a low power cycle heat sink (PCHS) and very small for a high PCHS. The CDF for events initiated by flooding in the Control Building is very small and the Reactor Building is extremely small. The estimated CDF for events initiated by flooding from all internal flood sources is very small for a low PCHS and for a high PCHS.

19.8.5.2 Logical Process Used to Select Important Design Features

The ABWR flooding probabilistic risk analysis used simplified event and fault trees to estimate the CDF due to postulated floods. This approach did not result in the calculation of the minimal cutsets which contribute to the CDF. Therefore, there was no calculation of importance parameters such as Fussell-Vesely or Risk Achievement. Therefore, the flooding analysis was systematically reviewed to identify important design features based on other factors. Since importance parameters were not available, the process used to determine the important features was the impact the feature would have on the results of the specific flood in question. If, by completing its function, the component either fully mitigated or prevented the flood or was required to allow some other component to mitigate the flood, then it was selected. Other features, such as sump pumps, that could mitigate some floods but could be backed up by other features were not selected.

19.8.5.3 Features Selected

Table 19.8-5 lists the features selected and the basis for each feature being considered important. These features met the criteria of either mitigating or preventing flooding or were required to allow some other feature to mitigate flooding.

Physical Separation of the Three Safety Divisions

The three safety division are physically separated by fire rated walls and floors. These walls and floors are also effective flood barriers. Entrances to rooms containing safety-related equipment on the first floor of the reactor and control buildings also have watertight doors. Watertight doors are also on all below-grade entrances to the reactor and control buildings from the service building. Cables penetrating the divisional rooms are sealed to prevent the propagation of fires. These seals are pressure tested and thus also serve as flood barriers.

Floor Drains

The reactor and control buildings are designed to mitigate potential flooding by diverting all flood waters to floors which contain sump pumps by the use of floor drains. The floor drains are sized to handle the largest potential flood source on the upper floors which is the fire protection water system. The floor drains are sized to ensure that water levels on the upper floors will not accumulate to levels high enough to damage important equipment.

Water Level Sensors in the RCW/RSW rooms

Water level sensors are installed in the turbine building condenser pit and the RCW rooms in the control building. These sensors are used to detect flooding in the rooms and send signals to trip pumps and close isolation valves in the affected systems. The sensors are arranged in a two-out-of-four logic. The control building has two sets of sensors (lower and upper) which measure the water level using diverse means to eliminate the potential for common cause failures. The sensors also send signals to the control room to alert the operator to a potential flooding condition so that appropriate manual actions can be taken to isolate the flooding source.

B3F Corridor

The corridor of the Reactor Building first floor has a volume that is sufficient to contain the largest Reactor Building sources which are the suppression pool and condensate storage tank (CST). Penetrations (except for water tight doors) in the divisional walls are at least 2.5 m above the floor level of 8200 mm. The corridor has two sump pumps but the analysis conservatively assumes that the sump pumps do not operate.

Anti-siphon Capability

The reactor service water (RSW) system contains anti-siphon capability (e.g., vacuum breakers, air break) to stop flooding in the event of a break in a RSW line in the reactor component cooling water (RCW) rooms in the control building. The anti-siphon capability will terminate RSW flow if the RSW pumps are tripped but the isolation valves in the affected division fail to close. The anti-siphon capability applies to both the RSW supply and return lines from/to the ultimate heat sink.

Motors and Motor Control Centers

Motors will be drip proof and MCCs will have NEMA Type 4 enclosures.

Ultimate Heat Sink

The ultimate heat sink will be designed such that water cannot gravity drain to the control building in excess of the allowed 4000 meters of RSW pipe from the isolation valves in the pump house (2000 meters each for supply and return).

RSW System

A maximum of 4000 meters of RSW piping is allowed between the RSW isolation valves at the pump house and the control building (2000 meters each for supply and return).

Overfill Lines in B1F Sump

The sumps on floor B1F of the reactor building contain overfill lines that are connected to the first floor of the reactor building (B3F). These overfill lines are designed to direct water to the first floor in the event that the sump pumps fail or cannot keep up with the flood rate. The lines penetrate secondary containment so water loop seals are included to maintain the integrity of the secondary containment.

Floods Originating in Turbine, Control, and Reactor Buildings

The screening analysis indicated that the flooding analysis only needed to address internal flooding from sources in the Turbine, Control, and Reactor Buildings. Other buildings do not contain equipment that can be used to achieve safe shutdown and flooding in those buildings cannot propagate to buildings which contain safe shutdown equipment. Although flooding originating in the Turbine Building could propagate through the Service Building and potentially enter the Control or Reactor Buildings if watertight doors fail or are left open, the analysis does not consider flooding to originate in the Service Building. The analysis addresses the potential for propagating of flooding through the Service Building.

Operator Check Watertight Doors are Dogged

The flooding analysis assumes that all watertight doors are closed and dogged to prevent floods from propagating from one area to another. The watertight doors are alarmed to alert security personnel that a watertight door is open but will not alarm to indicate that a door is not dogged. To guard against a door being left undogged, operators should check the doors every shift to assure that they are closed and dogged.

High Pressure or High Temperature Lines Not Routed Across Divisions

The flooding analysis assumes that high pressure or high temperature lines are not routed through floors or walls separating two different safety divisions. Piping penetrations are qualified to the same differential pressure requirements as the walls and floors they penetrate. This prevents the possibility of a system failure in one division from flooding and failing a different division.

19.8.6 Important Features from Shutdown Events Analyses

19.8.6.1 Summary of Analysis Results

A shutdown analysis was completed to evaluate the potential for core damage during shutdown (i.e., Modes 3, 4, and 5). The analysis focused on five areas identified by the NRC in NUREG 1449, "Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States", as having potentially high shutdown risk based on past experience with operating plants. The five areas are

(1) Decay heat removal

(2) Inventory control

(3) Containment integrity

(4) Reactivity

(5) Electrical power

Decay heat removal was evaluated probabilistically. The other areas were treated in a qualitative manner. A simplified maintenance model was used to calculate the core damage frequency (CDF) for loss of the operating decay heat removal pump, assuming certain minimum sets of available systems during shutdown. The assumption was that only these minimum sets and support systems were available and other systems were in maintenance. No credit was assumed for these other systems. In practice, not all of these other systems are expected to be in maintenance at the same time.

Several minimum sets were identified which met the CDF criterion. Many other minimum sets could have been evaluated, as well as other system configurations for shutdown conditions. A COL applicant will be able to choose from the configurations evaluated in this study or evaluate other configurations to show compliance to the shutdown CDF criterion.

19.8.6.2 Logical Process Used to Select Important Design Features

The analysis systematically evaluated potential risks during shutdown. Maintenance activities during shutdown result in more systems being unavailable than during normal operation. The simplified maintenance model assumed many systems were undergoing maintenance at the same time. Since systems are artificially assumed to be out of service and because of the way the analyses were structured, computing importance parameters such as Fussell-Vesely would not result in any meaningful conclusions. Therefore, the shutdown risk study did not lend itself to a quantitative evaluation of the importance of ABWR components for loss of decay heat removal during shutdown.

Since no quantitative measures are calculated to determine the importance of components associated with shutdown risk, the following qualitative basis was used. A component was considered to be “important” for a specified shutdown risk “category” (i.e., the five areas identified in NUREG 1449) if it was capable of preventing or mitigating identified shutdown accident scenarios associated with that category. Using this qualitative basis, the shutdown analysis was systematically reviewed to identify important design features. For example, isolation of the RPV on low water level mitigates loss of inventory control, so it was selected. The condenser hotwell, while it could be used for makeup during shutdown, was not selected because it will more than likely not be available due to maintenance and other systems such as the Residual Heat Removal (RHR) system in the low pressure core flood mode (LPFL) or high pressure core flooder (HPCF) are typically available for inventory control.

19.8.6.3 Features Selected

Table 19.8-6 lists the features selected as important for each category evaluated along with the reason the feature is important. The list includes both active (e.g., RHR pumps) and passive (e.g., shutdown cooling (SDC) nozzle above TAF) features.

Decay Heat Removal

Three features were selected for events involving loss of decay heat removal: RHR shutdown cooling (SDC), Reactor Service Water (RSW), and the Ultimate Heat Sink (UHS). The RHR system was selected because it is the preferred and normally used method of decay heat removal during shutdown. The three RHR divisions allow for one division to be in maintenance and a single failure in the operating division. The third division could then be used to cool the core. The RSW and the UHS were selected because of their fundamental support functions for all the decay heat removal systems. The Reactor Water Cleanup (CUW) System was not selected because it cannot remove all the decay heat by itself until several days following shutdown. Even then, two divisions of CUW are required which means that two divisions of RSW and RCW are also required.

The shutdown study concluded that boiling was an effective, although not preferred, method of decay heat removal for all modes including Mode 5 with the RPV head removed. In this case, injection systems such as HPCF are considered to be decay heat removal systems as they function to keep the core covered. Since these systems are primarily used for inventory control, they are included in that category.

Inventory Control

Four injection systems were selected: RHR(LPFL), CRD, HPCF, and AC-independent water addition (ACIWA). All of these systems are capable of ensuring that the core remains covered. Use of RHR(LPFL) and ACIWA require depressurization if the RPV pressure is high. The other features selected under Inventory Control either prevent or mitigate RPV drain down scenarios. Closure of all valves in lines connected to the RPV on low water level ensures that the core is not uncovered due to breaks in lines connected to the RPV or diversion of water from the RPV by the RHR or CUW systems. The permissives and inhibits associated with the RHR mode switch ensures that the proper valve line up is used for various modes of RHR operation. This minimizes the potential for diversion of water from the RPV. The RHR interlocks ensure that the low pressure RHR piping connected to high pressure systems is not inadvertently exposed to high pressure which could result in a LOCA. RPV level sensors inform the operator of the RPV level and actuate systems such as HPCF and RPV valve isolation to ensure that the core remains covered. A plug installed on the RIP diffuser during maintenance ensures that reactor coolant cannot leak out the RIP housing when the RIP motor, shaft, and bottom cover plate are removed.

Reactivity Control

Three reactivity control features were selected: RPS high flux trip (set down), CRD brake, and refueling interlocks. The RPS high flux trip (set down) protects the core from inadvertent power excursions during shutdown by inserting any withdrawn control rods if the power level reaches a preselected setpoint. The CRD brake prevents ejection of a CRD blade which could result in excessive power and core damage. When in the REFUEL mode, refueling interlocks prevent hoisting another fuel assembly over the RPV if a CRD blade has been removed.

Containment Integrity

Containment integrity during Mode 3 and in part of Mode 4 is preserved by automatic isolation of secondary containment on a high radiation signal. This will prevent or at least delay a potential release of radioactivity to the environs. The standby gas treatment system (SGTS) can function to process gasses before release to the atmosphere to reduce potential contamination.

Electrical Power

The features selected for electrical power include the three divisions of safety-related power physically and electrically independent, the four sources of onsite power (3 emergency diesel generators (EDGs) and the combustion turbine generator (CTG)), and the two independent offsite power sources. The electrical power systems include redundancy and diversity of sources. This allows some power sources to be in maintenance during shutdown and still have adequate sources to provide power when needed. Even if all offsite power is lost, the four onsite power sources can be used to power any safety or non-safety bus. This means that the ABWR can use alternate sources of decay heat removal (e.g., condensate pump) with only onsite power sources.

19.8.7 ABWR Features to Mitigate Severe Accidents

The ABWR has been designed to prevent the occurrence of a core damage accident. In fact, the probability of a core damage accident is extremely small. This represents an improvement in severe accident prevention when compared to current plants. In the extremely unlikely event of a core damage accident, the ABWR containment has been designed with specific mitigating capabilities. These capabilities not only mitigate the consequences of a severe accident but also address uncertainties in severe accident phenomena. The capabilities are listed below along with a discussion of the specific severe accident phenomena that the mitigation device is addressing. The severe accident issues addressed are consistent with the issues discussed in SECY 90-016.

AC-Independent Water Addition System

This system not only can play an important role in preventing core damage, it is the primary source of water for flooding the lower drywell should the core become damaged and relocate into the containment. The primary point of injection for the system is the LPFL header inside the vessel. Flow can also be delivered through the drywell spray header to the upper drywell. The drywell spray mode of this system not only provides for debris cooling, but it is capable of directly cooling the upper drywell atmosphere and scrubbing airborne fission products. This system has sufficient capacity to cover the core debris ex-vessel and provide debris cooling and scrub fission products released as a result of continued core-concrete interactions.

The system operating in the drywell spray mode will also reduce the consequences of a suppression pool bypass or containment isolation failure. This is due to the fission product removal function performed by this mode of operation. Fission products will be scrubbed by the sprays prior to leaving the containment.

The system has been sized to optimize the containment pressure response and slow the rate of containment pressurization. The system is capable of delivering water to the containment up to the setpoint pressure of the COPS system. The flow rate, nominally 0.06 m^3/s with no containment back pressure and 0.04 m^3/s at the COPS setpoint, is sufficient to allow cooling of the core debris, while maximizing the time until the water level reaches the bottom of the vessel, at which point it is turned off.

Lower Drywell Flooder

The lower drywell flooder system has been included in the ABWR design to provide alternate cavity flooding in the event of core debris discharge from the reactor vessel and failure of the AC-Independent Water Addition System. The thermally activated flooder valves are actuated by the melting of a fusible plug. The temperature set point for the plug is 533 K. The system consists of ten lines located about 4 m below the normal suppression pool water level discharging into the lower drywell about 1 m above the floor. The expected flooder flow is 10.8 kg/s per valve. Only two of the valves are required to open to remove decay heat energy and the energy from zirconium-water reaction and allow for quenching of the debris. The passive flooder will not open until after vessel failure. By flooding after the introduction of core material, the potential for energetic core-water interactions during debris discharge is minimized. The flooder will cover the core debris with water providing for debris cooling and scrubbing any fission products released from the debris due to core-concrete interactions.

Containment Overpressure Protection

The COPS consists of overpressure relief rupture disks mounted in a line which connects the wetwell airspace to the stack. This system will provide for a scrubbed release path in the event that pressure in the containment cannot be maintained below the structural limit. The system includes two reclosable valves which may be used to re-establish containment isolation as a part of post accident recovery. These valves should be normally open and be designed to fail open.

This controlled release will occur at a containment pressure of 0.72 MPa. The outer rupture disk of the COPS has a rupture differential pressure of less than 0.03 MPa. The setpoint of the COPS system is based on the competing goals of minimizing the probability of containment structural failure and maximizing the time of any fission product release. The setpoint was assumed to be reliable to within ±5% of the actuation pressure at nominal temperature. The effect of temperature on the rupture disk should be small, the analysis assessed the variability to be small.

The area of the rupture disk is designed to permit the COPS system to be effective in mitigating the pressure increase during an ATWS event in which the operator controls the injection flow. The minimum capacity of the COPS is 28 kg/s steam flow when the containment is at the actuation pressure. This provides ample margin to steam generation rates related to decay heat generation. Analysis of the blowdown of the containment following rupture disk operation indicates that the pool swell and the blowdown loads will not threaten the piping, and that significant entrainment will not occur.

This system is beneficial for several of the severe accident issues. In cases with continued core-concrete attack, or those with no containment heat removal operational, the containment will pressurize. The COPS provides a controlled release path preventing containment structural failure and mitigating fission product release. The COPS system reduces the effect of uncertainties in severe accident behavior, e.g. debris coolability, in the ABWR design.

Vessel Depressurization

The ABWR reactor vessel is designed with a highly reliable depressurization system. The nitrogen supply and battery capacity are sufficient to allow depressurization after RCIC failure during a long-term station blackout. This system plays a major role in preventing core damage. However, even in the event of a severe accident, the RPV depressurization system can prevent the effects of high pressure melt ejection. If the reactor vessel would fail at an elevated pressure, fragmented core debris could be transported into the upper drywell. The resulting heatup of the upper drywell could pressurize and fail the drywell. Parametric analyses performed in Attachment 19EA indicate that even in the event of direct containment heating, the probability of early drywell failure is low. The RPV depressurization system further decreases the probability of this failure mechanism.

Lower Drywell Design

The details of the lower drywell design are important in the response of the ABWR containment to a severe accident. Seven key features are described below.

(1) Sacrificial Concrete

The floor of the ABWR lower drywell includes a 1.5 meter layer of concrete above the containment liner. This is to ensure that debris will not come in direct contact with the containment boundary upon discharge from the reactor vessel. This added layer of concrete will protect the containment from possible early failure.

(2) Basaltic Concrete

The sacrificial concrete in the lower drywell of the ABWR will be a low gas content concrete. The selection of concrete type is yet another example of how the ABWR design has striven not only to provide severe accident mitigation, but to also address potential uncertainties in severe accident phenomena. Here, the uncertainty is whether or not the ex-vessel core debris can be cooled by flooding the lower drywell. For scenarios in which water in the lower drywell is unable to cool the core debris, the concrete type selected has approximately 4 weight percent calcium carbonate which will result in a very low gas generation rate. This translates into a long time to pressurize the containment. This is important because time is one of the key factors in aerosol removal.

(3) Pedestal

The ABWR pedestal is formed of two concentric steel shells with webbing between them. The space between the shells is filled with concrete. The thickness of the concrete between the shells is 1.64 m. A parametric study of core concrete interaction was performed which indicated a very small potential for pedestal failure even in the event of continued interaction. Furthermore, any potential failure will not occur for approximately one day.

(4) Sump Protection

The lower drywell sumps are protected by corium shields such that core debris will not enter them. This maximizes the upper surface area between the debris and the water and maximizes the potential to quench the core debris. The shields are made of alumina which is impervious to chemical attack from core-concrete interaction. The walls of the floor drain sump shield have channels which permit water flow, but which will not permit debris flow. The equipment drain sump shield has no such channels. The height and depth of the shields has been specified to ensure that debris will not enter the sumps in the long term.

(5) Increased Floor Area

The floor area of the lower drywell has been maximized to improve the potential for debris cooling. The minimum lower drywell floor area of 79 m^2 meets the ALWR Utility Requirements Document criterion of 0.02 m^2/MWth.

(6) Wetwell-Drywell Connecting Vents

The flow area between the lower and upper drywell has been designed in a way to allow adequate venting of gases generated in the lower drywell. The connecting vents flow area is 11.25 m^2. This is important when considering the steam generation rates associated with fuel-coolant-interactions in the lower drywell. The interconnection between the lower drywell and the wetwell is at elevation -4.55 m, 8.6 m above the floor of the suppression pool. Thus, approximately 7.2E5 kg of water must be added from outside the containment for the pool to overflow into the lower drywell.

The path from the lower to the upper drywell includes several 90 degree turns. This tortuous path enables core debris to be stripped prior to transport into the upper drywell minimizing the consequences from high pressure melt ejection. Also important when considering high pressure core melt scenarios, the configuration of the connecting vents will result in the transport of some core debris directly into the suppression pool. This is preferable to transport into the upper drywell and would result in the debris being quenched with only a slight increase in the suppression pool temperature.

(7) Solid Vessel Skirt

The vessel skirt in the ABWR does not have any penetrations which would allow the flow of water from the upper drywell directly to the lower drywell. This, in combination with other design features described above, ensures a very low probability that water is in the lower drywell before the time of vessel failure. Thus, large scale fuel-coolant interactions are precluded.

Inerted Containment

One of the important severe accident consequences is the generation of combustible gasses. Combustion of these gasses could increase the containment temperature and pressure. The ABWR containment will be inerted during operation to minimize the impact from the generation of these gasses.

Containment Isolation

The ABWR containment design has striven to minimize the number of penetrations. This impacts the severe accident response due to a smaller probability of containment isolation failure. All lines which originate in the reactor vessel or the containment have dual barrier protection which is generally obtained by redundant isolation valves. Lines which are considered non-essential in mitigating an accident isolate automatically in response to diverse isolation signals. Lines which may be useful in mitigating an accident have means to detect leakage or breaks and may be isolated should this occur.

Upgraded Low Pressure Piping

The low pressure piping in the ABWR has been upgraded to withstand higher pressure. This reduces the probability of an interfacing system LOCA and the severe accident consequences associated with such an event.

Drywell-Wetwell Vacuum Breakers

The ABWR contains eight vacuum breakers which provide positive position indication in the control room. Failure of the vacuum breakers to close as designed can potentially lead to increased source terms and early containment failure. The vacuum breakers have been located high in the wetwell to reduce potential loads occurring during pool swell. The analysis in the PRA assumes that the position switch which provides annunciation in the control room can sense a gap between the disk and the seating surface greater than 0.9 cm. Additionally, the vacuum breakers will be tested during periodic outages to ensure operability. The result of the vacuum breaker design in the ABWR is to reduce the potential for suppression pool bypass.

Residual Heat Removal System

The RHR system is the primary mechanism for the removal of decay heat from the containment. This system is capable of pumping saturated water up to the pressure of the COPS setpoint. Recovery of a single loop of RHR is adequate to remove decay heat in the long term. The RHR system also has a drywell spray functions which may be important in preventing high temperature failure of the containment in an accident in which debris is entrained to the upper drywell. The wetwell spray may be used to mitigate the effects of suppression pool bypass.

Overall Containment Performance

The design of the ABWR containment provides for holdup and delay for fission product release should the containment integrity be challenged. The design basis containment leak rate is 0.5% per day at containment design pressure. Leakage is expected to be of this magnitude in a severe accident. Long term containment pressurization is governed by the generation of decay heat and non-condensable gases. The primary source of non-condensable gas generation is metal-water reaction of the zirconium in the core. This is accommodated by a relatively large containment volume and a high containment pressure capability. The mitigating systems discussed above ensure that the decay energy results in steam production. The suppression pool absorbs this energy, resulting in very slow containment response which ensures ample time for fission product removal.

The containment strength was evaluated. The limiting structure is the drywell head. Service Level C was found to be greater than 0.77 MPa. This is adequate to withstand the generation of 100% metal water reaction. The median ultimate strength of the containment was found to be 1.025 MPa. Ultimate strength capability is important for very rapid containment challenges such as direct containment heating and rapid steam generation. Evaluation of both these phenomena indicate early containment failure from these mechanisms is unlikely.

Key Severe Accident Modeling Parameters

Table 19.8-7 provides a list of key severe accident modeling parameters. This list has been derived from the discussions presented above and from a variety of ABWR severe accident evaluations.

Table 19.8-1 Important Features from Level 1 Internal Events Analyses

Feature	Basis
Capability to operate RCIC for two hours without AC power (2.4.4), and ability to override switchover to makeup water source from CST to suppression pool (2.4.4). This defines requirement for station battery capability to provide RCIC control power for two hours (2.4.4). The importance of RCIC unavailabilities are addressed in Subsection 19K.11.1.	This system with this capability provides the only means available to provide core cooling with the reactor at high pressure and avoid core damage in the event of a station blackout.
Combustion turbine generator connectable to any of the three safety divisions and capable of powering one complete set of normal safe shutdown load (2.12.11). No plant support systems are needed to start or run the CTG (2.12.11). Safety-grade loads are to be added manually (2.12.1, 2.12.11). Manual transfer of CTG power to the condensate pumps is discussed in Subsection 19.9.19.	Provides a diverse source of emergency AC power as added defense against loss of offsite power and diesel generator failure events.
Operability of one high pressure core flooder (HPCF) loop independent of essential multiplexing system (2.2.6).	Provides an independent and diverse means of initiating emergency core cooling in the event of postulated common mode failures in the digital safety system logic and control (SSLC).
AC-independent Water Addition System, including a dedicated diesel (2.15.6) and manually operable valves (2.15.6), to provide a diverse means of low pressure water injection into the reactor vessel.	Provides an independent and diverse means of achieving emergency core cooling in the event of station power loss or failure of the engineered safety features to provide this function.
Sufficient cooling capacity available in the RCW system to provide seal and motor bearing cooling for ECCS core cooling pumps with one RCW and one RSW system pump in each loop in each division and two RCW heat exchangers in each division operating.	The redundant capability in each RCW/RSW division to successfully support ECCS functions substantially lowers the calculated CDF.
All piping systems, major systems components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) which extend outside the primary containment boundary are designed to the extent practicable to an ultimate rupture strength (URS) at least equal to full RCPB pressure (2.4.1, 2.4.2, 2.4.4, 2.2.2, 2.2.4, 2.6.1).	The designing of interfacing low pressure systems to URS equal to RCPB pressure reduces the possibility of an intersystem loss of coolant accident and consequently the possibility of a loss of coolant accident outside the containment.

Table 19.8-1 Important Features from Level 1 Internal Events Analyses (Continued)

Feature	Basis
Redundant and diverse CRD scram capability consisting of both hydraulic and electric run-in capabilities (2.2.2) with redundant and diverse scram signals from the RPS (2.2.7) and ARI logic (2.2.8).	The CRD scram system provides the first line of defense against ATWS events. In addition, the redundancy and diversity incorporated in the CRD scram system significantly reduces the probability of an ATWS.
Automatically initiated standby liquid control (SLC) system (2.2.4) and recirculation pump trip (2.2.8) to provide backup shutdown capability in event of failure to insert control rods.	The automatic SLC and recirculation pump trip provides backup shutdown capability to the CRDs which substantially reduce the calculated CDF associated with an ATWS event.
Three separated divisions of engineered safety features, each containing both high and low pressure emergency core cooling systems as well as the capability to remove decay heat. The integrity of divisions is important. No high pressure or high temperature piping lines should penetrate walls or floors separating two different safety divisions (Combination of: 2.4.1, 2.4.2, 2.4.4, 2.1.2, 2.11.3, 2.11.9, 2.12.1, 2.12.12, 2.12.13, 2.12.14). Piping penetrations should be qualified to the same differential pressure requirements as the walls or floors they penetrate (2.15.10, 2.15.12).	The separated divisions of ESF provides three complete divisions of redundant engineered safety features which are the bases for the low calculated CDF of the ABWR.
Automatic Depressurization System to provide access to low pressure core cooling injection systems (2.1.2). ADS reliability/availability is also discussed in Subsections 19K.4, 19K.8, and 19K.11.1 as well as in Table 19K-4.	The ADS provides a reliable means of depressurizing the reactor to permit core cooling with low pressure systems in the event high pressure systems fail.
Three emergency diesel generators, one dedicated to each of the three safety divisions and each capable of powering the complete set of normal safe shutdown loads in its division (2.12.13).	The three emergency diesel generators provide redundant sources of emergency AC power as added defense against loss of offsite power events.
Four divisions of self-tested Safety System Logic and Control instrumentation designed on the basis of two out of four actuation logic (3.4). See also Subsection 19K.11.1 and Table 19K-4.	The four division SSLC provides reliable defense against ATWS events as well as reliable initiation of ESF core cooling and heat removal systems.
Conduct of quarterly testing of the Essential Multiplexing System and the Safety System Logic and Control System.	This testing is conducted to discover faults that are not identified by the continuous self-test feature. The conduct of the quarterly testing substantially increases the reliability of the Essential Multiplexing System and the Safety System Logic and Control System and the subsequent contribution to the low calculated CDF.

Table 19.8-1 Important Features from Level 1 Internal Events Analyses (Continued)

Feature	Basis
Administrative actions to avoid common-cause failures noted in Subsection 19.9.8.	Reduce the potential for common-cause failures to disable safety systems.
HPCF pump capability to pump 171° C (340° F) water (2.4.2).	Insures continued pumping, even if containment pressure increases to the rupture disk setting.

Table 19.8-2 Important Features from Seismic Analyses

Feature	Basis
Seismic design of the containment (2.14.1) and reactor building (2.15.10) and assurance that future modifications or additions to internal structures meet the requirements of Subsection 3.8 if they are made in the vicinity of safety equipment.	Failure of seismic Category I structures could lead directly to core damage because of possible damage to ESF equipment. The Containment and the Reactor Building are the seismic Category I structures with the lowest HCLPFs.
Seismic qualification of the station batteries and cable trays (2.12.12).	DC power is required for all safety-related instrument and equipment control functions. Failure of the DC power system could lead directly to core damage.
Seismic qualification of the emergency AC power system diesel generators (2.12.13), 480V transformers (2.12.1), circuit breakers (2.12.1, 2.12.12, and motor control centers (2.12.1, 2.12.12).	In a severe seismic event, it is likely that offsite AC power will be lost and emergency AC power will be the only source of AC power. The components in the emergency AC power system with the lowest HCPLFs are the diesel generators, 480V transformers, circuit breakers, and motor control centers.
Seismic qualification of the plant service water system service water pumps, room air conditioners, and pump house (2.11.3).	In a severe seismic event, it is likely that offsite AC power will be lost and emergency AC power will be the only source of AC power. The plant service water system is required for diesel generator cooling and other cooling functions. The components in the service water system most sensitive to a seismic event are the service water pumps, room air conditioners, and pump house.
Seismic qualification of SLC system boron solution tank and SLC pumps (2.2.4)	In a severe seismic event, the ability to insert control rods may be impaired due to seismic deformation of the fuel channels and the SLC system may be the only means of reactivity control. The most sensitive components in the SLC system are the boron solution tank and the SLC pumps.
Seismic qualification of the ACIWA system including the pumps, valves, and water supply ([2.15.6 (SSE only)]. The collapse of the ACIWA building (shed) should not prevent the pumps from starting and running [2.15.6 (SSE only)]. All needed valves for system operation can be accessed and operated manually (2.15.6, 2.4.1).	ACIWA can provide either vessel injection or drywell spray using equipment that does not require AC power. In addition, support systems normally required for ECCS operation are not required for ACIWA operation. ACIWA is an important system in preventing and mitigating severe accidents.

Table 19.8-2 Important Features from Seismic Analyses (Continued)

Feature	Basis
Seismic qualification of the RHR heat exchangers (2.4.1). See also Subsection 19K.5.	Seismic failure of RHR heat exchangers could partially drain the suppression pool and flood the RHR rooms. RHR is needed for decay heat removal and water in the suppression pool would provide fission product scrubbing in the event of core damage.
Seismic walkdown	A seismic walkdown could identify seismic vulnerabilities which were not identified in the margins assessment.

Table 19.8-3 Important Features from Fire Protection Analyses

Feature	Basis
Fire detection and suppression systems are provided throughout the plant (2.15.6). Fire suppression systems include hand held fire extinguishers, water hoses, foam and fire sprinklers (2.15.6). FPS actuation is alarmed in the control room (2.15.6).	The use of these systems (not credited in the analysis) will make core damage frequency much less than the screening value.
The Remote Shutdown Panel with the ability to control HPCFB, four SRVs, and two divisions of RHR (2.2.6, 2.1.2). See also Subsections 19.9.11 and 19.9.12 for other actions from the panel.	The Remote Shutdown Panel provides an independent alternative means of achieving safe shutdown of the reactor in the event that the control room becomes uninhabitable due to a fire or other event.
The capability to operate the RCIC from outside the control room (2.4.4) and the capability to operate four SRVs from the remote shutdown panel (2.2.6, 2.1.2).	The capability to operate a redundant and diverse high pressure injection (RCIC) system and the capability to operate a redundant fourth SRV from outside the control room were required to meet the fire risk screening criterion.
Design and maintenance of divisional separation by three hour rated fire barriers of engineered safety features and their support systems including the intake structure (e. g., electrical power and cooling water). Subsection 9A.5.5 under “Special Cases—Fire Separation for Divisional Electrical Systems” lists the only areas of the plant where there is equipment from more than one safety division in a fire area. These should be the only areas where multiple divisions share the same fire area (2.15.10, 2.15.12).	The integrity of the divisional fire barrier separation is required to meet the fire risk screening criterion. This assures that a fire in one division will not cause equipment in another division to fail because of fire propagation between divisions.
Routing of piping or cable trays during the detailed design phase will conform with the fire area divisional assignment documented in the fire hazard analysis 2.15.10, 2.15.12)	This design feature assures that the routing of piping or cable trays will not invalidate the requirement that all safety divisions are separated by three hour fire barriers. The integrity of the divisional fire barrier separation is required to meet the fire risk screening criterion.
Design, maintenance and testing of smoke control systems (2.15.5c, 2.15.6)	The prevention of the spread of smoke, hot gasses, and fire suppressant from one fire division to another is implicit in the FIVE analysis as an important requirement to prevent adversely affecting safe shutdown capabilities, including operator actions.

Table 19.8-4 Important Features from Suppression Pool Bypass and Ex-Containment LOCA Analyses

Feature	Basis
DW-WW vacuum breakers (2.14.1). Wetwell/Drywell vacuum breaker bypass failure to close is discussed in Subsection 19K.11.16.	Failure of a DW-WW vacuum breaker to close provides a significant bypass from the drywell into the wetwell airspace following a drywell LOCA or if RPV failure occurs. This bypass pathway can release fission products directly to the atmosphere if high wetwell pressure causes the containment rupture disk to open. The consequence of a vacuum breaker failing to close and causing the rupture disk to open was evaluated in the PRA.
Redundant Main Steam Isolation Valves (MSIVs) (2.1.2). The MSIVs are pneumatic operated, spring close, fail-closed designs (2.1.2) actuated by redundant solenoids through two-out-of-four logic (2.4.3).	The MSL is very large compared to other bypass pathways and a failure of both MSIVs in one steamline to close would provide a large bypass pathway from the RPV to the turbine building. Therefore, the failure of the MSIVs to close would have a higher consequence from a given postulated event than other bypass pathways.
The SRV discharge lines are designed and fabricated to Quality Group C requirements (2.1.2) and the welds in the wetwell region above the surface of the suppression pool are non-destructively examined to the requirements of ASME Section III, Class 2 (2.1.2). Seismic qualification of the SRV discharge lines is discussed in Subsection 19K.5.	A break in one of these lines in the wetwell airspace could cause the containment rupture disk to open and result in a pathway directly from the RPV to the environment.
Normally closed sample lines and drywell purge lines. This item is also discussed in Subsection 19.9.18	If sample lines or purge lines are inadvertently left open a bypass pathway can exist.
Redundant and seismically qualified CUW system isolation valves (2.6.1), qualified to close under postulated break conditions [2.6.1, 1.2(4)]. Reliability of these isolation valves is discussed in Subsection 19K.11.15.	Minimize the potential for an ex-containment LOCA to lead to core damage and potential offsite release.
CUW remote manual shutoff valve. Designed to be closed by the operator from the control room following a CUW line break and failure of both CUW isolation valves to automatically close (2.6.1).	Reduce the probability for the operator having to control RPV water level below normal level following a CUW pipe break.

Table 19.8-4 Important Features from Suppression Pool Bypass and Ex-Containment LOCA Analyses (Continued)

Feature	Basis
Blowout panels in the RCIC and CUW divisional areas (2.15.10 As-built structural evaluation).	Failure of the blowout panels during an ex-containment LOCA due to a break in a RCIC or CUW line could result in the pressurization of these divisional areas that could impact equipment in adjacent areas and result in a second electrical division being unavailable.
The high availability of drywell or wetwell sprays to condense steam which bypasses the suppression pool is discussed in Subsection 19K.11.16.	

Table 19.8-5 Important Features from Flooding Analyses

Feature	Basis
Equipment for each safety division is located within compartments designed to prevent water from a flood from propagating from one division to another (2.15.10, 2.15.12). This includes features such as watertight doors and sealed cable penetrations (2.15.10, 2.15.12).	Assuming a flood has occurred and other mitigation features have failed, this single design feature prevents flooding in one division from affecting another division.
Floor drains in all upper floors of reactor and control buildings (2.9.1).	Assuming a flood has occurred and other mitigation features have failed, this single feature assures that flood waters on upper floors of the reactor and control buildings will flow to lower floors thereby preventing the failure of important equipment on that floor and allow other features on lower floors to mitigate the flood (e.g., sump pumps, watertight doors).
Water level sensors in RCW/RSW rooms (2.15.12) and logic in the control building to alert operator and trip RSW pump (2.11.19 Interface Req,) and close valves in affected RSW division (2.11.9).	Assuming a flood has occurred, the water level sensors and logic are the only automatic features that can identify and terminate flooding in the RCW rooms.
The reactor building corridor on floor B3F is large enough to contain the largest flood sources in the reactor building (condensate storage tank or suppression pool) (2.15.10).	Assuming a flood has occurred and other mitigation features have failed, this feature prevents any flood in the reactor building that flows to the corridor from affecting any safe shutdown equipment in the reactor building by isolating the water in the B3F corridor.
Anti-siphon capability in RSW systems. (2.11.9 Interface Requirement)	Anti-siphon capability will prevent a control building flood from continuing to siphon water after the pumps have been stopped. Failure of this capability could increase the chances of some floods leading to core damage.
Reactor Building sumps on floor B1F have overfill lines to the B3F corridor.	Assuming the failure of the sump pumps or a flood that exceeds the capacity of the sump pumps, these overfill lines prevent flood water in one division from propagating to another division. Loop seals are provided to preserve the integrity of the secondary containment.
Buildings other than the Turbine, Control, and Reactor Building do not contain equipment that can be used to achieve safe shutdown and flooding in those buildings cannot propagate to buildings which contain safe shutdown equipment (Multiple ITAAC entries define ABWR design).	The screening analysis indicated that the flooding analysis only needed to address internal flooding from sources in the Turbine, Control, and Reactor Buildings. If this is not the case, the basic flooding analysis could be invalidated.

Table 19.8-5 Important Features from Flooding Analyses (Continued)

Feature	Basis
A maximum of 4000 meters or RSW piping is allowed between RSW isolation valves at the pump house and the Control Building (2000 meters each for supply and return).	Following isolation of an RSW pipe break, draining of the water in the RSW piping into the Control Building will only affect equipment in one RCW division.
Operator check on each shift that watertight doors are closed and dogged.	A watertight door must be dogged to assure that it will provide full protection in the event of a flood.
High pressure or high temperature lines not routed through floors or walls separating two different safety divisions (2.15.10 Divisional Separation).	This single design feature prevents the failure of one division of a system ultimately resulting in the flooding and disabling of a second division.
The protection of electric motors and motor control centers from inadvertent spray or dripping from failing equipment is discussed in Subsection 19R.4.2.8 and Section 3.4.1.	
Avoidance of flooding due to premature or spurious actuation of the passive flooder valves is discussed in Subsection 19K.11.4.	

Table 19.8-6 Important Features From Shutdown Events Analyses

Feature	Basis
Decay Heat Removal	
Shutdown cooling (SDC) mode of the RHR system (2.4.1).	RHR(SDC) Is capable of both removing decay heat and ensuring that the core is covered with water. SDC is the normally used and preferred method of decay heat removal (DHR) during shutdown.
Reactor service water (RSW) system (2.11.9).	Failure of the RSW system would disable the principal RHR system. The RSW removes heat from the RHR and other systems and transfers it to the ultimate heat sink.
Ultimate heat sink (UHS) (4.1).	The UHS rejects decay heat to the environment from the RHR/RCW/RSW systems.
Inventory Control	
The low pressure core flood mode of the RHR system (2.4.1).	The low pressure core flood mode of the RHR system can supply makeup to the reactor with the reactor at low pressure.
The CRD system pumps which can supply water to the core through the CRD purge flow (2.2.2).	The CRD system pumps are capable of providing makeup to the reactor at high and low pressures to ensure the core is covered.
High pressure core flooders (HPCF) (2.4.2).	The HPCF is capable of providing makeup to the reactor at high and low pressure to ensure the core remains covered.
AC-independent Water Addition (ACIWA) System (2.15.6, 2.4.1).	The ACIWA can supply makeup to the reactor with the reactor at low pressure.
RPV Isolation on low water level (2.4.3).	The isolation of lines connected to the RPV on a low water level signal prevent uncovering the fuel for many potential RPV drain down events.
Permissives and inhibits associated with the RHR Mode Switch (2.7.1).	The permissives and inhibits associated with the RHR Mode switch ensures that valve line ups are correct for most RHR functions thereby preventing inadvertent diversion of water from the RPV.
RHR Valve Interlocks (2.4.1).	The RHR valve interlocks prevent low pressure RHR piping connected to high pressure systems from being exposed to high pressures.
RPV Level Indication (2.1.2).	The RPV level instrumentation informs the operator of RPV level and allows automatic initiation of ECCS pumps and closure of RPV isolation valves on low water level.

Table 19.8-6 Important Features From Shutdown Events Analyses (Continued)

Feature	Basis
RIP Diffuser Plug (2.1.3)	RIP maintenance during shutdown requires a temporary plug be installed in the RIP diffuser when RIP impeller, shaft and motor are removed. The plug is designed so it can not be removed unless the RIP motor housing bottom cover is in place.
Reactivity Control	
RPS High Flux Trip (Set Down) (2.2.7).	The RPS high flux trip automatically inserts withdrawn CRDs at a specified flux level to prevent criticality.
CRD Brake (2.2.2).	The brake system on the CRDs prevents ejection of a CRD which could cause criticality.
Refueling Interlocks (2.2.1, 2.5.5).	When the reactor Mode switch is placed in the REFUEL position, no fuel assembly can be hoisted over the RPV if a CRD blade has been removed.
Containment Integrity	
Automatic isolation of secondary containment (Modes 3 and 4) (2.4.3).	The automatic isolation of the secondary containment on a specified high radiation signal prevents release of radioactivity to the environs.
SGTS (2.14.4).	The SGTS processes gasses before release to the atmosphere.
Electrical Power	
Three physically and electrically independent divisions of safety-related power (2.12.1).	The three divisions of safety-related electric power allows for one division to be in maintenance and still mitigate a single active failure in another division.
Four onsite sources of AC power (three EDGs and one CTG) (2.12.13, 2.12.11).	The four sources of onsite AC power backs up offsite power and ensures power will be available to safe shutdown equipment.
Two independent offsite sources of AC power (2.12.1 Interface Req.).	Redundant offsite power sources allow for the loss of one offsite power source without losing power for decay heat removal during shutdown.

Table 19.8-7 Key Severe Accident Parameters

Parameter Description	Value	Relates to What Feature?	Cross Reference
Core Power	3926 MW	Containment Performance	ITAAC 1.2 - Description Only
El. of Top of Fuel	9.05 m	Containment Performance	
Normal Water Level	13.26 m	Containment Performance	
ADS Area	0.07m^2	Vessel Depressurization	
Containment Leak Rate	0.5% per day	Containment Performance	ITAAC 2.14.1
Containment Service Level C	0.77 MPa	Containment Performance	
Containment Ult. Strength	1.025 MPa	Containment Performance	
Total Zr in Core	72,550 kg	Containment Performance	
Sacrificial Concrete			
Calcium Carbonate Content	<4 weight percent	Basaltic Concrete	ITAAC 2.14.1
Height of Layer	1.5 m	Sacrificial Concrete	ITAAC 2.14.1
Pedestal Concrete Thickness	1.64 m	Pedestal	ITAAC 2.14.1
Compartment Volume			
Lower Drywell	1860 m^3	Containment Performance	
Upper Drywell	5490 m^3	Containment Performance	
Wetwell	9585 m^3	Containment Performance	
Floor Area			
Lower Drywell	79 m^2	Lower Drywell	ITAAC 2.14.1
Tolerance of Vacuum Breaker Position Switch	0.9 cm	Vacuum Breaker	ITAAC 2.14.1
Overflow Elevation			
LDW to Wetwell	-4.55 m	Lower Drywell	
UDW to Wetwell	7.35 m	Lower Drywell	

Table 19.8-7 Key Severe Accident Parameters (Continued)

Parameter Description	Value	Relates to What Feature?	Cross Reference
LDW to UDW vent area	11.3 m^2	Connecting Vents	ITAAC 2.14.1
Lower Drywell Flooder			
Elevation	–10.5 m	Lower Drywell Flooder	
Area per valve	0.0081 m^2	Lower Drywell Flooder	
Plug Melting Temperature	533 K	Lower Drywell Flooder	
Suppression Pool Mass	3.6 x 10^6 kg	Containment Performance	ITAAC 2.14.1
COPS			
Equivalent Flow Diameter of Disk	0.2 m (8 in.)	COPS	
Diameter of Piping	0.25 m (10 in.)	COPS	
Setpoint	0.72 MPa	COPS	ITAAC 2.14.6
Tolerance at nom. temp.	5%	COPS	ITAAC 2.14.6
Effect of temp. on setpoint	2% per 55.6°C	COPS	
Firewater Addition System			
Injection Locations	Vessel and Drywell	ACIWA	ITAAC 2.4.1, 2.15.6
Maximum Flow Rate	0.06 m^3/s	ACIWA	
Minimum Flow rate at COPS Setpoint	0.04 m^3/s	ACIWA	
Oxygen Concentration	<3.5% By Volume	Containment Inerting	Technical Specification LCO 3.6.3.2

19.9 COL License Information

A review was conducted to determine actions which will be completed by the COL applicant.

This section represents the results of that review.

19.9.1 Post Accident Recovery Procedure for Unisolated CUW Line Break

An unisolated reactor water cleanup system (CUW) line break, although very unlikely to occur (Subsection 19E.2.3.3), could lead to reactor building flooding and eventual depletion of ECCS water sources if the break can not be isolated. Attempting to control RPV water level in the normal range during post accident recovery operation could lead to a continuous coolant outflow through the break since the CUW suction nozzle and the RPV drain line connection to the suction line are below the normal RPV water level.

For a CUW break outside of the containment, the Secondary Containment Control Guideline of the symptom-based Emergency Procedure Guideline (EPGs), Appendix 18A, provides the appropriate initial operator actions for isolation of CUW, scram the reactor, and depressurization of the reactor. The RPV Control Guideline of the EPGs (Steps RC/P-5, RC/L-3) provides the direction for proceeding to cold shutdown in accordance with a procedure which the COL applicant will develop. This COL applicant procedure for post accident recovery will be developed using the following guidance:

(1) After RPV depressurization, attempt to close the CUW isolation valves and the CUW remote manual shutoff valve. If at least one of the three CUW valves can be closed, control RPV water level in the normal range and initiate shutdown cooling operation.

(2) If the CUW remote manual shutoff valve can not be closed and at least one of the two CUW isolation valves can not be closed, control RPV water level between the top of the active fuel and 38 cm above the top of the active fuel. (The RPV drain line connects to the CUW suction line at this elevation). When practical, enter the CUW room and/or the containment and affect the necessary repairs. When at least one of the two CUW isolation valves can be closed, control water level in the normal range and initiate shutdown cooling.

(3) When practical, enter the CUW room and/or the containment and affect the necessary repairs. When at least one of the two CUW isolation valves can be closed, control water level in the normal range and initiate shutdown cooling.

19.9.2 Confirmation of CUW Operation Beyond Design Bases

CUW can be used to remove decay heat under accident conditions by bypassing the regenerative heat exchanger as noted in Subsection 19.3.1.3.1(1)(b). This causes the nonregenerative heat exchanger to remove additional heat. However, this could lead to exceeding the design temperature limits of the CUW nonregenerative heat exchanger and some portions of the piping of the CUW and the reactor building cooling water (RCW) systems.

When the design of the CUW and RCW systems (including piping and support structures) is completed, the COL applicant must confirm that if the CUW is operating in the heat removal mode, the following areas will remain functional while operating outside their design basis temperature values:

(1) The CUW nonregenerative heat exchanger

(2) The CUW piping downstream of the regenerative heat exchanger

(3) The RCW piping downstream of the nonregenerative heat exchanger

(4) The feedwater piping downstream of CUW injection

(5) Piping supports for the above piping

When the CUW is used to remove decay heat by bypassing the regenerative heat exchanger, steps should also be taken to prevent boiling in the shell side of the nonregenerative heat exchanger (NRHX) by increasing the reactor building closed cooling water (RBCCW) flow through the NRHX. To accomplish this the plant emergency procedures should consider the following steps:

(1) Terminate RBCCW flow to the RHR heat exchangers.

(2) Bypass the hot water heat exchanger in the RBCCW line.

(3) Bypass the flow control valve which controls RBCCW flow through the NRHX.

19.9.3 Event Specific Procedures for Severe External Flooding

Internal flooding is addressed in Appendix 19R. The site selection process will take into account the worst case predicted flood. Then grade level and flood control methods (e.g., site grading) will be determined based on this predicted flood level. The grade level floor will be 0.3 meters above this predicted flood level. Therefore, external flooding should not be a major concern for the ABWR. To further reduce the susceptibility of external floods, plant and site specific procedures will be developed by the COL applicant for severe external flooding using the following guidelines:

(1) Check that the door between the turbine and service buildings is closed.

(2) Sandbag the external doors to the

  (a) Reactor building,

  (b) Control building,

  (c) Service building,

  (d) Pump house at the ultimate heat sink,

  (e) Diesel generator fuel oil transfer pits, and

  (f) Radwaste building.

(3) Close and dog all external water tight doors in the reactor and control buildings.

(4) Shut the plant down.

(5) Use power from the diesel generators or CTG if offsite power is lost.

Underground passages between buildings would not be affected because they are required to be watertight.

19.9.4 Confirmation of Seismic Capacities Beyond the Plant Design Bases

The seismic analysis assumed seismic capacities for some equipment for which information was not available. It is expected that these capacities can be achieved, but determination of actual seismic capacities must be deferred to the COL applicant when sufficient design detail is available. The actions specified in Subsection 19H.5 will be taken by the COL applicant.

19.9.5 Plant Walkdowns

The COL applicant shall develop procedures for the plant walkdown to seek seismic vulnerabilities which will be conducted by the COL applicant as outlined in Subsection 19H.5.1.

Similar walkdowns will be conducted by the COL applicant for internal fire and flooding events.

19.9.6 Confirmation of Loss of AC Power Event

The COL applicant will confirm the frequency estimate for the loss of AC power event (Subsection 19D.3.1.2.4). This review will address site-specific parameters (as indicated in the staff's licensing review basis document) such as specific causes (e.g., a severe storm) of the loss of power, and their impact on a timely recovery of AC power.

19.9.7 Procedures and Training for Use of AC-Independent Water Addition System

Specific, detailed procedures will be developed by the COL applicant for use of the AC-independent Water Addition System (including use of the fire truck) to provide vessel injection, wetwell spray, drywell spray, and suppression pool makeup water, if necessary. Training will be included in the COL applicant's crew training program.

The procedures to be developed by the applicant will address operation of the ACIWA for vessel injection or drywell spray operation. Operation of the ACIWA System in the vessel injection mode requires valves F005, F101, and F102 to be opened and valve F592 to be closed. Reactor depressurization to below ACIWA System operating pressure is required prior to ACIWA operation in the vessel injection mode. Operation of the ACIWA in the drywell spray mode requires valves F017, F018, F101, and F102 to be opened and valve F592 to be closed. These valves are shown on Figure 5.4-10. The diesel fire pump will start automatically when the ACIWA is properly aligned for vessel injection or drywell spray. If the normal firewater system water supply is unavailable, the alternate water supply can be made available b y opening the manual valve between the diesel driven fire pump and the alternate water supply. This valve is shown in Figure 9.5-4. If it is necessary to use a fire truck for vessel injection or drywell spray, valve F103 must be opened in addition to operation of the valves discussed above for ACIWA operation. The valve for operation of the ACIWA using the fire truck is also shown on Figure 5.4-10. All of the valves required for ACIWA operation are manually operable.

If it is necessary to operate the ACIWA, radiation levels may be elevated in the rooms where the valves required for ACIWA operation are located. The applicant will make dose rate calculations for the specific configuration being constructed. These calculation will include the specific piping layout, shielding considerations, the potential for systems within the room to have recently been operated and thus contain radioactive coolant, and any other factors that significantly affect the dose rates. These dose rate calculations will be considered in the development of the specific plant procedures for ACIWA operation.

19.9.8 Actions to Avoid Common-Cause Failures in the Essential Multiplexing System (EMUX) and Other Common-Cause Failures

To reduce the potential for significant EMUX common cause failures, (Subsection 19N.4.12), the COL applicant will take the following actions:

(1) To eliminate remote multiplexing unit (RMU) miscalibration as a credible source of EMUX common cause failure, administrative procedures will be established to perform cross-channel checking of RMU outputs at the main control room safety system logic and control instrumentation, as a final check point of RMU calibration work.

(2) To prevent any unidentified EMUX faults/failure modes (e.g., an undetected software fault) from propagating to other EMUX divisions, the plant operating procedures will include the appropriate detailed procedures necessary to assure that the ABWR plant operations are maintained in compliance with the governing Technical Specifications during the periods of divisional EMUX failure. This will assure that such unidentified faults are effectively eliminated as a credible source of EMUX common cause failure. These procedures will also include the appropriate symptom-based operator actions to assure that adequate core cooling is maintained in the hypothetical event of an entire EMUX system failure.

(3) To eliminate maintenance/test errors as a credible source of EMUX common-cause failure, administrative procedures will be established which will not permit the same technician to work on multiple divisions of the EMUX.

As noted in Subsection 19D.7, a maintenance procedure must be established so that if a sensor is found out of tolerance, before it is recalibrated, the calibration instrument is first checked. In addition, the same technician will not be allowed to calibrate sensors in different divisions.

19.9.9 Actions to Mitigate Station Blackout Events

It was necessary to make several assumptions in the assessment of plant performance under station blackout conditions as noted in Subsection 19E.2.1.2. The following actions will be taken by the COL applicant to confirm these assumptions:

(1) Confirm that the minimum condensate storage tank volume is 570 cubic meters.

(2) Develop battery loading profiles to define appropriate load shedding during station blackout to ensure that the RCIC System can be operated for approximately 8 hours (See Subsection 8.3.2.1.3.1).

(3) Perform analyses to confirm that RCIC room temperature will not exceed equipment design temperature without room cooling for at least 8 hours.

(4) Perform analyses to confirm that control room temperature will not exceed equipment design temperature for at least 8 hours without room cooling.

(5) Develop procedures for the emergency replenishment of gas supply for safety-related, pneumatically operated components. A discussion of the types of actions which could be taken is in Subsection 19E.2.1.2.2.2(2)(b).

(6) Develop procedures to provide backup DC power to ADS valves to keep the valves open as long as possible to keep the reactor vessel depressurized if such action was necessary during a Station Blackout. See the discussion in Subsection 19E.2.1.2.2.2(a).

In the detailed procedures which supplement the Emergency Procedure Guidelines, include the manual valve operation which is noted in Subsection 19.7.3(3a).

19.9.10 Actions to Reduce Risk of Internal Flooding

In the unlikely event of significant flooding from internal sources (addressed in Appendix 19R) such as the ultimate heat sink, suppression pool, condensate storage tank, or fire water system, actions will be completed by the COL applicant to ensure that the following can be performed to mitigate flooding in the turbine, control, and reactor buildings:

(1) Training on isolation of potential flooding sources.

(2) Maintenance of pump trip and valve isolation capability of potential unlimited flood sources should be controlled to assure that flood mitigation capability exists at all times. If pump trip and valve isolation capability is unavailable, procedures to monitor applicable piping lines for leakage must be implemented and replacement/repair of failed components must be completed as soon as possible or other mitigative features must be implemented.

(3) Sizing of floor drains must be adequate to accommodate all potential flood rates. In sizing the floor drains, the following considerations must be addressed:

(a) The maximum volume and flow rate of potential flood sources on each floor must be calculated based on ANSI/ANS 58.2, “Design Basis For Protection Of Light Water Nuclear Power Plants Against The Effects Of Postulated Pipe Rupture.”

(b) The floor drain sizing must be able to drain the highest flow rate in that area without allowing flood buildup to reach installed equipment in another area containing equipment from a different train or division (i.e. less than 200mm).

(c) The size and number of floor drains should address the probability of some drains becoming clogged with debris.

(4) Procedures for maintenance of watertight integrity of buildings and rooms especially during shutdown conditions.

(5) Procedure to ensure that if flooding occurs in an ECCS divisional room that the watertight door to the affected room will not be opened until watertight integrity of the remaining ECCS rooms is assured.

(6) Complete a site specific PRA-based analysis for potential flood sources, the potential for flooding in the UHS pump house, and required mitigation features.

(7) Procedure to open doors or hatches to divert water from safety-related equipment following postulated floods.

(8) Ensure that the design of the RSW System includes anti-siphon capability on both the supply and return lines to the UHS.

(9) Ensure that seals on radwaste tunnels between building are adequate to prevent interbuilding flooding.

(10) Ensure that the RSW pump house is designed to prevent interdivisional flooding and water in excess of 4000 meters of RSW piping cannot gravity drain to the control building.

19.9.11 Actions to Avoid Loss of Decay Heat Removal and Minimize Shutdown Risk

To reduce the potential for losing shutdown decay heat removal capability (addressed in Appendix 19Q), procedures will be prepared by the COL applicant for the following:

(1) Recovery of failed operating RHR System.

(2) Rapid implementation of standby RHR Systems if the initially operating RHR system cannot be restored.

(3) Ensuring that instrumentation associated with the following functions is kept available if the system is not in maintenance:

- RPV isolation valves,

- ADS,
- HPCF,
- LPFL,
- RPV water level, pressure, and temperature,
- RHR System alarms,
- EDG,
- Refueling interlock,
- Flood detection and valve/pump trip circuits.

(4) Use of alternate means of decay heat removal using non-safety grade equipment such as reactor water cleanup, fuel pool cooling, or the main condenser.

(5) Use of alternate means for inventory control using non-safety grade equipment such as AC-independent water addition, CRD pump (i.e., increasing CRD flow), and main feedwater and condensate.

(6) Recovery from loss of offsite power.

(7) Boiling as a means of decay heat removal in Mode 5 with the RPV head removed including available makeup sources.

(8) Conducting suppression pool maintenance, especially as it relates to reduced availability of ECCS suction sources.

(9) Fire/flood watches during periods of degraded safety division physical integrity.

(10) Ensuring that at least one division of safety equipment is not in maintenance and its physical barriers are intact at all times.

(11) Fire fighting during shutdown.

(12) Use of remote shutdown panel while the plant is shutdown.

To reduce other risks during shutdown, procedures will be prepared by the COL applicant for the following:

(1) Firefighting with part of the fire protection system in maintenance,

(2) Outage planning using guidance from NUMARC-91-06,

(3) Use of freeze seals and RIP and CRD replacement.

(4) Verification of correct fuel loading during refueling.

(5) Maintenance of secondary containment during Modes 3 and 4, when necessary.

19.9.12 Procedures for Operation of RCIC from Outside the Control Room

In the PRA fire analysis (Subsection 19M.6.2), credit is taken for operation of RCIC from outside the control room. The COL applicant will develop procedures and conduct training for such RCIC operation.

The procedure should be developed along the following lines:

(1) Station operation personnel and provide communication at areas for manual operation of the RCIC suction valves (CST suction and suppression pool suction), RCIC turbine trip and throttle valve, RCIC turbine steam admission valve, outboard steam isolation valve, RPV injection valve, turbine speed control panel, and the Remote Shutdown System.

(2) If the RCIC steam isolation valves are closed, open these valves from their MCCs. If necessary, disconnect power to the outboard steamline isolation valve and open it using the valve's manual handwheel.

(3) Disconnect or de-energize all control signals to and from the turbine.

(4) Close the turbine trip and throttle valve.

(5) Disconnect power to the motor-operated suction valve (CST or suppression pool, as required), steam admission valve, and manually open these valves using their handwheels.

(6) Use a portable speed sensing instrument to monitor turbine speed.

(7) Manually manipulate the trip and throttle valve and manually open the RPV injection valve using their handwheels. Control injection flow by manipulating the trip and throttle valve and operate the turbine below the overspeed trip value. If the turbine trips on overspeed, reset the trip and throttle valve, and manipulate this valve to operate the turbine.

(8) Monitor RPV water level at the Remote Shutdown System. Maintain RPV water level between Level 3 (low level) and Level 8 (high level).

19.9.13 ECCS Test and Surveillance Intervals

The test and surveillance intervals assumed in the PRA are documented in Tables 19D.6-1 through 19D.6-12. The COL applicant will develop a plan and implement procedures for identifying significant departures from these assumptions.

19.9.14 Accident Management

As noted in Section 19.11, the human actions for which credit has been taken in this PRA have been compiled (Subsection 19D.7) and checked against the emergency procedure guidelines. In addition, COL action items address human actions (Section 19.9). All of the human actions identified should be reviewed by the COL applicant so that detailed procedures can be developed and the appropriate training conducted.

Directions and guidance for operation of the Containment Overpressure System (COPS) shutoff valves should be developed. Appropriate care should be taken in the development of these procedures to ensure that the recovery of containment heat removal or containment sprays do not induce late containment structural failure. If a suppression pool water level of at least one meter above the top of the top horizontal connecting vent can be maintained following COPS operation, the COL applicant may wish to consider leaving the shutoff valves open until after recovery of Containment Heat Removal since the fission product release will be dominated by the initial noble gas release. In addition, the procedure for closure of the shutoff valves should include steps for the re-introduction of nitrogen into the containment. In developing accident mitigation strategies, the COL applicant may wish to examine the potential benefits of drywell spray operation if the containment fails in the drywell. Source term calculations, such as the one in Subsection 19E.2.2.8 indicate the release to the atmosphere may be substantially decreased by initiating drywell sprays after fission product release begins.

For human actions which are taken that rely on instrumentation which may be operating outside of the qualification range, the expected performance of the instrumentation should be determined and additional guidance provided to the operator if needed.

Accident management strategies should consider the potential for recriticality during the recovery. Recriticality could occur either as a result of boron dilution in an ATWS event or as a result of control blade relocation during the recovery of a badly damaged core. A possible strategy could be a caution for the operators and/or technical support staff to monitor the power level (perhaps indirectly via the rate of containment pressurization) and enter ATWS procedures as necessary.

19.9.15 Manual Operation of MOVs

As noted in Subsection 19.7.3 (3)(a), manual operation of MOVs can be used to improve the availability of decay heat removal. The COL applicant will implement procedures for such an operation.

19.9.16 High Pressure Core Flooder Discharge Valve

As noted in Subsection 19D.7.7.5, the HPCF loop B pump discharge valve is in the drywell. Plant procedures should include independent verification that the valve is locked-open following maintenance.

19.9.17 Capability of Containment Isolation Valves

To insure that containment isolation valve capability does not reduce the containment capability, the COL applicant will demonstrate that the stresses of containment isolation valves, when subjected to severe accident loadings of 0.77 MPa internal pressure and 260°C (500°F) temperature in combination with dead loads, do not exceed ASME Section III service level C limits. In addition, the ultimate pressure capability at 260°C (500°F) will be shown to be at least 1.03 MPa.

19.9.18 Procedures to Insure Sample Lines and Drywell Purge Lines Remain Closed During Operation

As noted in Subsection 19.8.4.3, it is important that these lines be normally closed during plant operation. The COL applicant will develop procedures and administrative controls to ensure the valves are normally sealed closed and that the purge valves have motive power to the valve operators removed.

19.9.19 Procedures for Combustion Turbine Generator to Supply Power to Condensate Pumps

The COL applicant will implement procedures for manual transfer of Combustion Turbine Generator (CTG) power to the condensate pumps. Condensate pump support systems (lube oil, cooling water) are also supplied power from the CTG.

19.9.20 Actions to Assure Reliability of the Supporting RCW and Service Water Systems

To assure the reliability of the RCW and Service Water Systems, the COL applicant will take the following action. At least each month, the standby pumps and heat exchangers are started and the previously running RCW and service water equipment is placed in a standby mode.

19.9.21 Housing of ACIWA Equipment

If AC-independent water addition (ACIWA) equipment is housed in a separate building, that building must be capable of withstanding site specific seismic events, flooding, and other site-specific external events such as high winds (e.g., hurricanes). The capability of the building housing the ACIWA equipment must be included in the plant-specific PRA.

19.9.22 Procedures to Assure SRV Operability During Station Blackout

To assure the operability of the SRVs during station blackout, the COL applicant will develop procedures for the use of the stored nitrogen bottles as discussed in Subsection 19E.2.1.2.2.2 (b).

19.9.23 Procedures for Ensuring Integrity of Freeze Seals

The COL applicant will provide administrative procedures to ensure the integrity of the temporary boundary when freeze seals are used. Mitigative measures will be identified in advance, and appropriate back-up systems will be made available to minimize the effects of a loss of coolant inventory (See Subsection 19Q.8).

19.9.24 Procedures for Controlling Combustibles During Shutdown

The COL applicant shall provide administrative procedures for controlling the combustibles and ignition sources during shutdown operations. (See Subsection 19Q.6 under “Fires During Maintenance”).

19.9.25 Outage Planning and Control

The COL applicant shall provide an outage planning and control program to ensure that the safety principle is clearly defined and documented (See Subsection 19Q.10).

19.9.26 Reactor Service Water Systems Definition

Service water systems modeled in the ABWR PRA are described and fault trees presented in Subsection 19D.6.4.2. These include the Reactor Building Cooling Water (RCW) System, Reactor Service Water (RSW) System, and the Ultimate Heat Sink (UHS). Those portions of the RSW System that are outside of the Control Building and the entire UHS are not in the scope of the ABWR Standard Plant. The COL applicant shall review RSW and UHS design configurations and performance capabilities against those assumed and modeled in Subsection 19D.6.4.2 and assess the impact of any differences on the ABWR PRA results.

19.9.27 Capability of Vacuum Breakers

The vacuum breaker seating material will be demonstrated to withstand the temperature profiles associated with the equipment survivability requirements specified in Subsection 19E.2.1.2.3.

19.9.28 Capability of the Containment Atmospheric Monitoring System

The COL applicant will demonstrate that the portion of the CAMS System which can be exposed to containment pressure can withstand the loading associated with the equipment survivability requirements specified in Subsection 19E.2.1.2.3.

19.9.29 Plant Specific Safety-Related Issues and Vendors Operating Guidance

The COL applicant shall address and incorporate plant-specific safety-related issues and the vendor's operating guidance on safe operations during shutdown (See Subsection 19Q.10 under "Shutdown Safety Issues").

19.9.30 PRA Update

A COL applicant referencing the ABWR certified design will review and, if necessary, update the design PRA to ensure that it bounds the site specific design (e.g. the ultimate heat sink) and that interface requirements of the standard design are satisfied. In addition, site characteristics such as river flooding, wind loadings, etc., will be compared to those assumed in the design PRA to ensure it is bounding. If the existing PRA is not bounding for site characteristics, then a risk based evaluation should be performed.

19.10 Assumptions and Insights Related to Systems Outside of ABWR Design Certification

The systems for which credit was taken which are outside of the ABWR design certification are those portions of the reactor service water (RSW) system outside of the control building including the safety related ultimate heat sink (UHS), the power cycle heat sink, parts of the offsite power system, and the fire truck which supplies the AC independent water addition system.

19.10.1 Reactor Service Water (RSW) System and Safety-Related Ultimate Heat Sink (UHS) Assumptions

The configurations of the RSW system and UHS as defined by ABWR system drawings and design performance specifications provided the bases for PRA fault tree modeling and evaluation. The total heat removal capacity of these configurations is sufficient to remove heat loads associated with emergency shutdown and post-LOCA core and containment cooling.

The design features and capacities of the RSW system are such that any one division can provide sufficient cooling capacity to remove decay heat provided that two RSW pumps, two reactor building cooling water (RCW) pumps, and three RCW heat exchangers in that division are in operation. In addition, one RCW and one RSW pump, and two RCW heat exchangers provide sufficient cooling capacity to support the core cooling (injection) function for ECCS equipment in a division. These assumptions were made in both internal event and seismic analyses. Developing a plan and implementing procedures for validating these capabilities are COL license information items.

Those portions of the RSW System that are outside of the control building are not in the ABWR scope and are described as interface requirements. Outside the control building, the pumps, strainers, valves, instruments, and controls are located in the UHS pump house. Piping connects those portions of the RSW system in the UHS pump house and the control building. Though not part of the certified design, these components are modeled in the Level 1 PRA based on RSW system drawings and specifications. Modeling is presented in Figure 19D.6-14 and component reliability assumptions are documented in Table 19D.6-6. These out of scope portions of the RSW system were modeled as an integral part of the RCW/RSW fault tree for each division. Reliability of the RCW/RSW system in each division was calculated in successfully supporting the ECCS injection function (single train success) and also for successfully supporting the heat removal function (two train success criterion).

It was assumed that structures which house RSW components include inter-divisional boundaries with a three-hour fire rating and inter-divisional flood control features to preclude flooding in more than one division. These features are noted in the RSW Interface Requirements, Subsection 2.11.9 (2).

All RSW isolation valves receive an automatic close signal on a high water level (1.5 meter) in the control building RSW/RCW rooms. In addition, anti-siphon capability ensures that RSW flow will stop if the RSW pumps are tripped and the isolation valves remain open. The reliability assumptions for RSW system components are contained in 19R.6.5.

The RSW System must be designed such that following a RSW line break and termination of RSW flow the RSW/RCW room of the affected division will not flood to a level exceeding 5 meters. In addition, the UHS cannot continuously flow into the control building by gravity alone.

19.10.2 Reactor Service Water (RSW) System and Safety-Related Ultimate Heat Sink (UHS) Insights

The design features and capabilities of the RSW System and UHS contribute to the reliability of decay heat removal and ECCS injection. If a transient is initiated by an internal event or a seismic event while the plant is at power, loss of heat removal is one potential threat which must be considered.

While the plant is shutdown and the containment is open, shutdown cooling and/or fuel pool cooling provides decay heat removal. Insights from the shutdown risk study in Appendix 19Q indicate that there are multiple means of removing decay heat during shutdown. Even if all decay heat removal systems fail, the core can be kept covered by injecting water into the reactor vessel using any of several systems and allowing the water in the RPV to boil. Appendix 19Q provides guidelines on what systems may be maintained during shutdown while still maintaining an acceptably low risk due to loss of the operating decay heat removal system.

The configuration and capabilities of the RSW System and UHS also contribute to the reliability of Emergency Core Cooling System performance by removing heat from the Reactor Building Cooling Water (RCW) System as described in the preceding subsection.

In the event of an RSW line leak in the control building RSW/RCW room, flood water level detectors alert the operator, trip the RSW pumps and close the isolation valves in the affected division. Insights from the flooding probabilistic risk assessment indicate that either the pump trip or isolation valve closure features (either automatically or due to operator action) must be successful in terminating the flood in order to reduce the risk from control building flooding. To ensure that pump tripping alone will result in termination of the flood, anti-siphon capability is included in the RSW System design on both the supply and return lines to the UHS.

The design of the RSW pump house must ensure that no more than one division of RSW will be affected by a break in a RSW line.

A control building flooding procedure should be prepared as part of the emergency plan.

19.10.3 Power Cycle Heat Sink Assumptions

These assumptions are noted in Table 19D.4-16. They relate to the ability to recover the heat sink given that it has been lost.

19.10.4 Power Cycle Heat Sink Insights

The circulating water pumps are tripped in the event of a turbine building flood. This trip is expected to be sufficiently reliable to assure a negligibly small addition to the inadvertent plant trip frequency. Beyond this observation, no special attention to the power cycle heat sink is needed from a PRA perspective.

19.10.5 Offsite Power Assumptions

These assumptions are noted in Subsection 19D.3.1.2.4. Credit is also taken for offsite power recovery and diesel generator recovery, based on operating experience. Most of these assumptions are more reflective of the offsite power grid than equipment at the plant. However, Subsection 8.2.3 (4) is an interface requirement to analyze the site specific incoming power line configuration relative to the PRA assumption. Switchyard equipment inspections are included in the PRA input to the reliability assurance program (Appendix 19K).

19.10.6 Offsite Power Insights

The ABWR has three separate safety-grade divisions of ECCS including one division with an RCIC which does not require AC power. The ABWR also has a combustion turbine generator that can supply AC power to the ECC systems in the event of a loss of offsite power and failure of all three diesel generators. Finally, the AC-independent water addition system can be used to maintain core cooling. Therefore, the results of the internal event and seismic event evaluations are not particularly sensitive to assumptions about offsite power.

19.10.7 Fire Truck Assumption

The fire truck provides a backup water source for the AC-independent water addition system. As noted in Subsection 19D.5.11.3.5.4, an overall high reliability for fire water injection was taken for transients.

19.10.8 Fire Truck Insights

The AC-independent water addition system was added to the original ABWR design to provide a diverse and seismically qualified means of adding water to the reactor vessel and spraying the drywell. Because of its importance, it is included in the PRA input to the reliability assurance program (Appendix 19K), and its use should be included in the applicants training program. The later is included as an action item in Subsection 19.9.7.

19.11 Human Action Overview

Several functions, performed manually in an earlier ABWR design, were automated to reduce the dependence on human actions. In addition, other studies were performed to provide an improved understanding of human actions in the PRA.

Sensitivity studies of the core damage frequency resulting from the level 1 analysis were conducted (Subsection 19D.7). From this study, four human actions after accident initiation were found to be the most important. They are actions taken to provide water injection to the reactor vessel if the several automatic injection features fail to accomplish this function.

In addition, the PRA was reviewed to compile a list of human actions which were assumed in other parts of the analysis (Subsection 19D.7). From this list and the above mentioned sensitivity studies, actions were identified which should be given consideration as being “CRITICAL TASKS” as defined by the human factors evaluation Design Acceptance Criteria, as noted in Subsection 18E.2. These human factors are listed and discussed in Subsection 19D.7.

The human actions lists were also reviewed to ensure consistency with the ABWR emergency procedure guidelines (Appendix 18A). This review is documented in Appendix 18F. Some of the actions are not appropriate for inclusion in the symptom based emergency procedure guidelines. These are included as COL license information in Section 19.9.

19.12 Input to the Reliability Assurance Program

The major results of the PRA were reviewed to determine the reliability and maintenance actions that should be considered by the COL applicant throughout the life of the plant. This review is documented in Appendix 19K.

The Level 1 analysis results were reviewed by examining two importance measures ("Fussell-Vesely" and "Risk Achievement Worth"). Individual systems and components were identified as being most important (Table 19K-1).

The balance of the PRA was reviewed (Subsections 19K.4 through 19K.10) to determine other important features not addressed in the Level 1 analysis.

The most important features thus identified were finally reviewed to determine appropriate maintenance and surveillance actions (Subsection 19K.11).

19.13 Summary of Insights Gained from the PRA

The PRA was conducted with several objectives in mind:

(1) To ensure that the PRA-related goals in the ABWR Licensing Review Bases established in 1987 were satisfied.

(2) To review and improve the design capability for potential weaknesses or relative vulnerabilities, not withstanding the achievement of the Licensing Review Bases goals.

(3) To identify the most important aspects of the design and its operation so that particular attention can be placed on these aspects during certification, detailed design and plant operation.

(4) To provide additional basic studies which were not anticipated when the Licensing Review Bases was established.

(5) To provide uncertainty/sensitivity studies of key results.

The objectives were achieved as noted in the following subsections.

19.13.1 Licensing Review Bases Goals

These goals were established to ensure that an appropriate balance between accident prevention and accident mitigation is achieved by ABWR. The goals (Table 19.6-1 provides a summary) focus on prevention (core damage frequency), mitigation (avoiding containment failure from several potential threats) and offsite consequences (as measured by offsite doses, consequences, conditional containment failure probability, and the Safety Goal Policy Statement).

Measurement against these goals and the features which are important in achieving the goals are discussed in detail in Section 19.6. The goals are satisfied, indicating a very robust design with an excellent balance between accident prevention and mitigation features.

19.13.2 The Search for Vulnerabilities

As noted in detail in Section 19.7, the PRA process was used extensively to improve the design, even though it could be argued that satisfying the goals of Section 19.6 was sufficient. Improvements were made in many areas, including for example: the automation of several accident prevention functions, the addition of a combustion turbine generator to improve power supply diversity, the addition of an AC-independent Water Addition System to improve accident prevention and mitigation, and the addition of two passive accident mitigation features (the lower drywell flooder and the containment overpressure protection system) which substantially address uncertainties associated with severe accident progression. Procedural improvements were also identified. Many other examples are cited in Section 19.7 to illustrate the manner in which PRA techniques or PRA considerations were used throughout the design process to improve the design.

19.13.3 The Most Important Aspects of the Design

The ABWR design and its operation was reviewed to determine the features and operator actions which are most important from a PRA perspective. Applying additional focus in these aspects can provide confidence that ABWR operation will be as accident resistant as characterized by the PRA. The most important features of the design are identified in Section 19.8.

The potential for human error was reviewed extensively (Section 19.11) to ensure that “CRITICAL TASKS” were identified for the human factors Design Acceptance Criteria and to ensure that human actions are covered by the emergency procedures guidelines or other, more specific procedures.

The PRA results were reviewed to determine which surveillance and maintenance activities are most important with respect to assuring that PRA assumptions will be valid throughout plant life (Section 19.12).

19.13.4 Additional Studies

Several additional studies which were not anticipated in the original Licensing Review Bases were conducted to further review and enhance the robustness of the ABWR design.

The potential for internal fires to lead to core damage is studied in Appendix 19M. The basic ABWR features of separating the three safety divisions into individual fire zones and the ability to control key systems from outside the control room are the major reasons that very low core damage frequencies are calculated.

Internal flooding is investigated in detail from both a deterministic and probabilistic perspective in Appendix 19R. Divisional and building separation along with other key flooding mitigation features are identified which lead to the conclusion that there is a very small threat posed by internal flooding. General guidelines for addressing the potential for severe external flooding are provided in Subsection 19.9.3.

A seismic analysis (Appendix 19I) was conducted to assess the potential for seismic events beyond the design basis to lead to core damage. It was determined that there is high confidence in a low failure probability, even at ground accelerations approximately two times the plant seismic design basis. Key components and their seismic capacities are identified so that the COL applicant can review the design capability against those assumed in this margins analysis.

An assessment of the potential for core damage to result from ABWR operations while shutdown is documented in Appendix 19Q. Potential precursor events are reviewed for their applicability to ABWR and several ABWR features are noted which reduce the risk from activities conducted while shutdown. A decay heat removal reliability study is conducted to provide input to the COL applicant as to which complements of decay heat removal and water addition systems could be kept available while shutdown to reduce the risk of core damage resulting from the loss of an operating RHR system.

19.13.5 Uncertainty and Sensitivity Studies

Following quantification of the level 1 PRA, a data uncertainty study was performed (Subsection 19D.10). The effect of data uncertainty is relatively minor. The most important contribution to the uncertainty is the RCIC maintenance activity. This activity is addressed in the PRA input to reliability assurance (Appendix 19K).

A comparison of the level 1 quantified results to those for Grand Gulf was also developed to document the major reasons for reductions in the frequency of the various accident classes (Subsection 19D.11). The sensitivity of the results to equipment outage times and surveillance intervals was also considered (Subsection 19D.9). The contribution of human errors was compared to the contribution from an operating plant in Subsection 19D.7.8.

Uncertainties associated with severe accident progression were examined in detail through the use of containment event trees supplemented by decomposition event trees. The latter were used to study the potential for different outcomes of various severe accident events. The results show that the ABWR design is very robust. Analysis of phenomena such as direct containment heating were performed which indicate that the probability of occurrence with significant magnitude to fail the containment is very small. The design is not sensitive to assumptions affecting debris coolability due to the high strength of the containment and the lower drywell pedestal design. The studies also demonstrated that the features of the ABWR design substantially reduced the uncertainty associated with many severe accident phenomena. In many areas, these studies were conducted in greater depth than studies with similar objectives reported in NUREG-1150 and its supporting documents. In addition, the basis for the judgments made is described in detail.

19.13.6 Systems and Effects Not Modeled in the PRA

19.13.6.1 Equipment Aging

Aging or other deterioration of cables, pipes, walls and structures is not directly addressed in the analysis or in the RAP. It is expected that routine maintenance and inspection of equipment for in service inspection requirements and plant walkdowns will identify deterioration of cables, pipes, walls and support structures to the extent that such deterioration would reduce the safety of the plant. It is assumed that detection of any deterioration of this equipment will lead to prompt corrective action to return the equipment to its as-designed condition.

19.13.6.2 Plant Control System and Control Room

The plant control system and control room are not directly modeled in the PRA, although the RPS and other risk significant systems are modeled. The control system impact on safety will be primarily through the potential to cause transients as initiating events. The ABWR control system is expected to be more reliable than control systems of operating BWRs, because of additional redundancy and frequent self-checking of control circuits and components. Therefore, it should not be a significant contributor to plant transients.

The control room is being designed with human factors considerations, so the ability of operators to take proper corrective action in abnormal situations will be greater than that in operating plants. The analyses have considered conservative values for operator actions, so the enhanced control room design is not expected to negatively impact plant safety and does not have to be explicitly modeled in the PRA.

19.13.6.3 Equipment Lubrication Systems

Equipment lubrication by active subsystems, including lube oil pumps, has been reviewed with regard to the possibility that several different loops or divisions of safety related equipment could be simultaneously disabled by a single failure. If lube oil pumps are used within a given division of a safety-related system, such as in the RHR System, they must be powered by the same electrical division that powers the pumps or they must require no power. (This statement does not apply to RCIC since RCIC lube oil cooling is mechanically driven by the turbine or pump shaft.) Thus, loss of one electrical division would only disable one division of the RHR system or another multi-division system. It is judged that detailed modeling of lubrication systems is not necessary, since the failure rate for a given equipment item includes the failure of its lubricating system.

19A Response to CP/ML Rule 10 CFR 50.34(f)

19A.1 Introduction

On January 15, 1982 (47 FR 2286) the NRC amended 10 CFR 50.34 to include paragraph (f), “Additional TMI-Related Requirements”. These additional requirements were directed to each applicant for a light-water-reactor construction permit or manufacturing license (CP/ML) whose application was pending as of February 16, 1982.

In its “Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation”, on April 13, 1983 (48 FR 16014), the NRC proposed to extend its policy such that future CP applications or reactivations of CP applications previously docketed also comply with the CP/ML rule. Finally, on August 8, 1985 the commission issued a revision to this proposed policy statement as “Policy Statement on Severe Accident Regarding Future Designs and Existing Plants”. This appendix reports General Electric’s responses for the ABWR Standard Plant to the CP/ML rule.

The responses demonstrate that the NRC requirements are satisfactorily fulfilled for the ABWR design. For each item, a summary of the NRC position is given and followed by a response. The response clarifies the issue as it pertains to the ABWR design and/or provides a listing of applicable Tier 2 sections, relevant correspondence, or other necessary documentation that may be referenced for complete clarification of our position. Where a particular requirement is not applicable to the ABWR Standard Plant, a statement to that effect is provided in the response.

For items that affect equipment outside the scope of the ABWR Standard Plant or utility operations and procedures, the response indicates that item will be addressed by the COL applicant. Otherwise, this appendix is complete in that all of the “Additional TMI-Related Requirements” approved for implementation by the NRC as listed in 10 CFR 50.34(f) have been favorably addressed where they apply to the ABWR design.

The bracketed item numbers at the end of each title correspond with the subsections in 10CFR50.34(f). Alphanumeric designations at the end of each “NRC Position” statement correspond to the related action plan items in NUREG-0718 and NUREG-0660 [provided in 10 CFR 50.34(f) for information only].

Table 19A-1 is provided as a convenient cross-reference which consolidates pertinent information associated with each of the 47 requirements. This includes the 10 CFR 50.34(f) subsection, the action plan numbers, the Appendix 19A subsection number, the item title, and the Tier 2 reference detailing resolution.

19A.2 NRC Positions/Responses

19A.2.1 Probabilistic Risk Assessment [Item (1) (i)]

NRC Position

Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. [II.B.8]

Response

The ABWR probabilistic risk assessment (PRA) was submitted as Appendix 19D.

19A.2.2 Auxiliary Feedwater System Evaluation [Item (1)(ii)]

NRC Position

Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWRs only) [II.E.1.1]:

(1) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(2) A design review of AFWS.

(3) An evaluation of AFWS flow design bases and criteria.

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.

19A.2.3 Impact of RCP Seal Damages Following Small-Break LOCA with Loss of Offsite Power [Item (1) (iii)]

NRC Position

Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. [II.K.2(16) and II.K.3(25)]

Response

This item is addressed in Subsection 1A.2.30.

19A.2.4 Report on Overall Safety Effect of PORV Isolation System [Item (1) (iv)]

NRC Position

Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCAs from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWRs only.) [II.K.3(2)]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.

19A.2.5 Separation of HPCS and RCIC System Initiation Levels [Item (1) (v)]

NRC Position

Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI System, and of providing that both systems restart on low water level. (For plants with High Pressure Core Flood Systems in lieu of high pressure coolant injection systems, substitute the words, “high pressure core flood” for “high pressure coolant injection” and (“HPCF” for “HPCI”) (Applicable to BWRs only). [II.K.3(13)]

Response

This item is addressed in Subsection 1A.2.22.

19A.2.6 Reduction of Challenges and Failures of Safety Relief Valves—Feasibility Study and System Modification [Item (1) (vi)]

NRC Position

Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWRs only.) [II.K.3(16)]

Response

This item is addressed in Subsection 1A.2.24.

19A.2.7 Modification of ADS Logic-Feasibility Study and Modification for Increased Diversity of Some Event Sequences [Item (1) (vii)]

NRC Position

Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modification that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWRs only.) [II.K.3(18)]

Response

This item is addressed in Subsection 1A.2.26.

19A.2.8 Restart of Core Spray and LPCI Systems on Low Level—Design and Modification [Item (1) (viii)]

NRC Position

Perform a study of the effect on all core-cooling modes under accident conditions of designing the core-spray and low pressure coolant in section systems to ensure that the systems will automatically re-start on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWRs only.) [II.K.3(21)]

Response

This item is addressed in Subsection 1A.2.27.

19A.2.9 Confirm Adequacy of Space Cooling Study for HPCS and RCIC [Item (1) (ix)]

NRC Position

Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and hi-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, “high pressure core spray” for “high pressure coolant injection” and “HPCS” for “HPCI”.) (Applicable to BWRs only.) [II.K.3(24)]

Response

This item is addressed in Subsection 1A.2.29.

19A.2.10 Verify Qualification of Accumulators on ADS Valves [Item (1) (x)]

NRC Position

Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety-related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWRs only.) [II.K.3(28)]

Response

This item is addressed in Subsection 1A.2.31.

19A.2.11 Evaluate Depressurization with Other Than Full ADS [Item (1) (xi)]

NRC Position

Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cool down. (Applicable to BWRs only.) [II.K.3(45)]

Response

The BWR Owners' Group sponsored a program to evaluate depressurization modes other than full actuation of the ADS. The results of this program were submitted to the NRC in a letter report from D. B. Waters, Chairman of BWR Owners' Group, to D. G. Eisenhut, Director (NRC), dated December 29, 1980. A summary of this evaluation follows.

The cases analyzed in the letter report above show that, based on core cooling considerations, no significant improvement can be achieved by a slower depressurization rate. A significantly slower depressurization will result in increased core uncovery times before ECCS injection. Furthermore, a moderate decrease in the depressurization rate necessitates an earlier action time to initiate ADS. Such an earlier actuation time has the negative impact of providing less time for the operator to start high pressure ECCS without obtaining a significant benefit to vessel fatigue usage. This earlier actuation time necessitates a higher initiation level which would result in an increased frequency of ADS actuation.

It should be noted that the ADS is not a normal core cooling system, but is a backup for the high pressure core cooling systems such as feedwater, RCIC or HPCF. If ADS operation is required, it is because normal and/or emergency core cooling is threatened. As a full ADS blowdown is well within the design basis of the RPV and the system is properly designed to minimize the threat to core cooling, no change in depressurization rate is required or appropriate.

19A.2.12 Evaluation of Alternative Hydrogen Control Systems [Item (1) (xii)]

NRC Position

Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of 10 CFR 50.34(f). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(1) A comparison of costs and benefits of the alternative systems considered.

(2) For the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of 10 CFR 50.34.

(3) For the selected system, preliminary design descriptions of equipment, function, and layout.

Response

The ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation. In fact, increasing amounts of hydrogen moves the primary containment oxygen concentration further from the flammable regime. The ABWR is also provided with permanently-installed recombiners which prevent the buildup of oxygen, due to radiolysis, from creating a potentially flammable mixture. Radiolysis is the only potential source of oxygen in the ABWR primary containment.

See Subsection 6.2.7.1 for COL license information pertaining to alternate hydrogen control.

19A.2.13 Long-Term Training Upgrade [Item (2) (i)]

NRC Position

Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs. (Applicable to construction permit applicants only.) [I.A.4.2]

Response

COL license information, see Subsection 19A.3.1. This will be addressed as part of simulator design which falls under operator training (Section 18.8.8).

19A.2.14 Long-Term Program of Upgrading of Procedures [Item (2) (ii)]

NRC Position

Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only.) [I.C.9]

Response

COL license information, see Subsection 19A.3.2.

19A.2.15 Control Room Design Reviews [Item (2) (iii)]

NRC Position

Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. [I.D.1]

Response

This item is addressed in Subsection 1A.2.2.

19A.2.16 Plant Safety Parameter Display Console (SPDS) [Item (2) (iv)]

NRC Position

Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. [I.D.2]

Response

This item is addressed in Subsection 1A.2.3.

19A.2.17 Safety System Status Monitoring [Item (2) (v)]

NRC Position

Provide for automatic indication of the bypassed and inoperable status of safety systems. [I.D.3]

Response

The ABWR Standard Plant design fully complies with Regulatory Guide 1.47 (Subsection 7.1.2.10.2). The automatic indication of bypassed and inoperable status of safety systems is, therefore, inherent in the design. Details on human factors are not addressed specifically, however, will be addressed by the COL applicant during the conduct of the HSI design implementation process described in Section 18.E.1.

19A.2.18 Reactor Coolant System Vents [Item (2) (vi)]

NRC Position

Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. [II.B.1]

Response

This issue is addressed in Subsection 1A.2.5.

19A.2.19 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation [Item (2) (vii)]

NRC Position

Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. [II.B.2]

Response

This item is addressed in Subsection 1A.2.6.

19A.2.20 Post-Accident Sampling [Item (2) (viii)]

NRC Position

Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 0.05 Sv to the whole-body or 0.50 Sv to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. [II.B.3]

Response

This item is addressed in Subsection 1A.2.7.

19A.2.21 Hydrogen Control System Preliminary Design [Item (2) (ix)]

NRC Position

Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal-water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1) (xii) of 10 CFR 50.34(f) is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: [II.B.8]

(1) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(2) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(3) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.

(4) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

Response

Per the response to Item (1)(xii), refer to Subsection 6.2.5 for a detailed description of the inerting and recombiner systems.

19A.2.22 Testing Requirements [Item (2) (x)]

NRC Position

Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transient without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. [II.D.11]

Response

This item is addressed in Subsection 1A.2.9.

19A.2.23 Relief and Safety Valve Position Indication [Item (2) (xi)]

NRC Position

Provide direct indication of relief and safety valve position (open or closed) in the control room. [II.D.3]

Response

This item is addressed in Subsection 1A.2.10.

19A.2.24 Auxiliary Feedwater System Automatic Initiation and Flow Indication [Item (2) (xii)]

NRC Position

Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWRs only.) [II.E.1.2]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.

19A.2.25 Reliability of Power Supplies for Natural Circulation [Item (2) (xiii)]

NRC Position

Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWRs only.) [II.E.3.1]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.

19A.2.26 Isolation Dependability [Item (2) (xiv)]

NRC Position

Provide containment isolation systems that: [II.E.4.2]

(1) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(2) For each non-essential penetration (except instrument lines) have two isolation barriers in series,

(3) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,

(4) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,

(5) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

Response

This item is addressed in Subsection 1A.2.14.

19A.2.27 Purging [Item (2) (xv)]

NRC Position

Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. [II.E.4.4]

Response

The ABWR primary containment vessel (PCV) operates with an inert atmosphere. During normal operation, all large valves in containment ventilation lines are closed with the exception of two large valves in the overpressure/protection where flow is prevented by a rupture disk in the piping.

Only small 50A (2-inch) pipe size nitrogen-makeup valves are opened during power operation. These are air-operated valves with rapid closure times, presenting little opportunity for substantial releases from the PCV in the event of a transient requiring containment isolation. Note that under the technical specifications, containment inerting and purging with the larger ventilation lines is permitted during power operation above 15% for limited periods at either end of the operating cycle. The process of purging the containment with air also serves to remove any potential activity for ALARA considerations prior to actual personnel entry into the PCV.

The large ventilation valves will be tested regularly and after any valve maintenance to assure that closing times are within the limits assured in the radiological design basis. These tests are part of the inservice test program detailed in Subsection 3.9. (See Subsection 19A.3.3 for COL license information.)

19A.2.28 Design Evaluator [Item (2) (xvi)]

NRC Position

Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe over cooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only.) [II.E.5.1]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type (B&W designed) reactors.

19A.2.29 Additional Accident Monitoring Instrumentation [Item (2) (xvii)]

NRC Position

Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. [II.F.1]

Response

This item is addressed in Subsection 1A.2.15.

19A.2.30 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling [Item (2) (xviii)]

NRC Position

Provide instruments that provide in the control room an unabiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs. [II.F.2]

Response

This item is addressed in Subsection 1A.2.16.

19A.2.31 Instrumentation for Monitoring Accident Conditions (Regulatory Guide 1.97) [Item (2) (xix)]

NRC Position

Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. [II.F.3]

Response

This item is addressed in Subsection 1A.2.17 and Subsection 7.5.

19A.2.32 Power Supplies for Pressurizer Relief Valves, Block Valves and Level Indication [Item (2) (xx)]

NRC Position

Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWRs only.) [II.G.1]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type reactors.

19A.2.33 Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable [Item (2) (xxi)]

NRC Position

Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWRs only.) [II.K.1(22)]

Response

This item is addressed in Subsection 1A.2.20.

19A.2.34 Analysis of Upgrading of Integrated Control System [Item (2) (xxii)]

NRC Position

Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only.) [II.K.2(9)]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type (B&W designed) reactors.

19A.2.35 Hand-Wired Safety-Grade Anticipatory Reactor Trips [Item (2) (xxiii)]

NRC Position

Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only.) [II.K.2(10)]

Response

This requirement is not applicable to the ABWR. It applies only to PWR-type (B&W designed) reactors.

19A.2.36 Central Water Level Recording [Item (2) (xxiv)]

NRC Position

Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWRs only.) [II.K.3(23)]

Response

In the ABWR design, the RPV water level wide range instruments and fuel zone instruments are utilized to provide this PAM indication. The four divisions of wide range instruments cover the range from above the core to the main steam lines. The two channels of fuel zone instruments cover the range from below the core to the top of the steam separator shroud. Two channels of enhanced water level indication are provided which use a wide range level transmitter, a fuel zone level transmitter, an RPV pressure transmitter and six temperature elements. The signals are input to a microprocessor which computes compensated level and provides a level signal to an indicator and a recorder. This design avoids the ambiguity of varying process and/or ambient temperatures, of a instrument line break, or boiling in an instrument line. If one of the enhanced water level indication channels fails, reliable indication of vessel level may be regained utilizing the 4 individual wide range indicators and the 2 individual fuel zone range level indicators as described below.

Evaluation has concluded that two channels of fuel zone level instrumentation provide adequate post accident monitoring capability. Post accident operator actions will be in accordance with detailed procedures developed based upon the BWR Owners' Group emergency operating procedure (EOP) guidelines. In the event the vessel water level is below the range of the wide range level (WRL) sensors (i.e., the water level is in the full zone range) and the two channels of fuel zone level instrumentation disagree, the EOPs instruct the operator to return the water level back up into the range of the instrumentation. Using the four divisions of WRL instruments, an unambiguous indication of vessel water level can be determined, despite a postulated failure of a single instrument channel or division, and the operator could safely continue the execution of appropriate accident instigation activities as defined by the EOPs.

19A.2.37 Upgrade License Emergency Support Facility [Item (2) (xxv)]

NRC Position

Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near site Emergency Operations Facility. [IIIA.1.2]

Response

The design features for the onsite Technical Support Center and the onsite Operational Support Center are provided in Subsection 13.3. The near site Emergency Operations Facility is provided by the COL applicant, Subsection 19A.3.4.

19A.2.38 Primary Coolant Sources Outside the Containment Structure [Item (2) (xxvi)]

NRC Position

Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. [III.D.1.1]

Response

This issue is addressed in Subsection 1A.2.34.

19A.2.39 Inplant Radiation Monitoring [Item (2) (xxvii)]

NRC Position

Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. [III.D.3.31]

Response

COL license information, personal monitoring radiation and portable instrumentation, training and procedures (Subsections 12.5.2, 12.5.3.1, 12.5.3.2, and 19A.3.5). Airborne radiation monitoring equipment (nonportable), Subsection 12.3.4.

19A.2.40 Control Room Habitability [Item (2) (xxviii)]

NRC Position

Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release, and make necessary design provisions to preclude such problems. [III.D.3.4]

Response

This item addressed in Subsection 1A.2.36.

19A.2.41 Procedures for Feedback of Operating, Design and Construction Experience [Item (3) (i)]

NRC Position

Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. [I.C.5]

Response

COL license information, see Subsection 19A.3.6.

19A.2.42 Expand QA List [Item (3) (ii)]

NRC Position

Ensure that the quality assurance (QA) list required by Criterion 11, App. B. 10 CFR Part 50 includes all structures, systems, and components important to safety. [I.F.1]

Response

Quality system requirements are identified in Table 3.2-1 Classification Summary. In addition, COL license information requirements, Section 1.9, ensure that quality system requirements will be provided during construction and operation.

19A.2.43 Develop More Detailed QA Criteria [Item (3) (iii)]

NRC Position

Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) Performing quality assurance / quality control functioning at construction sites to the maximum feasible extent; (C) Including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) Establishing criteria for determining QA programmatic requirements; (E) Establishing qualification requirements for QA and QC personnel; (F) Sizing the QA staff commensurate with its duties and responsibilities; (G) Establishing procedures for maintenance of “as-built” documentation; and (H) Providing a QA role in design and analysis activities. [I.F.2]

Response

(1) NEDO-11209-04A, “GE Nuclear Energy Quality Assurance Program Description”, conforms to this requirement. See Paragraph 1.1 on page 1-1.

(2) GE-NE services performed at the construction site are under the Owners' QA program. GE-NE provides QA program support to the Owner as described in NEDO-11209-04A, pages 1-3,1-7, 11-1, and 11-2.

(3) The GE-NE Nuclear Quality Assurance(NQA) is responsible for preparing the top level GE-NE quality policy and instructions for issue by the Vice President and General Manager, GE-NE. NQA is also responsible for preparing and issuing several GE-NE quality procedures. These documents are identified on pages 2-2 and 2-3 of NEDO-11209-04A.

In addition, NQA is responsible for developing, issuing, and controlling NEDO-11209.

The GE-NE line QA organizations are responsible for developing and documenting a quality system in compliance with GE-NE policies, instructions and procedures, and applicable codes, standards, and regulatory requirements. See NEDO-11209-04A, Section 1.3, "QA Functional Responsibilities" and Section 2.2 for typical line—QA procedure manuals.

(4) NEDO-11209-04A responds to each of the QA programmatic requirements of 10CFR50, Appendix B, and the requirements of the regulatory guides and industry standards identified in Table 2-1. In addition, the GE-NE QA program conforms to the requirements of the ASME Code.

(5) NEDO-11209-04A, Section 2-1, fourth paragraph, describes the qualification of training of GE-NE personnel who perform activities affecting quality. See also Subsection 1.4, "QA Personnel Responsibilities and Qualifications".

(6) The NRC has evaluated the GE-NE QA Program implementation for several years and has found that the program, including sizing of the QA staff, is being implemented satisfactorily. See NRC letters in Docket No. 99900403.

(7) NEDO-11209-04A, Section 17, describes the GE-NE commitments related to "as-built" documentation. The GE-NE commitments are further detailed on pages 2-10, 2-11, and 2-13 thru 2-15.

(8) NQA has the following responsibilities that are documented in NEDO-11209-04A, Subsection 1.3:

   (a) Develop GE-NE policies and procedures related to project and services management, engineering, manufacturing, procurement, field service and construction QA.

   (b) Conduct or participate in independent design reviews.

   (c) Conduct independent audits of the GE-NE design control program.

Based on the foregoing evaluation, it is demonstrated that the GE-NE QA program as described in NEDO-11209-04A, and as currently accepted by the NRC, includes full consideration of the matters identified in this item.

COL licensing information, see Subsection 19A.3.8.

19A.2.44 Dedicated Containment Penetrations Equivalent to a Single 3-Foot Diameter Opening [Item (3) (iv)]

NRC Position

Provide one or more dedicated containment penetrations, equivalent in size to a single 91 cm (3-foot) diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. [II.B.8]

Response

The Containment Overpressure Protection System is described in Subsection 6.2.5.2.6 and is analyzed in the PRA. The sizing of the system is developed in Subsection 19E.2.8.1.3 and precludes the need for a dedicated penetration equivalent in size to a single 91-cm (3-foot) diameter opening.

19A.2.45 Containment Integrity [Item (3) (v)]

NRC Position

Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: [II.B.9]

(1) (a) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 0.412 MPa (45 psig). Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(b) Subarticle NE-3220, Division 1, and Subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(1) and (Q (3)(v)(B)(1) of 10 CFR 50.34, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H St., NW., Washington, D.C.

(2) (a) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, Service Load Category.

(c) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

Response

(1) The containment design basis accident pressure is 0.412 MPa. The peak pressure resulting from 100% fuel-clad metal water reaction is about 0.618 MPa (Subsection 19E.2.3.2). The containment is capable of withstanding 0.618 MPa internal pressure together with dead load by meeting the code requirements (Subsection 19E.2.3.2).

(2) ABWR does not employ post accident inerting; thus, item (2) does not apply.

19A.2.46 Dedicated Penetration [Item (3) (vi)]

NRC Position

For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. [II.E.4.1]

Response

This item is addressed in Subsection 1A.2.13.

19A.2.47 Organization and Staffing to Oversee Design and Construction [Item (3) (vii)]

NRC Position

Provide a description of the management plant for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) Technical resources directed by the applicant; (C) Details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) Proposed procedures for handling the transition to operation; (E) The degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. [II.J.3.1]

Response

COL license information, see Subsection 19A.3.7.

19A.3 COL License Information

19A.3.1 Long-Term Training Upgrade

Simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs shall be provided. (Subsection 19A.2.13.) COL License Information regarding operator training is in Section 18.8.8.

19A.3.2 Long-Term Program of Upgrading of Procedures

A long-term program of upgrading procedures shall be established to begin during construction and following term program of upgrading procedures shall be established to begin during construction and follow into operation for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analysis, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Subsection 19A.2.14.) COL License Information is in Section 13.5.3.1.b.

19A.3.3 Purge System Reliability

A testing program shall be provided to ensure that the large ventilation valves close within the limits assured in the radiologic design bases. (Subsection 19A.2.27.)

19A.3.4 Licensing Emergency Support Facility

The COL applicant has a requirement to provide a near site Emergency Operational Facility (EOF) (See Subsection 19A.2.37).

19A.3.5 In-Plant Radiation Monitoring

Personal monitoring and portable instrumentation of in-plant radiation and airborne radioactivity as well as training and procedures appropriate for a broad range of routine and accident conditions shall be provided (Subsections 12.5.2, 12.5.3.1, 12.5.3.2, and 19A.2.39).

19A.3.6 Feedback of Operating, Design and Construction Experience

Administrative procedures for evaluating design and construction experience and for ensuring that applicable important industry experiences shall be provided in a timely manner to those designing and constructing the ABWR Standard Plant. (Subsection 19A.2.41) COL license information regarding incorporation of operator experience into training and procedures is found in Sections 13.2.3 and 13.5.3, respectively.

19A.3.7 Organization and Staffing to Oversee Design and Construction

A description of the management plan for design and construction activities shall be provided. It will include:

(1) Organizational and management structure singularly responsible for direction of design and construction for the plant

(2) Technical resources directed by the applicant referencing the ABWR design

(3) Details of the interaction of design and construction within the organization of the applicant referencing the ABWR design and the associated organization by which integration of the total project is ensured

(4) Procedures for handling the transition to operation

(5) The degree of top level management oversight and technical control will be exercised during design and construction including the preparation and implementation of procedures necessary to guide the effort (Subsection 19A.2.47)

19A.3.8 Develop More Detailed QA Criteria

Establish a quality assurance (QA) program in accordance with the requirements in Subsection 19A.2.43.

Table 19A-1 ABWR—CP/ML Rule Cross Reference

CP/ML Rule Section	Item Action Plan	Appendix Section	Title	Tier 2 Reference
(1) (i)	II.B.8	19A.2.1	Probabilistic Risk Assessment	Appendix 19D
(ii)	II.E.1.1	19A.2.2	Auxiliary Feedwater System Evaluation	Not Applicable (PWR Only)
(iii)	II.K.2(16) & II.K.3(25)	19A.2.3	Impact of RCP Seal Damages Following Small-Break LOCA with Loss of Offsite Power	Subsection 1A.2.30
(iv)	II.K.3(2)	19A.2.4	Report on Overall Safety Effect on PORV Isolation System	Not Applicable (PWR Only)
(v)	II.K.3(13)	19A.2.5	Separation of HPCF and RCIC System Initiation Levels	Subsection 1A.2.22
(vi)	II.K.3(16)	19A.2.6	Reduction of Challenges and Failures of Safety Relief Valves Feasibility Study and System Modification	Subsection 1A.2.24
(vii)	II.K.3(18)	19A.2.7	Modification of ADS Logic-Feasibility Study and Modification for Increased Diversity of Some Event Sequences	Subsection 1A.2.26
(viii)	II.K.3(21)	19A.2.8	Restart of Core Flood and LPCI Systems on Low Level-Design and Modification	Subsection 1A.2.27
(ix)	II.K.3(24)	19A.2.9	Confirm Adequacy of Space Cooling Study for HPCF and RCIC	Subsection 1A.2.29
(x)	II.K.3(28)	19A.2.10	Verify Qualification of Accumulators on ADS Valves	Subsection 1A.2.31
(xi)	II.K.3(45)	19A.2.11	Evaluate Depressurization with Other than Full ADS	Subsection 19A.2.11
(xii)	—	19A.2.12	Evaluation of Alternative Hydrogen Control Systems	Subsection 19A.2.12
(2) (i)	IA.4.2	19A.2.13	Long-Term Training Upgrade	Subsection 19A.3.1
(ii)	I.C.9	19A.2.14	Long-Term Program of Upgrading of Procedures	Subsection 19A.3.2/13.5.3.1
(iii)	I.D.1	19A.2.15	Control Room Design Reviews	Subsection 1A.2.2/18.8.1
(iv)	I.D.2	19A.2.16	Plant Safety Parameter Display Console	Subsection 1A.2.3/18.8.4
(v)	I.D.3	19A.2.17	Safety System Status Monitoring	Subsection 19A.2.17/18.8.9

Table 19A-1 ABWR—CP/ML Rule Cross Reference (Continued)

CP/ML Rule Section	Item Action Plan	Appendix Section	Title	Tier 2 Reference
(vi)	II.B.1	19A.2.18	Reactor Coolant System Vents	Subsection 1A.2.5
(vii)	II.B.2	19A.2.19	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	Subsection 1A.2.6
(viii)	II.B.3	19A.2.20	Post-Accident Sampling	Subsection 1A.2.7
(ix)	II.B.8	19A.2.21	Hydrogen Control System Preliminary Design	Subsection 19A.2.21
(x)	II.D.1	19A.2.22	Testing Requirements	Subsection 1A.2.9
(xi)	II.D.3	19A.2.23	Relief and Safety Valve Position Indication	Subsection 1A.2.10
(xii)	II.E.1.2	19A.2.24	Auxiliary Feedwater System Automatic Initiation and Flow (Indication	Not Applicable (PWR Only)
(xiii)	I.E.3.1	19A.2.25	Reliability of Power Supplies for Natural Circulation	Not Applicable (PW R Only)
(xiv)	II.E.4.2	19A.2.26	Isolation Dependability	Subsection 1A.2.14
(xv)	II.E.4.4	19A.2.27	Purging	Subsections 19A.2.27 and 19A.3.3
(xvi)	II.E.5.1	19A.2.28	Design Evaluator	Not Applicable (B&W Only)
(xvii)	II.F.1	19A.2.29	Additional Accident-Monitoring Instrumentation	Subsection 1A.2.15/18.8.13
(xviii)	II.F.2	19A.2.30	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	Subsection 1A.2.16/18.8.14
(xix)	II.F.3	19A.2.31	Instrumentation for Monitoring Accident Conditions (Regulatory Guide 1.97)	Subsection 1A.2.17 and Section 7.5
(xx)	II.G.1	19A.2.32	Power Supplies for Pressurizer Relief Valves, Block Valves and Level Indication	Not Applicable (PWR Only)
(xxi)	II.K.1(22)	19A.2.33	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW Systems not Available	Subsection 1A.2.20

Table 19A-1 ABWR—CP/ML Rule Cross Reference (Continued)

CP/ML Rule Section	Item Action Plan	Appendix Section	Title	Tier 2 Reference
(xxii)	II.K.2(9)	19A.2.34	Analysis of Upgrading of Integrated Control System	Not Applicable (P&W Only)
(xxiii)	II.K.2.(10)	19A.2.35	Hand-Wired Safety-Grade Anticipatory Reactor Trips	Not Applicable (P&W Only)
(xxiv)	II.K.3(23)	19A.2.36	Central Water Level Recording	Subsection 19A.2.26
(xxv)	III.A.1.2	19A.2.37	Upgrade License Emergency Support Facility	Subsection 19A.3.4
(xxvi)	III.D.1.1	19A.2.38	Primary Coolant Sources Outside the Containment Structure	Subsection 1A.2. 34
(xxvii)	III.D.3.3	19A.2.39	In-Plant Radiation Monitoring	Subsection 19A.3.5
(xxviii)	II.D.3.4	19A.2.40	Control Room Habitibility	Subsection 1A.2. 36
(3) (i)	I.C.5	19A.2.41	Procedures for Feedback of Operating, Design and Construction Experience	Subsection 19A.3.6/13.2.3.1 /13.5.3.3.f
(ii)	I.F.1	19A.2.42	Expand QA List	Subsection 19A.2.42
(iii)	I.F.2	19A.2.43	Develop More Detailed QA Criteria	Subsection 19A.2.43
(iv)	II.B.8	19A.2.44	Dedicated Containment Penetrations, Equivalent to a Single 3-foot Diameter Opening	Subsection 19A.2.44
(v)	II.B.8	19A.2.45	Containment Integrity	Subsection 19A.2.45
(vi)	II.E.4.1	19A.2.46	Dedicated Penetration	Subsection 1A.2.13
(vii)	II.J.3.1	19A.2.47	Organization and Staffing to Oversee Design and Construction	Subsection 19A.3.7

19B Resolution of Applicable Unresolved Safety Issues and Generic Safety Issues

Safety Issues Index

Title	NRC Priority	Tier 2 Subsection
Generic Issues		
A-1 Water Hammer	Resolved	19B.2.2 COL App.
A-7 Mark I Long-Term Program	Resolved	19B.2.3
A-8 Mark I Containment Pool Dynamic Loads—Long-Term Program	Resolved	19B.2.4
A-9 ATWS	Resolved	19B.2.5
A-10 BWR Feedwater Nozzle Cracking	Resolved	19B.2.6
A-13 Snubber Operability Assurance	Resolved	19B.2.7
A-24 Qualification of Class 1E Safety-Related Equipment	Resolved	19B.2.8
A-25 Non-Safety Loads on Class 1E Power Sources	Resolved	19B.2.9
A-31 RHR Shutdown Requirements	Resolved	19B.2.10
A-35 Adequacy of Offsite Power Systems	Resolved	19B.2.11
A-36 Control of Heavy Loads Near Spent Fuel	Resolved	19B.2.12 COL App.
A-39 Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	Resolved	19B.2.13
A-40 Seismic Design Criteria—Short-Term Program	Resolved	19B.2.14
A-42 Pipe Cracks in Boiling Water Reactors	Resolved	19B.2.15
A-44 Station Blackout	Resolved	19B.2.16
A-47 Safety Implications of Control Systems	Resolved	19B.2.17 COL App.
A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	Resolved	19B.2.18
B-10 Behavior of BWR Mark III Containments	Resolved	19B.2.19
B-17 Criteria for Safety-Related Operator Actions	Medium	Appendix 18A
B-36 Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	Resolved	19B.2.21
B-55 Improved Reliability of Target Rock Safety/Relief Valves	Medium	19B.2.22

Safety Issues Index (Continued)

Title	NRC Priority	Tier 2 Subsection
B-56 Diesel Reliability	High	19B.2.23
B-61 Allowable ECCS Equipment Outage Periods	Resolved	19B.2.24
B-63 Isolation of Low Pressure Systems Connected to the Reactor Coolant pressure Boundary	Resolved	19B.2.25
B-66 Control Room Infiltration Measurements	Resolved	19B.2.26
C-1 Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	Resolved	19B.2.27 COL App.
C-10 Effective Operation of Containment Sprays in a LOCA	Resolved	19B.2.28
C-17 Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	Resolved	19B.2.29 COL App.
New Generic Issues		
15 Radiation Effects on Reactor Vessel Supports	High	19B.2.30
23 Reactor Coolant Pump Seal Failures	High	19B.2.31
25 Automatic Air Header Dump on BWR Scram System	Resolved	19B.2.32
40 Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Resolved	19B.2.33
45 Inoperability of Instrumentation Due to Extreme Cold Weather	Resolved	19B.2.34
51 Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	Resolved	19B.2.35 COL App.
57 Effects of Fire Protection System Actuation on Safety-Related Equipment	Medium	19B.2.36
67.3.3 Improved Accident Monitoring	Resolved	19B.2.37
75 Generic Implications of ATWS Events at the Salem Nuclear Plant	Resolved	19B.2.38 COL App.
78 Monitoring of Fatigue Transient Limits for Reactor Coolant System	Medium	19B.2.39
83 Control Room Habitability	Possible Res.	19B.2.40
86 Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	Resolved	19B.2.41
87 Failure of HPCI Steam Line Without Isolation	Resolved	19B.2.42
89 Stiff Pipe Clamps	Medium	19B.2.43
103 Design for Probable Maximum Precipitation	Resolved	19B.2.44
105 Interfacing Systems LOCA at BWRs	High	19B.2.45 COL App.

Safety Issues Index (Continued)

Title	NRC Priority	Tier 2 Subsection
106 Piping and Use of Highly Combustible Gases in Vital Areas	Medium	19B.2.46
118 Tendon Anchorage Failure	Resolved	19B.2.48
124 Auxiliary Feedwater System Reliability	Resolved	19B.2.51
128 Electrical Power Reliability	Resolved	19B.2.52
142 Leakage Through Electrical Isolators in Instrumentation Circuits	Medium	19B.2.53
143 Availability of Chilled Water Systems	High	19B.2.54
145 Actions to Reduce Common Cause Failures	Resolved	19B.2.55 COL App.
153 Loss of Essential Service Water in LWRs	High	19B.2.57 COL App.
155.1 More Realistic Source Term Assumptions	Resolved	19B.2.58
Human Factors Issues		
HF.1.1 Shift Staffing	Resolved	18.8.2
HF.4.4 Guidelines for Upgrading Other Procedures	High	18.8.1 18E.1.7
HF.5.1 Local Control Stations	High	18.8.11
HF.5.2 Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	High	18.8.9
Issues Resolved With No New Requirements		
A-17 Systems Interaction in Nuclear Power Plants	Resolved	19B.2.59
A-29 Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage	Resolved	19B.2.60 COL App.
B-5 Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	Resolved	19B.2.61
C-8 Main Steamline Leakage Control Systems	Resolved	19B.2.61.1
29 Bolting Degradation or Failure in Nuclear Power Plants	Resolved	19B.2.62
82 Beyond Design Basis Accidents in Spent-Fuel Pools	Resolved	19B.2.63
113 Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	Resolved	19B.2.64
120 On-Line Testability of Protection Systems	Resolved	19B.2.49 COL App.

Safety Issues Index (Continued)

Title	NRC Priority	Tier 2 Subsection
121 Hydrogen Control for Large, Dry PWR Containments	Resolved	19B.2.50
151 Reliability of Anticipated Transient without Scram Recirculation Pump Trip in BWRs	Resolved	19B.2.56 COL App.
TMI Issues		
I.A.1.1 Shift Technical Advisor	Resolved	COL App.
I.A.1.2 Shift Supervisor Administrative Duties	Resolved	COL App.
I.A.1 3 Shift Manning	Resolved	COL App.
I.A.1.4 Long-Term Upgrading	Resolved	Appendix 18E
I.A.2.1(1) Qualifications-Experience	Resolved	COL App.
I.A.2.1(2) Training	Resolved	COL App.
I.A.2.1(3) Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	Resolved	COL App.
I.A.2.3 Administration of Training Programs	Resolved	COL App.
I.A.2.6(1) Revise Regulatory Guide 1.8	Resolved	COL App.
I.A.3.1 Revise Scope of Criteria for Licensing Examinations	Resolved	COL App.
I.A.4.1(2) Interim Changes in Training Simulators	Resolved	COL App.
I.A.4.2(1) Research on Training Simulators	Resolved	19A.3.1
I.A.4.2(2) Upgrade Training Simulator Standards	Resolved	19A.3.1
I.A.4.2(3) Regulatory Guide on Training Simulators	Resolved	19A.3.1
I.A.4.2(4) Review Simulators for Conformance to Criteria	Resolved	19A.3.1
I.C.1(1) Small-Break LOCAs	Resolved	COL App.
I.C.1(2) Inadequate Core Cooling	Resolved	COL App.
I.C.1(3) Transients and Accidents	Resolved	1A.2.1
I.C.2 Shift and Relief Turnover Procedures	Resolved	COL App.
I.C.3 Shift Supervisor Responsibilities	Resolved	COL App.
I.C.4 Control Room Access	Resolved	COL App.
I.C.5 Procedures for Feedback of Operating Experience to Plant Staff	Resolved	19A.3.6
I.C.6 Procedures for Verification of Correct Performance of Operating Activities	Resolved	COL App.
I.C.7 NSSS Vendor Review of Procedures	Resolved	COL App.

Safety Issues Index (Continued)

Title	NRC Priority	Tier 2 Subsection
I.C.8 Pilot-Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	Resolved	COL App.
I.D.1 Control Room Design Reviews	Resolved	1A.2.2
I.D.2 Plant Safety Parameter Display Console	Resolved	1A.2.3
I.D.3 Safety System Status Monitoring	Medium	19A.2.17
I.D.5(2) Plant Status and Post-Accident Monitoring	Resolved	19B.2.65
I.D.5(3) On-Line Reactor Surveillance System	Near Res.	19B.2.66
I.F.2(2) Include QA Personnel in Review and Approval of Plant Procedures	Resolved	19A.2.43
I.F.2(3) Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Resolved	19A.2.43
I.F.2(6) Increase the Size of Licensees' QA Staff	Resolved	19A.2.43
I.F.2(9) Clarify Organizational Reporting Levels for the QA Organization	Resolved	19A.2.43
I.G.1 Training Requirements	Resolved	1A.2.4
I.G.2 Scope of Test Program	Resolved	19B.2.67
II.B.1 Reactor Coolant System Vents	Resolved	1A.2.5 COL App.
II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	Resolved	1A.2.6
II.B.3 Post-Accident Sampling	Resolved	1A.2.7
II.B.4 Training for Mitigating Core Damage	Resolved	COL App.
II.B.8 Rulemaking Proceeding on Degraded Core Accidents	Resolved	19A.2.1 19A.2.2 19A.2.43 19A.2.45
II.D.1 Testing Requirements	Resolved	1A.2.9
II.D.3 Relief and Safety Valve Position Indication	Resolved	1A.2.10
II.E.4.1 Dedicated Penetrations	Resolved	1A.2.13
II.E.4.2 Isolation Dependability	Resolved	1A.2.14
II.E.4.4 Purging	Resolved	19A.2.27
II.E.6.1 Test Adequacy Study	Resolved	19B.2.68 COL App.
II.F.1 Additional Accident Monitoring Instrumentation	Resolved	1A.2.15
II.F.2 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	Resolved	1A.2.16
II.F.3 Instruments for Monitoring Accident Conditions	Resolved	1A.2.17
II.J.4.1 Revise Deficiency Reporting Requirements	Resolved	COL App.

Safety Issues Index (Continued)

Title	NRC Priority	Tier 2 Subsection
II.K.1(5) Safety-Related Valve Position Description	Resolved	1A.2.18 18.8.7
II.K.1(10) Review and Modify Procedures for Removing Safety-Related Systems from Service	Resolved	1A.3.2
II.K.1(13) Propose Technical Specifications Changes Reflecting Implementation of All Bulletin Items	Resolved	19B.2.69
II.K.1(22) Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	Resolved	1A.2.20
II.K.1(23) Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	Resolved	1A.2.21
II.K.3(3) Report Safety and Relief Valve Failures Promptly and Challenges Annually	Resolved	1A.3.4
II.K.3(11) Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	Resolved	19B.2.70
II.K.3(13) Separation of HPCI and RCIC System Initiation Levels	Resolved	1A.2.22
II.K.3(15) Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	Resolved	1A.2.23 COL App.
II.K.3(16) Reduction of Challenges and Failures of Relief Valves-Feasibility Study and System Modification	Resolved	1A.2.24
II.K.3(17) Report and Outage of ECC Systems—Licensee Report and Technical Specification Changes	Resolved	1A.2.25
II.K.3(18) Modification of ADS Logic—Feasibility Study and Modification for Increased Diversity for Some Event Sequences	Resolved	1A.2.26
II.K.3(21) Restart of Core Spray and LPCI Systems on Low Level-Design and Modification	Resolved	1A.2.27
II.K.3(22) Automatic Switchover of RCIC System Suction—Verify Procedures and Modify Design	Resolved	1A.2.28
II.K.3(24) Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	Resolved	1A.2.29
II.K.3.(25) Effect of Loss of AC Power on Pump Seals	Resolved	1A.2.30
II.K.3(27) Provide Common Reference Level for Vessel Level Instrumentation	Resolved	1A.2.21
II.K.3(28) Study and Verify Qualification of Accumulators on ADS Valves	Resolved	1A.2.31
II.K.3(30) Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	Resolved	1A.2.32

Safety Issues Index (Continued)

Title	NRC Priority	Tier 2 Subsection
II.K.3(31) Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	Resolved	1A.2.33
II.K.3(44) Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	Resolved	1A.2.33.1
II.K.3(45) Evaluate Depressurization with Other Than Full ADS	Resolved	19A.2.11
II.K.3(46) Response to List of Concerns from ACRS Consultant	Resolved	1A.2.33.3
III.A.1.1(1) Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	Resolved	COL App.
III.A.1.2(1) Technical Support Center	Resolved	19A.3.4
III.A.1.2(2) On-Site Operational Support Center	Resolved	19A.3.4
III.A.1.2(3) Near-Site Emergency Operations Facility	Resolved	19A.3.4
III.A.2.1(1) Publish Proposed Amendments to the Rules	Resolved	COL App.
III.A.2.1(2) Conduct Public Regional Meetings	Resolved	COL App.
III.A.2.1(3) Prepare Final Commission Paper Recommending Adoption of Rules	Resolved	COL App.
III.A.2.1(4) Revise Inspection Program to Cover Upgraded Requirements	Resolved	COL App.
III.A.2.2 Development of Guidance and Criteria	Resolved	COL App.
III.A.3.3(1) Install Direct Dedicated Telephone Lines	Resolved	COL App.
III.A.3.3(2) Obtain Dedicated, Short-Range Radio Communication Systems	Resolved	COL App.
III.D.1.1(1) Review Information Submitted by Licensee Pertaining to Reducing Leakage from Operating Systems	Resolved	1A.2.34
III.D.3.3(1) Issue Letter Requiring Improved Radiation Sampling Instrumentation	Resolved	19A.2.39
III.D.3.3(2) Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	Resolved	19A.2.39
III.D.3.3(3) Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	Resolved	19A.3.5
III.D.3.3(4) Issue a Regulatory Guide	Resolved	19A.3.5
III.D.3.4 Control Room Habitability	Resolved	1A.2.36

19B.1 Introduction

19B.1.1 Purpose

The ABWR has proposed technical resolutions of those Unresolved Safety Issues (USI) and medium and high priority Generic Safety Issues (GSI) which are identified in the version of NUREG-0933 through Supplement 15 (Reference 19B.1.1-1) and which are technically relevant to the ABWR design in accordance with 10 CFR 52.47(a)(iv). NUREG-0933 and associated correspondence (References 19B.1.1-2 and 19B.1.1-3) were reviewed and evaluated for the ABWR. The TMI issues satisfying Section II of NUREG-0800, Standard Review Plan, are addressed in Appendix 1A; and those satisfying 10 CFR 50.34(f) are addressed in Appendix 19A. The remaining issues are addressed in Subsection 19B.2.

The following guidelines were used in the review of NUREG-0933 to eliminate potentially non-relevant issues to the ABWR design:

(1) Priority rating of low, dropped, or not yet prioritized

(2) Operational, environmental, licensing, or other NRC impact with no plant design content

(3) No design content applicable to the ABWR design except for NRC identified issues

(4) Resolved with no new requirements except for ACRS and NRC selected issues

In addition, the NRC staff assisted in identifying relevant and current issues and resolutions. The group of issues remaining are identified in the Safety Issues Index and are evaluated in the referenced subsection. Where COL applicant is indicated in the Tier 2 subsection column, the issue is included in Subsection 19B.3 for the COL applicant to address and evaluate. These COL issues pertain to operating personnel including staffing, training, qualification and licensing; operating procedures including post accident operation, severe accident safety reviews, improved emergency preparedness and radiation effects and deficiency reporting; and assisting in the development of regulatory documentation. The COL applicant is required to provide the resolution for each issue as described below.

The documentation of the issue evaluation is comprised of four sections:

- ISSUE,
- ACCEPTANCE CRITERIA,
- RESOLUTION, and

- REFERENCES.

The ISSUES statement is a brief summary description of the issue. The ACCEPTANCE CRITERIA are taken from NUREG-0933 and GIMCS (Reference 19B.1.1-2) resolution references and where there is no formal NRC resolution, accepted industry codes and standards and good engineering practices. The RESOLUTION contains the technical resolution of the issue for the ABWR Standard Plant design. The REFERENCES identifies documentation other than Tier 2.

References

19B.1.1-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.1.1-2 “Generic Issue Management Control System” - Fourth Quarter FY-93, Update, Memorandum for James M. Taylor from E. S. Beckjord dated March 30, 1993.

19B.1.1-3 “Advanced Light Water Reactor Utility Requirements Document”, Volume II, Electric Power Institute, Advanced LWR Program.

19B.2 Safety Issues

19B.2.1 Issue, Acceptance Criteria and ABWR Resolution

19B.2.2 A-1: Water Hammer

Issue

Unresolved Safety Issue (USI) A-1 in NUREG-0933 (Reference 19B.2.2-1) addresses identifying the probable causes of water hammer and minimizing the susceptibility of fluid systems and components to water hammer by correcting design and operational deficiencies.

Water hammer is defined as a rapid deviation in pressure caused by a change in the velocity of a fluid in a closed volume. There are various types of water hammer, including steam condensation-induced water hammer, which occurs in the secondary side of a PWR steam generator at the connection to the feedwater line. This type of water hammer involves steam generator feedrings and piping. Water hammer has been observed in many fluid systems including residual heat removal, containment spray, service water, feedwater systems, and main steam lines. In addition to condensation-induced water hammer, other forms of initiating events which cause water hammer can occur, such as steam driven slugs of water, pump startup with partially empty lines, and rapid valve cycling.

Regardless of the initiating event, water hammer and the resulting fluid accelerations can cause damage to the affected fluid system. The level of severity of damage depends upon the event, and can range from minor damage such as overstressed pipe hangers to major damage to restraints, piping and components.

According to NUREG-0927 (Reference 19B.2.2-2), water hammer can be induced by operator/maintenance actions and by design inadequacies. Experience has shown that water hammer events reported on LERs are about equally divided between operator or maintenance actions and design deficiencies. The NRC implemented Standard Review Plan (SRP) changes relative to the design, operation, and maintenance of new plants to minimize the probability and effects of water hammer, and issued a Branch Technical Position (BTP) for pre-operational tests.

Acceptance Criteria

Reference 19B.2.2-1 concluded that USI A-1 was resolved by the publication of SRP sections:

SRP	Revision
3.9.3	1
3.9.4	2
5.4.6	3
5.4.7	3
6.3.1	2
9.2.1	3
9.2.2	2
10.3	3
10.4.7	3

Compliance with these SRPs becomes the acceptance criteria for resolving this issue.

Resolution

The ABWR design complies with the above listed SRPs and therefore the water hammer issue is resolved. Of all the ABWR systems, the systems discussed below are the only systems considered as having a potential for water hammer.

Potential water hammer conditions are prevented by implementation of the following analyses, design features, and pre-operational tests. Tier 2 section references are given.

(1) Water hammer evaluation is required for specific piping regions as follows:

(a) Condensate and Feedwater System. The feedwater lines are demonstrated to have low probability of failure from water hammer effects. Subsections 10.4.7.3, 3E.6.2.2, 3E.6.2.7, 20.3.10 RAI-10 Question-Response 430.89.

(b) Main steam lines are analyzed for dynamic loadings due to fast closing of the turbine stop valves. Subsection 5.4.9.1(4).

(c) All components of the main steam supply system are designed to accommodate the loads and stresses resulting from steam hammer. Subsection 10.3.3.

(2) Water hammer evaluation is part of the Leak Before Break (LBB) analysis consideration. [Subsection 3.6.3.2(5)] If the COL applicant applies the LBB analysis (Subsection 3.6), the systems identified by Table 3E.1-1 would be evaluated for water hammer. (Subsection 3.6.3, and Table 3E.1-1) These systems include main steam, feedwater RCIC, HPCF, RHR, and CUW. Relative to LBB, feedwater lines were demonstrated to have immunity to failure from water hammer effects, (Subsection 3E.6.2.7) RCIC, HPCF, and RHR Systems are precluded from water hammer by their keep-filled features and absence of fast acting valves.

(3) Applicable systems are filled with water, and kept filled with water, which prevents water hammer when pumps are started from a standby condition. Systems described in Tier 2 are as follows:

(a) Operating procedures will be developed so that all divisions of the Reactor Service Water (RSW) System are maintained full of water to prevent water hammer [Subsection 9.2.15.1.1(5)]

(b) Operating procedures will be developed so that all components of the Turbine Service Water (TSW) System are maintained full to prevent water hammer [Subsection 9.2.16.2.2(5)]

(c) Residual Heat Removal (RHR) [Subsections 5.4.1.1.4, 6.3.2.2.5, and 14.2.12.1.8(3)(m)]

(d) High Pressure Core Flooder (HPCF) [Subsections 6.3.2.2.5, and 14.2.12.1.10(3)(n)]

(e) Reactor Core Isolation Cooling (RCIC) [Subsections 5.4.6.2.5.1, 6.3.2.2.5, and 14.2.12.1.9(3)(n)]

(f) HVAC Emergency Cooling Water System [Subsection 9.2.13.1.2(6)]

(4) Condensation Induced Water Hammer (CIWH) for the ECCS systems (RHR, HPCF, and RCIC) was evaluated for the ABWR (Reference 19B.2.2-5). The conclusion was that the ECCS injection piping configuration was not susceptible to CIWH.

(a) For the RHR System low pressure flooder (LPFL) mode, the water in the sloped, but nearly horizontal, injection line flashes to steam during reactor depressurization. An analysis was performed that indicated about 80% of the water remained in the pipe after depressurization. Therefore, slow injection of cold water by the LPFL injection valve into the horizontal LPFL pipe partially filled with saturated water will not cause CIWH.

(b) For the HPCF System, in the event of a LOCA, the high pressure flooder spargers located inside the RPV shroud are immersed in a two-phase mixture. During the flashing period prior to HPCF initiation, the RPV is depressurizing and water in the piping can flash. A steam bubble can form at the piping's high point. For the HPCF high pressure system, injection begins within a few seconds, and the entrance of subcooled water could cause decompression inside the pipe. Any water slug accelerated from the reactor side towards the upstream piping will flash into a two-phase mixture because the water is in a saturated condition. A slug of two-phase mixture, which is highly compressible, colliding with another surface has been analyzed and found to produce a pressure pulse of the order of 6.8947E+04 Pa to 1.3789E+05 Pa. This analysis was done earlier for typical BWR-5 and BWR-6 piping using TRACB01 computer code. 1.3789E+5 Pa pressure pulses are not considered significant, and it is concluded that CIWH is not a problem for the HPCF injection piping.

(c) For the RCIC System during system initiation, the water level in the reactor is a Level 2 or higher, which is higher than the feedwater nozzle height. The fluid condition at the feedwater sparger is water when RCIC water is pumped into the vessel. Therefore, CIWH will not occur at the time of RCIC makeup water injection into the reactor vessel.

(5) Pre-operational tests are specified for the purpose of verifying the piping keep-fill methods are operational. Filled pipelines preclude water hammer associated with pump startup.

(a) RHR [Subsection 14.2.12.1.8(3)(m)]

(b) HPCF [Subsection 14.2.12.1.10(3)(n)]

(c) RCIC [Subsection 14.2.12.1.9(3)(n)]

References

19B.2.2-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.2-2 NUREG-0927, Revision 1, “Evaluation of Water Hammer Occurrence in Nuclear Power Plants”, U.S. NRC, March 1984.

19B.2.2-3 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.2-4 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.2-5 Jack Fox, GE, to Chet Poslusny, NRC, “Submittal Supporting Accelerated ABWR Review Schedule – Water Hammer Evaluation”, May 11, 1993.

19B.2.3 A-7: Mark I Long-Term Program

Issue

During testing for an advanced BWR containment system design (MARK III), suppression pool hydrodynamic loads were identified which had not been considered in the original design of the MARK I containment system. To address this issue, a MARK I Owners Group was formed and the assessment was divided into a short-term and long-term program. The results of the NRC staff’s review of the MARK I Containment Short-Term Program are described in NUREG-0408 (Reference 19B.2.3-6). The long-term program (LTP) was conducted to provide a generic basis to define suppression pool hydrodynamic loads and the related structural acceptance criteria, such that a comprehensive reassessment of each MARK I containment system would be performed. A series of experimental and analytical programs were conducted by the MARK I Owners Group to provide the necessary bases for the generic load definition and structural assessment techniques. The generic methods proposed by the MARK I Owners Group, as modified by the NRC staff’s requirements, will be used to perform plant-unique analyses, which will identify the plant modifications, if any, that will be needed to restore the originally intended margin of safety in the MARK I containment designs. This item was originally identified in NUREG-0371 (Reference 19B.2.3-5) and was later determined to be a Unresolved Safety Issue (USI).

Acceptance Criteria

The objectives of the LTP were to establish design basis (conservative) loads that are appropriate for the anticipated life of each Mark I boiling water reactor (BWR) facility (40 years) and to restore the originally intended design safety margins for each Mark I containment system. The principal thrust of the LTP has been the development of generic methods for the definition of suppression pool hydrodynamic loadings and the associated structural assessment techniques for the Mark I configuration.

Resolution

On the basis of the review of the experimental and analytical programs conducted by the Mark I Owners Group, the NRC staff concluded that, with one exception, the proposed suppression pool hydrodynamic load definition procedures, as modified by the NRC Acceptance Criteria in Appendix A of Reference 19B.2.3-1, will provide a conservative estimate of these loading conditions. The exception is the lack of an acceptable specification for the downcomer condensation oscillation loads. In addition, the staff requested confirmatory programs to justify the adequacy of the loading specifications in the following three areas:

(1) adequacy of the data base for specifying torus wall pressures during condensation oscillations,

(2) possibility of asymmetric torus loading during condensation oscillations, and

(3) effect of fluid compressibility in the vent system on pool-swell loads.

These programs were documented in Reference 19B.2.3-3. This report supplements the Mark I SER (NUREG-0661) by addressing the outstanding issues relating to the Mark I containment LTP, namely the downcomer condensation oscillation load definition and the confirmatory analyses and test programs that are intended to justify the adequacy of the load specifications.

The Mark I torus pool and vent configuration is not similar to the ABWR annular pool and vent design. The Mark I pool loads are not directly applicable to the ABWR because of these configuration differences and the results thus obtained cannot be used directly in the ABWR design. Nevertheless, since the line clearing phenomena for single SRV discharge conditions are the same, the results obtained from the Mark I Owner's Group program were used as a database for definition of the SRV loads that are applicable to the ABWR and utilized in resolution of Issue A-39 (Subsection 19B.2.13).

References

19B.2.3-1 NUREG-0661, "Safety Evaluation Report, Mark I Long Term Program, Resolution of Generic Technical Activity A-7", U.S. NRC, July 1980.

19B.2.3-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.3-3 NUREG-0661, Supplement 1, "Safety Evaluation Report for the MARK I Containment Long-Term Program", U.S. NRC, August 1982.

19B.2.3-4 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.3-5 NUREG-0371, “Task Action Plans for Generic Activities Category A,” U.S. NRC, 1978.

19B.2.3-6 NUREG-0408, *“Mark I Containment Short-Term Program”*, U.S. NRC, 1977.

19B.2.4 A-8: Mark II Containment Pool Dynamic Loads Long-Term Program

Issue

As a result of the GE testing program for the MARK III pressure-suppression containment program, new containment loads associated with a postulated LOCA were identified in 1975 which had not been explicitly included in the original design of MARK I and MARK II containments. These loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA event. Other pool dynamic loads previously unaccounted for result from the actuation of safety/relief valves (SRVs) in the MARK II containment. The review and evaluation of the MARK I loads were addressed in USI A-7 and SRV loads for all suppression-type containments were addressed in USI A-39 (Reference 19B.2.4-3). This item was originally identified in NUREG-0371 (Reference 19B.2.4-4) and was later determined to be a USI.

Acceptance Criteria

The NRC established an acceptance criteria for Mark II LOCA-Related Pool Dynamic Loads addressing pool swell loads, condensation oscillation loads, and chugging loads (Reference 19B.2.4-1, Appendix A, and Reference 19B.2.4-2). The original design of the Mark II containment system considered only those loads normally associated with design-basis accidents. These included pressure and temperature loads associated with a LOCA, seismic loads, dead loads, jet impingement loads, hydrostatic loads due to water in the suppression chamber, overload pressure test loads, and construction loads. However, since the establishment of the original design criteria, additional loading conditions have been identified that must be considered for the pressure-suppression containment-system design.

Resolution

As described in Subsection 3B.4.2.1, the ABWR pool swell response calculations to quantify pool swell loads were based on a simplified, one-dimensional analytical model which was reviewed and approved by the NRC staff (Reference 19B.2.4-5). Since the ABWR vent design system utilizes horizontal vents (like Mark III containments) rather than vertical, additional studies were performed to assure the applicability of the Mark II model to the ABWR.

This issue is resolved for the ABWR.

References

19B.2.4-1 NUREG-0808, "MARK II Containment Program Evaluation and Acceptance Criteria", U.S. NRC, August 1981.

19B.2.4-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.4-3 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.4-4 NUREG-0371, "Task Action Plans for Generic Activities Category A", U.S. NRC, 1978.

19B.2.4-5 NEDE-21544-P, "Mark II Pressure Suppression Containment Systems: Analytical Model of the Pool Swell Phenomena", U.S. NRC, December 1976.

19B.2.5 A-9: Anticipated Transients Without Scram, (ATWS)

Issue

This issue, A-9 (Reference 19B.2.5-1), addresses the concern that the reactor can attain safe shutdown after incurring an anticipated transient (such as a loss of feedwater, loss of condenser vacuum, or loss of offsite power) with a failure of the reactor protection system to shutdown the reactor. The technical report on ATWS (WASH-1270) (Reference 19B.2.5-2) discussed the probability of an ATWS event as well as an appropriate safety objective for the event. In 1975 the staff published a status report on each vendor analysis which included guidelines on analysis models and ATWS safety objectives. This issue was resolved by the NRC with the publication of a final rule, 10 CFR 50.62 (Reference 19B.2.5-3).

Acceptance Criteria

The acceptance criteria for the resolution of this issue is that the reactor must be capable of reaching a safe shutdown condition as identified in 10 CFR 50.62 after incurring an anticipated transient and a failure to scram. Specifically, 10 CFR 50.62 requires the BWR to have automatic recirculation pump(s) trip, an alternate rod insertion system and an automatic standby liquid control system.

Resolution

For ATWS prevention / mitigation for the ABWR, the following are provided:

- An ARI system diverse and independent of the reactor protection system,
- Electric insertion of the fine motion control rod drives which is also diverse and independent of the reactor protection system,

- Automatic recirculation pump trip, and
- Automatic initiation of the standby liquid control system.

These features are described in Section 15.8 and fulfill the requirements of 10 CFR 50.62 to resolve this issue for the ABWR, and the details are discussed in Reference 19B.2.5-4.

References

19B.2.5-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.5-2 WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Reactors", U.S. NRC, September 1973.

19B.2.5-3 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

19B.2.5-4 NEDE-31096-A, GE Licensing Topical Report, "ATWS-Response to NRC ATWS Rule, 10CFR50.62", February, 1987.

19B.2.6 A- 10: BWR Feedwater Nozzle Cracking

Issue

Inspections of operating BWRs conducted up to April 1978 revealed cracks in the feedwater nozzles of 20 reactor vessels. Most of these BWRs contained 4 nozzles with diameters ranging from 250A to 300A (10 inches to 12 inches). Although most cracks range from 12.7 to 19.05 mm (0.5 inch to 0.75 inch) in depth (including cladding), one crack penetrated the cladding into the base metal for a total depth of approximately 38.1 mm (1.5 inch).

It was determined that cracking was due to high cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low feedwater temperature when flow is unsteady and intermittent. Once initiated, the cracks enlarge from high pressure and thermal cycling associated with startups and shutdowns. This item was originally identified in NUREG-0371 and was later determined to be an unresolved safety issue (USI) (References 19B.2.6-1 and 19B.2.6-2)

Acceptance Criteria

The acceptance criteria are based on developing a design that provides protection to the feedwater nozzle from the water temperature fluctuations. The feedwater nozzles experience thermal stress because the incoming feedwater is colder than that in the vessel. It is much colder during startups before the feedwater heaters are in service and during shutdown after the heaters have been taken out of service. Turbulent mixing of the hot water returning from the steam separators and dryers and the incoming cold feedwater causes thermal stress cycling in the nozzle bore unless it is thoroughly protected by the sparger thermal sleeve.

In previous designs bypass leakage past the junction of the thermal sleeve and nozzle safe end has been the primary source of cold water impinging on the nozzle bore. A secondary source is the layer of water that sheds off after being cooled by contact with the outer surface of the sleeve.

Resolution

The welded double sleeve design gives a low fatigue usage factor in the nozzle bore and at the inner nozzle corner. The design protects the nozzle from fluctuating temperatures and, therefore, the issue of high cycle fatigue in the feedwater nozzle has been resolved for the ABWR.

The ABWR utilizes a double feedwater nozzle thermal sleeve as can be seen on Figure 19B-1. An inner thermal sleeve leading the cooler feedwater to the feedwater sparger is welded to the nozzle safe end. The welded thermal sleeve design was adopted to assure that there is no leakage of cold feedwater between the thermal sleeve and the safe end. A secondary thermal sleeve is placed concentrically in the annulus between the inner thermal sleeve and the nozzle bore to prevent cold water that may be shedding from the outside surface of the inner sleeve impinge on the nozzle bore and the inside nozzle corner.

The material of the nozzle forging is SA-508, Class 3 low alloy steel and that of the safe end is SA-508, Class 1 carbon steel. The carbon steel safe end is welded to the nozzle forging with a carbon steel weld. The nozzle itself has no cladding.

Welded thermal sleeves have been successfully used in at least three domestic reactors and in BWR/5s in Japan since 1977. The welded double thermal sleeve with no cladding inside the nozzle is considered an improvement of the welded single sleeve design in that the outer thermal sleeve provides additional protection against high cycle fatigue in the nozzle bore and the inside nozzle corner. The double thermal sleeve as applied to the ABWR has not been used in earlier plants although Monticello and Tsuruga (Japan) are using similar designs.

The ABWR feedwater nozzle and thermal sleeve design does not correspond to any design mentioned in Table 2 of NUREG-0619. The closest design is considered to be “Welded, clad removed (spargers have top mounted elbows)”. Hence, the proposed program for ISI of the ABWR feedwater nozzles and spargers is based on this design. Based upon programs approved by the NRC allowing relief from periodic PT inspections, the following program is proposed:

UT examination from the external surface of the nozzle safe ends, nozzle bores and nozzle blend radius every second outage. If indications are found in the safe ends, evaluate per Section XI of the ASME Code. If recordable indications are interpreted as cracks in any nozzle, proceed with repair as outlined in NUREG-0619, Paragraph 4.3.2.3.

Visual inspection of flow holes and welds in sparger arms and sparger tees every fourth outage.

Visual inspection of accessible areas of the nozzles from the ID surface on the same schedule as core internal components.

It is believed that UT examination of the nozzle bore using advanced techniques give better results than PT inspection of accessible areas. This method has successfully been tried out on several domestic reactors. Depending upon actual operating experience, it may be possible to extend the period between UT examinations.

Reference

19B.2.6-1 NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", U.S. NRC, November 1980.

19B.2.6-2 NUREG-0371, "Task Action Plans for Generic Activities (Category A)", U.S. NRC, November 1978.

19B.2.7 A-13: Snubber Operability Assurance

Issue

Generic Safety Issue (GSI) A-13 in NUREG-0933 (Reference 19B.2.7-1), addresses snubber selection and operability for safety related systems and components by identifying the need for:

(1) A consistent means of determining snubber operability through standardized functional testing.

(2) A set of criteria for selection and specification.

(3) Preservice and inservice inspection programs.

Snubbers are utilized primarily as seismic and pipe whip restraints at operating plants. Their safety function is to operate as rigid supports for restraining the motion of systems or components under dynamic load conditions such as earthquakes and severe hydraulic transients, e.g., pipe breaks.

According to NUREG-0933, a substantial number of Licensee Event Reports (LERs), concerning snubber operability, were issued by utilities. A review of these LERs showed that a variety of methods were employed to determine the operability of the snubbers and that different types of snubbers were used for systems with similar configurations.

Acceptance Criteria

The acceptance criteria for the resolution of GSI A-13 is that the design, specification, installation, and in-service operability of snubbers must meet the intent of the guidance given in SRP Section 3.9.3 (Reference 19B.2.7-2).

Specifically, during the design of safety systems or components for which snubbers are to be used, sufficient consideration should be given as to their unique application, i.e., their response to normal, upset, and faulted conditions and the effect of these responses on the associated system and/or component.

Resolution

For the ABWR design, snubbers are minimized by using design optimization procedures. However, where required, snubber supports are used as shock arrestors for safety-related systems and components. Snubbers are used as structural supports during a dynamic event such as earthquake or pipe break, but during normal operation act as passive devices which accommodate normal expansions and contractions without resistance.

Assurance of snubber operability for the ABWR design is provided by incorporating analytical, design, installation, in-service, and verification criteria to meet the intent of the draft Regulatory Guide (Reference 19B.2.7-3) as described in Subsection 3.9.3.4.1(3). The elements of snubber operability assurance include:

(1) Consideration of load cycles and travel that each snubber will experience during normal plant operating conditions.

(2) Verification that the thermal growth rates of the system do not exceed the required lock-up velocity of the snubber.

(3) Appropriate characterization of snubber mechanical properties in the structural analysis of the snubber-supported system.

(4) For engineered, large bore snubbers, issuance of a design specification to the snubber supplier, describing the required structural and mechanical performance of the snubber with respect to: activation level, release rate, spring rate, dead band, and drag as specified in the draft Regulatory Guide SC-708-4 (Reference 19B.2.7-3). Subsequent verification that the specified design and fabrication requirements were met.

In summary, during the design of safety-related systems or components for which snubbers are to be used, sufficient consideration is given as to their unique application, (i.e., their response to normal, upset and faulted conditions and the effect of these responses on the associated system and / or component). Thus the design, specification, installation, and in-service operability of snubbers meets the intent of SRP Section 3.9.3 and this issue is resolved for the ABWR design.

References

19B.2.7-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.7-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S.NRC.

19B.2.7-3 DRAFT Regulatory Guide (SC-708-4), February 1981.

19B.2.8 A-24: Qualification of Class 1E Safety Related Equipment

Issue

Safety Issue A-24 in NUREG-0933 (Reference 19B.2.8-1) addresses the adequacy of environmental qualification methods and acceptance criteria for Class 1E electrical equipment.

The Nuclear Regulatory Commission (NRC) initially required license applicants to qualify all safety-related equipment to IEEE Std 323-1974 (Reference 19B.2.8-2). Some of the industry qualification methods and concepts proposed in accordance with this standard, such as testing margins, aging effects, and the simulation of worst case environments, were not resolved to the satisfaction of the NRC. It was therefore decided that a generic approach should be developed under A-24 to expedite the review and assessment of equipment qualification methods used by vendors.

All major Nuclear Steam Supply Systems (NSSS) vendors and architect engineers submitted topical reports on their methods of environmental qualification which were reviewed by the NRC and the results documented in NUREG-0588 (Reference 19B.2.8-3). In a subsequent rulemaking, 10 CFR 50.49 (Reference 19B.2.8-4) established the requirement for an environmental qualification program for Class 1E electrical equipment together with rules for its content. References 19B.2.8-2 and 19B.2.8-3 comprise the bases for the rules. Regulatory Guide 1.89 was then revised (Reference 19B.2.8-5) to described an acceptable method for complying with 10 CFR 50.49.

Dynamic and seismic qualification of Class 1E electrical equipment was not included in the scope of 10 CFR 50.49. Existing dynamic and seismic qualification requirements are identified in Regulatory Guide 1.100 (Reference 19B.2.8-6).

Acceptance Criteria

The acceptance criteria for the resolution of Issue A-24 is that safety-related electrical equipment shall be environmentally qualified in accordance with 10 CFR 50.49, and dynamically and seismically qualified in accordance with the acceptance criteria of Regulatory Guide 1.100.

Resolution

The ABWR safety-related electrical equipment is environmentally qualified in accordance with 10 CFR 50.49 as described in Subsection 3.11 and dynamically and seismically qualified in accordance with Regulatory Guide 1.100 as described in Subsection 3.10.

References

19B.2.8-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.8-2 IEEE Std. 323, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations", Institute of Electrical and Electronics Engineers.

19B.2.8-3 NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", U.S. NRC, July 1981.

19B.2.8-4 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plant", Office of the Federal Register, National Archives and Records Administration.

19B.2.8-5 Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants", U.S. NRC.

19B.2.8-6 Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants", U.S. NRC.

19B.2.9 A-25: Non-Safety Loads on Class 1E Power Sources

Issue

Generic Safety Issue (GSI) A-25 in NUREG-0933 (Reference 19B.2.9-1), addresses the potential safety degradation of a Class 1E Power system caused by its connection to a non-safety-related power source or load.

There are two approaches to assuring the reliability of the safety-related system Class 1E power supplies for future plants. The first approach is to allow only Class 1E loads to be connected to Class 1E power supplies. [In previous designs, non-safety electrical equipment was connected to Class 1E power supplies (i.e., the emergency diesel generators) to provide a source of power during loss-of-offsite power (LOPP) events.]

The second approach is to limit the connection of non-safety-related electrical equipment to the Class 1E power systems and assure that when this equipment is connected to the Class 1E power systems that the equipment and the connections conform to the requirements for independence, electrical isolation, and physical separation. These requirements are identified in IEEE Standard 384 (Reference 19B.2.9-2), and guidance is provided in Regulatory Guide 1.75 (Reference 19B.2.9-3). [Supplemental information on Class 1E safety systems may be found in IEEE Standard 603, IEEE Standard 279, and IEEE Standard 308, (References 19B.2.9-4, 19B.2.9-5 and 19B.2.9-6, respectively).]

Both industry and the NRC, through IEEE Standard 384 and Regulatory Guide 1.75, have determined that these design requirements provide an acceptable means of achieving an adequate level of reliability for the Class 1E power supplies. Therefore, a commensurate level of safety for the safety systems is assured.

Acceptance Criteria

The acceptance criteria for the resolution of GSI A-25 is that the reliability and level of safety of Class 1E power sources and the safety systems which they supply may not be degraded by the sharing of loads between safety-related systems and non-safety-related systems.

Specifically, the second approach, identified in the issue statement, shall be used in establishing an acceptable level of reliability and safety for Class 1E power sources and safety-related systems.

This shall be accomplished by assuring that the interface between safety-related and non-safety-related equipment on Class 1E power sources and safety-related systems is adequately controlled by meeting the independence, electrical isolation, and physical separation requirements identified in IEEE Standard 384 and other applicable standards (References 19B.2.9-2, and 19B.2.9-4 through 19B.2.9-6, respectively) taking into consideration the guidance provided in Regulatory Guide 1.75.

Resolution

The ABWR design assures the reliability and safety of the Class 1E power sources and safety-related systems by a highly selective connection (i.e., only one subsystem) of non-safety-related equipment and strict control of the interface between this subsystem and Class 1E power system. Each safety related system conforms to the requirements of IEEE Standard384 (Reference 19B.2.9-2) and meets RG 1.75 (Reference 19B.2.9-3) and addresses IEEE Standard 279 (Reference 19B.2.9-5).

The ABWR design incorporates three independent Class 1E diesel generators (DGs) and a non-Class 1E combustion turbine generator (CTG). The CTG is designed to automatically and independently assume the plant investment protection (PIP) loads, should a LOPP event occur. This is in much the same manner as the DGs assume the Class 1E loads for the same event. Therefore, it is not necessary for the Class 1E buses to assume the PIP loads. (See Subsections 8.2.1 and 8.3.1.)

The ABWR design excludes non-Class 1E from the Class 1E busses, with the exception of the fine-motion control rod drive (FMCRD) subsystem, the associated AC standby lighting system, and the associated DC emergency lighting system. The reliability of the FMCRD subsystem is enhanced for the anticipated transient without scram (ATWS) event by using Class 1E power for the drive motors.

Class 1E load breakers in the switchgear are part of the isolation scheme between the Class 1E power and the non-Class 1E FMCRD loads. In addition to the normal overcurrent tripping of these load breakers, zone selective interlocking (ZSI) is provided between them and the upstream Class 1E bus feed breakers. The Class 1E load breakers, in conjunction with the ZSI feature, provides the needed isolation between the Class 1E bus and the non-Class 1E loads. (See Subsection 8.3.1.1.1 for more details on this feature relative to the FMCRD power circuits.)

Since both the safety systems and their Class 1E power supplies conform to the requirements of IEEE Standard 384 and meet the intent of Regulatory Guide 1.75, an acceptable level of safety exists for both the safety systems and their Class 1E power supplies.

Therefore, this issue is resolved for the ABWR.

References

19B.2.9-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.9-2 IEEE Standard 384, “Criteria for Separation of Class 1E Equipment and Circuits”, The Institute of Electrical and Electronics Engineers, Inc.

19B.2.9-3 Regulatory Guide 1.75, “Physical Independence of Electric Systems”, U.S. NRC.

19B.2.9-4 IEEE Standard 603, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations”, The Institute of Electrical and Electronics Engineers, Inc.

19B.2.9-5 IEEE Standard 279, “Criteria for Protection Systems for Nuclear Power Generating Stations”, The Institute of Electrical and Electronic Engineers, Inc.

19B.2.9-6 IEEE Standard 308, “Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations”, The Institute of Electrical and Electronic Engineers, Inc.

19B.2.10 A-31: Residual Heat Removal (RHR) Shutdown Requirements

Issue

Unresolved Safety Issue (USI) A-31 in NUREG-0933 (Reference 19B.2.10-1), addresses the safe shutdown of the reactor, following an accident or abnormal condition other than a Loss of Coolant Accident (LOCA), from a hot standby condition (i.e., the primary system is at or near normal operating temperature and pressure) to a cold shutdown condition. Considerable emphasis has been placed on long-term cooling which is typically achieved by the residual heat removal system which starts to operate when the reactor coolant pressure and temperature are substantially lower than the hot-standby values.

Even though it may generally be considered safe to maintain a reactor in a hot-standby condition for a long time, experience has shown that there have been abnormal occurrences that required long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment, after a shutdown resulting from an accident or abnormal occurrence, is an important safety function. It is essential that a power plant be able to go from hot-standby to cold-shutdown conditions subsequent to any accident or abnormal occurrence condition.

Acceptance Criteria

The acceptance criterion for the resolution of USI A-31 is that the RHR system shall be designed so that the reactor can be brought from a “Hot Standby” to a “Cold Shutdown” condition as described in SRP Section 5.4.7, (Reference 19B.2.10-2).

Specifically, the RHR system shall meet the intent of the following functional requirements with respect to cooldown:

(1) The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems. These systems shall satisfy 10 CFR 50 Appendix A (Reference 19B.2.10-3) General Design Criteria (GDC) 1 through 5, and 34.

(2) The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak connection, and isolation capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) the system function can be accomplished assuming a single failure.

(3) The system shall be capable of being operated from the control room with either onsite or offsite power available. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable, if suitably justified.

(4) The system(s) shall be capable of bringing the reactor to a cold shutdown condition, with either offsite or onsite power available, within a reasonable period of time following a shutdown, assuming the most limiting single failure.

In addition to the functional requirements listed above, there are certain additional requirements for the RHR system including, pressure relief, pump protection, test and operation.

Resolution

The Residual Heat Removal (RHR) system is composed of three electrically and mechanically independent divisions, except for the outboard containment isolation valves, which are in different electrical divisions than the inboard valves, designated as A, B, and C with each division containing the necessary piping, pumps, valves, and heat exchangers (Subsection 5.4.7).

One of the basic design functions of the RHR system is shutdown. Shutdown cooling to remove decay and sensible heat from the reactor, which also includes the safety-related requirements that the reactor must be brought to a cold shutdown condition using safety grade equipment (Subsection 5.4.7.1.1.7).

The design basis for the RHR Shutdown Cooling subsystem is that it is manually activated by the operator from the control room following insertion of the control rods and normal blowdown to the main condenser (Subsection 5.4.7.1.1.7).

For emergency operations where one of the RHR loops has failed, the RHR system is capable of bringing the reactor to the cold shutdown condition of 373 K (100°C) within 36 hours following reactor shutdown with any two of the three divisions. The subsystem can maintain or reduce this temperature further so that the reactor can be refueled and serviced (Subsection 5.4.7.1.1).

The RHR system is part of the Emergency Core Cooling (ECCS) System, and therefore is required to be designed with redundancy, piping protection, power separation, and other safeguards as required of such systems (Section 6.3).

Shutdown suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of isolation and shutdown following a loss of offsite power (Subsection 9A.5.5.14).

The RHR system is designed to meet General Design Criteria (GDC) 1, 2, 3, 4, and 5 for quality assurance, protection against natural phenomenon, environmental and internally generated missiles, pipe breaks, seismic effects, and fires (Subsection 5.4.7.1.6).

The RHR Shutdown Cooling System is designed to meet the intent of SRP Section 5.4.7, Revision 3, with respect to providing a means of bringing the reactor plant from hot standby to cold shutdown under all accident or abnormal occurrence conditions, as described above.

Therefore, this issue is resolved for the ABWR (Subsection 5.4.7.1.1.7).

References

19B.2.10-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.10-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.10-3 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.11 A-35: Adequacy of Offsite Power System

Issue

Issue A-35 in NUREG-0933 (Reference 19B.2.11-1) concerns the protection of safety-related equipment from the effects of a sustained undervoltage condition or a rapid rate of decay of the frequency of the offsite power source as well as interaction effects between offsite and onsite power sources. Associated testing requirements are also addressed.

The plant operator historically has performed transient and steady-state stability analyses of the offsite power system which were documented in the Safety Analysis Report (SAR). However, abnormal occurrences at several operating plants indicated that a sustained undervoltage condition of the offsite power source not detected by the existing loss of voltage relays could result in a failure of redundant safety-related equipment.

The NRC therefore evaluated the power systems of operating plants to determine the susceptibility of safety-related electrical equipment to:

(1) A sustained undervoltage condition on the offsite power source.

(2) A rapid rate of decay of the offsite power source frequency.

(3) Interaction for the offsite and onsite power sources.

An additional factor evaluated was

(4) The adequacy of testing requirements.

New criteria relative to factors (1), (3) and (4) above were issued in Branch Technical Position (BTP) PSB-1 “Adequacy of Station Electric Distribution System Voltages”, which was incorporated in SRP Section 8.3.1, Appendix A (Reference 19B.2.11-2). Frequency decay [factor (2)] was found not to be a significant safety issue.

Acceptance Criteria

The acceptance criteria for the resolution of USI A-35 is that the design and capability for test and calibration of the undervoltage protection schemes for the Class 1E buses of the onsite power system (while connected to the offsite power source) shall conform to the guidance of BTP PSB-1 in Appendix A of SRP Section 8.3.1.

Specifically, a second level of voltage protection shall be provided for Class 1E equipment in addition to the existing protection based on detecting the complete loss of offsite power to the Class 1E buses. The second level shall have two separate time delays before alerting the control room operator and automatically separating the Class 1E buses from the offsite power source, respectively. Duration of the time delays shall ensure protection from sustained low voltage while avoiding disconnection from the offsite source due to short term transients such as motor starting. The undervoltage protection scheme shall have the capability of being tested and calibrated during power operation. Voltage levels at the safety related buses shall be optimized for the maximum and minimum load conditions that are expected, throughout the anticipated range of offsite power source voltage variation. Technical Specifications are to include limiting conditions of operation, surveillance requirements, and protection equipment setpoints.

Resolution

The conceptual design of an offsite power system and station switchyard(s) for the ABWR design is given in Section 8.2. The interface requirements will ensure that the switchyard(s) provide redundant offsite power feed capability to the nuclear unit, consisting of two preferred power circuits, each capable of supplying the necessary safety loads and other equipment.

The ABWR onsite power systems are described in Section 8.3, and include three redundant and independent 6.9kV Class 1E safety buses. The incoming source breakers trip upon loss of normal power, and emergency power is provided to each Class 1E bus by separate and independent diesel generator (DG) units. A combustion turbine generator automatically assumes the plant investment protection loads, but can be used to manually provide back-up power for any Class 1E bus, should a DG fail or be out of service.

The Class 1E AC Power Systems are described in Subsection 8.3.1.1. Protection against degraded voltage is specifically addressed in Subsection 8.3.1.1.7(8). The protection schemes are designed according to the recommendations of IEEE Standard 741 (Reference 19B.2.11-3), which is consistent with the guidance of BTP PSB-1.

The ABWR Standard Plant Class 1E auxiliary power system is designed in compliance with General Design Criterion (GDC) 18 (Reference 19B.2.11-4) so that inspection, maintenance, calibration and testing can be carried out with a minimum of interference with operation of the nuclear unit, as described in Subsection 8.3.1.1.5.3. On-line testing is greatly enhanced by the design, which utilizes three independent Class 1E divisions. Indication of the system unavailability is provided in the control room.

A Technical Specification establishes limiting conditions for operations, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the undervoltage protection sensors and associated time delay devices.

Protection of the Class 1E power supplies to safety-related equipment from the effects of an undervoltage condition of the offsite power source thus conforms to the guidance of BTP PSB-1, and this issue is, therefore, resolved for the ABWR Standard Plant design.

References

19B.2.11-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.11-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.11-3 ANSI / IEEE 741, “Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations”, Institute of Electrical and Electronics Engineers, Inc.

19B.2.11-4 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.12 A-36: Control of Heavy Loads Near Spent Fuel

Issue

Issue A-36 in NUREG-0933 (Reference 19B.2.12-1), addresses the consequence of dropping heavy loads on spent fuel. Overhead cranes are used to lift heavy objects in the vicinity of spent fuel. If the heavy object, such as a spent-fuel shipping cask or shielding block, were to fall on to spent fuel, there could be a release of radioactivity to the environment that could exceed 10 CFR 100 guidelines. This issue was resolved by the NRC with the publication of NUREG-0612 (Reference 19B.2.12-2) and SRP Section 9.1.5 (Reference 19B.2.12-3).

Acceptance Criteria

The acceptance criteria for the resolution of Issue A-36 is that the overhead heavy load handling systems shall be designed to provide the equipment, procedures and operator training such that no credible drop can cause a release of radioactivity, a criticality accident, an inability to cool fuel within the reactor vessel or spent-fuel pool, or prevent a safe shutdown of the reactor. Where applicable the design shall conform to the industrial and electrical codes, the relevant requirements of General Design Criteria 2, 4, and 61 of 10 CFR 50, Appendix A (Reference 19B.2.12-4) and NUREG-0612.

Resolution

The ABWR design addresses the above criteria as follows:

(1) A transportation routing study will be made of all planned heavy load handling moves to evaluate and minimize safety risks. The study will require the COL to establish the heavy load handling safe load paths and routing plans (Subsections 9.1.5.5, 9.1.5.8 and 9.1.6.6).

(2) The major heavy load handling equipment components (i.e., cranes, hoists, etc.) will be provided with an operating instruction and maintenance manual for reference and utilization by operations and maintenance personnel for use in operating procedures, maintenance procedures and operator training programs. The handling equipment operating procedures will comply with the requirements of NUREG-0612, Subsection 5.1.1(2) (Subsections 9.1.5.4, 9.1.5.8 and 9.1.6.6).

(3) Crane inspections and testing will comply with the requirements of ANSI B30.2 and NUREG-0612, Subsection 5.1.1(6). The COL applicant will provide the heavy load handling system and equipment inspection and test plans (Subsections 9.1.5.6, 9.1.5.8 and 9.1.6.6).

(4) The equipment handling components, including the reactor building crane and the refueling machine crane, used over the fuel pool are designed to meet the single failure proof criteria of NUREG-0554 (Reference 19B.2.12-8). Redundant safety interlocks and limit switches are provided to prevent transporting heavy loads other than spent fuel by the refueling machine crane, over any spent fuel that is stored in the spent-fuel storage pool (Subsections 9.1.5.2.1 and 9.1.5.5).

(5) The reactor vessel head lifting strongback and the dryer/separator lifting strongback are designed in accordance with the acceptable factors of safety. This is in accordance with ANSI-N14.6 (Reference 19B.2.12-5) and in accordance with NUREG-0612 (Subsection 9.1.4.2.5).

(6) The heavy load handling system is designed in accordance with relevant requirements of GDC 2, 4, and 61 and the guidance of References 19B.2.12-2, and 19B.2.12-5 through 19B.2.12-7. The ABWR design is for a single unit; therefore, GDC 5 is not applicable (Subsection 9.1.5.1 and Section 3.1).

The acceptance criteria for this safety issue are met and, therefore, the issue is resolved for the ABWR design.

References

19B.2.12-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.12-2 NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants", U.S. NRC.

19B.2.12-3 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.12-4 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants", Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.12-5 ANSI-N14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 KG) or More for Nuclear Materials", American National Standards Institute.

19B.2.12-6 ANSI/ANS-57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants", American Nuclear Society.

19B.2.12-7 ANSI/ANS-57.1, "Design Requirements for Light Water Reactors Fuel Handling Systems", American Nuclear Society.

19B.2.12-8 NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants", U.S. NRC.

19B.2.13 A-39: Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits

Issue

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching due to high local pool temperatures. This issue addresses GE MARK I, II, and III containments.

Acceptance Criteria

The acceptance criteria set forth for quencher discharge loads are applicable only to the cross-quencher configuration described in Attachment A to Appendix 3B of General Electric Standard Safety Analysis Report II (GESSAR II), Revision 1. Deviation from this configuration shall be reviewed on a plant-unique basis. And acceptability of suppression pool temperature limit(s) shall be based on conformance with the resolution of the issue specified in Section 5 of NUREG-0783 (Reference 19B.2.13-1).

Resolution

Safety/Relief Valves (SRVs) are utilized in a BWR pressure suppression system to provide pressure relief during certain reactor transients. SRV steam flow is routed through discharge lines into the pressure suppression pool, where it is condensed. Each discharge pipe is fitted at the end with a device called a quencher to promote heat transfer during SRV actuation between the high temperature compressed air and steam mixture and the cooler water in the suppression pool. This enhances heat transfer while providing a low amplitude oscillating pressure in the pool and eliminates concern over operation at a high suppression pool temperature. For ABWR plants, the discharge device is an X-quencher such as has been used in prior plants (Section 3B.2.1).

Following the actuation of a SRV, water contained initially in the discharge line is rapidly discharged through the X-Quencher discharge device attached at the end of the SRV discharge line. A highly localized water jet is formed around the X-Quencher arms. The hydrodynamic load induced outside a sphere circumscribed around the quencher arms by the quencher water jet is not significant. This is the first phase of loading on the suppression pool boundary due to the SRV blowdown. There are no submerged structures located within the sphere mentioned above in the ABWR arrangement. The induced load for submerged structures located outside the circumscribed sphere by the quencher arm is negligible and is ignored (Section 3B.5.4).

After the water discharge, the air initially contained in the discharge line is forced into the suppression pool under high pressure. The air bubbles formed interact with the surrounding water and produce oscillating pressure and velocity fields in the suppression pool. This pool disturbance (air-clearing) gives rise to hydrodynamic loads which are the second phase of SRV blowdown loading on submerged structures in the pool and on the pool boundary (Section 3B.5.4).

The final stage of SRV blowdown is the steady steam flow phase. Submerged structure and pool boundary loading is from condensing steam jet oscillations at the quencher (Reference 19B.2.13-1).

This issue was resolved with the issuance of SRP Section 6.2.1.1.C (Reference 19B.2.13-2). NUREG-0763 (Reference 19B.2.13-3), NUREG-0783 (Reference 19B.2.13-1), and NUREG-0802 (Reference 19B.2.13-4) were also issued for Mark I, II, and III containments, respectively, and the load definition for dynamic loads on submerged structures was developed on the basis of cumulative information that was described in Issues A-7, A-8, and B-10 (Subsections 19B.2.3, 19B.2.4, and 19B.2.19, respectively). The load definition methodology for defining the SRV air bubble loads on submerged structures will be consistent with that used for prior plants.

Therefore, this issue is resolved for the ABWR.

References

19B.2.13-1 NUREG-0783, “Suppression Pool Temperature Limits for BWR Containments,” U.S. NRC, November 1981.

19B.2.13-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.13-3 NUREG-0763, “Guidelines for Confirmatory In-plant Tests of Safety Relief Valve Discharges for BWR Plants”, U.S. NRC, May 1981.

19B.2.13-4 NUREG-0802, “Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments”, U.S. NRC, October 1982.

19B.2.14 A-40: Seismic Design Criteria Short-Term Program

Issue

Issue A-40 in NUREG-0933 (Reference 19B.2.14-1) addresses short-term improvements in seismic design criteria.

The seismic design sequence for recently designed plants included many conservative factors. Although it is believed that the overall sequence was adequately conservative, certain aspects may not have been conservative for all plant sites. The objective of A-40 was to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternative approaches where desirable, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan (Reference 19B.2.14-2), where justified.

Studies were conducted, and the results were documented in NUREG/CR-1161 (Reference 19B.2.14-3) with specific recommendations for changes in seismic design requirements. In addition, an NRC/Industry workshop was held to discuss the complex and controversial subject of soil-structure interaction (SSI) analysis. The adequacy of seismic design of large, above ground, vertical, safety-related tanks was also of concern to the NRC.

Standard Review Plan (SRP) sections were then revised (Revision 2) with the following principal areas of change: Section 2.5.2, updated to reflect the current NRC staff review practice; Section 3.7.1, design time history criteria; Section 3.7.2, development of floor response criteria, damping values, SSI uncertainties, and combination of modal responses; and Section 3.7.3, seismic analysis of above ground tanks, and Category 1 buried piping.

The NRC concluded in NUREG-1233 (Reference 19B.2.14-4) that these revisions would reflect the current state-of-the-art in seismic design in the licensing process.

Implementation of the SRP revisions is expected to contribute to a more uniform and consistent licensing process and is not expected to have significant impact on recently designed plants.

Acceptance Criteria

The acceptance criterion for the resolution of A-40 is that future nuclear power plants shall conform to the seismic design acceptance criteria and guidance of Revision 2 to SRP Sections 2.5.2, Vibratory Ground Motion; 3.7.1, Seismic Design Parameters; 3.7.2, Seismic System Analysis; and 3.7.3, Seismic Subsystem Analysis.

Specifically, these SRP Sections respectively cover review of the site characteristics and earthquake potential, the parameters to be used in seismic design, methods to be used in seismic analysis of the overall plant, and methods to be used in seismic analysis of individual systems or components.

Resolution

The design ground motions, site envelope soil parameters, and system and subsystem analyses criteria and methods described in Sections 2.3.2.22, 3.7.1, 3.7.2 and 3.7.3 meet the intent of Revision 2 of the corresponding SRP sections, except that the OBE is not a design requirement for the ABWR. Elimination of the OBE from the design in advanced reactors is consistent with Policy Issue SECY-93-087 (Reference 19B.2.14-5).

This issue is, therefore, resolved for the ABWR Standard Plant design.

References

19B.2.14-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.14-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.14-3 NUREG/CR-1161, “Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria”, U.S. NRC, May 1980.

19B.2.14-4 NUREG-1233, Regulatory Analysis for USI A-40, “Seismic Design Criteria”, U.S. NRC, September 1989.

19B.2.14-5 Policy Issue SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” April 1993.

19B.2.15 A-42: Pipe Cracks in Boiling Water Reactors

Issue

Issue A-42 in NUREG-0933 (Reference 19B.2.15-1), addresses the past occurrences of intergranular stress corrosion cracking (IGSCC) in BWR austenitic steel components. Safe ends, short transition pieces between vessel nozzles and the piping, that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication, were in the late 1960's found to be susceptible to IGSCC.

Acceptance Criteria

The acceptance criteria for the resolution of A-42 are that IGSCC resistant materials and fabrication techniques to minimize sensitization shall be used. In addition, the ABWR water shall be maintained at the lowest practically achievable impurity levels. Furthermore, the material and fabrication techniques shall comply with the guidelines of NUREG-0313 (Reference 19B.2.15-2).

Resolution

For the ABWR, IGSCC resistance is achieved through the use of Type 316 stainless steel and compliance with the guidelines of NUREG-0313. All materials are supplied in the solution heat treated condition. During fabrication, any heating operations (except welding) between 700 and 1255 K (427 and 982°C) are avoided, unless followed by solution heat treatment. The ABWR water is maintained at the lowest practically achievable impurity levels to minimize its corrosion potential.

In summary, only stainless steel type 316 material is used and all austenitic steel components are fabricated in accordance with NUREG-0313.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.15-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.15-2 NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," US. NRC, July 1977, (Revision 1) July 1980, (Revision 2) January 1988.

19B.2.16 A-44: Station Blackout

Issue

The total loss of AC power (that is, the loss of AC power from both the off-site and on-site sources) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core is dependent on the availability of systems that do not require AC power and on the ability to restore offsite or onsite AC power before other means of cooling the core are lost. The concern is that a prolonged station blackout might result in a core damage accident (Reference 19B.2.16-1).

Acceptance Criteria

The acceptance criteria for the resolution of this issue for evolutionary ALWRS is compliance with:

- SECY-90-016–Evolutionary LWR Certification Requirement (Reference 19B.2.16-1)
- NRC Commissioner Policy Statement Certification Requirement (Reference 19B.2.16-2)
- 10 CFR 50.63, “Loss of all Alternating Current Power” (Reference 19B.2.16-3)
- Regulatory Guide 1.155, “Station Blackout” (Reference 19B.2.16-4)
- NUMARC-87-00 Guidelines and Technical Basis for Resolution of SBO (Reference 19B.2.16-5)
- EPRI-URD–Utility Requirements for Evolutionary LWRS (Reference 19B.2.16-6)

Resolution

The ABWR design satisfies the acceptance criteria by demonstrating (in Appendix 19E.2.1.2.2) that the ABWR can withstand a station blackout without core damage or loss of containment integrity through the use of the combustion turbine generator as an alternate AC source. Station blackout is also addressed in Appendix 1C to show compliance with applicable regulations.

Therefore, this issue is resolved for the ABWR.

References

19B.2.16-1 SECY-90-016, “Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements,” January 12, 1990.

19B.2.16-2 Letter J. Taylor to S. Chilk, “Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements”, June 26, 1990.

19B.2.16-3 10 CFR 50.63, “Loss All Alternating Current Power (Station Blackout–SBO)”, July 21, 1988.

19B.2.16-4 Regulatory Guide 1.155, “Station Blackout.”

19B.2.16-5 NUMARC-87-00, “Guidelines and Technical Basis for NUMARC Initiation Addressing Station Blackout at LWR’s”, Plus Supplemental Q/A, January 4, 1990.

19B.2.16-6 “Advanced Light Water Reactor Utility Requirements Document, Volume II; EPRI”, July 1990.

19B.2.17 A-47; Safety Implications of Control Systems

Issue

This issue, A-47, concerns the potential for accidents or transients (e.g., overpressure, overfilling, reactivity events) being made more severe as a result of control system failures including control and instrumentation power support faults. These failures or malfunctions may occur independently or as a result of an accident or transient and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed. The NRC evaluated the effects of control system failures on three operating plants and identified the concern of steam generator overfill by feedwater. Subsequently, GL89-19 (Reference 19B.2.17-3) was issued to require all operating plants and plants under construction to provide automatic feedwater overfill protection.

Acceptance Criteria

The acceptance criteria for resolution is that the plant shall provide automatic reactor vessel overfill protection, and that plant procedures and technical specifications shall include provisions to verify periodically the operability of the overfill protection to assure that automatic overfill protection is available to mitigate main feedwater overfeed events during reactor power operation. Also, the system design and setpoints shall be selected with the objective of minimizing inadvertent trips of the main feedwater system during plant startup, normal operation, and protection system surveillance.

Resolution

The reactor vessel overfill protection is described in Subsection 7.7.1.4(9) and Figure 7.7-8. Plant procedures will be developed by the COL applicant. As a matter of good design practice for maximum availability, the feedwater system design and setpoints will be selected to minimize inadvertent trips for all modes of operation. This system, with fault tolerant digital controllers and self-test and on-line diagnostics, is described in Subsection 7.7.1.4.

Feedwater control (FDWC) system uses the non-Class 1E level transmitter signals in providing Level-8 trip instrumentation and part of the diverse ATWS trip logic. For this reason, reliance on the Class 1E level transmitters by FDWC would partially defeat the diversity of the ATWS design. In addition, three channels of reactor vessel water level high instrumentation are provided as input to a two-out-of-three logic to minimize inadvertent trips of the main feedwater system.

The LCO of the feedwater trip instrumentation is identified in Technical Specifications 3.3.4.2. Operation restrictions are provided in technical specifications to assure the overfill protection availability. Incorporated in these restrictions is a periodic evaluation of the feedwater trip instrumentation condition which considers such availability items as correct channel sensor operation and proper channel function, calibration and logic system operation. Additionally, the COL applicant will be required to incorporate these requirements into plant operating procedures.

Therefore, this issue is resolved for the ABWR.

References

19B.2.17-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.17-2 NUREG-1217, “Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants”, June 1989.

19B.2.17-3 Generic Letter 89-19, “Request for Action Related to Resolution of USI A-47, Pursuant to 10CFR50.54(f)”, U.S. NRC, September 20, 1989.

19B.2.18 A-48: Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Issue

In the unlikely event of a degraded core accident (or following a LOCA) in a light water reactor plant, it is postulated that the result is the release of large quantities of combustible gases, principally hydrogen, that may accumulate inside the primary reactor containment. These gases would be the result of:

(1) A metal-water reaction involving the fuel element cladding. Hydrogen in significant quantity can be formed as a result of the reaction of zirconium fuel cladding at high temperature with steam.

(2) The radiolytic decomposition of the water in the reactor core and the containment sump.

(3) The corrosion of certain construction materials by the spray solution.

(4) Any synergistic chemical, thermal, and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

Acceptance Criteria

Because of the potential for significant hydrogen generation as a result of an accident, 10 CFR 50.44, “Standards for Combustible Gas Control System in Light-Water-Cooled

Power Reactors" (Reference 19B.2.18-1), and General Design Criterion 41, "Containment Atmosphere Cleanup", in Appendix A to 10 CFR Part 50 (Reference 19B.2.18-2), requires that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

Paragraph (f)(2)(ix) of 10 CFR 50.34, "Contents of Applications; Technical Information" (Reference 19B.2.18-4), requires that provision be made for a hydrogen control system that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal-water reaction.

An inerted containment and the provision for permanently installed hydrogen recombiners are acceptable as hydrogen control measures.

Resolution

The issue of a large amount of hydrogen being generated and burned within containment was resolved as stated in the NRC document SECY 89-122 dated April 19, 1989 (Reference 19B.2.18-3). This issue covers hydrogen control measures for recoverable degraded core accidents for all BWRs. Extensive research in this area has led to significant revision of the Commission's hydrogen control regulations, given in 10 CFR 50.44, published December 2, 1981.

The ABWR containment is inerted and per 10 CFR 50.34 (f)(2)(ix) can withstand the pressure and energy addition from a 100% fuel-clad metal-water reaction. However, in the ABWR, there are no design-basis events that result in core uncovery or core heatup sufficient to cause significant metal-water reaction. Section 6.2.5.3 states that this is equivalent to the reaction of the active clad to a depth of 5.842E-3 mm (0.00023 inches) or 0.72% of the active clad.

Therefore, this issue is resolved for the ABWR.

References

19B.2.18-1 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors", Office of the Federal Register, National Archives Records Administration.

19B.2.18-2 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants."

19B.2.18-3 SECY-89-122, "Resolution of Unresolved Safety Issue A-48, Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment", April 1989.

19B.2.18-4 10 CFR 50.34, "Contents of Applications; Technical Information", Office of the Federal Register, National Archives Records Administration.

19B.2.19 B-10: Behavior of BWR Mark III Containments

Issue

Evaluation and approval is required of various aspects of the MARK III containment design which differs from the previously reviewed MARK I and MARK II designs. The task involves the completion of the staff evaluation of the MARK III containment and documentation of the method used to validate the analytical models and assumptions needed to predict the containment pressures in the event of a LOCA (Reference 19B.2.19-1).

Following a postulated LOCA, escaping steam forces the steam-air mixture out of the drywell into the wetwell. This action results in pool swell and loads from vent clearing, jets, chugging, impact of water, impact from froth impingement, pool fallback, condensation, and containment pressure.

The concern is that these loadings may damage structures and components located within the wetwell. Although many of these structures (e.g., walkways) are by themselves not related to safety, the various ECCS systems take suction from the wetwell and, therefore, damage in the wetwell may affect the performance of the ECCS (Reference 19B.2.19-2).

Acceptance Criteria

On the basis of certain large-scale tests conducted between 1973 and 1979, the General Electric Company developed LOCA-related hydrodynamic load definitions for use in the design of the standard Mark III containment. The NRC staff and its consultants have reviewed these load definitions and their bases and conclude that, with a few specified changes (Reference 19B.2.19-2), the proposed load definitions provide conservative loading conditions.

The staff approved acceptance criteria for LOCA-related hydrodynamic loads are provided in NUREG-0978, Appendix C (Reference 19B.2.19-2). The following describes how the acceptance criteria are applied. The staff will review each applicant's use of the NRC acceptance criteria for applicability to their plant design. Mark III applicants for a construction permit (CP) need only furnish a commitment to use the staff's acceptance criteria in the design of their containment. Mark III applicants for an operating license (OL) will be required to show how the NRC acceptance criteria were applied and to justify any deviations taken. For both CP and OL applicants, the information required shall be submitted in a timely manner to allow for the evaluation to be included in the plant's Safety Evaluation Report, or supplements thereto (Reference 19B.2.19-2).

The ABWR horizontal vent confirmatory test program was performed to obtain data which could be used to determine condensation oscillation and chugging loads for design evaluation of containment structures. The test matrix included tests at conditions which produce bounding loads and additional tests to examine the sensitivity of the loads to system parameters. The test specifically documents work performed, including general evaluation of the test data and the specification of procedures which can be used to define containment loads.

Resolution

The ABWR design utilizes a horizontal vent system, which is similar to the Mark III design, but includes some ABWR design features. These unique features include pressurization of the wetwell airspace, the presence of a lower drywell, the smaller number of horizontal vents (76.2 cm ABWR vs. 304.8 cm Mark III), extension of horizontal vents into the pool, vent submergence, and suppression pool width, as described in Subsection 3B.1.2.

The ABWR horizontal vent test (HVT) program produced test data to confirm and define condensation oscillation (CO) and chugging (CH) loads for design application. The test demonstrated that a blowdown test facility can be constructed to be very rigid and thereby eliminate fluid-structure interaction effects. It was also shown that a scaled test facility can be used to obtain condensation data for full-scale design application. Most important, an extensive data base which can be used for confirmation of ABWR CO and CH loads was obtained

A spectrum of postulated loss-of-accidents (LOCAs) is considered in assessing the design adequacy of the ABWR containment system. Each of the accident conditions is described in Subsection 3B.2.2. The load definition methodology for defining the LOCA induced loads on submerged structures is consistent with the methodology used for prior plants, as described in Section 3B.5. The ABWR is designed to meet the NRC acceptance for Mark III LOCA-related pool dynamic loads.

Therefore, this issue is resolved for the ABWR plants.

References

19B.2.19-1 NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D)”, US. NRC, June 1978.

19B.2.19-2 NUREG-0978, “Mark III LOCA-Related Hydrodynamic Load Definition”, U.S. NRC, February 1984.

19B.2.19-3 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.19-4 NEDC-31393, “Containment Horizontal Vent Confirmatory Test Part 1 – Final Report”, March 1987.

19B.2.20 B-17: Criteria for Safety-Related Operator Actions

Issue

This issue, B-17, involves developing criteria for safety-related operator action (SROA) during the response to or recovery from transients and accidents. The criteria would include a determination of actions that shall be automated in lieu of operator action and the development of a time criterion for SROA. Specifically, to be determined for PWRs, is whether or not to require an automatic switchover from the injection mode to the recirculation mode following a LOCA. This issue is not resolved and it has a medium priority. (Reference 19B.2.20-1)

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-17 is that the plant transient response time (i.e., time required for safety systems or operator to act) shall be increased over current plants to improve operability, and that the plant design shall permit increased operator response time, including a determination on the need for automatic actuation. Required time before the operator must act shall be not less than 20 minutes with a target of 30 minutes, assuming a single failure. Best estimate methodology shall be used for analysis to show safety limits are not exceeded. Operational inputs should be obtained from experienced operators.

Resolution

The ABWR design satisfies the Acceptance Criteria concerning automation of safety-related operator actions and operator response times. The ABWR resolution is the same as the ALWR resolution. For example, the ABWR design requires no operator action earlier than thirty minutes for any design basis accident as described in responses to questions 420.81, 420.82, 420.83, 430.26 in Subsections 20.3.8, 20.3.2 on operator performance under the range of Loss-of-Coolant accidents. The ABWR design by incorporating the RHR heat exchanger in the ECCS injection loop has eliminated the need for operator actions for several accidents/transients (Subsection 5.4.7.1.1.1). In fact, even in the long term, operator action is only required for one situation—initiation of containment cooling (Subsection 5.4.7.1.1.6). This is a relatively simple action and some delay in this action should have no adverse consequences, thus eliminating the need to automate this function. In addition, advance Cathode Ray Tubes (CRTs) in the control room shall be utilized for monitoring and alarm functions for safety-related and non-safety-related systems (References 19B.2.19-2, 19B.2.19-3, and 19B.2.19-4). To achieve this goal, information displays, controls and other interface devices in the control room and other plant areas are designed and implemented with good human factors engineering and in compliance with pertinent regulations regarding separation and independence (Section 18.2).

Therefore, this issue is resolved for the ABWR.

References

19B.2.20-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.20-2 EPRI NP-4361, “Power Plant Alarm Systems: A Survey and Recommended Approach For Evaluating Improvements”, December 1985.

19B.2.20-3 EPRI NP-5693P, “Evaluation of Alternative Power Plant Alarm Presentations.”

19B.2.20-4 EPRI NP-3448, “A Procedure For Reviewing and Improving Power Plant Alarm Systems”, April 1984.

19B.2.20-5 ANSI/ANS 58.8, “Time Response Design Criteria for Nuclear Safety Related Operator Actions”, American Nuclear Society.

19B.2.21 B-36: Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems

Issue

Issue B-36 in NUREG-0471 (Reference 19B.2.21-1) addresses technical advances in design, testing and maintenance criteria for atmosphere cleanup system air filtration and adsorption unit for engineered safety features systems and for normal ventilation systems.

Any technological advances leading to better methods and/or standards for these design, testing and maintenance criteria for these systems in light-water-cooled nuclear power plants need to be documented for NRC staff guidance and technical positions.

However no new design requirements have been established by the Nuclear Regulatory Commission (NRC) other than those of Revision 1 of Regulatory Guide (RG) 1.140 (Reference 19B.2.21-3) and Revision 2 of Regulatory Guide 1.52 (Reference 19B.2.21-2).

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-36 are contained in RG 1.140, Revision 1, and RG 1.52, Revision 2. Both deal with design, testing, and maintenance criteria for air filtration and adsorption units of light-water-cooled nuclear power plants. RG 1.140 specifically applies to the non-safety-related normal ventilation exhaust system. RG 1.52 covers the criteria for post-accident engineered safety features.

Resolution

The filter systems required to perform safety-related functions following a design basis accident are the standby gas treatment system (SGTS) and the control room habitability system as described in Sections 6.4 (Habitability Systems) and 9.4.1.1 (Control Room Habitability Area HVAC), and Subsection 6.5.1 (Engineered Safety Features Filter Systems). The SGTS consists of two parallel and redundant filter trains. Each filter train is designed to have a HEPA filter installed at both inlet and outlet sides of the charcoal adsorber. The CRHA HVAC system is provided with redundant divisions. Each division consists of an emergency filtration unit. A HEPA filter is also provided before and after the charcoal adsorber of each emergency filtration unit. The HEPA filters of these systems will be tested periodically with DOP using the installed instrumentation in conformance with the guidance of SRP Table 6.5.1-1 and as described in Appendix 6B, for SGTS, and Appendix 9D, for CRHA HVAC systems and test connections as required by RG 1.52. Additionally, both of these systems address RG 1.52 as described in Subsection 6.5.1.3.5, Appendix 6A (compliance with RG 1.52), Subsection 9.4.1.1.7 (RG 1.52 Compliance Status), and Appendix 9C.

Air filtration and adsorption units are not required for normal ventilation on ABWR, since there are no requirements for safety-related adsorption units in normal operations, except for the incinerator off-gas exhaust which is directed to a separate monitor vent (Subsection 9.4.6.5.3). Therefore, RG 1.140 is not applicable.

Thus, Issue B-36 is resolved for ABWR.

References

19B.2.21-1 NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D)”, U.S. NRC, June 1978.

19B.2.21-2 Regulatory Guide 1.52, “Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants”, U.S. NRC, March 1978.

19B.2.21-3 Regulatory Guide 1.140, “Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants”, U.S. NRC.

19B.2.22 B-55: Improved Reliability of Target Rock Safety/Relief Valves

Issue

Many of the valves in BWR main steam pressure relief systems are Target Rock safety/relief valves, and a significant number of failures of these valves have occurred. Failures include valves:

(1) Failing to open properly on demand.

(2) Opening spuriously and then failing to reseat properly.

(3) Opening properly and then failing to reseat properly.

The failure of a pressure relief system valve to open on demand results in a decrease in the total available pressure-relieving capacity of the system. Spurious openings of pressure relief system valves, or failures of valves to properly reseat after opening, can result in inadvertent reactor coolant system blowdown with unnecessary thermal transients on the reactor vessel and the vessel internals, unnecessary hydrodynamic loading of the containment systems' pressure-suppression chamber and its internal components, and potential increases in the release of radioactivity to the environs. In addition, if the valve also serves as part of the ADS, a degradation of the capability of the ADS to perform its emergency core cooling function could result.

Acceptance Criteria

In the late 1970s, the NRC staff concluded that the inadvertent blowdown events that had occurred as a result of malfunctions of pressure relief system valves had neither significantly affected the structural integrity or capability of the reactor vessel or its internals or the pressure-suppression containment system, nor resulted in any significant radiation releases to the environment. Even if such events were to occur more frequently, there would not likely be any significant effects. Issue B-55 in NUREG-0933 (Reference 19B.2.22-1) requires that the performance of these valves be under continual surveillance and the consequences of their failures be subject to review.

Resolution

Main steam safety/relief valves for ABWR service will be similar to the dual-function direct-acting SRV types currently in service in GE BWR/5 and BWR/6 plants. These SRV types have demonstrated improved in-service performance and reliability as compared to pilot-operated safety/relief valves used on earlier BWR models.

The B-55 issue is not applicable to the ABWR. The ABWR uses a direct acting safety/relief valve design described in Subsection 5.2.2.4.1. This design does not have a pilot stage such as that present in the Target Rock pilot operated safety/relief valve. Therefore, the typical mechanisms which cause the pilot valve to open spuriously and to fail to open properly are not applicable to the ABWR design. It is these mechanisms which have caused the most serious concerns with the Target Rock safety/relief valve performance. By adopting a direct acting safety/relief valve design, these most serious concerns are eliminated in the ABWR.

The B-55 issue is only applicable to those operating BWRs with Target Rock pilot-operated safety/relief valves. GE has identified the principal cause of the most significant concern with these Target Rock pilot operated safety/relief valve and has developed a modification to greatly improve the performance of this valve model.

References

19B.2.22-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.22-2 Memo from Robert Kirkwood to Robert L. Baer, Engineering Issues Branch, Division of Safety Issue Resolution, Office of Nuclear Regulatory Research, dated on September 2, 1992.

19B.2.23 B-56: Diesel Reliability

Issue

Issue B-56 in NUREG-0933 (Reference 19B.2.23-1), addresses emergency diesel generator reliability. The reliability goal identified in NSAC-108, (Reference 19B.2.23-2) for emergency diesel generator startup, is between 0.95 and 0.975 per demand.

Typical onsite electrical distribution systems for plants use diesel generators as an emergency source of power. These emergency power sources supply safety-related equipment, which is used to prevent or mitigate accidents, in the event of a loss of offsite power.

Because of the safety significance of the emergency diesel generators, limiting conditions for operation (LCOs) were developed and placed in the plant technical specifications. These LCOs require periodic testing. Licensee Event Reports (LERs) sent to the NRC document problems encountered during periodic testing of the emergency diesel generators (to demonstrate operability). As discussed in NUREG-0933, a review of the LERs conducted by the NRC revealed that a diesel generator’s starting reliability is, on the average, about 0.94 per demand. Thus, the NRC determined that there was a need to upgrade the reliability of emergency diesel generators. A new reliability of between 0.95 and 0.975 per demand for emergency diesel generator design, operation and periodic testing, was established in Regulatory Guide 1.9, Revision 3 (Reference 19B.2.23-3).

The specific emergency diesel generator starting reliability identified in Regulatory Guide 1.155 (Reference 19B.2.23-4) is the same as in Regulatory Guide 1.9, Revision 3 (i.e., it ranges from 0.95 to 0.975 per demand). The resolution of a related Issue A-44, Station Blackout, addresses the plant response to station blackout conditions.

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-56, is that emergency diesel generator design, operation, and periodic testing shall ensure, as a minimum, a starting reliability of 0.95 per demand, as identified in Regulatory Guides 1.9, Revision 3, 1.155 and 1.160.

Resolution

The ABWR Standard Plant design includes an onsite electrical distribution system which employs three redundant and independent Class 1E load group divisions. The Class 1E safety loads are capable of being supplied power, in decreasing priority, from the unit main turbine generator, either of two offsite power sources, the emergency diesel generators (DGs), and the combustion turbine generator (CTG) (Figure 8.3-1).

Each of the three Class 1E divisions can be supplied with emergency standby power from an independent DG. The DG is designed and sized with sufficient capacity to operate all the needed Class 1E loads powered from its respective Class 1E divisional bus. Furthermore, each division can be manually supplied from the non-Class 1E CTG, which is diverse from the DGs. The reliability of the CTG is comparable to that of the DG (Section 9.5.11).

Each DG is specified to start reliably and, with present technology, industry experience has shown that a starting reliability of 0.986 per demand may be achieved as identified in the EPRI ALWR Utility Requirements Document (Reference 19B.2.23-5). The time required for the DG to attain rated voltage and frequency, and to begin accepting load, has been eased from 13 to 20 seconds after receipt of a start signal. This reduces their starting stress and contributes to improved reliability over the life of the units. The extended time is still within the limiting case for opening of the RHR valves [Subsection 8.3.1.1.8.2(4)].

A variety of tests are performed to assure DG reliability and operability. In addition to factory tests, a number of pre-operational and onsite acceptance tests and periodic tests are conducted on each DG system. These tests are identified in Subsection 8.3.1.1.8.2, and in the technical specifications. Also, conditions for operation are imposed to ensure continual reliability.

In summary, the ABWR Standard Plant design utilizes three independent diesel generators as emergency power sources, which are incorporated in the onsite electrical distribution system, and which have a diverse backup (i.e., the CTG).

The onsite electrical distribution system meets the intent of the guidance given in Regulatory Guides 1.9, Revision 3, 1.155 and 1.160.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.23-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.23-2 NSAC-108, "Reliability of Emergency Diesel Generators at U.S. Nuclear Plants", Electric Power Research Institute, September 1986.

19B.2.23-3 Regulatory Guide 1.9, Revision 3, “Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used as Onsite Electrical Power Systems at Nuclear Power Plants”, U.S. NRC.

19B.2.23-4 Regulatory Guide 1.155, “Station Blackout”, U.S. NRC.

19B.2.23-5 “Advanced Light Water Reactor Utility Requirements Document”, Volume II, Chapter 11; EPRI, April 1989.

19B.2.23-6 Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”, U.S. NRC.

19B.2.24 B-61: Allowable ECCS Equipment Outage Periods

Issue

Issue B-61 in NUREG-0933 (Reference 19B.2.24-1) addresses the potential for an overall reduction in the core damage frequency of a plant by reducing the frequency of surveillance testing and reducing permissible outage times for safety-related ECCS equipment.

Historically, ECCS equipment outage times and surveillance testing were not established by analysis. Instead, these test requirements were developed using engineering judgment and equipment operating, performance testing, and maintenance histories. After development, these test requirements were incorporated into the plant Technical Specifications as Limiting Conditions for Operation (LCOs).

Studies performed for the NRC on operating reactors indicate that from 30 to 80 percent of the ECCS system unavailability was due to testing, maintenance and allowed outage periods. The NRC is evaluating whether overall ECCS unavailability and resulting core damage frequency can be reduced by extending the intervals between testing and maintenance of equipment. Intervals can be extended within a range in which equipment unavailability due to testing and maintenance is reduced more than the predicted equipment unavailability due to failure is increased. Probabilistic risk assessment (PRA) methods would be used to determine the optimum intervals between ECCS equipment tests. Surveillance intervals optimized in this manner would then be used in LCOs (See Subsection 6.3).

As a part of this program a computer code (References 19B.2.24-2 and 19B.2.24-3) has been developed for the time dependent unavailability analysis. This code, using generic data from the Interim and National Reliability Evaluation Programs (IREP and NREP, respectively), will be used to verify the capability of the code to determine optimum surveillance intervals and resulting overall risk reduction. The costs and benefits can then be assessed.

Because the NRC evaluation of this issue has not yet been completed the initial LCOs for a future plant design may continue to be based on current industry practice without prejudicing later optimization when the methods and requirements have been confirmed. The overall plant PRA should take the initial LCOs into account, to establish a base against which to measure the effects of later optimization.

Acceptance Criteria

The acceptance criterion for the resolution of Issue B-61 for future plant designs is that the Technical Specification LCOs surveillance periods and allowable completion times of ECCS equipment shall be developed in accordance with current industry practice.

The LCOs surveillance periods and completion times shall be accounted for in the overall plant PRA required by 10 CFR 52.47 (Reference 19B.2.24-4). Any subsequent proposed changes to the LCOs' provisions for ECCS surveillance shall be demonstrated to be within the results of an existing PRA (see Section 6.3).

Resolution

The ABWR design incorporates many design enhancements to improve the operation and safety of the plant, and the most significant advances are in the area of engineered safety features. The ECCS conforms to all licensing requirements and good design practices of isolation, separation and common mode failure considerations.

In order to meet the above requirements, the ECCS network has built-in redundancy so that adequate core cooling can be provided, even in the event of specific failures. Each system of ECCS, including flow rate and sensing networks, is capable of being tested during plant operation, including logic required to automatically initiate component action. Provisions for testing the ECCS network components (electrical, mechanical, hydraulic and pneumatic, as applicable) are installed in such a manner that they are an integral part of the design.

The PRA uses a system fault tree approach to quantify system accident sequences which result in severe core damage. Data related to the engineered safety features used in the quantification includes:

(1) Component failure rates

(2) Component repair times and maintenance frequencies

(3) Component inspection and test times and frequencies

(4) Allowable equipment outage times

The data is used in accordance with the guidance in NUREG/CR-2815 (Reference 19B.2.24-5), and basic failure rate data is obtained from the ERPI ALWR Requirements Document (Reference 19B.2.24-6) supplemented with other nuclear sources. Maintenance and repair times are calculated as outlined in NUREG/CR-2815. The inspection and test times and frequencies are as specified in ABWR STS Section 3.5. Several LCO completion times for the ABWR were determined based on relative comparisons of PRA estimated frequencies for conditional core damage.

The PRA demonstrates that the ABWR design meets the industry goal of 1.0 x E-5 core damage frequency per reactor year (Reference 19B.2.24-6) and indicates that the initial LCOs are consistent with this goal. The owner-operator may refine the LCOs to achieve further risk reduction or increased operational flexibility provided that the resulting overall risk is shown to be within the results of the PRA.

This issue is, therefore, resolved for the ABWR.

References

19B.2.24-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.24-2 NUREG-0193, "FRANTIC – A Computer Code for Time Dependent Unavailability Analysis", U.S. NRC, October 1977.

19B.2.24-3 NUREG/CR-1924, "FRANTIC II – A Computer Code for Time Dependent Unavailability Analysis", U.S. NRC, April 1981.

19B.2.24-4 10 CFR 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Reactors", Office of the Federal Register, National Archives and Records Administration.

19B.2.24-5 NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide", Brookhaven National Laboratory, January 1984.

19B.2.24-6 "Advanced Light Water Reactor Utility Requirements Document – Volume II, Chapter 1: Overall Requirements, Appendix A: PRA Key Assumptions and Groundrules", Electric Power Research Institute, Draft, April 1987.

19B.2.25 B-63: Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary

Issue

Issue B-63 in NUREG-0933 (Reference 19B.2.25-1) addresses the need to ensure the integrity (i.e., leak-tightness) of boundary valves installed between high pressure (HP) (i.e., the Reactor Coolant System pressure boundary) and low pressure (LP) safety-related systems, during plant operation by performing periodic in-service testing.

The ASME Code, Section III, (Reference 19B.2.25-3) controls the design, fabrication, and initial testing of boundary and relief valves. During operation, the ASME Code, Section XI, specifies boundary and relief valve testing requirements to assure continued valve integrity.

Because of the importance of the HP to LP interface for safety-related systems, the NRC reviewed and updated SRP Section 3.9.6 by issuing Revision 2 (Reference 19B.2.25-2). This SRP references and endorses the ASME Code, Section XI (for the in-service testing of the boundary valves).

(A related issue, which also discusses the integrity of the HP to LP interface between safety-related systems, is Issue 105, “Interfacing Systems LOCA.”)

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-63 is that the periodic inservice testing of the HP to LP system boundary valves shall meet the intent of SRP Section 3.9.6, Revision 2. Because SRP 3.9.6, Revision 2, endorses the requirements of the ASME Code, Section XI, periodic testing of these valves shall be performed in accordance with the code.

Specifically, these boundary valves shall comply with the requirements of the applicable IWV subarticles identified within Section XI of the ASME Code. This compliance shall include the appropriate classification and / or categorization of safety-related valves and the development of the proper test procedures for pre-operational and periodic inservice valve testing.

Resolution

All pressure containing components including all high pressure to low pressure safety-related system boundary valves used in the Advanced Boiling Water Reactor (ABWR) Standard Plant design are identified as Safety Class 1, 2, or 3, and are designed, manufactured, and tested in accordance with the guidelines of the ASME Code, Section III. (See Subsections 3.2.1, 3.2.2, and 3.2.3 for Seismic Classification, Quality Group Classifications, and Safety Classifications, respectively. Table 3.2-1 provides a cross-reference between safety and code classifications.)

Boundary valves will be periodically inservice tested in accordance with the provisions of ASME Code, Section XI, to assure operational integrity as well as to Subsection IWV requirements for each valve category. Code Class 1, 2, and 3 valves will be categorized according to Subarticle IWV-2100. Valve test requirements and valve performance testing frequency are listed in the Subsections 3.9.6, 3.9.6.2, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3.

In summary, the High Pressure and Low Pressure system boundary interface valves are designed, manufactured, pre-operational tested, and in-service tested according to the guidelines of the ASME Code and satisfy the intent of SRP Section 3.9.6, Revision 2.

Therefore, Generic Safety Issue B-63 is resolved for the ABWR design.

References

19B.2.25-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.25-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.25-3 “ASME Boiler and Pressure Vessel Code”, Sections III (Nuclear) and XI, American Society of Mechanical Engineers.

19B.2.26 B-66: Control Room Infiltration Measurements

Issue

Issue B-66 in NUREG-0933 (Reference 19B.2.26-1) addresses maintenance of the control room in a safe habitable condition under accident conditions by providing adequate protection for the plant operators against airborne radiation and toxic gases.

The rate of air infiltration into the control room is a significant factor in maintaining habitability, and the NRC measured air exchange rates in selected operating reactor plant control rooms to improve the data base for evaluating its effects.

No new design requirements were established by the NRC as a result of this and other work related to control room habitability in an accident. However, more specific review procedures were incorporated in SRP Sections 6.4.1, 9.4.1 and 15.6.5.5 (Reference 19B.2.26-2), including the habitability review provisions of TMI Action Plan Item III.D.3.4 (Reference 19B.2.26-1) regarding analyses of toxic gas concentrations and operator exposures from airborne radioactive material and direct radiation, to ensure more effective implementation of existing requirements.

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-66 is that the control room ventilation and air-conditioning systems be designed to maintain the room’s environment within acceptable limits for the operation, testing and maintenance of the unit controls and for uninterrupted safe occupancy during normal and accident conditions. Specifically, these systems shall be designed to meet the intent of the guidance given in SRP, Sections 6.4, 6.5.1, 9.4.1 and 15.6.5.5.

Resolution

The ABWR main control area envelope is heated, cooled, ventilated and pressurized with respect to the atmosphere and adjacent areas are maintained at positive pressure with respect to the atmosphere by a system mixing recirculated air with filtered outdoor air. It is designed to ensure that the operators can remain in the main control area envelope and take actions to operate the plant safely under normal conditions and maintain it in a safe condition during and following an accident. There are two air intakes on the top floor side walls of the control building, one on each end. Redundant radiation monitoring sensors in each air intake warn operators of airborne contamination, and cause the CRHA HVAC system to switch automatically to an emergency system employing HEPA and charcoal filters for cleanup.

This control room habitability area heating, ventilating and air-conditioning (CRHA HVAC) system is designed:

- With redundancy to ensure operation in an emergency with a single, active failure;
- For radiation exposure limits not exceeding the guidelines of 10 CFR 50, Appendix A, General Design Criterion 19 (Reference 19B.2.26-3), for any of the Chapter 15 DBAs;
- With provisions to detect and remove smoke and airborne radioactive material;
- To provide a controlled temperature and pressurized environment for continued operation of safety-related equipment under accident conditions;
- Protection from toxic chemical and chlorine releases.

In addition, the safety-related components of the CRHA HVAC system are operable during loss of offsite power conditions using divisional onsite power from the diesel generators and safety-related batteries. Provisions are also made for periodic tests of the emergency filtration unit fans and filters. The high-efficiency particulate air (HEPA) filters of the CRHA HVAC system will be tested periodically with dioctyl phthalate (DOP) smoke. The charcoal filters will be periodically tested with an acceptable gas for bypasses. The system ductwork and housings, which are of welded construction, will be periodically tested for unfiltered inleakage in accordance with ASME N510.

This ABWR CRHA HVAC and its design bases are described in Section 6.4, 6.5.1 and Subsection 9.4.1.

Since the control room is monitored, pressurized and filtered by the above described systems, and since the NRC requirements and the guidance for their design are met, the issue of air infiltration is resolved for the ABWR.

References

19B.2.26-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.26-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S.NRC.

19B.2.26-3 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Office of the Federal Register, National Archives Records Administration.

19B.2.27 C-1: Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Issue

Item C-1 in NUREG-0933 (Reference 19B.2.27-1), addresses concerns regarding the long-term capability of hermetically-sealed instruments and equipment which must function in post-accident environments. NUREG-0471 (Reference 19B.2.27-2) was developed because of these concerns.

Certain classes of instrumentation incorporate seals. When safety-related components within containment must function during post-LOCA accident conditions, their operability is sensitive to the ingress of steam or water. If the seals should become defective as a result of personnel errors in the maintenance of such equipment, such errors could lead to the loss of effective seals and the resultant loss of equipment operability. The establishment of a basis for confidence that sensitive equipment has a seal during the lifetime of the plant is needed.

Acceptance Criteria

The NRC has undertaken a program to reevaluate the qualification of all safety-related electrical equipment at all operating reactors. As part of this program, more definitive criteria for environmental qualification of safety-related electrical equipment have been developed by the staff. The Division of Operating Reactors’ “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors” (DOR Guidelines) were completed in November 1979. The Guidelines are intended as a screening device to catch those pieces of equipment which might have qualification problems. In addition, for reactors under licensing review, the staff has issued NUREG-0588 (Reference 19B.2.27-3). The staff intends to evaluate the qualification of all electrical safety equipment in operating plants pursuant to the Guidelines. If problems arise, the staff shall resolve them using NUREG-0588 as a guide for their judgment.

On May 27, 1980 the NRC issued Commission Memorandum and Order CLI-80-21 (Reference 19B.2.27-4) ordering that the above two documents form the requirements which licensees and applicants must meet in order to satisfy those aspects of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4, (Reference 19B.2.27-6) which relate to the environmental qualification of safety-related electrical equipment. The order established an implementation schedule which set a goal that all safety-related electrical equipment in all operating plants be qualified to the DOR Guidelines or NUREG-0588 by no later than June 30, 1982.

Resolution

Environmental qualification of safety-related equipment is described in Section 3.11.

Safety-related equipment located in a harsh environment must perform its proper safety function during normal, abnormal, test, design basis accident and post-accident environments as applicable. A list of all safety-related electrical and mechanical equipment that is located in a harsh environment area will be included in the Environmental Qualification Document (EQD) to be prepared as indicated in Subsection 3.11.6.1.

Environmental conditions for the zones where safety-related equipment is located are calculated for normal, abnormal, test, accident and post-accident conditions and are documented in Appendix 3I. Environmental conditions are tabulated by zones, contained in the referenced building arrangements.

Safety-related electrical equipment that is located in a harsh environment is qualified by test or other methods as described in IEEE 323 (Reference 19B.2.27-5) and permitted by 10 CFR 50.49(f), "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants "(Reference 19B.2.27-6).

The qualification methodology is described in detail in the NRC approved Licensing Topical Report on GE's environmental qualification program (Reference 19B.2.27-7). This report also addresses compliance with the applicable portions of the General Design Criteria of 10 CFR 50, Appendix A, and the Quality Assurance Criteria of 10 CFR 50, Appendix B. Additionally, the report describes conformance to NUREG-0588, and Regulatory Guides and IEEE Standards referenced in Section 3.11 of NUREG-0800 (Reference 19B.2.27-8). The COL applicant will address issues identified in the Generic Safety Issue C-1 and provide a list of impacted safety-related components whose operabilities are sensitive to the ingress of steam or water in a harsh environment within containment. The COL applicant will also evaluate the provision of design features for assuring long-term capabilities of these components in post-accident environments based on NUREG-0588, as required in Subsection 3.11.6.

In summary, the safety-related electrical equipment is qualified in accordance with NRC Guidance, including NUREG-0588; and, therefore, this item is resolved for the ABWR Standard Plant design.

References

19B.2.27-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.27-2 NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)", U.S. NRC, June 1978.

19B.2.27-3 NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment”, U.S. NRC, July 1981.

19B.2.27-4 NRC Memorandum and Order CLI-80-21, docketed May 27, 1980.

19B.2.27-5 IEEE Standard 323, “Qualifying Class 1E Equipment for Nuclear Power Generating Stations”, The Institute of Electrical and Electronic Engineers, Inc.

19B.2.27-6 10 CFR 50, “Domestic Licensing of Production and Utilization Facilities”, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.27-7 NEDE-24326-1-P, “General Electric Environmental Qualification Program,” Proprietary Document, January 1983.

19B.2.27-8 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S.NRC.

19B.2.28 C-10: Effective Operation of Containment Sprays in a LOCA

Issue

Issue C-10 in NUREG-0933 (Reference 19B.2.28-1) is concerned with the effectiveness of various containment spray solutions in removing airborne radioactive materials present in the containment after a loss-of-coolant accident (LOCA). Also of concern is the possible damage to equipment in the containment caused by the solutions in an inadvertent actuation of the spray system.

After the TMI accident it became evident that previous regulatory assumptions as to the forms and timing of the release of radioactive iodine in an accident causing fuel damage were probably unduly conservative. The NRC and industry therefore reviewed experimental data and industry practice with regard to controlling the pH of spray solutions, which have to be borated to prevent boron dilution of reactor coolant, so as to ensure removal of radioactive iodine and particulates from the containment atmosphere and also to minimize corrosion in the safeguards systems during subsequent long term cooling. Some additives commonly used for pH control also have the potential to damage containment equipment if the spray system is unintentionally actuated, and make the resulting cleanup effort more difficult.

It was concluded that during the initial stage of an accident the removal efficiency of containment spray containing no dissolved iodine is essentially independent of the pH (for pH values less than 6.5) of the spray solution, but that while recirculating containment spray after the initial stage of the accident it is desirable to maintain the pH of the containment sump solution high enough to prevent re-release of absorbed iodine. Also at this time, as previously discussed in Branch Technical Position (BTP)

MTEB 6-1 attached to Revision 2 of SRP Section 6.1.1 (Reference 19B.2.28-2), the pH should be high enough to preclude stress corrosion cracking of austenitic stainless steel materials used in emergency safeguards systems. The NRC therefore issued Revision 2 of SRP Section 6.5.2 (Reference 19B.2.28-2). This revision endorses the industry standard ANSI/ANS 56.5-1979, “PWR and BWR Containment Spray System Design Criteria” (Reference 19B.2.28-3), with the proviso that the standard’s requirements for spray solution pH control need not be followed.

Acceptance Criteria

The acceptance criteria for the resolution of Issue C -10 is that the containment spray system shall be designed to meet the requirements of General Design Criteria 41, 42 and 43 (Reference 19B.2.28-4) related to fission product removal, periodic inspection, and functional testing, respectively, by conforming to the guidance of SRP Section 6.5.2, Revision 2. Specifically, the system design shall consider the appropriate criteria of ANSI/ANS 56.5-1979 except that the requirements of this standard for any spray additive or other pH control system need not be followed. The design shall minimize the probability of inadvertent actuation of the system and of consequent damage to equipment in the containment. The aqueous solution collected in the containment sump after completion of ECCS injection shall be maintained at an equilibrium pH of no less than 7.0 for long-term iodine retention and the protection of austenitic stainless steel materials from stress corrosion cracking in accordance with the guidance of BTP MTEB 6-1. Pre-operational tests of the containment spray system shall be specified to demonstrate that it meets the design requirements for an effective fission product scrubbing function, and technical specifications shall specify appropriate limiting conditions of operation.

Resolution

The Residual Heat Removal (RHR) system provides two independent containment spray cooling systems (on loops B and C) each having a common header in the wetwell and a common spray header in the drywell and sufficient capacity for containment depressurization by removing heat and condensing steam in both the drywell and wetwell air volumes following a LOCA. The drywell sprays also function to provide removal of fission products released during a LOCA as well as in the event of failure of the drywell head. The RHR system pumps water from the suppression pool, through the RHR heat exchangers into the wetwell and drywell spray spargers in the primary containment.

The drywell spray mode is initiated by operator action post-LOCA in the presence of high drywell pressure, and is terminated by operator action. Also, drywell spray is terminated automatically as the RHR injection valve starts to open, (which results from a LOCA and reactor depressurization). The wetwell spray mode is initiated by operator action, and is terminated automatically by a LOCA or terminated by operator action.

The water in the 304L stainless-steel-lined suppression pool is maintained at high purity (low corrosion attack) by the Suppression Pool Cleanup (SPCU) System. In the event of a LOCA, the SPCU function is automatically terminated to accomplish containment isolation. The pH range (5.3-8.9) is maintained to minimize any corrosive attack on the pool liner (304L SS) over the life of the plant. The post-LOCA aqueous phase pH in all areas of containment will have a flat time history (i.e., the liquid coolant will remain at its design basis pH throughout the event). The use of organic coatings within the containment has been kept to a minimum. The major use of such coatings is on the carbon steel containment liner, internal steel structures and equipment inside the drywell and wetwell. The epoxy coatings are specified to meet the requirements of Regulatory Guide 1.54 and are qualified using the standard ANSI tests, including ANSI N101.4. All safety-related equipment in the containment is environmentally qualified, and protected against spray actuation (Section 3.11).

The system design adheres to the appropriate criteria guidelines of ANSI/ANS 56.5-1979. Application of accepted human factors principles and methodologies to the RHR System instrumentation and controls design minimizes the possibility of inadvertent actuation as a result of operator error (Subsection 18.3.1). Pre-operational testing for operability is performed on the RHR Containment Spray Subsystem (Subsection 14.2.12.1.8). Technical Specifications/Limiting Conditions for Operation (LCOs) of the RHR Containment Spray Subsystem and the Primary Containment System are given in Chapter 16, Section 3.6.

It should be noted that credit is not taken for any fission product removal provided by the drywell and wetwell spray portions of the RHR system. The quantity of fission products released into the environment following postulated accidents is controlled by the standby gas treatment system (SGTS) that has the redundancy and capability to filter the gaseous effluent from the primary and the secondary containment.

The ABWR Design fulfills the requirements of General Design Criteria 41, 42, and 43 relating to fission product removal, periodic inspection, and functional testing by conforming to the criteria guidelines of SRP Section 6.5.2, Revision 2 (Subsections 3.1.2.4.12.2, 3.1.2.4.13.2, and 3.1.2.4.14.2).

In summary, the ABWR design meets the intent of the criteria guidelines of SRP Section 6.5.2, Revision 2, and BTP MTEB 6-1 in order to fulfill the function of reducing the concentration of radioactive iodine and particulates in the containment atmosphere during and after a LOCA, while also minimizing the probability of initiating stress corrosion cracking of stainless steel in the safeguard systems. Design features also minimize the probability of inadvertent actuation of the RHR Containment Spray subsystem or the SGTS, thus minimizing possible damage to safety-related equipment in the containment. Technical Specifications/LCOs are also provided.

Issue C-10 in NUREG-0933 is, therefore, resolved for the ABWR Standard Plant design.

References

19B.2.28-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.28-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.28-3 ANSI / ANS 56.5, “PWR and BWR Containment Spray System Design Criteria”, American National Standards Institute.

19B.2.28-4 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Office of the Federal Register, National Archives and Records Administration.

19B.2.29 C-17: Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Issue

NUREG-0471 Item C-17 (Reference 19B.2.29-1) discusses the Interim Acceptance Criteria for Solidification agents for radioactive solid wastes.

Acceptance Criteria

The acceptance criteria for the resolution of C-17 is under development. This NUREG-0471 (Reference 19B.2.29-1) task involves the development of criteria for acceptability of radwaste solidification agents to properly implement a process control program for the packaging of diverse plant wastes for shallow land burial.

Resolution

10 CFR Part 61 was published in the Federal Register on December 27, 1982 (47 FR 57446) and includes Section 61.56 which addresses waste characteristic (Reference 19B.2.29-2). BTP ETSB 11-3 on waste form has been developed under TMI Action Plan Item IV.C.1. The ABWR is committed to meeting the requirements in 10 CFR Part 61, Reference 19B.2.29-3 (Subsection 11.4.1.2).

The COL applicant shall demonstrate that the wet waste solidification processes and the spent resin and sludge dewatering processes will result in products that comply with 10 CFR 61.56. A process control program (PCP) shall be provided for the processes employed.

This procedure will encourage the development and use of additional acceptable methods of solidifying radioactive waste solids in the future.

Thus, this item has been resolved for the ABWR.

References

19B.2.29-1 NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)", U.S. NRC, June 1978.

19B.2.29-2 Memorandum for T. Speis from J. Funches, "Prioritization of Generic Issues – Environmental and Licensing Improvements", February 24, 1983.

19B.2.29-3 10 CFR 61.56, "Licensing Requirements for Land Disposal of Radioactive Waste."

19B.2.30 15: Radiation Effects on Reactor Vessel Supports

Issue

Issue 15 in NUREG-0933 (Reference 19B.2.30-1), addresses the potential for failure of the reactor vessel support structure (RVSS) due to a combination of low temperature and neutron flux irradiation embrittlement.

Neutron irradiation of structural materials used in the RVSS causes embrittlement that may increase the potential for propagation of pre-existing cracks or flaws within these materials. The potential for brittle fracture of these materials is typically measured in terms of their nil ductility transition temperature (NDTT). As long as the operating environment in which a material is used has a temperature that is significantly higher than the NDTT of the material, no failure by brittle fracture would be expected. Many materials, when subjected to neutron irradiation, experience an upward shift in the NDTT, i.e., they become more susceptible to brittle fracture. This effect must be accounted for in the design and fabrication of the RVSS.

During 1988, new data was developed for the RVSS materials at Oak Ridge National Laboratory (ORNL) (References 19B.2.30-2 and 19B.2.30-3). This data indicated that neutron flux at low temperatures caused greater embrittlement of the materials used in the RVSS than previously anticipated. This increased material embrittlement or "upward shift" in NDTT reduces the fracture toughness of these materials and, under certain specific and conservative transient conditions such as an earthquake or large-break Loss of Coolant Accident, could conceivably result in the failure of the supports thus permitting the reactor vessel to move.

As a result of the ORNL work, the NRC re-prioritized this issue and is reviewing its regulatory position relative to low temperature and neutron flux radiation embrittlement.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 15 is that the material integrity for the RVSS shall be maintained for the design lifetime of the plant.

Specifically, the design of the reactor vessel supports shall address irradiation effects (including low temperature and neutron flux) and the attendant material embrittlement, and be designed to restrain the reactor vessel under the combined Safe Shutdown Earthquake and branch line pipe break loadings in accordance with the stress and deflection limits established in Section III of the ASME Code (Reference 19B.2.30-4).

Resolution

The RVSS for the ABWR is described in Subsections 5.3.3.1.4.1 and 3.9.1.4.2 and shown in Figure 5.3-2. The RVSS consists of a support skirt bolted to the support pedestal. The skirt is located below the core beltline and slightly below the core support plate. As such, the skirt is in a region of low neutron flux which is further reduced since the ABWR water flow region between the vessel shroud and vessel wall is almost 40cm wider than previous BWRs. Therefore, neutron embrittlement of the skirt is well below any current or potential future limitations. A bounding analysis of neutron flux in these regions is given in Subsection 5.3.3.1.4.7. The value in this analysis of 6 x 10^{17} neutron/cm^2 can be compared to the bounding expected value for the skirt welds of 3 x 10^{14} neutron/cm^2 for a 60 year exposure.

References

19B.2.30-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.30-2 ORNL/TM-10444, “Evaluation of HFIR Pressure Vessel Integrity Considering Radiation Embrittlement”, Oak Ridge National Laboratory, 1988.

19B.2.30-3 ORNL/TM-10966, “Impact of Radiation Embrittlement on the Integrity of Pressure Vessel Supports for Two PWR Plants”, Oak Ridge National Laboratory, 1989.

19B.2.30-4 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear), American Society of Mechanical Engineers.

19B.2.31 23: Reactor Coolant Pump Seal Failures

Issue

This issue deals with the high rate of Reactor Coolant Pump (RCP) seal failures that challenge the makeup capacity of the ECCS in PWRs. However, operating experience indicates that the leak test for major RCP seal failures in BWRs is smaller. The smaller leak rate, larger RCIC, HPCI, and feedwater makeup capabilities, and isolation valves on the RCP loops negate the potential problem in BWRs.

Acceptance Criteria

Not applicable. Issue does not apply to BWRs.

Resolution

The ABWR wet motor Reactor Internal Pumps (RIPs) as described in the ABWR Subsection 5.4.1 do not include seals. This feature is further described in ABWR Subsection 1A.2.30.

Therefore, Issue 23 is resolved for ABWR.

References

19B.2.31-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.32 25: Automatic air Header Dump on BWR Scram System

Issue

This issue concerns the slow loss of control air pressure in the scram system of BWRs. Air pressure dropping at a certain rate will first allow some of the Control Rod Drive (CRD) scram outlet valves to open slightly, thus filling the scram discharge volume with water but allowing little or no control rod movement. Eventually, the rods will try to scram but the scram will be impaired in a manner similar to what happened at Browns Ferry Unit 3 on June 28, 1980 (Reference 19B.2.32-1). Meanwhile, the dropping air pressure can cause a transient (e.g., via feedwater controller lockup) which would normally call for a scram.

Acceptance Criteria

The acceptance criteria for this issue is specific to the scram discharge volume design and is not applicable to the ABWR. See the resolution discussion that follows.

Resolution

For the ABWR fine motion control rod drive (FMCRD) design, scram water is discharged through the drive directly into the reactor vessel. There is no scram discharge volume as used in previous BWR designs employing the locking piston control rod drive (LPCRD). Consequently, the common mode failure or impairment of scram associated with loss of control air pressure and filling of the scram discharge volume is not applicable to the ABWR.

The analogous concern for the ABWR design is that the slow loss of control air pressure in the scram air header can allow some of the scram accumulators to leak to the reactor. This action could deplete the accumulators' charge and impair or prevent their capability to scram the connected control rods, unless specific design features are provided to prevent or mitigate its occurrence. The ABWR design does provide protection against this event by incorporating the following features:

(1) A scram air header low pressure alarm to alert the operator of a low pressure condition in the header (Figure 4.6-8, Sheet 2)—The setpoint value is chosen

to be greater than the pressure at which the scram valves could start to open in order to allow the operator the opportunity to take corrective action.

(2) A rod block and alarm initiated by low pressure and a scram initiated by low-low pressure in the common header supplying the charging water to the scram accumulators—All the accumulators will have sufficient water volume to scram their associated control rods as long as the CRD System pump maintains the pressure in the charging header above the minimum required accumulator charging pressure, even if multiple scram valves are leaking. The pressure in the header will drop only if the total scram valve leakage flow is greater than the capability of the charging pump to provide make-up and maintain system pressure. If this should occur, instrumentation located in the charging header will sense the loss of pressure and signal the RCIS to initiate a rod block and alarm at a predetermined low pressure setpoint. If pressure degrades even further, it will signal the RPS to initiate an immediate scram at a predetermined low-low pressure setpoint. The low-low pressure setpoint value is based on the minimum accumulator charging pressure. This automatic feature protects the capability to safely shut down the plant by assuring that scram occurs while adequate accumulator charge is still available (Subsection 4.6.1.2.4.3).

In summary, the ABWR incorporates design features to prevent the loss or impairment of the scram function due to a slow loss of control air in the scram system. The first is a low pressure alarm to alert the operator to trouble in the scram air header; the second is an accumulator charging header low pressure scram to automatically shut down the plant before the accumulators are depleted.

Therefore, this issue is resolved for the ABWR design.

References

19B.2.32-1 "Report on the Browns Ferry 3 Partial Failure to Scram Event on June 28, 1980", U.S. NRC, July 30, 1980.

19B.2.33 40: Safety Concerns Associated with Pipe breaks in the BWR Scram System

Issue

If a break or leak exists or develops in the scram discharge volume (SDV) piping during a reactor scram, this would result in the release of water and steam at 373.15 K (212°F) into the reactor building at a maximum flow rate of 3.469E-2 m^3/s (550 gpm) and is postulated to result in 100% relative humidity in the reactor building. The principal means of isolating this break would be to close the scram exhaust valves which are located on the hydraulic control units; however, this is dependent upon the ability to reset scram, which cannot be absolutely ensured immediately following the scram. Therefore, a rupture of the SDV could result in the unisolable break outside of primary containment, which is postulated to threaten emergency core cooling equipment by flooding areas in which this equipment is located and by causing ambient temperature and relative humidity conditions for which this equipment is not qualified.

Acceptance Criteria

NUREG-0803 (Reference 19B.2.33-1) provides guidance to ensure SDV pipe integrity, detection capability, mitigation capability and qualification of the emergency equipment to the expected environment.

Resolution

For the ABWR fine motion control rod drive (FMCRD) design, scram water is discharged through the drive directly into the reactor vessel. There are no CRD withdraw lines or SDV as used in previous BWR designs employing the locking piston control rod drive (LPCRD). Consequently, the issue of SDV isolation provisions as addressed in NUREG-0803 (Reference 19B.2.33-1) is not applicable to the ABWR design.

In addition, for protection against a scram insert line break, the ABWR FMCRD design incorporates a ball-check valve located in the FMCRD flange housing at the point of connection of the insert line with the drive scram port. In the event of a rupture of the insert line, the ball-check valve will close to prevent reactor vessel flow out of the break. This feature is the same as used by the LPCRD in previous BWR designs.

For these reasons, this issue is resolved for the ABWR design.

References

19B.2.33-1 NUREG-0803, “Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping”, U.S. NRC, August 1981.

19B.2.34 45: Inoperability of Instrumentation Due to Extreme Cold Weather

Issue

Generic Safety Issue (GSI) 45 in NUREG-0933 (Reference 19B.2.34-1), addresses the potential for safety-related equipment instrument lines to become inoperable as a result of freezing or reaching the precipitation (i.e., solidification) point of the sensing fluids.

Typical safety-related systems employ pressure and level sensors which use small bore instrumentation lines. Most operating plants contain safety-related equipment and systems, parts of which are exposed to the ambient environment. These lines generally contain liquid (e.g., borated water) which is susceptible to freezing. Where systems or components and their associated instrumentation are exposed to sub-freezing temperatures, heat tracing and/or insulation is used to minimize the effects of cold temperatures.

These sensing and instrumentation lines are of concern because, should they freeze, they may prevent a safety-related system or component from performing its safety function. For example, an incident occurred at a plant wherein the heat tracing system surrounding sensing lines and level transmitters for the Refueling Water Storage Tank (RWST) failed during sub-freezing weather. The failure of the heat tracing systems resulted in freezing of the sensing lines and associated level transmitters causing a loss of all four RWST instrumentation channels, which could have resulted in the failure of the Emergency Core Cooling System, thus jeopardizing plant safety.

Because of the possibility of a safety-related system failure, the NRC issued additional guidance given in Regulatory Guide 1.151 (Reference 19B.2.34-2) to supplement the existing guidance and requirements which include the Standard Review Plan (SRP) Section 7.1, 10 CFR 50, Appendix A, and industry standard ISA-S67.02 (References 19B.2.34-3, 19B.2.34-4, and 19B.2.34-5, respectively). Regulatory Guide 1.151 specifically addresses the prevention of safety-related instrument sensing line freezing and includes design issues such as diversity, independence, monitoring and alarms.

Acceptance Criteria

The acceptance criterion for the resolution of GSI 45 is that the fluid in safety-related equipment instrument sensing lines shall be protected from freezing and maintained above the precipitation point.

The protection of safety-related equipment instrument sensing lines from freezing can be accomplished by providing environmental control systems which meet the requirements of 10 CFR 50, Appendix A (GDCs); industry standard ISA-S67.02; the intent of Regulatory Guide 1.151; and SRP Sections 7.1 (Revision 3), 7.1, Appendix A (Revision 1), 7.5 (Revision 3), and 7.7 (Revision 3).

Also, should environmental control prove to be limited, alternative forms of sensing line protection such as heat tracing and/or insulation can be used. (The use of heat tracing and/or insulation is not anticipated for the ABWR Standard Plant design; however, it is an acceptable alternate to environmental control.)

Resolution

The ABWR Standard Plant incorporates instrument sensing lines in safety-related systems and components. All safety-related systems and components used in the ABWR Standard Plant design, including instrument sensing lines, are located in temperature controlled environments which are maintained above the freezing (or precipitation) point of the contained fluid. The temperatures of these environments are not expected to be less than 283 K (10°C), as shown in Appendix 3I. In addition, the ABWR is committed to meet the requirements of Regulatory Guide 1.151 (Table 1.8-20), which endorses and augments ISA-S67.02.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.34-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.34-2 Regulatory Guide 1.151, "Instrument Sensing Lines", U.S. NRC.

19B.2.34-3 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition", U.S. NRC.

19B.2.34-4 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants", Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.34-5 ISA-S67.02, "Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants", Instrument Society of America.

19B.2.35 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

Issue

Issue 51 in NUREG-0933 (Reference 19B.2.35-1), identifies the susceptibility of the Station Service Water System (SSWS) to fouling which leads to plant shutdowns and reduced power operation for repairs.

The SSWS cools the Component Cooling Water System (CCWS) through the Component Cooling Water Heat Exchangers and rejects the heat to the ultimate heat sink (UHS) during normal, transient, and accident conditions. The CCWS in turn provides cooling water to those safety-related components necessary to achieve a safe reactor shutdown, as well as to various non-safety reactor auxiliary components.

Acceptance Criteria

Elimination of the possible effects of fouling of the service water system and ultimate heat sinks is a design goal of the ABWR. The COL applicant is given specific requirements and guidance on achieving this goal, including instruction to consider designs and new requirements which further mitigate the fouling effects. Additionally, the COL applicant is directed to investigate the problem with ice as a flow blockage mechanism and to dispose of and/or dissolve such ice as required.

Resolution

A review of operating plant experience shows that the most prevalent problems with plant cooling water systems are due to the corrosion and fouling caused by poor quality service water. In spite of a variety of water treatment schemes and use of expensive material, the wide range of harsh chemistry, silt and biological content result in a need for continuous maintenance of equipment. In order to make a significant operational improvement in this area, the ABWR requirements for plant cooling water systems will include the following (Reference 19B.2.35-2).

- Direct service water will not be used for component cooling. A closed loop component cooling water system will be utilized to transfer heat from the component heat loads via a heat exchanger to the service water system and ultimate heat sink. This minimizes the number of pieces of equipment which are in contact with the problem-causing service water and focuses the problem on the component cooling water heat exchanger.
- The COL applicant shall treat raw service water as needed to reduce the effect of mud, silt, or organisms.
- The COL applicant shall provide materials for piping, pumps, and heat exchangers that offer greater resistance to the range of probable water chemistry conditions.
- The COL applicant shall make provisions to facilitate the inspection of service water piping and replace sections of piping during plant life.

The COL applicant shall provide sufficient redundancy of makeup pumps for the ultimate heat sink so that makeup capabilities are not unduly reduced when one pump malfunctions. The need for a safety grade makeup shall be established in conjunction with establishing UHS water volume, as specified in Regulatory Guide 1.27 (Reference 19B.2.35-3).

The COL applicant shall provide the safety related portions of these systems to meet the design bases during a loss of offsite power. These systems shall be designed to perform their cooling function assuming a single active failure in any mechanical or electrical system.

Therefore, this issue, 51, is resolved for ABWR.

References

19B.2.35-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.35-2 “Advanced Light Water Reactor Utility Requirement Document” (Volume II), EPRI.

19B.2.35-3 Regulatory Guide 1.27, “Ultimate Heat Sink for Nuclear Power Plants.”

19B.2.35-4 Enclosure 1, Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment.”

19B.2.36 57: Effects of Fire Protection Systems Actuation on Safety-Related Equipment

Issue

Generic Safety Issue (GSI) 57 in NUREG-0933 (Reference 19B.2.36-1), addresses the potential for safety-related equipment to become inoperable because of water spray from the fire protection system. IE Information Notice 83-41 (Reference 19B.2.36-2) identified experiences in which actuation of fire suppression systems caused damage to safety-related equipment.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 57, is that the fire protection system be designed to preclude damaging safety-related equipment and rendering the equipment inoperable. In addition, the fire protection system shall be designed to meet 10 CFR 50, Appendix A (GDC 3) (Reference 19B.2.36-3), which states in part:

> "Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of those structures, systems, and components."

Resolution

The ABWR is designed to prevent the inadvertent actuation of fire protection systems and to limit the effects of water spray onto safety-related equipment. The only safety-related equipment located in areas protected by automatic fire suppression systems are the emergency diesel generators and their associated auxiliary equipment. The automatic fire suppression systems protecting the safety-related equipment are of a highly reliable pre-action automatic sprinkler type. Actuation of these sprinklers requires the detection of a fire by infra-red and/or rate of heat detectors, and the opening of the fusible link sprinkler heads. Furthermore, each division has its own dedicated detection and actuation equipment for the control of the fire sprinklers in that divisional area. Two actuation signals are required to initiate the fire suppression system, the first of which will annunciate an alarm to alert the operator to any potential problems. In addition the operator has the capability of terminating the flow of fire suppressant locally by manual isolation valves.

In order to prevent damage to other equipment due to flooding from the discharge of a sprinkler system, equipment is elevated and floor drains are provided.

The basic physical layout of the ABWR and the selection of systems is such to enhance the tolerance of the ABWR plant to fire. The systems are designed and located such that there are three independent and physically safety-related divisions, any one of which is capable of bringing the plant to a safe shutdown in the event of a fire. For design purposes it is assumed that a fire in a division results in the immediate loss of function of the entire division. Even with this conservative assumption, the two remaining independent safety-related divisions are available for full utilization by the operator.

The ABWR is designed in accordance with 10 CFR 50 Appendix A (GDC 3), Reference 19B.2.36-3, to minimize the adverse effects of fire. Since the automatic fire protection systems are designed to preclude inadvertent actuation and in the event of an improbable inadvertent actuation the effects are limited to a single division, this issue is resolved for the ABWR (Section 9.5 and Appendix 9A).

References

19B.2.36-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.36-2 IE Information Notice 83-41, “Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment”, June 22, 1983.

19B.2.36-3 10 CFR 50 Appendix A, “General Design Criteria”, Office of the Federal Register, National Archives and Records Administration.

19B.2.37 67.3.3: Improved Accident Monitoring

Issue

This Generic Safety Issue addresses the weaknesses in the accident monitoring of the type observed at the Ginna steam generator event (steam generator isolation, reactor coolant pumps trip, thermal shock from cold high pressure injection water). The weaknesses identified were:

(1) non-redundant monitoring of RCS pressure;

(2) failure of the position indication for the steam generator relief and safety valves; and

(3) the limited range of the charging pump flow indicator for monitoring charging flow during accidents.

These conditions make it more difficult for correct operator action in response to such events. Subsequently, the NRC Staff prepared and issued Regulatory Guide (RG) 1.97, Revision 2, which was implemented at Ginna (Reference 19B.2.37-1).

Acceptance Criteria

The acceptance criteria for the resolution of this item is based on the full implementation of the post-accident monitoring requirements of RG 1.97 (Reference 19B.2.37-3) and NUREG-0737 TMI Action Plans into the design of the ABWR.

Resolution

The ABWR has implemented into its basic design RG 1.97 requirements and the TMI action plan requirements of NUREG-0737 and NUREG-0737, Supplement 1 (Reference 19B.2.37-1). Refer to Subsections 7.5.1.1 and 18.2, and Table 7.5-2. The ABWR design is in full compliance with the latest issue of RG 1.97; and, this Issue, 67.3.3, is resolved for ABWR.

References

19B.2.37-1 NUREG-0737, "Clarification of TMI Action Plan Requirements", U.S. NRC, November 1980.

19B.2.37-2 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.37-3 Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", U.S. NRC.

19B.2.38 75: Generic Implications of ATWS Events at Salem Nuclear Plant

Issue

On two occasions, Salem Unit 1 failed to scram automatically due to failure of both reactor trip breakers to open on receipt of an actuation signal. In both cases the unit was successfully tripped by manual action. The failure of the breakers has been attributed to excessive wear due to improper maintenance of the undervoltage relays which receive the trip signal from the protection system and cause mechanical action to open the breakers.

Failure to scram (also commonly referred to as anticipated transient without scram, ATWS) could result in unacceptable consequences (Reference 19B.2.38-1).

Acceptance Criteria

The acceptance criteria for the resolution of this issue is that:

- The plant must have a program for a post-trip review of unscheduled reactor shutdowns.
- The plant must have a program for safety-related equipment classification and vendor interface.
- The plant must have a program for post-maintenance operability testing.
- The plant must have a program to control vendor-related modifications, preventative maintenance and surveillance for reactor trip breakers.

These acceptance criteria are described in Generic Letter 83-28 (Reference 19B.2.38-2) and NUREG-1000 (Reference 19B.2.38-3).

Resolution

The reactor protection (trip) system (RPS) design provides the capability for the ABWR to satisfy the NRC requirements indicated in Generic Letter 83-28 and in NUREG-1000.

Execution of the programs in the Acceptance Criteria fall primarily into the phase of operations and maintenance that are the responsibility of the COL applicant. However, Section 3.2 provides the safety-related classification of principal components for the second criterion of the Acceptance Criteria.

Therefore, this issue, 75, is resolved for ABWR.

References

19B.2.38-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.38-2 Generic Letter No. 83-28, “Required Actions Based on Generic Implication of Salem ATWS Events”, July 8, 1983.

19B.2.38-3 NUREG-1000, “Generic Implications of ATWS Events at the Salem Nuclear Power Plant”, Volumes 1 & 2, April 1983, August 1983.

19B.2.39 78: Monitoring of Fatigue Transient Limits for Reactor Coolant System

Issue

Generic Safety Issue (GSI) 78 in NUREG-0933 (Reference 19B.2.39-1), addresses the concern that for a number of older Operating Plants, there are no Technical Specification (TS) requirements for monitoring the actual number of transient occurrences. In addition, environmental effects were not taken into account in the design bases for Reactor Coolant Pressure Boundary (RCPB) components. Environmental effects on fatigue resistance of material are not explicitly addressed in the ASME Code, Section III, (Reference 19B.2.39-2) Design Fatigue curves. Therefore, an assessment of the increase in Core Damage Frequency (CDF) due to environmental effects on fatigue resistance of material should be performed.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 78 are that operating plants implement TS to monitor plant transients, and environmental effects on the fatigue life of ASME Code, Section III, Class 1 carbon steel piping be considered in accordance with Section 3.9.3.1.1.7.

Resolution

For the ABWR, Technical Specification 5.7.2.9 requires the monitoring of plant transients to ensure that RCPB components are maintained within their design limits. Environmental effects are included in the design bases for ABWR RCPB components. The calculated CDF includes the environmental effects on fatigue resistance of materials.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.39-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.39-2 "ASME Boiler and Pressure Vessel Code", Section III (Nuclear), American Society of Mechanical Engineers.

19B.2.40 83: Control Room Habitability

Issue

Safety Issue 83 in NUREG-0933 (Reference 19B.2.40-1) is a concern over the loss of control room habitability following an accident release of external airborne toxic or radioactive material or smoke. Such a loss could impair or cause loss of the control room operators' capability to safely control the reactor and could lead to a core damaging accident.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 83 is to verify that the control room design is adequate to prevent the loss of habitability of the control room during an accident. The design must meet the guidance given in Standard Review Plan (SRP) Sections 6.4, 9.4.1, and 15.6.5.5 (Reference 19B.2.40-2). The design must be in accordance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 2, 4, and 19 (Reference 19B.2.40-3); and ASME AG-1 and AG-1a (Reference 19B.2.40-5).

Resolution

The ABWR main control room habitability system is described in Subsection 9.4.1 and Section 6.4. The control room is a structure which is important to safety and is designed to withstand the effects of natural phenomena, missiles and postulated accidents in accordance with GDC 2 and 4. The design of the control room [and its heating, ventilation and air conditioning (HVAC) system] permits safe occupancy during abnormal conditions. Radiation exposure of control room habitability area personnel through the duration of any one of the postulated design basis accidents does not exceed the guidelines set by GDC 19, i.e., 50 mSv whole body radiation exposure. Smoke and toxic gas protection is provided as described in Subsection 6.4.4.2 by the use of non-combustible materials, purging by the HVAC, individual respirators, and site-specific considerations of potential chemical releases. The control room Engineered Safety Feature filter trains shall be designed, tested and maintained to comply with the applicable provisions of Regulatory Guide 1.52 (Reference 19B.2.40-4), as described in Subsection 9.4.1.1.7. Fire protection is provided by alarm systems, fire hoses and portable fire extinguishers (Subsections 9.5.1 and 9A.4.2). Testing and inspection requirements are identified in Subsection 6.4.5.

Since the control room design prevents the loss of control room habitability during accident conditions, and since all of the NRC requirements and guidance are met, this issue is resolved for the ABWR.

References

19B.2.40-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.40-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.40-3 10 CFR 50 Appendix A, “General Design Criteria for Nuclear Power Plants”, Office of the Federal Register, National Archives and Records Administration.

19B.2.40-4 Regulatory Guide 1.52, “Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants.”

19B.2.40-5 ASME AG-1 “Code on Nuclear Air and Gas Treatment and ASME AG-1a” Addenda.

19B.2.41 86: Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping

Issue

Issue 86 in NUREG-0933 (Reference 19B.2.41-1), addresses the past occurrences of intergranular stress corrosion cracking (IGSCC) in BWR recirculation loop piping and its impact on the integrity of the reactor coolant pressure boundary.

Cracking in large-diameter piping resulting from IGSCC could result in a loss-of-coolant accident.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 86 are that IGSCC resistant materials and fabrication techniques to minimize sensitization shall be used. In addition, the ABWR water shall be maintained at the lowest practically achievable impurity levels.

Furthermore, the material and fabrication techniques shall comply with the guidelines of NUREG-0313 (Reference 19B.2.41-2).

Resolution

For the ABWR, IGSCC resistance is achieved through the use of Type 316 stainless steel and compliance with the guidelines of NUREG-0313. All materials are supplied in the solution heat treated condition. During fabrication, any heating operations (except welding) between 800 K (427°C) and 1255 K (982°C) are avoided, unless followed by solution heat treatment. The ABWR water is maintained at the lowest practically achievable impurity levels to minimize its corrosion potential.

In summary, only stainless steel Type 316 material is used and the piping is fabricated, tested and installed in accordance with ASME Code, Section III, (Reference 19B.2.41-3) and NUREG-0313. Also, the owner-operator is required to comply with ASME Code, Section XI, (Reference 19B.2.41-3) for the performance of inservice inspection.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.41-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.41-2 NUREG-0313, “Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping”, U.S. NRC, July 1977, (Revision 1) July 1980, (Revision 2) January 1988.

19B.2.41-3 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear) and Section XI, American Society of Mechanical Engineers.

19B.2.42 87: Failure of HPCI Steam Line Without Isolation

Issue

This issue concerns a postulated break in the High Pressure Coolant Injection (HPCI) System steam supply line and the uncertainty regarding the operability of the HPCI steam supply line isolation valves under the postulated conditions (Reference 19B.2.42-1). A similar situation can occur in the Reactor Water Cleanup (CUW) System.

The HPCI steam supply line has two containment isolation valves (MOVs) in series: one on the inside and one on the outside of the containment. Both are normally open in order for the HPCI system to perform its function. The CUW also has two normally open containment isolation valves (MOVs) which must remain open if the system is to perform its function.

The operation of the valves is tested periodically without steam. Also, due to flow limitations at the valve manufacturer's facilities, only the opening characteristics are tested under operating conditions. Therefore, according to the NRC, the capability of the valves to close when exposed to the forces created by the flow resulting from a break downstream has not been demonstrated.

Furthermore, NRC sponsored testing has increased the concern over whether MOVs can reliably be expected to operate under design basis (i.e., pipe break) conditions.

Under a contract from the NRC, Idaho National Engineering Laboratory (INEL) conducted tests on six MOVs. The tests showed that all six valves required more force to open and close at the line break flow rates than was predicted. Two of the conditions tested were full guillotine breaks in the CUW and HPCI systems. These test results were reported at an NRC sponsored meeting on April 18, 1990 which prompted the NRC to issue Generic Letter 89-10 (Reference 19B.2.42-2).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 87 is defined in Generic Letter 89-10 which requires adequately sized actuators for MOVs, verification of correct thrust and torque settings, and a program for testing, inspection and maintenance of MOVs under differential pressure, temperature and flow conditions so as to provide assurance that they will function when subjected to design basis conditions.

Resolution

The ABWR does not have an HPCI system. It does have an CUW system and a Reactor Core Isolation Cooling (RCIC) system which may fall under this issue.

The ABWR addresses the concerns and issues identified in GL 89-10 (specifically the method of assessment of the loads, the method of sizing the actuators, and the setting of the torque and limit switches) in Subsections 3.9.3.2 and 3.9.6.2.2.

Thus, compliance with GL 89-10 resolves Issue 87 for the ABWR design.

References

19B.2.42-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.42-2 Generic Letter No. 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance (with Supplements 1-4)", February 12, 1992.

19B.2.43 89: Stiff Pipe Clamps

Issue

Issue 89 in NUREG-0933 (Reference 19B.2.43-1), addresses the concern that for operating plants, the effects of stiff pipe clamps were assumed to be negligible and were not explicitly considered in the piping design. For some applications, there is a concern that certain piping system conditions coupled with specific stiff pipe clamp design requirements could result in interaction effects that should be evaluated in order to determine the significance of the induced pipe stresses.

The ASME Code, Section III, (Reference 19B.2.43-2) requires that the effects of attachments in producing thermal stresses, stress concentrations and restraints on pressure retaining members be taken into account in checking for compliance with stress criteria. Attachments to piping are generally categorized as integral and non-integral attachments. Lugs welded to the pipe wall are an example of integral attachments. Clamps used for attaching hangers, struts and snubbers to the pipe by bolting are non-integral attachments. Piping design reports specifically address local stresses at integral attachments, such as lugs. Any additional stresses induced in the pipe by non-integral, clamp bolted attachments, are not included in the piping design report.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 89 is that the effect of stiff pipe clamps on piping stresses should be considered in the piping system design. For stiff pipe clamps installed on straight runs of pipe or on bends with a radius of at least five pipe diameters, the pipe clamp induced stresses can be considered negligible and explicit consideration is not required. This acceptance criteria is based upon analysis performed by GE.

In the 1980's, GE performed calculations for typical stiff pipe clamps used on BWR Main Steam and Recirculation piping systems. For each system, the stiff pipe clamps were installed on straight pipe or on bends with a radius of at least five pipe diameters. The purpose of these calculations was to evaluate the additional stresses at clamp locations due to the following:

(1) Differential thermal expansion of the pipe and clamp,

(2) Discontinuity stress in the pipe from internal pressure restraint,

(3) Thermal gradient through the pipe wall in the vicinity of the pipe clamp, and

(4) External loads due to dynamic events such as earthquake.

Maximum incremental primary stresses were less than 25% of the primary stress allowables, and maximum incremental secondary stresses were less than 40% of secondary stress allowables. The stresses at the clamp locations excluding clamp induced stresses were less than 30% of the ASME Code, Section III, allowables. The total primary and secondary stresses, including clamp induced stresses, were less than 70% of allowable stress.

The governing stress locations occurred at piping branch connections, elbows and shear lugs, they did not occur at stiff pipe clamp locations. The stress intensification that occurs at elbows, branch connections and shear lugs is much greater than that which occurs at stiff pipe clamps. Therefore, when the additional clamp induced stresses are included, the peak piping system stresses do not occur at the clamp locations. Based on these calculations, it was concluded that explicit consideration of clamp induced piping stresses is not required when the clamps are installed on straight pipe or on bends with a radius of at least five pipe diameters.

Resolution

For the ABWR, the following stiff pipe clamp parameters will be very similar to those for the BWR stiff pipe clamps evaluated in the calculations summarized above:

- Stiff pipe clamp geometry and material properties
- Pipe schedule and material properties
- Support rated loads less than or equal to 2.26E+05 N
- Piping system operating pressures and temperatures and operating transients
- Piping stresses at branch connections and elbows much greater than at stiff clamp locations

Therefore, it can be concluded that the governing ABWR piping stresses will not occur at stiff pipe clamp locations. For the ABWR, the piping design specifications shall require that stiff pipe clamps be installed on straight runs of pipe or on bends with a radius of at least five pipe diameters. The pipe clamp induced stresses for NSS piping can then be considered negligible and do not warrant explicit consideration. The piping design specifications shall require that if stiff clamps are used on other than NSS piping, the stresses they induce will be considered.

This issue is resolved for the ABWR.

References

19B.2.43-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.43-2 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear) and Section XI, American Society of Mechanical Engineers.

19B.2.44 103: Design for Probable Maximum Precipitation

Issue

Issue 103 in NUREG-0933 (Reference 19B.2.44-1) addresses the accepted methodology used for determining the design flood level for a particular reactor plant site. Accurate determination of the design flood level for a specific reactor site is necessary in order to ensure adequate protection of safety-related equipment against possible site flooding.

Reactor plant sites are designed to accommodate maximum flood level because flooding could disable safety-related equipment. Historically estimating design flood levels for specific reactor plant sites has been based upon input data for probable maximum flood (PMF) provided by the U.S. Army Corp. of Engineers for the specific site. The guidance identified in the Standard Review Plan (SRP) Sections 2.4.2, Revision 3, and 2.4.3, Revision 3 (Reference 19B.2.44-2); and GL 89-22 (Reference 19B.2.44-7) is used in predicting design flood levels. Furthermore, general requirements are defined in General Design Criterion (GDC) 2 (Reference 19B.2.44-3). The SRPs state that “design basis flood levels” incorporate the most severe historical environmental data with “sufficient margin”. What is considered to be “sufficient margin” and procedures for estimating PMFs are identified in Regulatory Guides 1.59 and 1.102; and ANSI/ANS 2.8 (References 19B.2.44-4, 19B.2.44-5, and 19B.2.44-6).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 103 is that the site chosen for a commercial nuclear generating facility shall be designed to accommodate a maximum expected flood from precipitation without jeopardizing the safe operation of the facility, in accordance with the guidance given in SRP 2.4.2, Revision 3; SRP 2.4.3, Revision 3; and GL 89-22. Also, the facility design, including structures, systems, and components important to safety, shall meet the criteria specified in 10 CFR 50, Appendix A (GDC 2).

Resolution

The ABWR is designed to meet the requirements of GDC 2 as described in Subsection 3.1.2. This ABWR design is based upon a set of assumed site-related parameters. These parameters were selected to envelope most potential nuclear power plant sites in the United States. A summary of the assumed site design parameters, including maximum flood level, is given in Section 2.0, Table 2.0-1, and Section 3.4.

Detailed site characteristics based upon historical site specific environmental data will be provided by the site owner-operator for any specific application. The site owner-operator will review and evaluate these characteristics to ensure compliance with the enveloping assumptions of Tables 2.0-1 and 3.4.1.

Since the ABWR is designed in accordance with GDC 2 for the most severe expected environment conditions, including flooding, tornado, hurricane etc. and meets the intent of SRP Section 2.4.2, Revision 3; SRP Section 2.4.3, Revision 3; and GL 89-22 with respect to plant design, this issue is resolved for the ABWR design.

References

19B.2.44-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.44-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.44-3 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants”, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.44-4 Regulatory Guide 1.59, “Design Basis Floods for Nuclear Power Plants”, U.S. NRC.

19B.2.44-5 Regulatory Guide 1.102, “Flood Protection for Nuclear Power Plants”, U.S. Nuclear Regulatory Commission.

19B.2.44-6 ANSI / ANS 2.8, “Standard for Determining Design Basis Flooding at Power Reactor Site”, American Nuclear Society.

19B.2.44-7 GL 89-22, “Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Plants Due to Recent Change in Probable Maximum Precipitation Criteria”, National Weather Service, October 19, 1989.

19B.2.45 105: Interfacing Systems LOCA at BWRs

Issue

In all currently operating light water reactors, there are a number of high / low pressure interfaces between the reactor coolant pressure boundary (RCPB) and connected systems. Thus there are systems in BWRs that are designed for a pressure lower than that of the primary system. For example, the BWR primary system operates at about 7.0 MPa, while the Residual Heat Removal (RHR) System can operate at pressures up to 3.53 MPa. Isolation valves (at least two) and piping to the primary system are designed for about 8.73 MPa. The discharge of the BWR RHR System, which also functions as a low pressure injection system, passes through testable check valves prior to returning to the reactor coolant system.

The common concern in the above issue is that either tests that require valve actuation, valve leakage, or multiple valve failures could result in a system pressure that exceeds the design pressure of the low pressure emergency cooling or decay heat removal systems causing them to fail from overpressure.

Risk calculations on existing plants suggest there may be a need for improved protection against the potential for overpressurization of some emergency cooling and decay heat removal systems (Reference 19B.2.45-1).

Acceptance Criteria

Reference 19B.2.45-2 indicated that future ALWR designs like the ABWR should reduce the possibility of a LOCA outside containment by designing (to the extent practicable) all systems and subsystems connected to the reactor coolant system (RCS) to an ultimate rupture strength (URS) at least equal to full RCS pressure.

Reference 19B.2.45-3 found that for the ABWR the design pressure for the low-pressure piping systems that interface with the RCPB should have the following criteria to achieve satisfactory retention of the full 7.17 MPa reactor pressure on an ultimate rupture strength basis.

(1) The design pressure for the low-pressure piping systems that interface with the RCPB pressure boundary should be equal to 0.4 times the normal operating RCPB pressure of 7.17 MPa.

(2) The minimum wall thickness of the low-pressure piping should be no less than that of a standard weight pipe.

(3) The remaining components in the low-pressure systems should also be designed to a design pressure of 0.4 times the normal operating reactor pressure [i.e., 2.93 MPa]. This is accomplished for the ABWR by the revised boundary symbols of the P&IDs to the 2.91 MPa design pressure, which includes all the piping and components associated with the boundary symbols.

(4) A Class 300 valve is adequate for ensuring the pressure of the low-pressure piping system under full reactor pressure.

(5) The design is to be in accordance with the ASME Code, Section III, Subarticle NC/ND-3600.

(6) Periodic surveillance and leak rate testing are required of the pressure isolation valves per Technical Specification requirements as a part of the In-Service Inspection (ISI) program.

Resolution

The ABWR design was evaluated and upgraded to comply with the above criteria. Criteria numbered 1 through 4 were accepted and implemented in Tier 2 documentation primarily by indicating the design pressure and design features on the system P&IDs (Piping and Instrument Diagrams). Criteria 5 and 6 were originally part of the ABWR design, and no upgrade was required to comply.

The increased design pressure was extended, forming an URS region extending outward from the RCPB, to the extent practicable. The following items form the basis of what constitutes practicality and set forth the test of practicality used to establish the boundary limits of URS for the ABWR:

- It is impractical to design large tank structures to the URS design pressure that are vented to atmosphere and have a low design pressure. Tanks included in this category are:

  – Condensate storage tank,

  – Standby Liquid Control System (SLCS) main tank,

  – Low Conductivity Waste (LCW) receiving tank,

  – High Conductivity Waste (HCW) receiving tank,

  – Fuel Pool Cooling (FPC) System skimmer surge tank, and

  – Fuel Pool Cooling (FPC) System spent-fuel storage pool and cask pit.

  These are termed low pressure sinks for the purposes of this discussion. The suppression pool is also a low pressure sink that is impractical to upgrade its pressure since it is part of the containment structure, which is designed to contain the most severe LOCA.

- It is impractical to consider a disruptive open flow path from reactor pressure to a low pressure sink. As a consequence, the furthest downstream valve in such a path is assumed closed (with nominal leakage) so that essentially all of the static reactor pressure is contained by the URS upgrade.

- It is impractical to design piping systems (that are connected to a low pressure sink) to URS design pressure when the piping is always locked open to a low pressure sink by locked open valves. Nominal leakage past the last closed valve is the only pressure source that could pressurize the line, and that line is locked open to the low pressure sink vented to atmosphere.

As implied above, boundary limits of the URS design are established assuming slow rates of leakage between high and low pressure regions. A key assumption to understanding the establishment of the boundary limits from the above practicality basis is that only static pressure conditions are considered. Static conditions result by assuming that the last valve in the URS region adjacent to a low pressure sink remains closed, yet considering a slow leak rate that would not over pressurize the down stream piping and components. Thus, the dynamic pressurization effects, violent high flow transients, and temperature escalations are precluded that would occur if the full RCPB pressure was connected directly to the low pressure sink. This means only static pressurization with small leak flow needs to be considered, and low pressure sinks do not pressurize because the path to the sink is open.

The following twelve systems, interfacing directly or indirectly with the RCPB, were evaluated and upgraded.

(1) Residual Heat Removal (RHR) System

(2) High Pressure Core Flooder (HPCF) System

(3) Reactor Core Isolation Cooling (RCIC) System

(4) Control Rod Drive (CRD) System

(5) Standby Liquid Control System (SLCS)

(6) Reactor Water Cleanup (CUW) System

(7) Fuel Pool Cooling Cleanup (FPC) System

(8) Nuclear Boiler System (NBS)

(9) Reactor Recirculation System (RRS)

(10) Makeup Water Condensate (MUWC) System

(11) Makeup Water Purified (MUWP) System

(12) Radwaste System (LCW Receiving Tank, HCW Receiving Tank).

The detailed system evaluation for ISLOCA is provided in Attachment 3MA.

The low pressure piping boundaries were upgraded to URS pressures and extend to the last closed valve connected to piping interfacing a low pressure sink, such as the suppression pool, condensate storage tank or other open configuration (identified pool or tank). The lines from the URS boundary to the low pressure sink do not pressurize because the path to the sink is open. Each interfacing system's piping was reviewed for upgrading. For some systems, with low pressure piping and normally open valves, the valves were changed to lock open valves to ensure an open piping pathway from the last URS boundary to the tank or low pressure sink.

In addition to the above 12 systems, the following two systems were identified as requiring an Interfacing System LOCA (ISLOCA) evaluation.

(1) Condensate, Feedwater and Condensate Air Extraction (F, FDW, AO) System

(2) Sampling (SAM System)

However, these two systems are not in sufficient detail to perform an ISLOCA evaluation. Therefore, an ISLOCA evaluation for these two systems is cited in Tier 2 as requirements for the COL applicant.

The periodic surveillance testing of the ECCS injection valves that interface with the reactor coolant system might lead to ISLOCA conditions if their associated testable check valve was stuck open. To avoid this occurrence, the RHR, HPCF, and RCIC motor-operated injection valves will only be tested during low pressure shutdown operation. This practice follows from the guidance given by Reference 19B.2.45-4.

Although the following is not a new design feature, the RHR shutdown cooling suction line containment isolation valves are also only tested during shutdown operation. These valves are interlocked against opening for reactor pressure greater than the shutdown cooling setpoint approximately 1.03 MPaG.

In summary, based on the NRC staff's new guidance cited in References 19B.2.45-2 through 19B.2.45-5, the ABWR is in full compliance. For ISLOCA considerations, a design pressure of 1.91 MPa and pipe having a minimum wall thickness equal to standard grade has been provided as an adequate margin with respect to the full reactor operating pressure of 7.17 MPa by applying the guidance recommended by Reference 19B.2.45-2. This design pressure was applied to the low pressure piping at their boundary symbols on the P&IDs; and, therefore, impose the requirement on the associated piping, valves, pumps, tanks, instrumentation and all other equipment shown between boundary symbols. A note was added to each URS upgraded P&ID requiring pipe to have a minimum wall thickness equal to standard grade. Upgrading revisions were made to 12 systems.

References

19B.2.45-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1- 15)", U.S. NRC, April 1993.

19B.2.45-2 Dino Scaletti, NRC, to Patrick Marriott, GE, "Identification of New Issues for the General Electric Company Advanced Boiling Water Reactor Review", September 6, 1991.

19B.2.45-3 Chester Poslusny, NRC, to Patrick Marriott, GE, “Preliminary Evaluation of the Resolution of the Intersystem Loss-of-Coolant Accident (ISLOCA) Issue for the Advanced Boiling Water Reactor (ABWR) – Design Pressure for Low-Pressure Systems”, December 2, 1992, Docket No. 52-001.

19B.2.45-4 James M. Taylor, NRC to The Commissioners, SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements”, January 12, 1990, page 8, paragraph 7.

19B.2.45-5 James M. Taylor, NRC, to The Commissioners, SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs”, April 2, 1993.

19B.2.46 106: Piping and the Use of Highly Combustible Gases in Vital Areas

Issue

Generic Safety Issue (GSI) 106 in NUREG-0933 (Reference 19B.2.46-1) was initiated to address the issue of combustible or explosive mixtures of gases accumulating in buildings containing safety-related equipment at a nuclear power plant and the potential disablement of these safety-related equipment if the accumulated gas mixtures are inadvertently ignited.

Except for hydrogen, most combustible gases are used in limited quantities and for relatively short periods of time at a nuclear power plant. Hydrogen is used as a coolant for electric generators and reactor water chemistry in both BWRs and PWRs, and is usually stored in high pressure storage vessels, and supplied to various systems through standard piping. The concern is the potential for leakage, accumulation of combustible or explosive mixtures of gases, and the inadvertent ignition of the gas. The ensuing combustion or explosion could damage or cause failure of safety-related equipment, thereby contributing to a possibly significant increase in the core-melt probability of the plant.

Generic Letter 93-06 (Reference 19B.2.46-2) discusses Issue 106 and risks from failures of hydrogen system lines and components.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 106 are that systems for delivery of hydrogen or other combustible gases be designed to preclude:

(1) large release and accumulation of combustible or explosive mixtures of gases, and

(2) combustion and explosions which could damage or cause failure of safety-related equipment.

This can be accomplished either by designing the piping systems to preclude failure, providing means to detect and limit the amount of hydrogen leakage and accumulation in the event of a piping system break or large leak, or locating safety-related equipment in areas that are not susceptible to damage from combustion and explosions.

Furthermore, the designer shall follow the guidance described in SRP Section 9.5-1 (Reference 19B.2.46-3) and modified BTP CMEB 9.5-1, Part C.5.d(5).

Resolution

The ABWR design uses hydrogen for the Hydrogen Water Chemistry (HWC) System and the main generator bulk hydrogen supply system. These systems are non-nuclear, non-safety-related and are located in the turbine building which is a non-safety-related structure in a non-vital area.There is no significant amount of hydrogen or other highly combustible gases in any vital area for ABWR design.

There are no safety-related systems or components located within the turbine building (Subsection 10.2.2.1) and there are no non-fail-safe safety-related services provided from or through the turbine building. Occurrences in the turbine building which may disable the turbine building's non-safety-related systems, capable of providing safe shutdown, do not disable the reactor building's safety-related equipment which provide safe shutdown (Subsection 9.5.1.1.1).

The arrangement of buildings at the facility, and location of building doors and the bulk hydrogen storage tanks is such that damage to buildings containing safety-related equipment due to combustion of hydrogen or an explosion is unlikely (Subsection10.2.2.2).

The prevention and mitigation of hydrogen combustion and explosions are discussed in Subsections 9.3.9 and 10.2.2. The wind and tornado loading and missile protection for buildings containing safety-related equipment are presented in Sections 3.3 and 3.5, respectively.

Therefore, this issue is resolved for the ABWR.

References

19B.2.46-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.46-2 Generic Letter 93-06, Research Results of Generic Safety Issue 106, “Piping and the Use of Highly Combustible Gases in Vital Areas”, October 25, 1993.

19B.2.46-3 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.47 Not Used

19B.2.48 118: Tendon Anchorage Failure

Issue

Generic Safety Issue (GSI) 118 in NUREG-0933 (Reference 19B.2.48-1), addresses the failure of lower vertical tendon anchor heads in a PWR prestressed concrete containment structure. The failures appear to have been caused by hydrogen stress cracking. The hydrogen is liberated by zinc in the presence of water. Quantities of water ranging from a few cubic centimeters (a few ounces) to about 5.7E+3 cm^3 (1.5 gallons) were found in the grease caps.

Acceptance Criteria

For the ABWR design, the primary containment structure consists of a reinforced concrete design. Since the prestressed concrete containment design is not used in the ABWR Standard Plant design, the tendon anchorage failure issue is not applicable; therefore, no acceptance criteria are needed.

Resolution

For the ABWR design, the primary containment structure is of a reinforced concrete design; therefore, Issue 118 is not applicable.

References

19B.2.48-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.49 120: On-Line Testability of Protection Systems

Issue

Issue 120 was established to examine the on-line (at-power) testability of protection systems and the possibility that some plants might not provide complete testing capability. Protection systems consist of the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS) (Reference 19B.2.49-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 120 is compliance with General Design Criterion (GDC) 21, “Protection System Reliability and Testability”, of Appendix A to 10 CFR 50 (Reference 19B.2.49-5). Supplementary guidance is provided in Regulatory Guides 1.22 and 1.118 (References 19B.2.49-2 and 19B.2.49-3), and IEEE Standard 338 (Reference 19B.2.49-4) to ensure that protection systems (including logic, actuation devices, and associated actuated equipment) will be designed to permit testing while the plant is operating without adversely affecting the plant’s operation. These requirements apply to both the RPS and the ESFAS. Existing Standard Technical Specification indicate that it is desirable to test all protection systems every six months.

Resolution

In the ABWR design the RPS and ESFAS can be tested during reactor operation by six separate tests. The first five tests are primarily manual tests and, although each individually is a partial test, when combined with the sixth test they constitute a complete system test. The sixth test is a self test of the safety system logic and control which automatically tests the complete system, excluding sensors and actuators. Online testability of protection systems is explained in Subsection 7.1.2.1.6. In the ABWR design, all actuation logic is solid state and in software. Periodic surveillance testing is required by Chapter 16, LCO 3.3.1.1 for the SSLC sensor instrumentation, LCO 3.3.1.2 for the RPS and the MSIV actuation, and LCO 3.3.4 for the ESF actuation.

Automatic system self-testing occurs during a portion of every periodic transmission period of the data communication network. Since exhaustive tests cannot be performed during any one transmission interval, the test software is written so that sufficient overlap coverage is provided to prove system performance during tests of portions of the circuitry, as allowed in IEEE 338 (Reference 19B.2.49-4).

On line testability of protection systems may fall into the Operational Reliability Assurance Program (O-RAP). The COL applicant will specify the policy and implementation procedures for the O-RAP, as described in Subsection 17.3.9.

Therefore, this issue, 120, is resolved for ABWR.

References

19B.2.49-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.49-2 Regulatory Guide 1.22, “Periodic Testing of Protection System Actuation Functions.”

19B.2.49-3 Regulatory Guide 1.118, “Periodic Testing of Electric Power and Protection Systems.”

19B.2.49-4 IEEE Standards 338, “Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems.”

19B.2.49-5 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” Office of the Federal Register, National Archives and Records Administration.

19B.2.50 121: Hydrogen Control for Large, Dry PWR Containments

Issue

This issue, 121, concerns the control of hydrogen concentrations in large, dry PWR containments during and after a degraded core accident. In December 1984, the staff recommended that rulemaking with regard to this issue could be safety deferred due to the greater inherent capability of these containments to accommodate large quantities of hydrogen. Ongoing NRC experimental and analytical programs are intended to provide data to support a final recommendation on whether safe shutdown equipment is likely to survive a hydrogen burn (Reference 19B.2.50-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 121 is that the control of hydrogen generated in the containment in a degraded core accident shall meet the requirements of 10 CFR 50.34(f) (Reference 19B.2.50-2) on limiting the distributed hydrogen concentration to 10%, on limiting combustible concentrations, and on maintaining safe shutdown equipment and containment integrity.

Resolution

This issue does not apply to BWRs and pressure suppression containment. Also, the ABWR primary containment is inerted and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation.

References

19B.2.50-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.50-2 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants”, Office of the Federal Register, National Archives and Records Administration.

19B.2.51 124: Auxiliary Feedwater System Reliability

Issue

Issue 124 in NUREG-0933 (Reference 19B.2.51-1) addresses Auxiliary Feedwater System reliability and availability and its impact on reducing core-melt frequency in PWRs.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 124 is that the Auxiliary Feedwater System shall be designed for a high degree of reliability (i.e., using reliability analyses the system shall attain 0.0001 to 0.00001 unavailability per demand).

Resolution

This issue, 124, is not applicable to BWRs and is, therefore, resolved for ABWR.

References

19B.2.51-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.52 128: Electrical Power Reliability

Issue

Issue 128 in NUREG-0933 (References 19B.2.52-1, 19B.2.52-6, and 19B.2.52-7), addresses the reliability of on-site electrical systems.

The minimum acceptable DC power system is comprised of two physically independent divisions which supply DC power for control and actuation of redundant safety-related systems. Questions have been raised concerning the position of regulatory staff, including the application of the single failure criterion for assuring a reliable DC power supply. These concerns stem from the dependence on DC power of the decay heat removal systems required for long-term heat removal. Failure of one DC division would generally result in a reactor scram which then would require removal of decay heat. The frequency of reported single DC division failures gives rise to the concern that the second DC division may not be available.

Two of the specific reasons for the concern that safety-related power may be unreliable are also addressed by this issue. One is that some operating nuclear power plants do not have technical specifications or administrative controls governing operational restrictions for Class 1E 120 VAC vital instrument buses and associated inverters. Without such restrictions these power sources could be out of service indefinitely and thereby may place certain safety systems in a situation where they could not meet the single failure criterion. The other is that the design of some plants do not provide interlocks to prevent the inadvertent closure of the single tie breaker between the 4160 V Class 1E buses.

Acceptance Criteria

The acceptance criteria for the resolution of GSI 128 are encompassed in three interrelated issues, i.e., A-30, 48 and 49, which are summarized as follows:

The acceptance criteria for Issue A-30 are:

(1) Non-safety-related loads shall be placed on completely separated DC power supplies.

(2) Class 1E (safety-related) DC power systems shall be divided into physically and electrically independent systems.

(3) Bus tie breakers between DC systems shall not be used.

(4) Operation and maintenance procedures and/or Technical Specifications governing maintenance and out-of-service shall be implemented.

The acceptance criterion for Issue 48 is that administrative controls or Technical Specifications shall be provided to govern operational restrictions for Class 1E 120 VAC vital instrument buses and associated inverters.

The acceptance criterion for Issue 49 is that the bus tie breakers, if provided between Class 1E emergency buses, shall be redundant and physically separated and open as a condition of operability of the redundant Class 1E electrical distribution system.

Resolution

The ABWR safety-related DC power system design as listed below provides reliable DC buses for safety-related electrical loads and meets the acceptance criteria specified by the resolution of Issue A-30. See Subsection 8.3.2 for as discussion of compliance.

(1) Does not supply power to any non-Class 1E loads, with the exception of the associated DC emergency lighting as described in Subsection 9.5.3.2.3.1.

(2) Consists of four separate and independent DC battery systems.

(3) Does not contain any direct bus ties between DC battery systems. However, it does contain two standby battery chargers. Each standby battery charger is capable of supplying one of two divisional DC systems. Redundant key locked breakers are provided to prevent manual paralleling between divisions. No automatic connections are provided between DC divisions.

(4) COL applicants are required to provide administrative controls for standby battery charger operation and Technical Specifications Sections 3.8.4, 3.8.5, 3.8.6 and 3.8.10 are provided for operational restrictions and allowable out of service times.

The ABWR design meets the acceptance criterion specified above for the resolution of Issue 48 by the system design and Technical Specifications. As described in Subsection 8.3.1.1.4.2, the ABWR design consists of four separate and independent Class 1E 120 VAC vital instrument buses with their respective inverters. There are no bus ties between the four divisions. Operational restrictions are provided in Technical Specifications Sections 3.8.7, 3.8.8, 3.8.9, and 3.8.10 to assure the onsite Class 1E AC and DC power distribution system availability and thus an uninterruptable power source for safety-related systems and components. The Technical Specifications include specific requirements regarding a periodic evaluation of the onsite power system bus condition which considers such availability items as correct breaker and the alignment and adequate bus voltage.

The ABWR design meets the acceptance criteria as stated above for the resolution of Issue 49. The ABWR Class 1E diesel generator bus design does not contain bus tie breakers between Class 1E divisions. However, it is possible to manually cross-connect the Class 1E diesel buses through the combustion turbine generator (CTG) connections since the ABWR design does have the capability of providing power to each diesel bus from the CTG. To cross-connect any two diesel buses, at least four circuit breakers must be at close positions and at least one circuit breaker must be racked in prior to closing. Each diesel generator is provided with synchronizing equipment for paralleling with offsite power supplies. The normal and alternate feeder breakers to the diesel buses are interlocked to prevent paralleling offsite circuits. See Subsections 8.2.1.3 and 8.3.1.1.6 for a discussion of compliance. As discussed in the resolution of Issue 48, the Technical Specification Sections 3.8.9 and 3.8.10 will include operational restrictions and periodic evaluation of correct breaker alignment of Class 1E onsite power distribution system.

Additionally, the ABWR Class 1E power distribution system design as described in Subsection 8.3 fully complies with the IEEE 308 (Reference 19B.2.52-5) and 603 (Reference 19B.2.52-4).

In summary, the ABWR design for the electrical power system avoids the problems described in this issue. Each division of the engineered safety systems has emergency onsite sources of AC and DC power, and at least two connections for offsite power, all of which are separate and independent. There are three divisions of decay heat removal, each with its own emergency AC and DC power source.

This issue is considered resolved for the ABWR.

References

19B.2.52-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.52-2 NUREG-0305, “Technical Report on DC Power Supplies in Nuclear Power Plants”, U.S. NRC, July 1977.

19B.2.52-3 NUREG-0666, “A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants”, U.S. NRC, April 1981.

19B.2.52-4 IEEE Standards 603, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations”, The Institute of Electrical and Electronics Engineers, Inc.

19B.2.52-5 IEEE Standard 308, “Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations”, The Institute of Electrical and Electronic Engineers, Inc.

19B.2.52-6 NRC Letter to All Holders of Operating Licensees, “Resolution of Generic Issue A-30, ‘Adequacy of Safety-Related DC Power Supplies,’ Pursuant to 10 CFR 50.54(f) (Generic Letter 91-06)”, April 29, 1991.

19B.2.52-7 NRC Letter to All Holders of Operating Licenses, “Resolution of Generic Issues 48, ‘LCOs for Class 1E Vital Instrument Buses,’ and 49, ‘Interlocks and LCOs for Class 1E Tie Breakers’ Pursuant to 10 CFR 50.54(f) (Generic Letter 91-11)”, July 18, 1991.

19B.2.53 142: Leakage Through Electrical Isolators in Instrumentation Circuits

Issue

Electronic isolators are used to maintain electrical separation between safety and non-safety-related electrical systems in nuclear power plants, preventing malfunctions in the non-safety systems from degrading performance of safety-related circuits. Isolators are primarily used where signals from Class-1E safety-related systems are transmitted to non-Class 1E control or display equipment.

There are a number of devices which may qualify as electrical isolators in a nuclear power plant, including fiber optic and photo-electric couplers, transformer-modulated isolators, current transformers, amplifiers, circuit breakers, and relays. These isolators are designed and tested to prevent the maximum credible fault applied in the transverse mode on the non-Class 1E side of the isolator from degrading the performance of the safety-related circuits (Class-1E side) below an acceptable level.

This issue was identified by the staff in June 1987 and arose from observations made during Safety Parameter Display System (SPDS) evaluation tests that demonstrated, for electrical transients below the maximum credible level, a relatively high level of noise could pass through certain types of isolation devices and be transmitted to safety-related circuitry. In some cases, the amount of energy that can pass through the isolator may be sufficient to damage or seriously degrade the performance of Class 1E components, while, in other cases, electrically-generated noise on the circuit may cause the isolation device to give a false output.

Due to the fact that there are a great number of each type of isolator in the field, this issue would require the staff to determine the extent to which potentially susceptible isolators are used in nuclear power plants and to identify the systems in which they are used. An NRC bulletin to all licensees to provide input on these questions would be necessary.

Acceptance Criteria

Assuming that the staff determines from the licensee responses to the proposed bulletin that a potential problem exists, a research program consisting of two major objectives would have to be initiated to develop the solution to this issue. The first objective would be to develop test procedures and acceptance criteria for isolators that licensees could use to determine the adequacy of installed isolators. The second objective would involve development of appropriate hardware fixes that could resolve the issue.

Therefore, with a reliable data base the final step in the solution to this issue would be the issuance of a generic letter to licensees with the following guidelines for:

(1) inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems;

(2) repair / replacement of isolators that fail the tests, including description of acceptable hardware fixes to the isolators; and

(3) implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

Issue 142 must meet the requirements of the Licensing Review Bases (LRBs) Criteria on isolators [the LRBs are contained in a letter from T. Murley of NRC to R. Artigas of G.E., dated August 7, 1987 (Reference 19B.2.53-3)].

Resolution

Fiber optic data links are the only type of isolation device used in the ABWR for electrical isolation of logic level and analog signals between protection divisions and from protection divisions to non-safety-related equipment (Subsection 7A.3). Subsection 7A.3 resolves issues regarding the Licensing Review Basis Criteria on isolators.

Maximum credible electrical faults applied at the outputs of isolation devices do not apply to fiber optic systems. The maximum credible fault is cable breakage causing loss of signal transmission. Faults cannot cause propagation of electrical voltages and currents into other electrical circuitry at the transmitting or receiving ends. Conversely, electrical faults originating at the input to the fiber optic transmitter can only damage the local circuitry and cause loss or corruption of data transmission; damaging voltages and currents will not propagate to the receiving end.

Fiber optic isolation devices are expected to have less difficulty than previous isolation devices in complying with all qualification requirements due to their small size, low mass, and simple electronic interfaces. The basic materials and components, except for the fiber optic cable itself, are the same as those used in existing, qualified isolation devices.

When using fiber optic devices as Class 1E isolation devices, only the input side of the transmitting device and output side of the receiving device use electrical power. The low voltage power supplies for these devices use the same power source as the logic that drives the isolating device. For ABWR safety systems, this power is:

(1) Divisional 120 Volt Vital AC (UPS) – For Reactor Protection System (RPS) logic and Main Steam Isolation Valve (MSIV) logic.

(2) 125 Volt Plant DC Power Supply – For ECCS logic and Leak Detection and Isolation System logic.

The isolating devices used for ABWR are similar to the Group 1 types referred to in Reference 19B.2.53-2. They are of the long fiber optic cable design, so transmitting and receiving ends are separated by a significant distance [typically one meter (several feet) to several hundred meters (several hundred feet)]. These types of designs had the best isolating characteristics of the various isolators compared in the NUREG study (Reference 19B.2.53-2).

Typically, the electrical-to-optical interfaces are part of the general logic processing equipment within a channel and do not reside in separate isolator units. The fiber optic interfaces receive the protection from EMI and surge currents designed into the logic equipment (for example, power supply decoupling, shielding, filtering, single-point signal common connection to chassis ground, and chassis ground connection to ground bus). The equipment will undergo EMI and surge testing to the standards identified in the NUREG or equivalent.

The results of the NUREG tests show that the fiber optic type of isolators exhibited no or very little effects from the major fault and lightning surge tests. Only surge and EMI tests applied to the isolator power supplies caused damage to the isolator input side, mainly because of the output and input supplies sharing a common, commercial AC power line. For the ABWR, RPS and ESF functions are supplied from different plant power sources (120 Volt Vital AC and 125 VDC, respectively). The low voltage DC supplies fed from these sources are highly regulated and filtered. Thus, isolator circuits are isolated from most power source transients.

See Subsection 19B.3.2 for COL license information pertaining to testing of isolators.

Therefore, this issue is resolved for the ABWR.

References

19B.2.53-1 Memorandum for B. Morris from B. Sheron (NRC Staff), “Proposed Generic Issue on Leakage Through Electrical Isolators”, June 23, 1987.

19B.2.53-2 NUREG/CR-3453, “Electronic Isolators Used in Safety Systems of U.S. Nuclear Power Plants”, U.S. NRC, March 1986.

19B.2.53-3 A letter from T.E. Murley of NRC to R. Artigas of G.E., “Advanced Boiling Water Reactor Licensing Review Bases”, dated August 7, 1987.

19B.2.54 143: Availability of Chilled Water Systems and Room Cooling

Issue

In recent years, several nuclear power plants have experienced problems with safety system components and control systems that were caused by a partial or total loss of heating, ventilating, and air conditioning (HVAC) systems. Many of these problems exist because of the desire to provide increased fire protection and the need to avoid severe temperature changes in equipment control circuits. Since the Browns Ferry fire, considerable effort has been expended to improve the fire protection of equipment required for safe shutdown. Generally, this improvement has been made by enclosing the affected equipment in small, isolated rooms. The result has been a significant increase in the impact of the loss of room cooling. Plant control and safety have improved with the introduction of electronic integrated circuits; however, these circuits are more susceptible to damage from severe changes in temperature caused by the loss of room cooling.

It is believed that failures of air cooling systems for areas housing key components, such as residual heat removal pumps, switchgear, and diesel generators, could contribute significantly to core-melt probability in certain plants. Because corrective measures are often taken at the affected plants once such failures occur, the impact of these failures on the proper functioning of air cooling systems may not have been considered. Thus, plants with similar inherent deficiencies may not be aware of these problems.

Operability of some safety-related components is dependent upon operation of HVAC and chilled water systems to remove heat from the rooms containing the components. If chilled water and HVAC systems are unavailable to remove heat, the ability of the equipment within the rooms to operate as intended cannot be assured (Reference 19B.2.54-1).

Acceptance Criteria

The impact of loss of room cooling is an important design consideration for the ABWR. Under these circumstances, a key design objective is to ensure that ABWR safety-related equipment will still operate reliably during the period of loss of room cooling. The following criteria will establish an acceptable ABWR design.

(1) An evaluation of the dependencies or non-dependencies of safety-related equipment on HVAC cooling shall be performed. This evaluation will include assessments of room heat load and heatup rates, and establish equipment operating conditions. The capability of the equipment to withstand these conditions without loss of function shall be established.

(2) For equipment found to be significantly dependent on HVAC cooling, an assessment of the HVAC system reliability shall be performed. PRA analyses will be carried out to assess plant risk and determine whether any modifications are necessary.

(3) Corrective design measures shall be identified where necessary to reduce plant risk.

Resolution

The ABWR design uses large chilled-water systems to provide essential environmental cooling, which in turn includes cooling of the solid-state electronic components. The performance of chilled-water systems under varying accident loads and during loss-of-offsite-power events, and their ability to operate after a prolonged station blackout are evaluated. The ABWR design features which address the acceptable criteria include the following:

- As part of actions to mitigate station blackout events, COL applicants are required to perform an analysis to confirm that RCIC room temperature will not exceed equipment design temperature without room cooling for at least 8 hours as stated in Subsection 19.9.9.(3).

- Table 3I-13 defines the most severe thermodynamic environment conditions at which other injection systems (HPCF, RHR/LPFL) equipment are qualified in their designs. These temperature extremes are expected to occur at about 6 hours into the post postulated accident periods without room cooling. If power is lost, the HVAC systems which provide room cooling for these plant areas will not be available for 10 minutes until power is recovered by the combustion turbine generator. The capability of these equipment to withstand the temperature environment which will develop during this 10 minute period is assured for the ABWR design.

- Detailed design specifications for ABWR safety-related equipment will specify the room conditions under which equipment must operate without room cooling. Room heat assessments will be performed to establish environmental conditions for equipment specification (Subsection 3I.3.2.1).

- Potential modifications including procedure changes or hardware changes evaluated through PRA analyses to ensure acceptable plant risk.

Despite the few extreme events which would cause loss of room cooling, the ABWR design incorporates several safety-related HVAC systems which provide room cooling under most circumstances. These systems include:

- R/B Secondary Containment Safety-Related Equipment HVAC—providing 14 fan-coil units for safety-related equipment rooms, including the 3 divisions of ECCS pump rooms (Subsection 9.4.5.2).

- R/B Safety-Related Electrical Equipment HVAC—3 divisions each with 2-100% supply/exhaust fans, 1 air conditioning unit (Subsection 9.4.5.4).

- R/B Safety-Related Diesel Generator HVAC—2 supply fans per division (Subsection 9.4.5.5).

- HVAC Emergency Cooling Water 3 Divisions—provides chilled water to R/B electrical equipment HVAC, C/B HVAC, and C/R habitability HVAC (Subsection 9.2.13).

The reliability and availability of these safety-related HVAC systems will be specified in detailed design to ensure a controlled environment for operation of safety-related equipment.

References

19B.2.54-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1- 15)”, U.S. NRC, April 1993.

19B.2.55 145: ACTIONS TO REDUCE COMMON CAUSE FAILURES

Issue

Issue 145 is concerned that common cause failures can be a major cause of a system failure. The TMI-2 and David Besse incidents were examples of scenarios involving common cause failures (Reference 19B.2.55-1).

Effective maintenance is important to ensure that design assumptions and margins in the original design basis are maintained. In the design of nuclear power plants, an important safety margin is the redundancy of equipment to perform safety functions. This redundancy, however, can be degraded by common cause failures. Therefore, defense against such failures (by root cause analyses and investigations) over the life of the plant is an important part of the licensee's maintenance program.

The NRC has published Regulatory Guide 1.160 (Reference 19B.2.55-3) to implement the maintenance rule, 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants “(Reference 19B.2.55-2).

Acceptance Criteria

The acceptance criteria for the resolution of Issue 145 is to demonstrate compliance with the maintenance rule, 10 CFR 50.65.

Resolution

Compliance with 10 CFR 50.65 will be the responsibility of the COL applicant.

In addition, the ABWR design demonstrates in Chapter 19 its capability to respond to system interactions and common cause failures (Subsection 19.2.3.4).

Actions to reduce common cause failures may fall into the Operational Reliability Assurance Program (O-RAP). The COL applicant will specify the policy and implementation procedures for the O-RAP as described in Subsection 17.3.9.

Therefore, this issue, 145, is resolved for the ABWR.

References

19B.2.55-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.55-2 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", Office of the Federal Register, National Archives and Records Administration.

19B.2.55-3 Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", U.S. NRC.

19B.2.56 151: Reliability of Anticipated Transient Without Scram Recirculation Pump Trip (ATWSRPT) in BWRs

Issue

Issue 151 in NUREG-0933 (Reference 19B.2.56-1) addresses the issue of the reliability of the ATWS RPT in BWRs. Issue 151 specifically identifies a reliability problem with GE's type AKF-25 circuit breaker and trip hardware [actually a type AKF-2-25 breaker, per NRC's IE Notice 87-12 (Reference 19B.2.56-2)].

Acceptance Criteria

The acceptance criterion for the resolution of Issue 151 is the use of reactor recirculation system pump trip hardware or method that is more reliable than the previously used AKF-2-25 breaker hardware or method.

Resolution

The design for the ABWR reactor recirculation system and RPT method and hardware is completely different from the previously designed BWR reactor recirculation systems and RPT trip methods. The design is more diverse and redundantly reliable. Rather than using only two recirculation pumps and the associated single RPT breakers, the ABWR will use ten pumps and multiple pump and RPT trip logic, circuits and hardware. Adjustable speed drive (ASD), recirculation internal pumps (RIPs) are used. The ABWR RPT trip hardware (not yet specifically identified) will be completely different. The ABWR does not use AKF-2-25 circuit breakers in the RPT logic circuits. Instead of using AKF-2-25 breaker switching hardware to provide a RPT, RFC controller switching and ASD gate inverter turn-off circuit hardware provides the RPT [Subsections 7.7.1.3(7) and 7.7.1.3(8)].

Thus, by diversity and redundancy in design, the ABWR addresses and resolves Issue 151.

This issue, 151, may fall into the Operational Reliability Assurance Program (O-RAP). The COL applicant will specify the policy and implementation procedures for O-RAP, as described in Subsection 17.3.9.

References

19B.2.56-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.56-2 IE Information Notice 87-12, “Potential Problems with Metal Clad Circuit Breakers, General Electric Type AKF-2-25”, U.S. NRC, February 13, 1987.

19B.2.57 153: Loss of Essential Service Water in Light-Water Reactors

Issue

The Essential Service Water (ESW) system at a nuclear power plant supplies cooling water to transfer heat from various safety-related and non-safety-related systems and equipment to the ultimate heat sink of the plant. Under Issue 153, the staff will examine all potential causes for ESW system unavailability, except those that are considered to be resolved by implementing the resolutions addressed in (Generic Letter (GL) 89-13 (Reference 19B.2.57-1), such as biofouling, sediment, corrosion, and erosion (Issue 51). The safety concerns of this issue include partial or complete loss of ESW system functions resulting from common causes (such as icing of the intake structure), degradation of the ESW system, design deficiencies, and procedural or maintenance errors. A complete loss of the ESW system could lead to a core-melt accident, posing a significant risk to the public.

The NRC evaluation of this issue has not yet been completed.

Acceptance Criteria

The ESW system is needed in every phase of plant operations and, under accident conditions, supplies adequate cooling water to systems and components that are important to safe shutdown or to mitigate the consequences of the accident. Under normal operating condition, the ESW system provides component and room cooling (mainly via the component cooling water system). During shutdown it also ensures that the residual heat is removed from the reactor core. The ESW system may also supply makeup water to fire protection systems, cooling towers, and treatment systems at a plant.

The design features for the essential service water (ESW) system are summarized as follows:

- Performance Requirements
  - The ESW system will be designed to meet the required heat loads.
  - The ESW system will be provided with two pumps and two heat exchangers per division.

- The plant designer will provide analyses for all potential operating conditions that properly account for uncertainties.

■ System Arrangement

- The ESW system will be divided into approximately three equal-sized divisions.
- A division will be made up of independent piping systems, each with pumps, heat exchangers, strainers, controls and instrumentation, power supplies, and associated equipment required for regulating system flow.

In addition, the ESW design shall address partial or complete loss of ESW system functions resulting from common causes, degradation of the ESW system, design deficiencies, and procedural or maintenance errors. The plant designer should provide an assessment of these potential failure modes and their associated contributions to the core damage frequency and should identify dominant accident sequences.

Resolution

The ABWR Reactor Service Water (RSW) system removes heat from the Reactor Building Cooling Water (RCW) system and transfers that heat to the Ultimate Heat Sink (UHS). The RSW system is provided in three divisions. Each division has two pumps which send cooling water to three RCW heat exchangers. Normally one pump and two heat exchangers are operating in each division. When heat removal requirements increase, the remaining pump and heat exchanger are automatically put into operation. If additional heat removal capacity is needed, some of the non-safety-related cooling loads may be taken out of operation.

In case of failure which disables any of the three RSW divisions, the other two divisions meet plant safety shutdown requirements (Subsection 9.2.15 and Table 9.2-5).

The ABWR RSW system divisions are physically and electrically separated from each other. This reduces the potential effects of common causes. Normally, each division is operating at all times with the capability to put into service the remaining pump and heat exchanger at any time. Margin is provided in pump flow capacity (and in RCW heat exchanger heat removal capacity). Periodic testing of these components will be performed and corrective action taken when needed (Subsections 9.2.11.4 and 9.2.15.1.4).

Several potential causes of RSW system degradation are site dependent. The RSW system is designed to prevent this degradation from occurring. Additionally, the COL applicant will provide the following system design features for those portions of the system which are not in the ABWR standard plant scope:

■ Adequate NPSH for the pumps at low UHS water levels.

- Low point drains and high point vents.
- Prevention of organic fouling (using methods such as trash racks, biocide treatment or thermal backwashing, as required).
- Component material selection suited to site water conditions.
- Protection against flooding, spraying, steam impingement, pipe whip, jet forces, missiles, fire and the effect of failure of any non-Seismic Category I equipment.

If required, recirculation of warm water through the intake structures will be provided to reduce the likelihood that ice will block cooling water flow (Subsections 9.2.5.4 and 9.2.15.2). Also system degradation is minimized by periodic testing and inspection to insure integrity and functional capability (Subsections 9.2.111.4 and 9.2.15.1.6).

The RSW pumps and pump house will be designed by the COL applicant, who will consider and reduce the effects of procedural and maintenance errors.

When the future plant-specific design is prepared, another assessment will be made of potential failure modes and their associated contributions to the core damage frequency and the dominant accident sequences will be identified.

These issues are resolved for the ABWR through the design features of the RSW system, the system design features, and the Operational Reliability Assurance Activities (Subsection 17.3.9) which will be provided by the COL applicant.

References

19B.2.57-1 Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”, July 18, 1989.

19B.2.58 155.1: More Realistic Source Term Assumptions

Issue

Current siting regulations (10 CFR Part 100) require that an accidental fission product release from the core into containment be assumed and that its offsite radiological consequences be evaluated against guideline doses given in Part 100. The postulated source term is derived from TID-14844 (Reference 19B.2.58-1) and is contained in Regulatory Guides 1.3 and 1.4. The regulatory guides specify a release into containment of 100% of the core inventory of noble gases and 50% of the iodine fission products. Half of the iodine is assumed to deposit on interior surfaces assuming instantaneous appearance within containment and that the iodine is predominately in elemental form (I_2).

Use of the TID-14844 source term has not been restricted to evaluation of plant mitigation features and site suitability. Regulatory applications of the source term are broad, including use as the basis for

(1) The post-accident environment for which safety-related equipment should be qualified.

(2) Post-accident habitability requirements for the control room.

(3) Post-accident sampling systems and accessibility.

A substantial amount of information has been developed to update knowledge about LWR severe accidents and behavior of fission products that could be released into containment. Studies have confirmed that although the TID-14844 source term is substantial and that its use has resulted in a high level of plant capability, the present recipe can be substantially improved.

In their staff requirements memorandum (SRM) dated January 25, 1991, the Commission approved the plan proposed by the staff to revise Part 100 to delete the source term and dose calculations and to directly specify site criteria; to issue (in parallel) an interim revision to Part 50 to retain the present source term and dose calculation (but not for siting purposes); to update the TID-14844 source term; and, in a second rule-making phase, to incorporate severe accident and revised source term insights for future plants. In their SRM dated April 11, 1991, the Commission requested the staff to make recommendations on the values of releases into containment (to update TID-14844), to provide a discussion of the status of EPRI's comparable values, and to discuss the use of the updated source term in evaluations of existing and future plants.

Acceptance Criteria

The acceptance criteria for GSI 155.1 is that the plant shall be designed to ensure that the dose commitment to the public in the event of a licensing design basis accident shall be within those limits prescribed by existing regulations based upon the limitations of 10 CFR 100.

Resolution

The ABWR is being designed and analyzed to the existing Regulatory Guides, Standard Review Plans, and General Design Criteria which are based upon TID-14844 (e.g., Regulatory Guide 1.3, Standard Review Plan 15.6.5). The use of revised source terms based upon NUREG-1465 (Reference 19B.2.55-2) is premature for the ABWR based upon the lack of clarification of what is a design basis event under the revised source terms and lacking adequate guidance from the Commission as to acceptable methods and conditions, i.e., revised regulatory guides and standard review plans.

References

19B.2.58-1 DiNunno, J.J. et al, "Calculation of Distance Factors for Power and Test Reactor Sites", Technical Information Document 14844, March 23, 1962.

19B.2.58-2 Soffer, L. et al, "Accident Source Terms for Light-Water Nuclear Power Plants", NUREG-1465, Draft Report for Comment, U.S. NRC, June 1992.

19B.2.59 A-17: Systems Interactions in Nuclear Power Plants

Issue

Unresolved Safety Issue (USI) A-17 in NUREG-0933 (Reference 19B.2.59-1) addresses the concern that inconspicuous or unanticipated interdependencies may exist between systems and may result in a degradation of the predicted capability of safety systems in an accident or transient, in particular from flooding and water intrusion.

In its regulatory analysis in NUREG-1229 (Reference 19B.2.59-4), the NRC concluded that for future plants the existing Standard Review Plans (SRPs) (Reference 19B.2.59-5) in general cover Adverse System Interactions (ASIs) of concern, except for the areas of internal flooding and water intrusion. A flooding event could cause a transient and also disable the equipment needed to mitigate the consequences of the event. NUREG-1174 (Reference 19B.2.59-6) provided guidance in this area and references NRC Information Notices regarding operating plant experiences. The NRC plans to develop an SRP relative to flooding and water intrusion, but otherwise not issue new requirements. In the meantime, the NRC recommends that plant designers keep current on lessons learned from operating experience as reported in LERs, and that the Probabilistic Risk Assessment (PRA) required for a future plant be also considered as a tool to help uncover flooding and water intrusion ASIs.

Acceptance Criteria

The acceptance criterion for the resolution of USI A-17 is that attention shall be paid in the detailed plant design to detecting and minimizing the potential for ASIs. Future plants must encompass the full spectrum of potential system interactions from operating plant experience and new design evaluations. The objective is to preserve the means for reaching and maintaining a safe hot shutdown.

Resolution

The A-17 Systems Interaction issues/concerns have been long recognized as being critically important to safe and reliable plant design, operation and maintenance activities. A significant amount of sometimes splintered and sometimes coordinated efforts have taken place during the last ten years. These include:

(1) NRC Research Studies—A series of NUREG reports were directed at achieving a systematic way to address system interactions (e.g. Diagraphic approaches).

(2) EPRI Research Studies—A series of reports evaluated plant system dependencies and interactions, common mode or cause failure methodologies, operator error profiles, etc.

(3) Office for Analysis and Evaluation of Operational Data (AEOD) Studies—On a continuing basis, the AEOD has searched for common cause high frequency, high consequence, etc. events aspects. They have also addressed man-machine interactions. Their review of Licensing Event Report Evaluation (LERE) and event inspection evaluations have provided the data for their findings and conclusions.

(4) Nuclear Safety Analysis Corporation (NSAC) and Institute of Nuclear Power Operations (INPO) Operating Experience Feedback and Generic Safety Issue Tracking—For over 10 years these organizations have systematically and comprehensively evaluated LERs, significant events, maintenance and operator anomalies and other events of interest. Special emphasis is placed on operator error-plant anomalies aspects by INPO.

(5) NSSS—Vendors have tracked equipment performance and failure aspects for over 25 years. System interactions are an integral part of the root cause analysis in these evaluations.

(6) PRA Evaluations—A full spectrum of PRA analyses now exist for each plant. These analyses can be macro rather than micro in assessing ASIs. However, the recent expanded use and application of PRA for SSFIs, 10 CFR 50.59s, etc. demand detailed inquiries of SI efforts.

(7) NRC-Staff/Region Information—A wide spectrum of operating experience feedback is available in Generic Letters (GLs), Information Notices (INs), etc. In summary, a lot of information is available, distributed and utilized by the designer, the analyst, the safety evaluator and reviewer, and the plant operator. This information supplements and compliments the previous or original design basis insights. Most unique or important operating experience feedback is associated with system interaction elements.

Plant system designers have found a number of critically important prevention—mitigation—accommodation ASIs avoidance attributes. Several attribute groups are mentioned below for example purposes.

(1) Separation Criteria—Physical, Electrical, Mechanical, Environmental

(2) Kind/Type Criteria—Redundant, Diverse, Reliable and Available

(3) Failure Aspects—Fail As-Is, Fail Safe, Fail Recoverable, Single Active/Passive Failures, Common-Cause/Mode Failures, Fail Alarm

(4) Protective Action—Auto vs. Manual, Auto Reset, No Operator Involvement (Errors of Omission), Limited Operator Involvement (Errors of Commission)

(5) Maintenance Aspects—Limited Replacement or Repair and Replacement (R&R), Staggered Testing / Calibration / Inspection, One-on-One Signoff, Diverse Crews, No At-Power Maintenance

(6) Diverse Phenomena and Responses—Built-in Inherencies, Gravity-Driven Responses, Extended / Enhanced Capability, Time-Independent Responses, Non-Electro / Mechanical Response, Self-Powered Capabilities

The designer now has more evaluation tools to work with relative to ASIs. Detailed Failure Modes and Effects Analyses (FMEAs), Fault Tree Analyses (FTAs), Event Tree Analyses (ETAs), System Dependency Charts, Common-Mode Failures (CMFs), Common-Cause Failures (CCFs), Common-Mode Probabilistic Failures (CMPFs), etc. are but a few of the new tools available to use in ASI evaluations. Simulator and Emergency Operating Procedures (EOPs) audit critiques provide insights into man-machine aspects (e.g. operator error patterns, recovery alternatives, etc.). More precise root cause analysis techniques are available and demanded by regulatory requirements [e.g. Kepner Tregoe (KT), Event Sequence Plots, etc.]. PRA and operating experience feedback give the designer a feel for which component, sequence, or operator action is critical, sensitive, difficult, time related, etc.

The ABWR is very much like prior BWRs. Many BWR plant features are designed into the plant explicitly to avoid unwanted, unacceptable or unknown ASIs. The major items include:

- Utilize an Operating Experience Proven Design
- Multiple Fission Product Barriers
- Inherent Shutdown Features and Mechanisms
- Redundant and Diverse Containment Features
- Redundant and Diverse ESF Network
- Redundant and Diverse I&C Protection Network
- Redundant and Diverse Safe Shutdown Capabilities

ABWR unique features (in addition to other plants) to prevent, mitigate and accommodate ASIs include:

- More Redundant, Diverse and Independent Decay Heat Removal Systems (DHRSs) Capabilities (Subsections 5.4.7 and 6.3.2.2.4)
- More Redundant, Diverse and Independent RPV and Containment Make-up and Cooling Capabilities (Subsections 6.3.1.1.1 to 6.3.1.1.4)
- More Redundant, Diverse and Independent Power Sources (Subsections 8.1.2.1 and 8.1.2.2)
- More Redundant, Diverse and Independent Operator Action Capabilities (Subsection 18.4.2)
- More Redundant, Diverse and Independent and Fault-Tolerant I&C Protection Nature (Appendix 7C)
- More Secure and Protected ESF Housing from Fire and Flood Aspects (Appendices 19M and 19R, and Subsection 9.5.1)
- More Secure and yet accessible ESF Housing for Inspection and Maintenance (Subsection 19K.11.1)

The ABWR design utilizes most of the ASI avoidance attributes described and cited above. The ABWR has been extensively evaluated both deterministically and probabilistically. This design is based on over 25 years of successful operating experience. The plant designers had access to the extensive lessons learned and feedback over the last ten years. The design has been reviewed and evaluated over the last 5 years by the world's foremost safety experts (ACRS, NRC Staff, GE Staff, Utility Staffs, DOE and Consultants) for a spectrum (both broad and deep) of inquiry. The plant reflects proven technology and accepted design standards and requirements. The plant design addresses system interactions at three levels—prevention, mitigation and accommodation.

Therefore, this issue is resolved for the ABWR.

References

19B.2.59-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U. S. NRC, April 1993.

19B.2.59-2 NUREG/CR-3922, “Survey and Evaluation of System Interaction Events and Sources”, U. S. NRC, January 1985.

19B.2.59-3 NUREG/CR-4261, “Assessment of System Interaction Experience at Nuclear Power Plants”, U. S. NRC, June 1986.

19B.2.59-4 NUREG-1299, “Regulatory Analysis for Resolution of USI A-17”, U. S. NRC, August 1989.

19B.2.59-5 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U. S. NRC.

19B.2.59-6 NUREG-1174, “Evaluation of Systems Interaction in Nuclear Power Plants – Technical Findings Related to Unresolved Safety Issue A-17”, U. S. NRC, May 1989.

19B.2.60 A-29: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Issue

Issue A-29 in NUREG-0933 (Reference 19B.2.60-1) addresses the susceptibility of nuclear power plants to industrial sabotage, the resulting risk to plant safety, and the countermeasures to assure an acceptable level of protection.

Consideration should be given to sabotage during the design phase of the plant. The goal would be to achieve an acceptable level of protection of a plant to industrial sabotage by emphasizing design features which reduce the likelihood of the plant incurring damage from industrial sabotage, both internal and external.

Acceptance Criteria

The acceptance criteria for the resolution of Issue A-29 is that plants shall be designed to be resistant to the effects of internal and external sabotage through prevention, deterrence and mitigation.

Specifically, plant safety-related systems and components required for the safe operation and shutdown of the plant shall be designed for protection against and mitigation of sabotage.

Resolution

The ABWR design will mitigate the acts of sabotage through physical separations in the plant arrangement of independent, engineered safety systems, and the design and location of barriers to resist threats (refer to Sections 9.5, 3.4, and 3.6).

Appendix 19C describes and analyzes the ABWR design features that reduce the risk from postulated insider sabotage.

In addition, the ABWR design includes various methods of access control to prevent intrusion as well as provide detection during a breach of the system. Specifically, Subsection 13.6.3 describes the physical protection systems and controls for compliance with 10 CFR 73.55 (Reference 19B.2.60-2).

The design of the decay heat removal system provides an inherent resistance to sabotage by its protection against tornado missiles, winds, earthquakes and floods.

GE has performed an analysis of the ABWR design for vulnerability to sabotage as discussed in the DFSER, and recommends that the COL applicant should perform a sabotage vulnerability analysis per (Revision 1 of ABWR Utility Rights Document Vol. II, Chapter 9, Section 5221) to optimize system designs and compatibility of plant arrangement and system design from insider and outsider threats and, that before fuel loading, the COL applicant should confirm conformance to the ABWR design features identified in Subsection 19C.4 enhancing resistance of the ABWR to sabotage. Further, during cold shutdown, that the provisions of (SECY-91-029) dealing with procedures for access will be in effect.

In summary, the ABWR design is highly resistant to sabotage, because of the features described which protect against internal and external sabotage.

Therefore, this issue is resolved for the ABWR.

References

19B.2.60-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.60-2 10 CFR 73.55, “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage”, Office of the Federal Register, National Archives Records Administration.

19B.2.61 B-5: Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

Issue

Generic Safety Issue (GSI) B-5 in NUREG-0933 (Reference 19B.2.61-1) identifies two concerns relating to containment design. First that sufficient information is not available to predict the behavior of two-way reinforced concrete slabs; and second, that the structural design of a steel containment vessel subjected to unsymmetrical dynamic loadings may be governed by the instability of the shell.

(1) Ductility of Two-Way Slabs and Shells

The first concern was originally identified in NUREG-0471 (Reference 19B.2.61-2) and involved concern over the lack of information related to the behavior of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension (resulting from in-plane loads), flexure, and shear. If structures (concrete slabs) were to fail (floor collapse or wall collapse) due to loading caused by a loss-of-coolant-accident (LOCA) or high-energy-line break (HELB), there would be a possibility that other portions of the reactor coolant system or safety-related systems could be damaged. Such loads would be caused by very concentrated high-energy sources causing direct impact on the structures of concern. The damage could lead to an accident sequence resulting in the release of radioactivity to the environment.

Because of NRC and industry concern, the American Concrete Institute addressed these dynamic loads by establishing the methodology identified in the Appendix C Commentary to ACI 349 (Reference 19B.2.61-3).

(2) Buckling Behavior of Steel Containments

The second concern, also identified in Reference 19B.2.61-2, involves concern over the lack of a uniform, well-defined approach for design evaluation of steel containments. The structural design of a steel containment vessel subjected to unsymmetrical dynamic pressure loadings may be governed by the instability of the shell. For this type of loading, the current design verification methods, analytical techniques, and the acceptance criteria may not be as comprehensive as they could be. Section III of the ASME Code (Reference 19B.2.61-4) does not provide detailed guidance on the treatment of buckling of steel containment vessels for such loading conditions.

Moreover, this Code does not address the asymmetrical nature of the containment shell due to the presence of equipment hatch openings and other penetrations. Regulatory Guide 1.57 recommends a minimum factor of safety of two against buckling for the worst loading condition provided a detailed rigorous analysis, considering in-elastic behavior, is performed.

On the other hand, the 1977 Summer Addendum of the ASME Code permits three alternate methods, but requires a factor of safety between 2 and 3 against buckling, depending upon applicable service limits.

However, NUREG-0933 states that the issue was resolved and no new requirements were established.

Acceptance Criteria

The acceptance criteria for part 1 of this issue is that the design of safety-related concrete structures shall meet the ductility requirements of ACI 349, as supplemented by RG 1.142 (Reference 19B.2.61-5).

The acceptance criteria for part 2 of this issue is that the buckling design of steel portions of containment vessels (i.e., ABWR reactor closure head) shall meet provisions of NE-3222 or code case N-284 of the ASME code.

Resolution

The design of ABWR safety-related concrete structures (other than containment) is based on ACI 349 as supplemented by RG 1.142.

Part 1 of this issue is thus resolved for the ABWR.

The ABWR containment is a reinforced concrete structure and it is designed according to ASME Code, Section III, Division 2, Subsection CC. The steel components (reactor closure head not backed by concrete) of the containment vessel are designed in accordance with to ASME Code, Section III, Subsection NE, including the buckling provisions as stated in the acceptance criteria above.

Part 2 of this issue is, thus, resolved for the ABWR.

References

19B.2.61-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.61-2 NUREG-0471, “Generic Task Problem Descriptions (Categories B, C, and D)”, U.S. NRC, June 1978.

19B.2.61-3 ACI 349, “Code Requirements for Nuclear Safety Related Structures”, American Concrete Institute.

19B.2.61-4 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear), Division I, Subsection NE, American Society of Mechanical Engineers.

19B.2.61-5 Regulatory Guide 1.142, “Safety Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)”, U.S. NRC.

19B.2.61.1 C-8: Main Steam Line Leakage Control Systems

Issue

Dose calculations indicated that operation of the main steam isolation valve leakage control system (MSIVLCS) required for some BWRs could result in higher offsite accident doses than if the system were not used and the integrity of the steam lines and condenser was maintained. The calculations for accidents with a TID-14844 (Reference 19B.2.61.1-2) source indicated a potential increase in offsite releases of iodine by two to three orders of magnitude for proper operation of a MSIVLCS, when compared to the calculations of releases assuming the steam system intact and MSIV leakage is eventually released through the condenser. Therefore, use of the MSIVLCS recommended in Regulatory Guide 1.96 (Reference 19B.2.61.1-3) could increase the overall risk to the public. After an extensive evaluation of alternative solutions, it was decided that Regulatory Guide 1.96 was acceptable, and the issue was resolved with no new requirements (Reference 19B.2.61.1-1).

Acceptance Criteria

This issue was resolved with no new requirements. However, the requirements of GDC 54 and the guidance of RG 1.96 are applicable to the ABWR. RG 1.96 describes a method of implementing GDC 54 with regard to design of a leakage control system for the MSIVs of BWRs to ensure that total radiological effects do not exceed guidelines of 10 CFR 100 in the event of a postulated design basis LOCA. RG 1.96 states that the isolation function of the MSIVs should be supplemented by a leakage control system (LCS), or if an alternative method is used it must be approved by the NRC staff. RG 1.96 indicates that a leakage control system would not be required if the main steam line leakage path can be relied on to remain intact and capable of providing significant dose reduction factors for postulated accident conditions.

Resolution

The ABWR main steam line leakage path is designed to remain intact and capable of providing significant dose reduction factors for postulated accident conditions. The design of the ABWR main steam leakage path is described in Subsection 3.2.5.3. The main steam lines and all branch lines 65A (2-1/2 inches) in diameter and larger are designed to withstand the safe shutdown earthquake; the main steam and bypass lines at the turbine that are not safety-related, are analyzed to demonstrate their structural integrity under the safe shutdown earthquake loading. The condenser anchorage is seismically analyzed to demonstrate that it does not fail. The radiation results of the main steamline leakage analysis (Subsection 15.6.5) are given in Tables 15.6-13 and 15.6-14, for offsite and control room dose evaluations, respectively, and are within current regulatory guidelines. The COL applicant will recalculate iodine removal credit on the basis of the specific design characteristics of main steamlines, drains, and main condenser, as outlined in Subsections 15.6.5.5.1.2 and 15.6.5.5.1.3. The ABWR alternative design approach has been approved by the NRC staff.

Therefore, upon approval of the alternative design approach, this issue is resolved.

References

19B.2.61.1-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.61.1-2 TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites”, U.S. Atomic Energy Commission, March 23, 1962.

19B.2.61.1-3 Regulatory Guide 1.96, “Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants”, U.S. NRC.

19B.2.62 29: Bolting Degradation or Failure in Nuclear Power Plants

Issue

Issue 29 in NUREG-0933 (Reference 19B.2.62-1) addresses bolting degradation within safety-related components and support structures and its impact on the integrity of the reactor coolant pressure boundary.

The most crucial bolting applications are those constituting an integral part of the primary pressure boundary such as closure studs and bolts on reactor vessels and reactor coolant pumps. Degradation of these bolts or studs could result in the loss of reactor coolant. Other bolting applications such as component support and embedment anchor bolts or studs are essential for withstanding transient loads created during abnormal or accident conditions.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 29 are that proven bolting designs, materials, and fabrication techniques shall be employed. Reactor coolant pressure boundary (RCPB) bolting, component support bolts and embedment anchor bolts or studs shall meet the requirements of ASME Code, Section III; NUREG-1339; EPRI NP-5769; and GL 91-17 (References 19B.2.62-2, 19B.2.62-3, 19B.2.62-4 and 19B.2.62-5, respectively). Also, for RCPB bolting the owner-operator shall use established industry practice in developing maintenance, assembly, and disassembly procedures. Furthermore, for RCPB and its support bolting, inservice inspection shall meet the requirements of ASME Code, Section XI (Reference 19B.2.62-2).

Resolution

Bolting degradation of RCPB bolts is primarily an operating plant issue since most of the degraded bolts have resulted from poor maintenance practices. Bolting integrity is assured by the designer through the initial specification of proven bolting materials, installation requirements, and by the owner-operator through the use of acceptable maintenance and inspection practices.

For the ABWR design, only proven materials for the specific application and environment are employed, having been selected after evaluation of the potential for corrosion wastage and intergranular stress corrosion cracking. Also, the RCPB components and their integral bolts, including the reactor vessel, reactor coolant pumps and piping are fabricated, tested, and installed in accordance with ASME Code, Sections III and XI; and NUREG-1339, EPRI NP-5769 and GL 91-17 (References 19B.2.62-3, 19B.2.62-4 and 19B.2.62-5, respectively). Finally, the owner-operator must perform periodic inservice inspection in accordance with ASME Code, Section XI. In addition, for critical pressure boundary applications such as the reactor vessel head closure, redundant seals and leak monitoring further assure the integrity of the RCPB.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.62-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.62-2 “ASME Boiler and Pressure Vessel Code”, Section III (Nuclear) and Section XI, American Society of Mechanical Engineers.

19B.2.62-3 NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants”, U.S. NRC, June 1990.

19B.2.62-4 EPRI NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants, Electric Power Research Institute”, April 1988.

19B.2.62-5 Generic Letter 91-17, “Generic Safety Issue 29, ‘Bolting Degradation or Failure in Nuclear Power Plants ‘”, U.S. NRC, October 17, 1991.

19B.2.63 82: Beyond Design Basis Accidents In Spent-Fuel Pools

Issue

Issue 82 in NUREG-0933 (Reference 19B.2.63-1) addresses the potential for a beyond-design-basis accident in which the water is drained out of the spent-fuel pool. In such an event the discharged fuel from the last two refuelings may have sufficient decay heat to melt, ignite the zircaloy cladding and release fission products to the atmosphere.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 82 is that the design of the spent-fuel pool, storage racks, fuel pool cooling and cleanup system and the load handling equipment in the spent-fuel pool area shall meet applicable current requirements, i.e., the guidance of the Standard Review Plan (SRP) Sections 9.1.2 – 9.1.5 (Reference 19B.2.63-2) and Regulatory Guide 1.13 (Reference 19B.2.63-3).

Resolution

The ABWR design includes a spent-fuel storage facility, a fuel pool cooling and cleanup system and a fuel handling system that meet the intent of Regulatory Guide 1.13 and SRP 9.1.2 - 9.1.5 as described in Subsection 9.1. A brief summary of the design features relating to the Regulatory Guide and the SRP follows.

- The spent-fuel pool and storage racks are Seismic Category I structures. The spent-fuel pool is in the Reactor Building which is also Seismic Category I. There are no non-seismic systems, high or moderate energy pipes, or rotating machinery located in the vicinity of the spent-fuel pool or cask loading area on the refueling floor.

- The Reactor Building protects the fuel and spent-fuel pool from tornadic winds and the missiles generated by these winds. The Reactor Building also prevents turbine missiles from effecting the spent-fuel pool.

- Interlocks prevent the movement of heavy loads over the spent-fuel pool. Heavy loads, defined such that if inadvertent operations or equipment malfunction either separately or in combination, could cause:

  (1) a release of radioactivity,

  (2) a criticality accident, or

  (3) the inability to cool fuel within the spent-fuel pool.

- The Standby Gas Treatment System limits the potential release of radioactive iodine and other radioactive materials from the Reactor Building which encloses the spent-fuel pool.

- The travel of the Reactor Building crane which handles heavy loads, including the fuel casks, is limited by interlocks to preclude movement over the spent-fuel storage pool.

- No inlets, outlets or drains are provided that might permit the pool to be drained below a safe shielding level. Lines extending below this level are equipped with siphon, breakers, check valves, or other suitable devices to prevent inadvertent pool drainage.

- A level switch is provided in the spent-fuel pool to alarm locally and in the control room on either high or low level. The Fuel Handling Area Ventilation Exhaust Radiation system monitors the offgas radiation level in the fuel handling area ventilation exhaust duct. A high-high radiation trip results in the initiation of the Standby Gas Treatment System and in the isolation of the secondary containment (including closure of the containment purge and vent valves, and closure of the Reactor Building ventilation exhaust isolation valves).

- The Fuel Pool Cooling and Cleanup (FPC) System provides the primary means of maintaining the water level in the spent-fuel pool utilizing a connection to the Condensate System. The Suppression Pool Cleanup (SPCU) System can be used as a backup. Both the FPC and SPCU Systems are Seismic Category I designs. Additionally, connections from the RHR System to the FPC System provide a Seismic Category I, safety-related makeup capability to the spent-fuel pool. The FPC System from the RHR connections to the spent-fuel pool are Seismic Category I and safety related. The RHR system connections will be protected from the effects of pipe whip, internal flooding, internally generated missiles, and the effects of a moderate pipe rupture. Furthermore, fire water can be used to supply water from the fire protection system to the spent-fuel pool via the RHR system or fire hoses.

Since the acceptance criteria are met for the spent-fuel storage facility, this issue is resolved for the ABWR.

References

19B.2.63-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)," U.S. NRC, April 1993.

19B.2.63-2 NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition."

19B.2.63-3 Regulatory Guide 1.13, "Design Objective for Light-Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

19B.2.64 113: Dynamic Qualification Testing of large Bore Hydraulic Snubbers

Issue

Issue 113 in NUREG-0933 (Reference 19B.2.64-1) addresses the need for requirements for dynamic qualification testing of large bore hydraulic snubbers [>2.224E5 Newtons (>50 kips) load rating]. Qualification tests of large bore hydraulic snubbers typically utilize a shutoff valve in place of the snubber control valve. To assure operability of the snubber control valves when subjected to dynamic loads, testing should be performed to determine the operational characteristics of the snubber control valve.

Acceptance Criteria

The acceptance criteria for the resolution of Issue 113 for the ABWR design are the performance of dynamic tests in accordance with Subsection 3.9.3.4.1 (3). The dynamic load tests identified specifically for large bore hydraulic snubbers (LBHS) are to be performed in addition to the dynamic tests required for mechanical and hydraulic snubbers.

Resolution

Mechanical and hydraulic snubbers will only be used for piping systems when dynamic supports are required at locations where large thermal displacements prohibit the use of rigid supports.

Large bore hydraulic snubbers (LBHS) will only be used as piping restraints. Mechanical and hydraulic snubbers including LBHS are tested to insure that they can perform as required during seismic and other dynamic loading events. These tests are described in Subsection 3.9.3.4.1(3). Additional dynamic cyclic load tests are required for LBHS to assure operability of the snubber control valves when subjected to dynamic loads. This requirement is specified in Subsection 3.9.3.4.1(3)(C).

The acceptance criteria for this issue are met, therefore, the issue is resolved for the ABWR design.

References

19B.2.64-1 NUREG-0933, "A Prioritization of Generic Safety Issues (with Supplements 1-15)", U.S. NRC, April 1993.

19B.2.65 I.D.5(2) Plant Status and Postaccident Monitoring

Issue

The issue addressed is documented in TMI Action Plan, and focuses on the need to improve the ability of nuclear power plant control room operators to prevent, diagnose, and properly respond to accidents by using the full information provided to them (Reference 19B.2.65-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue I.D.5(2) is that plant status and post-accident monitoring is in compliance with Regulatory Guide (RG) 1.97 (Reference 19B.2.65-2).

Resolution

The ABWR design of its information systems (important to safety) provide information for manual initiation and control of safety systems. These systems provide indication to the control room that plant safety functions are being accomplished and provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents. It is designed to perform as described in Subsection 7.5 and is in compliance with RG 1.97 (Reference 19B.2.65-2).

Therefore, this issue, I.D.5(2), is resolved for the ABWR.

References

19B.2.65-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980.

19B.2.65-2 Regulatory Guide 1.97, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs condition During and Following an Accident”, U.S. NRC.

19B.2.66 I.D.5(3) On-Line Reactor Surveillance System

Issue

NUREG-0933 (Reference 19B.2.66-1), Generic Safety Issue (GSI) Item I.D.5(3) addresses the TMI issue of an “On-Line Reactor Surveillance System.” This issue specifically concerns detecting abnormal reactor core internal's noise associated with on-line reactor operation, e.g., detecting loose internal reactor parts.

Acceptance Criteria

The acceptance criteria for the resolution of GSI-I.D.5(3) is that, based on the on-going generic BWR programs, it is concluded that the technical resolution of this issue has been identified (see Reference 19B.2.66-1).

Resolution

The primary cause of core vibration is high and turbulent reactor water recirculation flow. To detect such vibration, the ABWR design incorporates a reactor vessel loose parts monitoring system (LPMS), as described in Subsection 4.4.3 that complies with NRC's Regulatory Guide 1.133 (Reference 19B.2.66-2) requirements. In addition, with the redesign for the ABWR reactor core internals, i.e., core fuel supports, fuel boxes and instrument channel's etc., problem reoccurrence has essentially been eliminated. The LPMS and other ABWR instrumentation systems will continue to monitor various reactor operational parameters, e.g., reactor core vibration, neutron flux patterns and stability; and thus, any problem recurrence would be quickly detected prior to any adverse core effects which might result. Furthermore, when compared to most other BWR's, the ABWR design incorporates ten small, rather than two large, reactor water recirculation pumps and these are in vessel type pumps. This arrangement is designed to more uniformly distribute core flow, and thus, reduce any flow turbulence that might lead to the loosening of reactor internal core parts.

Therefore, this issue, I.D.5(3), is resolved for the ABWR.

References

19B.2.66-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.66-2 Regulatory Guide 1.133, “Loose Parts Detection Program for the Primary System of Light-Water Cooled Reactors”, U.S. NRC.

19B.2.67 I.G.2: Scope of Test Program

Issue

The major thrust of TMI Action Plan I.G is to use the preoperational and startup test programs as a training exercise for the operating crews.

In contrast to this, Item I.G.2 calls for a more comprehensive test program to search for anomalies in a plant’s response to a transient. The safety significance of this issue lies in the early discovery of anomalies of unanticipated plant behavior. When a plant responds to a transient in an anomalous or unanticipated manner, the result may be an accident caused directly by the new phenomena, or by the surprise or confusion on the part of the operators (Reference 19B.2.67-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue I.G.2 is compliance with Standard Review Plan (SRP) Chapter 14 (Reference 19B.2.67-2), and Regulatory Guide 1.68 (Reference 19B.2.67-3).

Resolution

The ABWR will have a test program to evaluate and demonstrate, to the extent possible, that the operating group is knowledgeable about the plant and procedures and fully prepared to operate the facility in a safe manner as described in Chapter 14. Subsection 14.2.7 identifies Regulatory Guide 1.68 and other applicable regulatory guides used in the development of test programs.

Therefore, this issue is resolved for ABWR.

References

19B.2.67-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.67-2 NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition”, U.S. NRC.

19B.2.67-3 Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants”, U.S. NRC.

19B.2.68 II.E.6.1: Test Adequacy Study

Issue

The purpose of this TMI Action Plan (Reference 19B.2.68-1) is to establish the adequacy of current requirements for safety-related valve testing. It recommends a study which would result in recommendations for alternate means of verifying performance requirements.

Acceptance Criteria

The acceptance criteria for the resolution of Issue II.E.6.1 include the following four parts:

(1) Investigation of in-situ testing of pressure isolation valves (PIVs) under Issue 105

(2) In-situ testing and surveillance of check valves will be performed to ensure their adequacy under design basis and required operating conditions

(3) Compliance with the thermal overload protection provisions of Regulatory Guide 1.106 (Reference 19B.2.68-4) for motor-operated valves (MOVs)

(4) Compliance with the recommendations of GL 89-10 (Reference 19B.2.68-2) for in-situ testing of motor-operated valves

Resolution

In-situ testing of PIVs, including check valves, is addressed in the resolution of Issue 105 (Subsection 19B.2.45). The COL applicant is to perform periodic surveillance and leak rate testing of PIVs per the ABWR Technical Specifications as part of the IST program.

With regard to in-situ testing and surveillance of safety-related check valves, Subsection 3.9.6.2.1 requires the COL applicant to perform in-situ full-flow testing, in addition to the ASME Code, Section XI, in-service testing requirements. Additionally, the COL applicant will use advanced non-intrusive techniques to assess valve degradation and performance. The COL applicant will also develop a program which establishes the frequency and extent of disassembly and inspection of check valves.

As indicated in Table 1.8-20, the ABWR will comply with the guidance of Regulatory Guide 1.106 (Reference 19B.2.68-4) regarding the application of thermal overload protection devices that are integral with the motor starter for electric motors on MOVs.

The COL applicant will need to address the concerns and issues identified in GL 89-10 (Reference 19B.2.68-2) for MOVs prior to plant startup (Subsection 3.9.6.2.2).

Valve performance is critical to the successful functioning of a large number of the plant safety systems. In-service testing of safety-related valves will be performed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1, 6 and 10, as described in Subsection 3.9.6. Subsection 3.9.6 lists the in-service testing parameters and frequencies for the safety-related valves. The reason for each code defined testing exception or justification for each code exemption request is noted in the description of the affected valve. Valves having a containment isolation function are also noted in the listing.

Details of the in-service testing program, including test schedules and frequencies, will be reported in the in-service inspection and testing plan which will be provided by the applicant referencing the ABWR design. The plan will integrate the applicable test requirements for safety-related valves including those listed in the technical specifications (Chapter 16) and the containment isolation system. This plan will include baseline pre-service testing to support the periodic in-service testing of the components. Depending on the test results, the plan will provide a commitment to disassemble and inspect the safety-related valves when limits of the OM Code are exceeded. The primary elements of this plan, including the requirements of Generic Letter 89-10 (Reference 19B.2.68-2) for motor operated valves, are delineated in Subsection 3.9.6.

Therefore, this issue is resolved for the ABWR.

References

19B.2.68-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980, Revision 1, August 1980.

19B.2.68-2 NRC Letter to All Licensees of Operating Power Plants and Holders of Construction Permits for Nuclear Power Plants, “Safety-Related Motor-Operated Valve Testing and Surveillance (Generic Letter No. 89-10) – 10 CFR 50.54(f)”, June 28, 1989.

19B.2.68-3 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.68-4 Regulatory Guide 1.106, “Thermal Overload Protection for Electric Motors on Motor-Operated Valves.”

19B.2.69 II.K.1(13) Propose Technical Specification Changes Reflecting Implementation of all Bulletin Items

Issue

Issue II.K (Measures to Mitigate Small - Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents) has the objective of improving the capability to mitigate the consequences of small-break accidents and loss-of-feedwater events. Nine Inspection and Enforcement (IE) bulletins were issued to operating plants with twenty-eight requirements [Task II.K and Table C.1, NUREG-0660 (Reference 19B.2.69-1)] for reviews of plant design and operation.

Issue II K.1(13) is one of the twenty-eight requirements of the overall issue. It is directed at implementing the Technical Specification changes that would be required from other changes made to respond to all IE bulletin items.

Acceptance Criteria

The acceptance criteria for the resolution of Issue II.K.1(13) is compliance with 10 CFR 50.36, Technical Specifications (Reference 19B.2.69-2), and the interim “Proposed Policy Statement on Technical Specification Improvements for Nuclear Power” (Reference 19B.2.69-3).

Resolution

The ABWR demonstrates in Chapter 15, Accident Analysis, the capability to respond to the full spectrum of line breaks and loss-of-feedwater accidents without loss of containment or significant core damage. Chapter 16 sets forth the restrictions on plant operation required to control the transients and abnormal events of Chapter 15 to ensure conformance with the NRC rules identified in the Acceptance Criteria for this issue.

Accordingly, the analyses of Chapter 15 and the operational conditions and limitations of Chapter 16 ensure that the ABWR fulfills the intent of Issue II.K.1(13).

References

19B.2.69-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980.

19B.2.69-2 10 CFR 50.36, “Technical Specifications”, Office of the Federal Register, National Archives and Records Administration

19B.2.69-3 Federal Register Notice 52FR3788, “Proposed Policy Statement on Technical Specification Improvements for Nuclear Power”, February 1987.

19B.2.70 II.K.3(11): Control Use of PORV Supplied by Control Components, Inc. Until Further Revision Complete

Issue

Issue II.K, “Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents,” has the objective of improving the capability to mitigate the consequences of small-break accidents and loss-of-feedwater events. For this issue, the Bulletins and Orders (B&O) Task Force conducted generic reviews of systems reliability, emergency procedures, and operator training as documented in NUREG-0626 (Reference 19B.2.70-2) and the NRC issued some 32 recommendations for the BWR [Task II.K and Table C.3, NUREG-0660 (Reference 19B.2.70-1)] for reviews of plant design and operations.

Issue II.K.3(11) is one of the 32 BWR recommendations of the Bulletins and Orders Task Force. It requires all plants to justify the use of PORVs (Power Operated Relief Valves) supplied by Control Components, Inc. that had failed during testing.

Acceptance Criteria

The acceptance criteria for the resolution of Issue II.K.3(11) is compliance with 10 CFR 50. Appendix A, General Design Criterion 15, “Reactor Coolant System Design”, and the applicable codes and standards governing safety/relief valves (SRV).

Resolution

The ABWR demonstrates in Chapter 15 the capability to respond to the full spectrum of line breaks and loss-of-feedwater accidents without loss of containment or significant core damage.

Section 5.2 describes the overpressure protection provided by the SRVs performing an overpressure relief valve function, an overpressure safety valve function, or an Automatic Depressurization system (ADS) function.

The SRV for the ABWR is not a Power Operated Relief Valve by Control Components, Inc. It is a spring-loaded safety valve for the safety valve function with a pneumatic cylinder/piston for power operation in the ADS and relief function.

Subsection 3.9.3.2.4.2 describes the qualification by type test of the SRVs to IEEE 344, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"(Reference 19B.2.70-3), for operability during a dynamic event.

Therefore, this issue, II.K.3(11), is resolved for the ABWR.

References

19B.2.70-1 NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident", U.S. NRC, May 1980.

19B.2.70-2 NUREG-0626, "Staff Report on the Generic Assessment of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Boiling Water Reactors Designed by the General Electric Company", U.S. NRC, January 1980.

19B.2.70-3 IEEE Standard 344, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

19B.2.71 II.K.3(27): Provide Common Reference Level For Vessel Instrumentation

Issue

Issue II.K., "Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents," has the objective to perform systems reliability and to effect changes in emergency operating procedures and operator training to improve the capability to mitigate such accidents.

The concern in Issue II.K.3(27) is that different reference points of the various reactor vessel water level instruments could cause operator confusion. Either the bottom of the vessel or the active fuel were considered to be reasonable reference points (Reference 19B.2.71-1).

Acceptance Criteria

The acceptance criteria for the resolution of Issue II.K.3(27) is to confirm that the ABWR design has a common zero reference for all water level indications.

Resolution

The resolution of this issue, II.K.3(27), for the ABWR is accomplished by setting a common reference for the reactor vessel water level at the top of the active fuel as shown on Figure 5.1-3 and as described in Section 7.7.

Therefore, this issue, II.K.3(27), is resolved for the ABWR.

References

19B.2.71-1 NUREG-0933, “A Prioritization of Generic Safety Issues (with Supplements 1-15)”, U.S. NRC, April 1993.

19B.2.72 III.D.3.3(1): Issue Letter Requiring Improved Radiation Sampling Instrumentation

Issue

10 CFR Part 20 provides criteria for control of exposures of individuals to radiation in restricted areas, including airborne iodine. Since iodine concentrates in the thyroid gland, airborne concentrations must be known in order to evaluate the potential dose to the thyroid. If the airborne iodine concentration is overestimated, plant personnel may be required to perform operations functions while using respiratory equipment, which sharply limits communication capability and may diminish personnel performance during an accident. The purpose of this issue is to improve the accuracy of measurement of airborne iodine concentrations.

Acceptance Criteria

Airborne iodine concentrations must be accurately determined throughout the plant under accident conditions.

Resolution

Item III.D.3.3(1) which concerns in-plant radiation monitoring is resolved in Subsection 12.3.4 which also references each area detector location on the plant layout drawings for each building (Figures 12.3-56 through 12.3-73) as well as the specific area radiation channels for each building, the detector map location, the channel sensitivity range, and the local alarm areas (Tables 12.3-3 through 12.3-7).

References

19B.2.72-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980.

19B.2.72-2 NUREG-0737, “Clarification of TMI Action Plan Requirements”, U.S. NRC, November 1980.

19B.2.73 III.D.3.3(2): Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment

Issue

NUREG-0660 (Reference 19B.2.73-1) is a guideline to improve nuclear power plant worker radiation protection to allow workers to take effective action to control the course and consequences of an accident, as well as to keep exposures as low as reasonably achievable (ALARA) during normal operation and accidents.

Acceptance Criteria

This issue required the NRR to set criteria requiring licensees to evaluate in their plants the need for additional survey equipment and radiation monitors in vital areas and requiring, as necessary, installation of area monitors with remote readout. The NRR evaluated the need to specify the minimum types and quantities of portable monitoring instrumentation, including very high dose rate survey instruments. Operating reactors were reviewed for conformance with Standard Review Plan (SRP) Section 12.3.4, “Area Radiation and Airborne Radioactivity Monitoring Instrumentation”. The NRR revised the SRP Sections 12.5 and 12.3.4 to incorporate additional monitor requirement criteria.

Resolution

Item III.D.3.3(2) which concerns licensees evaluate the need for additional radiation survey equipment is resolved in Subsection 12.3.4. This item also concerned the need to specify the minimum types and quantities of portable monitoring instrumentation, including very high dose rate survey instruments. As noted in Subsections 12.5.2, 19A.2.39 and 19A.3.5, COL applicants will provide the portable instruments in operating reactors that accurately measure radio-iodine concentration in plant areas under accident conditions.

References

19B.2.73-1 NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, U.S. NRC, May 1980.

19B.2.73-2 NUREG-0737, “Clarification of TMI Action Plan Requirements”, U.S. NRC, November 1980.

19B.3 COL License Information

19B.3.1 COL Applicant Safety Issues

The COL applicant shall provide resolutions for the issues identified as COL applicant in the Safety Issues Index consistant with the documentation format discussed in Subsection 19B.1.1.

19B.3.2 Testing of Isolators

As established in Section 7A.3, the COL applicant is required to establish a test program for fiber optic-type isolators used between safety-related and non-safety-related systems. If other types of isolators are used (those subject to electrical leakage due to maximum credible electrical faults), the COL applicant shall implement the required testing, inspection, and replacement isolators when needed (See Subsection 19B.2.53).

19C Design Considerations Reducing Sabotage Risk

This Section is not included in DCD (Refer to SSAR Appendix 19C, Amendment 33).

19D Probabilistic Evaluations

This Section is not part of the DCD (Refer to SSAR).

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19E Deterministic Evaluations

19E.1 Introduction

This appendix documents evaluations which are deterministic in nature. These evaluations were conducted to provide an insight into performance within the plant boundaries and outside the plant boundaries.

Subsection 19E.2 focuses on the containment performance for several specific accident challenges and develops input for the offsite consequence analysis.

Subsection 19E.3 focuses on offsite consequence analysis with the CRAC code to allow a measurement against consequence related goals. Primary inputs come from the MAAP-ABWR analysis of Subsection 19E.2 and the containment event trees in Subsection 19D.5.

19E.2 Deterministic Analysis of Plant Performance

19E.2.1 Methods and Assumptions

This subsection summarizes the methods and assumptions that were used in evaluating the Reactor Pressure Vessel (RPV) and containment responses and determining the resulting source term. The Modular Accident Analysis Program (MAAP) (Reference 19E.2-1) was the primary tool used to determine the fission product source terms. Included in this subsection is a brief description of the code, the basic assumptions about the ABWR configuration, a discussion of those phenomena not explicitly modeled in the MAAP analysis, and the definition of the base case.

19E.2.1.1 Code Description

MAAP was used to determine the vessel and containment responses and the source terms for the ABWR under severe accident conditions. MAAP3.0B was modified to model the configuration of the ABWR. An overview of MAAP3.0B is provided below, followed by a discussion of the changes made in the code to model the ABWR. This new version of the code will be referred to as MAAP-ABWR.

19E.2.1.1.1 MAAP3.0B

MAAP is a computer code developed as a 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 reactor pressure vessel (RPV) failure, and 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 is an integrated code which tracks the progression of hypothetical accident sequences from a set of initiating events to either a safe, stable and coolable state or containment structural failure and fission product release to the environment. MAAP models a wide spectrum of phenomena including steam flashing, water inventory loss, core heatup, cladding oxidation and hydrogen evolution, fission product release from the degraded fuel rods and their transport to the containment and beyond, molten core slump into the lower plenum of the RPV, vessel failure, corium-concrete interactions and further release and transport of fission products. MAAP models all of the engineered safety systems such as emergency core cooling, automatic depressurization, safety relief valves, and decay heat removal. MAAP also allows the user to model operator behavior and deviations in system operation.

MAAP has a modular structure in which separate subroutines are dedicated to modeling specific regions and physical phenomena. The main program directs the program execution through several high level subroutines. The program calls a sequence of system and region subroutines at each time step. These subroutines, in turn, call phenomenology subroutines as required. The simulation of an entire accident sequence does not require any user intervention during the running of the program. A set of built-in property-library subroutines provide physical properties.

(1) High Level Subroutines

The high level subroutines include the main program, the input and output subroutines, the data storage and retrieval subroutines, and the numerical integration subroutines. Also included in the high level subroutines is a controlling routine, BWROP, which allows user interventions that describe the actions occurring during an accident sequence. The high level subroutines pass global variables by common blocks (not argument lists) and do not contain physical models for the reactor plant. The time integration subroutines, INTGRT and DIFFUN control the time steps and call system and region subroutines at each time step during an accident transient.

(2) System and Region Subroutines

The system and region subroutines include the EVENTS subroutine which sets the event flags (Boolean variables) giving the status of the system and the status of operator interventions. The event flags control code execution. Region subroutines, one for each physical region of the reactor system, define the differential equations for the conservation of internal energy and mass. Other systems subroutines examine the inter-region gas flow rates and calculate the core temperatures and fuel-cladding-coolant interactions. The systems and region subroutines pass global variables by common blocks and operate on them by calling the phenomenology subroutines.

(3) Phenomenology Subroutines

The phenomenology subroutines describe the rates of the physical processes occurring in each region of the reactor plant model. The phenomenology subroutines pass variables by argument lists, and generally do not use or alter global variables. The phenomenology subroutines are generic in nature and can be called by any of the region subroutines or by other phenomenology subroutines.

(4) Property-Library Subroutines

The property-library subroutines give the physical properties (e.g., specific heat and saturation pressure) of the important materials. These subroutines use argument lists to pass variables and do not have side effects on global variables. Property subroutines are called by the phenomenology subroutines.

19E.2.1.1.2 ABWR Modifications

Several modifications to the MAAP3.0B code were required to adequately model the ABWR. The starting point for the modifications was the MAAP3.0B Mark II models. The modified version of the code is referred to below as MAAP- ABWR. Specific ABWR features which required code changes are listed below.

(1) Containment Configuration

The ABWR configuration is different than previous BWR configurations. MAAP-ABWR models the flow paths between the containment compartments correctly. The high level subroutine DIFFP was modified. The affected regions are:

(a) Suppression Pool Configuration

The ABWR suppression pool configuration required changes in the models to accurately reflect the relationship between water level and volume. The ABWR suppression pool is modeled by applying the Mark III pool model. The affected subroutines are system routine INITAL and phenomenological routine M3POOL.

(b) Lower Drywell

Several alterations were required in order to model the ABWR lower drywell. Flow paths were added to model the vacuum breakers from the wetwell, the vents to the suppression pool and overflow from the suppression pool through the wetwell drywell connecting vents. Core concrete attack in the lower drywell region can result in penetration of the pedestal to the wetwell / drywell connecting vents. When penetration occurs, flow between the lower drywell region and the suppression pool will occur. Models for this flow were incorporated which employ a user supplied concrete penetration limit. The PEDSTL region subroutine was affected, as was the PDFP region fission product subroutine.

(c) Upper Drywell

This region required the removal of the flow path which represented the vacuum breaker in the Mark II model, and the addition of steam and gas venting to the suppression pool via the lower drywell. Affected subroutines are the DRYWEL region and DWFP fission product region subroutines.

(d) Wetwell

The wetwell fission product transport subroutine WWFP was modified to correctly model the ABWR.

(e) Horizontal Vents

The M3VENTA phenomenological subroutine model for the horizontal vents in a Mark III were applied to model the horizontal vents connecting the wetwell / drywell vents and the wetwell in the ABWR.

(2) RHR Heat Exchangers

ABWR has a heat exchanger in each of the three RHR loops. Previously, heat exchangers were modeled in only two loops of the RHR system. Addition of the third heat exchanger required a change in the ECCS system subroutine.

(3) LOCA Location

MAAP-ABWR directs the flow from all LOCA breaks into the upper drywell. However, since there is a small possibility of LOCAs which blowdown into the lower drywell, MAAP-ABWR allows the user to input the RPV Failure Event Code to simulate this event. This change was accomplished by modifying the high level subroutine BWROP and the region subroutine EVENTS.

(4) Recirculation Pump Trip

In the ABWR, four of the Recirculation Pumps (RIPs) trip on either High Vessel Pressure or on Level 3, with the remaining six RIPs tripping on Level 2. MAAP-ABWR allows the user to input these different setpoints. Region subroutine BWRVSL was modified to allow this capability.

(5) Evaporation from a Pool Surface

The evaporation model in MAAP3.0B was found to be non-conservative for the ABWR. The problem arises when the firewater system is used or the passive flooder operates and water from the wetwell floods the lower drywell. The vapor pressure in the lower drywell is much below the saturation point since there was no water in this region prior to water addition. Therefore, steam will begin to evaporate off the surface of the pool in the lower drywell.

In MAAP3.0B, the water in the suppression pool had to heat to the boiling point before evaporation was permitted off the surface of the pool. In MAAP-ABWR, the vapor pressure is conservatively assumed to rise to saturation in two time steps. This model was applied to the wetwell, upper and lower drywells. The PEDSTL, DRYWEL and BWM2WW region subroutines were affected.

19E.2.1.2 ABWR Configuration Basis

19E.2.1.2.1 ABWR Configuration Assumptions

This subsection provides a description of the assumptions which were made about the configuration and systems of the ABWR. These assumptions were made where the design detail was not yet available or was outside the scope of this submittal: for example, the type of concrete to be used in the plant is not specified in the certified design.

(1) Condensate Storage Tank. The configuration for the condensate storage tank is assumed to be consistent with the description in Subsection 19.9.9. This is sufficient to satisfy the station blackout performance requirements discussed in Subsection 19E.2.1.2.2.

(2) Not Used

(3) Type of Concrete Used for Containment. Limestone Sand concrete was assumed to be used for all portions of the containment building except the lower drywell floor. This assumption will affect the conduction of heat into the containment walls. However, since concrete has very low thermal diffusivity there will be no negative impact on containment performance. Limestone Sand concrete is representative of the concrete which might be used in much of the United States. Basaltic concrete, with a calcium carbonate content of approximately 4 weight percent was assumed for the lower drywell floor.

(4) Not Used

(5) Battery loading profiles will be developed to define appropriate load shedding during Station Blackout (Subsection 19E.2.1.2.2.2(3)). This item has been identified as COL license information in Subsection 19.9.9.

(6) RCIC room temperature will not exceed equipment design temperature without room cooling for at least 8 hours (Subsection 19E.2.1.2.2.2(5)). This item has been identified as COL license information in Subsection 19.9.9.

(7) Control room temperature will not exceed equipment design temperature for at least 8 hours without room cooling (Subsection 19E.2.1.2.2.2(6)). This item has been identified as COL license information in Subsection 19.9.9.

(8) Operator action during station blackout is consistent with the EPGs as specified in Subsection 19E.2.1.2.2.4.

19E.2.1.2.2 Performance During Station Blackout With Failure of the Combustion Turbine Generator

19E.2.1.2.2.1 Summary

A station blackout is defined as the loss of offsite electrical power and the unavailability of onsite AC electrical power (i.e., failure of diesel generators, in most cases). During this period the important plant performance characteristics to be considered are maintenance of core cooling and containment integrity.

The primary means by which the ABWR copes with a station blackout is use of the combustion turbine generator (CTG). The analyses summarized in this subsection show that the ABWR can withstand a station blackout with failure of the CTG without core damage or loss of containment integrity for a period of approximately 8 hours. If AC power is still unavailable beyond this period, core cooling by the RCIC system is assumed to be lost. However, the ACIWA system may be able to prevent core damage. This accident sequence is discussed in Subsection 19E.2.2.3.

The key requirements of core cooling and primary containment vessel (PCV) integrity are treated separately below.

19E.2.1.2.2.2 Core Cooling

The reactor core isolation cooling (RCIC) system provides water to the reactor vessel during a station blackout with failure of the CTG. The following areas are considered to assure RCIC functionality during this event:

(1) Reactor monitoring function

(2) Steam supply to the RCIC turbine

(3) DC battery capacity

(4) Water source inventory (condensate storage tank or suppression pool)

(5) RCIC room temperature

(6) Control room(s) temperature

Each of these functions is addressed below.

(1) Reactor Monitoring Function.

The reactor monitoring of vessel water level and pressure is performed using local detectors with control room indication. Instrument power supply is from the station batteries as either DC or constant voltage constant frequency (CVCF) sources.

(2) Steam Supply to the RCIC Turbine.

The reactor vessel is the source of energy for the RCIC turbine which operates the RCIC pump, maintaining vessel water level. The RCIC turbine will isolate (i.e., trip) at low pressure 0.446 MPa. However, since the operator will be maintaining vessel pressure near 6.619 MPa in accordance with the emergency procedure guidelines (EPGs), there will be more than adequate RCIC turbine pressure for operation. The RPV pressure will be controlled manually at this level (by opening 1 or more SRVs) below the first SRV setpoint to avoid SRV cycling. SRV operability during station blackout is dependent on a DC supply source and a nitrogen supply and these are evaluated in the following discussions. It should be noted that the SRVs will cycle on the spring setpoint if the operator fails to manually control pressure.

(a) Availability of DC Power for SRV Solenoids.

Based on the following evaluation, it is concluded that there is ample DC power for operating SRV solenoids.

The control power for six of the 18 SRVs is taken from the Division I battery. The valves have been considered as part of the load on the Division I battery for purposes of calculating the time the RCIC would be operable during station blackout. This evaluation leads to the conclusion that the 4000 ampere hour capacity of the Division I battery is sufficient for approximately 8 hours of coping during station blackout with failure of the combustion turbine generator.

Of the remaining 12 SRVs, 6 have their control power supply on the Division II battery and 6 are on the Division III battery. Each of these batteries have a capacity of 3000 ampere hours. Since Divisions II and III would normally be shut down during a station blackout situation with failure of the CTG, these batteries and their associated power distribution equipment would be available to supply power to the SRVs if necessary.

The ambient temperature for Divisions II and III batteries should remain acceptable as there would be very little load on these batteries during station blackout. For this reason, ambient temperature rise due to the lack of HVAC should not be a problem for the batteries and their associated equipment.

Based on the above, Divisions II and III DC supplies should be available on an intermittent basis for use in operating SRVs, as desired. The 6000 ampere hour total capacity of the two batteries would be adequate for many days of operation beyond the approximately 8 hour capability of Division I.

Further, eight of the 18 SRVs are used for the ADS function and thus have alternate power sources. Five of the eight can be supplied by either of two divisions (Divisions I or II). The other three can be supplied by any of three divisions. Control power for each of the ten SRVs which are not used for the ADS function is supplied by one division (four from Division I, three from Division II, and three from Division III). Thus the ability to control reactor pressure is very reliable.

(b) SRV Operability and High Pressure Containment Conditions.

The SRV actuators can open the SRVs without assistance from internal steam pressure when the makeup pneumatic supply is available to maintain the minimum required differential pressure. The SRV accumulators used for the ADS function (Figure 19E.2-1) shall have sufficient pressure and capacity to fully open the SRVs at 0.860 MPaA pressure in the containment, and additional gas is available from outside the containment to ensure the pressure control and depressurization function. The 0.86 MPaA containment pressure is based on the Containment Overpressure Protection System setpoint of 0.72 MPaA $\pm$ 5% (per section 6.2.5.2.6) plus a maximum pressure difference of 0.1 MPa between the wetwell and drywell.

The normal supply of N_2 gas to the SRVs from the atmospheric control system outside the containment may shut off due to low pressure caused by loss of AC power to the heaters or heating boiler which is used to gasify the liquid N_2 supply. However, there is a backup supply of N_2 gas from stored bottles at 14.8 to 5.96 MPa (maximum to minimum) pressure which can be used to open the SRVs in the ADS system.

Use of the stored nitrogen bottles requires operator action to manually open a closed supply valve at the valve location. Gas is then fed to the SRV actuators through the DC powered ADS solenoid valves inside the containment automatically. The ADS supply lines from the N_2 bottles should also be isolated from the normal N_2 supply to other systems by local manual closure of the motor operated crosstie valves which are otherwise inoperable on AC power loss.

The high pressure gas from the N_2 bottles is automatically reduced to the normal required pressure by a self-actuated pressure regulating valve. If the SRVs do not open with the pressure supplied by the self-actuated pressure regulating valve [for example, if containment pressure was equal to 0.860 MPa or if somewhat less than the normal required pressure were supplied], the operator could adjust the setpoint of the pressure regulating valve above the normal required pressure at the local station.

The capacity of a group of ten 45 liter high pressure N_2 gas bottles at 5.96 MPa minimum pressure is about 16 times that needed to open the 8 ADS SRVs, each of which has an actuator piston volume of 16.4 liters (1000 cubic in). Additionally, there are 10 other N_2 bottles that can be valved into service by local manual operation. After the 8 ADS valves are opened there is sufficient N_2 gas to account for at least 7 days leakage from the valve actuators, after which the N_2 bottles must be replaced to hold the ADS valve open. Based on the foregoing, it is concluded that the ADS valves can be operated to depressurize the reactor on loss of normal AC power supplies with the containment at 0.86 MPa. The operator has to manually close and open valves at the valve locations to supply nitrogen from outside the containment to open the 8 SRVs used for the ADS function and to hold them open when the pressure in the RPV drops to near containment pressure.

(3) DC Battery Capacity.

The Division I DC battery will be sized to be capable of operating the RCIC system for approximately 8 hours assuming the expected loading profiles for station blackout with failure of the CTG. These loading profiles will assume acceptable battery area environmental conditions and load shedding, when necessary, and will be defined in detail as the ABWR design progresses.

(4) Water Source Inventory.

The primary water source for the RCIC System is the condensate storage tank (CST) which has been sized to provide sufficient inventory for a minimum of 8 hours in combination with the suppression pool. In the event the CST became depleted, the backup source is the suppression pool. The suction source switches to the suppression pool automatically on high suppression pool level. The RCIC system must be manually overridden to assure that the suction revert to the condensate storage tank to limit heating of the containment.

(5) RCIC Room Temperature.

Failure of the AC power to the room cooling will allow the RCIC room temperature to rise. The ABWR plant will be designed to prevent the room temperature from reaching the equipment design temperature of 340 K (151°F), starting at the normal room temperature of 313 K (104°F), for at least 8 hours.

(6) Control Room Temperatures.

The safety-related equipment required to function during station blackout with failure of the CTG and located in the main, lower and computer control rooms will be designed for a maximum operating temperature of 331 K (136°F). The ABWR plant will be designed to prevent the control room temperature from reaching this equipment design temperature for at least 8 hours, starting at the normal room temperature of 299 K (79°F).

19E.2.1.2.2.3 Primary Containment Vessel (PCV) Integrity

Containment pressure and temperature analyses were performed to determine the containment atmospheric conditions after 8 hours of station blackout conditions with failure of the CTG assuming event initiation at 100% thermal power. An analysis was performed which assumed the RCIC suction was taken from the condensate storage tank for the duration of the event. The drywell and wetwell pressure and temperature were calculated to be less than their design basis of 0.411 MPa and 444 K (340°F) (drywell) / 377 K (219°F) (wetwell) after 8 hours. Therefore, PCV integrity is maintained.

19E.2.1.2.2.4 Operator Actions

The loss of normal AC power will lead to indirect turbine trip and reactor scram due to high condenser pressure on loss of circulating water. The subsequent loss of feedwater will cause the RPV to isolate on low water level. Failure of the emergency diesel generators to initiate and failure of the combustion gas turbine will leave the RCIC system as the only source of makeup water to the core. The RCIC system will automatically restore the RPV water level. Operator actions are specified in the EPGs to control the RCIC system and maintain the RPV level between Level 3 and Level 8.

In addition, the operator will be instructed to maintain RPV pressure below the high pressure scram setpoint to avoid SRV cycling by controlling 1 or more SRVs manually. The PCV pressure and temperature will not approach design values for at least 8 hours. Failure of the RCIC (core uncovery) will require the operator to blowdown through the SRVs when the heat capacity temperature limit is exceeded or the water level falls below the top of the active fuel and thereby avoid a high pressure as the core melts.

19E.2.1.2.2.5 Recovery Following Restoration of AC Power

All equipment necessary for restoration of power is located external to the primary and secondary containments in the reactor building. With the exception of the control building, all heat generating sources external to secondary containment are shutdown during station blackout so that the rooms should be at temperatures which allow restart of the support systems under their automatic or manual modes following restoration of AC power. Temperatures in the control building should be such that restart can be accomplished by the operators from the control room. Also, restart could be initiated from the remote shutdown panel or even by local control at the motor control centers and switchgear. Following restoration of power and initiation of the reactor cooling water system, the ECCS areas of secondary containment will be cooled by their safety grade room coolers so normal operation of the safe shutdown systems could be restored. The turbine building electrical systems and the non-safety-related secondary cooling system provide a backup means of restoring cooling to the ECCS equipment areas within secondary containment.

19E.2.1.2.2.6 Conclusions

The ABWR plant is being designed to be capable of maintaining core cooling and containment integrity for at least 8 hours following the loss of offsite and onsite AC electrical power including the combustion turbine generator. This capability assessment follows the general criteria of:

(1) Assuming no additional single failures

(2) Realistic analytical methods and procedures

A summary of the key plant parameters, design basis values and capability assessment is shown in Table 19E.2-2. Note that the response of the ABWR containment to this event would be successful even if the design basis values were exceeded, as long as the ultimate capability were not exceeded.

19E.2.1.2.3 Equipment Survivability

The requirements for equipment survivability are derived from two sources. 10CFR50.34(f) specifies the conditions required for an analysis in which the 100% of the active fuel cladding is oxidized. Additional requirements for demonstrating the survivability of equipment needed to mitigate a severe accident are specified in SECY-90-016. In order to meet these requirements, three categories of events were considered. The first category consists of one event which responds to the requirements of 10CFR50.34(f) paragraphs (2)(ix)(C) and (3)(v). A non-mechanistic scenario is modelled which results in the requisite oxidation but which follows the rules of design basis analysis. The other two categories respond to the requirements of SECY-90-016. The second category consists of events representing the frequency dominant events ending in in-vessel recovery. Similarly, category three is made up of events representing the frequency dominant events ending in ex-vessel recovery. Together the events in categories two and three represent the vast majority of the core damage frequency.

The list of required instrumentation and equipment was derived from reviews of the safe shutdown equipment list, the EPGs, the PRA, and the severe accident analysis. The list of required equipment varies for the three categories of events described above. The capability of each piece of identified equipment was then compared to the environmental conditions for the appropriate category of events. In reviewing the equipment capability, the environmental qualification standards for assessing compliance to 50.49 were not used as a strict measure. Rather, they were used to provide a measure of confidence that the equipment would survive the expected conditions.

19E.2.1.2.3.1 Definition of Survivability Profiles

For each of the three categories of events, a set of curves representing the bounding environmental conditions for that category were developed for use in evaluating the equipment and instrumentation survivability. These conditions were then compared to the equipment capabilities to provide a measure of confidence that the necessary equipment would survive the expected conditions. It is important to note that the

ABWR containment is inerted for all of the events described below. Therefore, there is no containment challenge due to hydrogen burning or detonation.

The basis for each category of events is provided below along with a brief summary of the event progression.

19E.2.1.2.3.1.1 10CFR50.34(f) Category

This category corresponds to an event which could result in the conditions of 10CFR50.34(f)(2) (ix), which specifies that core cooling is degraded sufficiently to result in the generation of 100% oxidation of the active cladding. Core cooling is then recovered before the vessel fails. The PRA has confirmed the results of previous studies which show that the core damage frequency is dominated by accidents initiated from transients. Table 19.3-5 indicates that a very small percentage of all core damage events are initiated by LOCA. Therefore, a transient initiated event is specified for this evaluation.

Best estimate analyses do not result in oxidation of 100% of the active cladding. In order to simulate the hypothetical event, MAAP-ABWR was run using a multiplier to non-mechanistically generate oxidation of the active cladding. Additionally, ECCS was cycled on and off to produce the requisite amount of hydrogen for 100% metal-water reaction. The event progresses as follows:

- An isolation event occurs.
- All core injection is assumed to fail.
- Drywell and wetwell sprays are initiated 30 minutes after the initiation of the accident, water flow is directed through the RHR heat exchanger.
- The core begins to heat up and zirconium begins to oxidize.
- ECCS is recovered.
- Additional hydrogen is generated as the core is quenched.
- Vessel water level is recovered, terminating the event.

Curves representing the environmental conditions during this event are shown in Figures 19E.2-26a through 19E.2-26e. The vessel pressure remains within the range of normal operating pressures for the duration of the accident. Therefore, a curve of the vessel pressure is not included here.

19E.2.1.2.3.1.2 Severe Accidents Recovered In-Vessel

This category is designed to represent the dominant in-vessel recovery sequences. There are four credible sequences of this type. The events are LCHP-IV-N-N, LCLP-IV-N-N, LCLP-IV-R-N, and SBRC-IV-N-N.

In the SBRC-IV-N-N sequence, the RCIC operates for several hours before it fails, due to the loss of sufficient battery power for RCIC controls. As discussed in Subsection 19E.2.2.3(1), the firewater system can be used to prevent core damage in this instance. The probability associated with the successful use of the firewater addition system in the development of the containment event trees is consistent with prevention of core damage. However, this possibility was not modeled in the core damage event trees. Therefore, for consistency, no credit was taken for the prevention of core damage. Nonetheless, the sequence evaluated for SBRC-IV-N-N would not be expected to have core damage. Thus, it is excluded from further consideration for the purpose of assessing equipment survivability.

All of the remaining events in this category are initiated by transients with a presumed loss of core cooling at initiation. The core gradually uncovers and heats up. Some core damage occurs, but core cooling is recovered and the vessel does not fail. In two of the sequences (those in which the seventh character is N), containment cooling is recovered before the rupture disk opens, while in one (with the seventh character R) the rupture disk opens to prevent containment failure. The curves shown in Figures 19E.2-27a through 19E.2-27f represent the bounding environmental conditions for this category of events.

19E.2.1.2.3.1.3 Severe Accidents which Progress Ex-Vessel

This category is designed to represent the dominant ex-vessel sequences. There are six credible sequences of this type. The events are LCHP-FS-N-N, LCHP-FS-R-N, LCLP-FS-N-N, LCLP-FS-R-N, LCLP-PF-N-N, and LCLP-PF-R-N. For the high pressure melt sequences (LCHP), it is known that the drywell spray system is available since the sequence does not result in a penetration overtemperature failure (i.e., the seventh character in the sequence is not P). For the low pressure scenarios, the use of the firewater addition system cannot be distinguished from sequences with the passive flooder. Therefore, both methods of mitigation are considered.

The details of the core melt progression are discussed in Subsections 19E.2.2.1 and 19E.2.2.2. In general the accident progression is as follows:

- A transient results in scram and containment isolation.
- All core cooling is lost and the vessel water level fails, resulting in core uncovery.
- The core melts and vessel breach occurs.
- For the high pressure scenario, debris may be entrained from the lower drywell, so drywell sprays are used to cool the containment and quench the core debris.
- For the low pressure scenario, either the firewater addition system or the passive flooder may be used to cool the molten core debris.

This category is characterized by core melt and vessel failure. As the fuel melts, the gas in the vessel heats up. The containment response is characterized by pressurization due to steam and non-condensible gas generation. When the vessel fails, high temperatures are generated in the drywell for a short period of time due to the introduction of core debris in the lower drywell. High pressure events have significantly different characteristics than low pressure events. Therefore, the resulting environmental conditions are broken down into two sets of bounding profiles. The curves shown in Figures 19E.2-28a through 19E.2-28f represent the bounding environmental conditions for the high pressure category of events. The curves in Figures 19E.2-29a through 19E.2-29f represent the bounding environmental conditions for the low pressure category of events.

19E.2.1.2.3.2 Identification of Required Equipment and Instrumentation

Three primary sources were used to identify the equipment and instrumentation required for the mitigation of either the 10CFR50.34(f) event or a severe accident. 10CFR50.34(f) requires that the equipment required for safe shutdown and containment isolation be considered, while the equipment and instrumentation required to survive severe accident conditions may be extracted from the discussion of the accident sequences in Subsection 19E.2.2. Additionally, all instrumentation which monitor plant variables required for operator actions were reviewed.

19E.2.1.2.3.2.1 Requirements for 10CFR50.34(f)

Safe shutdown is defined in 10CFR50.2 for non-DBA events as hot shutdown. In addition, 10CFR50.34(f) requires that containment integrity be demonstrated. Thus, the critical functions of reactivity control, vessel inventory control, containment isolation and containment integrity were considered.

The functions of reactivity control and containment isolation are required in the very early stages of an accident, during which all parameters are well within their design basis values. Therefore, since the survival of equipment to support these functions is assured, this equipment is not considered here, although the continued maintenance of containment integrity is considered.

The 10CFR50.34(f) event does not impact the secondary containment in excess of the impact of design basis events. Therefore, equipment located in the secondary containment is not considered in this review.

The core cooling function can be performed by the HPCF, the RCIC, or, following depressurization of the vessel, the LPFL mode of RHR or the firewater addition system (ACIWA). The operability of both LPFL and ACIWA will be demonstrated to satisfy equipment survivability for severe accident. Therefore, the survivability of HPCF and RCIC will not be considered.

Maintenance of containment integrity requires that isolation valves remain closed, and that excessive leakage does not occur through the containment penetrations. For the 10CFR50.34(f) event, the RHR system is used to prevent containment overpressurization.

The required instrumentation was developed from Table 7.5-8 which contains a list of all variables required for manual actions. These are obtained from a review of the events included in Chapter 15 as well as the EPGs, as discussed in Subsection 7.5.2.1. In one case, the action which would be specified by the variable is also required if the operator cannot determine the status of the variable. The neutron flux measurement indicates if the reactor is critical. If that the reactor has not been scrammed during an accident, the operator is required to scram the reactor. This same action is specified if the operator cannot determine the neutron flux. Thus, the instrumentation to determine neutron flux is not included as required for survivability.

The exhaust fan radiation monitor is used during normal operation. Upon sensing high radiation levels, the normal exhaust path is isolated and flow is directed through the Standby Gas Treatment System. Isolation also occurs if the monitor fails. For the classes of accidents considered here, the containment will be isolated. Therefore, this instrument will not be affected by the event. Further, the monitor is not a post accident monitoring device, so its survivability is not an issue.

Based on the above discussion, the equipment and instrumentation list contained in Table 19E.2-29 will be used in assessing the survivability for the 10CFR50.34(f) event.

19E.2.1.2.3.2.2 Requirements for Severe Accidents

As discussed above a review of the PRA and severe accident analysis was done to determine the set of equipment required for accident mitigation. Both in-vessel and ex-vessel scenarios were considered. The survivability of all equipment which is used in the development of the containment event trees or in the severe accident analysis is addressed. It is noted for clarification that, although the RCIC system is discussed in the development of the severe accident analysis, it is only used before core damage occurs. This ensures the proper initial conditions for the accident. Therefore, the survivability of RCIC is not addressed.

In-vessel recovery sequences occur when ECCS fails initially and a source of vessel injection is subsequently recovered or activated prior to vessel failure. Since the mean time to recovery for ECCS is approximately 19 hours and core cooling must be recovered within approximately 1 hour of the initiation of the accident, the in-vessel recovery sequences are dominated by cases in which the reactor is blown down and the firewater addition system is used to provide core cooling. In the long term, the RHR system must also be recovered to provide containment heat removal. Therefore, only these systems are considered for equipment survivability.

The instrumentation considered for equipment survivability for severe accidents was derived from the 10CFR50.34(f) instrumentation list developed in Subsection 19E.2.1.2.3.2.1. This ensures that all instrumentation considered in the severe accident analysis is accounted for, since all operator actions for severe accidents have been included in the Emergency Procedure Guidelines. The list contains more instruments than are actually considered in the severe accident analysis. For example, no actions in the PRA or severe accident analysis are based on the wetwell pressure. As in the case of 10CFR50.34(f) instrumentation, the neutron monitoring function is not required to survive the event for either in-vessel or ex-vessel accidents.

For ex-vessel accidents, it is not necessary for the SRVs or the in-vessel instrumentation to survive past the time of vessel failure. Thus, although very high temperatures persist in the vessel for the duration of an ex-vessel accident, the depressurization function, RPV water level instrumentation and RPV pressure instrumentation must only survive approximately one hour after core uncovery.

19E.2.1.2.3.3 Equipment Required For Accident Mitigation

For each required system identified in Table 19E.2-29, the components of the system which are located inside the containment are identified in the discussion which follows. Components which are located outside of containment and are not exposed to containment process fluid, are excluded from the discussion since neither the 10CFR 50.34(f) event nor a severe accident will cause significant changes outside of the containment itself. The basis for equipment survivability is also provided for each piece of critical equipment.

Stainless steel components such as piping, spargers and quenchers will not be threatened by the conditions in the containment. Therefore no further consideration of those components will be given in this discussion.

The valve actuation cabling within the primary containment is composed of concentric-lay coated copper. All of the cabling inside containment is housed in insulation which is a flame retardant cross-linked polyethylene. Additionally, the insulated cable is housed within a thermoplastic chlorinated polyethylene (Hypalon) jacket which provides protection from severe radiation environments. Analysis performed by ORNL (Reference 19E.2-33) shows that the insulation and jacket begins to lose its chemical composition at 673 K (752°F). Finally, eighty percent of the actuation cabling located inside containment is enclosed within metal conduit which further shields the cabling from severe environments. Therefore survival of the cabling for the environments considered is not a concern.

(1) Depressurization System

During a core damage event, the SRVs must be able to remain open during the in-vessel phase of the accident to ensure that any potential vessel failure occurs at low pressure. After vessel failure, SRV operability is not required.

Inside of the primary containment the depressurization system consists of the following equipment and instrumentation:

(a) Nitrogen supply

(b) Nitrogen supply line

(c) Valve actuation cabling

(d) Piping and quenchers

(e) Safety relief valves

(f) SRV solenoid

(g) Temperature and position monitoring instrumentation

For the 10CFR50.34(f) and core melt scenarios with in-vessel recovery, the safety relief valves must survive or fail in the open position for the duration of the event. For the ex-vessel cases, the safety-relief valves must survive only until the vessel fails. Vessel temperature, pressure, and radiation profiles for the ex-vessel cases fall below those for the in-vessel cases. Hence, the in-vessel cases provide an upper bound for this analysis.

The SRVs are held open with a nitrogen actuator. The nitrogen supply is located outside of containment. As discussed in Subsection 19E.2.1.2.2, the nitrogen supply will be adequate to assure SRV operability over a full range of hypothetical accidents.

The nitrogen supply line consists of piping, valves, and condensation tanks, none of which will be threatened by the containment environment. The survivability of the piping and condensation tanks is discussed above. The valves are rated to a pressure and temperature of 1.8 MPa and 444 K, respectively. The 10CFR50.34(f) and in-vessel scenarios drywell thermodynamic loads do not exceed these conditions. The ex-vessel scenario drywell loads do not exceed these conditions prior to vessel failure, after which equipment required for vessel depressurization is no longer needed. Additionally, the integrity of the valve actuation cabling, system piping, and quenchers within the containment will not be adversely affected during the accident as discussed above.

The vessel pressure does not pose a problem because it remains within design limits. Comparison to radiation qualification limits are based on two day integrated dose rates. The equipment integrated radiation doses are below the equipment qualification integrated dose rates of 2.0E+8 R and 2.0E+9 R for gamma and beta radiation, respectively, as set forth in Table 3I-16.

During the early part of the event, the temperature of the process gas exceeds the SRV design limit. This will not pose a problem for several reasons. First, organic material is only located in the solenoids, which are far enough removed from the process fluid that overheating from that source should not occur. The remainder of the valve is steel. Any deformation that might occur tends to stretch the valves outward due to internal pressure. This deformation does not restrict the valve from relieving pressure. Valve closure is not required. Therefore, valve failure in the open position is acceptable. Finally, the SRV pressure relief capacity is substantially oversized for this event. Thus, vessel depressurization requires the opening of only one valve. Deformation of all 18 SRVs in a manner which would prevent vessel depressurization is not credible.

For the 10CFR50.34(f) and in-vessel scenarios, the drywell temperature and pressure never exceed the SRV design limits. For the ex-vessel scenario, the drywell temperature and pressure remain below the design limits until the vessel depressurizes. These conditions do not pose a problem for SRV survivability.

SRV temperature and position monitoring instrumentation used during depressurization are not needed for accident mitigation. Therefore, their survival is not discussed here.

(2) Residual Heat Removal System (RHR)

In modelling the 100% metal-water reaction scenario (10CFR50.34(f)), the RHR removes heat from containment through drywell and wetwell sprays. This decreases the wetwell airspace pressure enough to avoid COPS activation, eliminating the potential for containment breach. Inside of the primary containment the RHR system consists of piping, spargers, motor-operated valves (MOVs), and check valves. The integrity of the system piping and valve actuation cabling within the containment will not be adversely affected during the accident as discussed above. The valves are rated for 8.6 MPa and 575 K. Drywell loads do not exceed these conditions for any of the event categories analyzed here. Additionally, heat transfer in the long pipe runs allows the process fluid to remain within survivability limits.

Table 6.3-9 contains pressure and temperature design parameters for the RHR system components.

For the in-vessel core melt scenarios, the LPFL mode of the RHR system may provide low pressure in-vessel core injection which eventually quenches the molten core. Core injection in the core flooder mode utilizes piping from the suppression pool, an RHR heat exchanger, and an injection valve. None of these components are located inside the vessel. The heat exchanger is located outside of primary containment. The injection valve sees maximum ambient drywell conditions of 400 K and 0.7 MPa. This maximum drywell temperature is below the thermal qualification limit of 455 K. Furthermore, the maximum drywell pressure exceeds neither the pump suction piping design pressure of 2.8 MPa, nor the pump discharge piping design pressure of 3.4 MPa.

The RHR system is needed to remove decay heat from the containment during an ex-vessel transient to avoid COPS activation, as is the case in three of the six ex-vessel events identified. As with the 10CFR50.34(f) scenario, the RHR functions in the containment cooling mode which may involve drywell sprays. The drywell spray mode consists of piping, spargers, valves and valve actuation cabling. As above, only the sparger and piping are located inside of the containment, therefore survivability is not threatened. The pressure in the suppression pool exceeds the system qualification pressure of 0.31 MPa. However, the piping is nominally capable of withstanding pressures approximately 2.5 times that based on an implied safety factor (Appendix 3M). The suppression pool temperature slightly exceeds the system qualification temperature of 377 K but this is not a concern because the RHR system component discussed here do not contain organic material.

The process fluids that are used in the RHR system come from either the suppression pool or the RPV. The suppression pool is limited to a maximum pressure of 0.72 MPa by the COPS system. Furthermore, the suppression pool temperature never rises above 430 K, a temperature reached during some ex-vessel accident scenarios. The RHR suction and discharge piping used for suppression pool cooling are rated to a pressure of 0.31 MPa and a temperature of 377 K. As discussed above, however, the piping is nominally capable of withstanding pressures up to 2.5 times the rated pressure. The high suppression pool water temperature does not pose a problem for RHR system components because they contain no organic material. In the shutdown cooling mode, the RHR loop isolates from the RPV at 0.9 MPa. In the low pressure core injection mode, the RHR loop isolates from the RPV at 3.0 MPa. In-board of the isolation valves all components are rated to a pressure of 8.6MPa and a temperature of 575 K. Because of these ratings, severe vessel conditions do not threaten RHR survivability. Since the reactor pressure will not increase after RHR activation, overpressurization will not occur.

(3) Firewater System

The firewater system may be called upon to inject water into the vessel for a severe accident with in-vessel recovery or for the 10CFR50.34(f) event or through the drywell sprays during a severe accident which progresses ex-vessel. The system is manually initiated. All flow in the system is from outside the containment. Thus, accumulation of radioactive material in the firewater pumping system will not occur. All components of the firewater system are outside of the containment and will not be significantly affected during a severe accident. Inside the containment, the firewater system utilizes RHR valves, piping and spray headers which were discussed in (2).

(4) Passive Flooder

The passive flooder may be needed to provide a water flow path from the suppression pool to the lower drywell after vessel failure. The flow path is opened as a direct result of high temperatures in the lower drywell which occur after debris relocation from the vessel. This system does not contain any active systems, instrumentation or controls. Additionally, the system components are not hindered from performing their functions due to high radiation levels which might exist in the lower drywell after debris relocation from the vessel. Therefore, the system is expected to operate under the required conditions.

(5) Containment Overpressure Protection System (COPS)

The COPS may be needed during a severe accident to relieve high containment pressure. No credit is taken for the COPS system for the 100% metal-water reactor event. The system contains piping, a rupture disk and two valves which are normally open and fail open. To relieve containment pressure, the rupture disk must burst. Activation will not be adversely affected by the radiation in the wetwell airspace during a severe accident. The sensitivity of rupture disk activation to wetwell temperature is discussed in Subsection 19E.2.8.1.2.

(6) Vacuum Breakers

The vacuum breakers may need to open during an accident to relieve high differential pressures between the wetwell and drywell. Vacuum breakers are passive in nature and have no instrumentation or control other than position indication, which is not essential for operations. The vacuum breakers are located on stub tubes high in the wetwell, a location which subjects them to the temperature loads in the wetwell airspace. During the three scenarios considered here, the differential pressures between the wetwell and the drywell do not exceed the design basis. The gas temperature in the wetwell exceeds the equipment qualification limit of 377 K by approximately 60 K. The valves are composed of steel and organic seals. There is no concern that the temperature in this area will degrade the capabilities of the steel. Tests at Sandia (Reference 19E.2-32) have shown that organic materials are capable of withstanding temperatures exceeding 600 K. The seats for these valves will be selected to survive the temperatures of the accident environment. Per Subsection 6.2.1.1.4.1, the COL applicant will be responsible for ensuring that the vacuum breakers are shielded from pool swell loads. Finally, there is no direct means for debris to reach the vacuum breakers. Therefore, they are not expected to be adversely affected during any of the accidents considered.

(7) RIP Vertical Restraints

The vertical restraints on the RIPs prevent the pumps from being ejected if the RIP attachment welds are destroyed during a core melt event. The restraints are attached to the outer vessel surface and do not experience the severe conditions within the vessel during core melting. Therefore, the integrity of the vertical restraints is not jeopardized during the in-vessel phase of the accident. Since the restraints are metallic the containment conditions will not lead to failure if vessel failure occurs.

(8) Containment Isolation Valves

The containment isolation valves will close very early in the event when all of the parameters are well within the design basis values. The valves will remain closed for the duration of the event. All of the valves attached to the primary system are rated to a pressure of 8.6 MPa. Therefore they will not be threatened by a severe pressure environment during an accident. The remaining containment isolation valves are rated to pressures above the COPS pressure limit of 0.72 MPa. Thus, they will not be threatened by the pressure environment.

The air supply to the air actuated valves automatically closes on a containment isolation signal. Therefore, the valve can not activate and reopen, even if the elastomers in the solenoid fail due to high temperature. Power for motor-operated valves (MOVs) is controlled outside of containment. During a severe accident, the power is shut off, preventing the possibility of MOV self-actuation due to shorting in a high temperature drywell environment. In addition, MOVs are self-locking, indicating they will not relax and allow leakage during the course of the accidents considered here. Since metal seats are specified for the check valves, a severe accident environment will not degrade the valve seats.

(9) Containment Structure

Extensive work has been done to demonstrate containment survivability. The reactor pedestal is considered to have a very high probability of survival as discussed in Attachment 19EC. All three scenarios considered here have a very high probability of containment integrity as discussed in Appendix 19F.

(10) Containment Penetrations

The containment penetrations consist of mechanical and electrical fixed penetrations as well as operable penetrations. The survivability of these penetrations is discussed in Subsection 19F.3.2.2. The fixed mechanical penetrations are constructed from steel and will not be affected by the conditions in the drywell. The fixed electrical penetrations contain organic seals that can be affected by high temperatures. However, tests at Sandia National Lab (Reference 19E.2-32) have shown that the penetrations maintain their integrity to a temperature of 644 K and 1.025 MPaG. This pressure is beyond the COPS activation pressure and the temperatures in the drywell do not exceed 644 K. Therefore the electrical penetrations will maintain their leak tightness for the bounding severe accident conditions.

The operable penetrations consist of two types, pressure-seating and pressure-unseating. The operable penetration seals may degrade if the drywell temperatures exceed 533 K. However, the pressure-unseating operable penetrations remain in metal to metal contact up to a pressure of 0.46 MPa. Therefore, any leakage will be within design limits for pressures at or below 0.46 MPa, even if the seal temperature limit has been exceeded. Since the remaining operable penetrations are pressure-seating they will have only a fraction of the leakage of the pressure-unseating penetrations as discussed in Subsection 19F.3.2.2. The bounding profiles show that the operable penetration temperature limit of 533 K is only exceeded for high pressure ex-vessel recovery scenarios as shown in Figure 19E.2-28d. For these scenarios the containment pressure does not exceed 0.46 MPa after the temperature limit has been exceeded, as evidenced in 19E.2-28a. Therefore, containment leaktightness will not be degraded.

The radiation loads on the penetrations are below the TID-14844 limits so radiation is not a concern.

(11) Recombiners

The recombiner system is needed in a long-term accident (order of days) to ensure that the oxygen concentration does not reach flammability limits. The recombiners are located outside of the primary containment. Piping is used to remove and return fluid to the primary containment. Therefore, the process fluid provides the only significant impact on this system. Since the supply and return lines are isolated during the early part of an event, the recombiners are not subjected to the primary containment thermodynamic loads until days later, after accident recovery when the environment is not as severe. At this time, recovery from a postulated accident might occur in a much less severe environment. Additionally, the integrated radiation doses will be well below the design basis values. Therefore, the recombiners will survive these accident scenarios.

(12) Pressure and Water Level Instrumentation

The pressure sensors used to measure both water level and pressure in the vessel and in the containment are located outside of containment. The conditions in the vessel and containment are monitored via pressure taps. The pressure sensors will not see the higher vessel or primary containment temperature and radiation doses due to the significant length-to-diameter ratio of the piping used in these sensors. The integrated radiation gamma dose for the pressure sensors is slightly over the equipment qualification limit set forth in Table 3I-16. However, the radiation limits set for design basis events are extremely conservative. Therefore, there is reasonable assurance that the sensors will survive this condition. Furthermore, the sensors are capable of withstanding very high overpressure events, on the order of 14 MPa, indicating that there is no possibility of damage from high containment pressures.

(13) Temperature Instrumentation

The GE standard practice is to use thermocouples rated to 575 K and 14 MPa. These ratings are well above the drywell and wetwell thermodynamic loads experienced during a postulated severe accident. Therefore, operation of the thermocouples should not be adversely affected. Comparison to radiation qualification limits are based on two day integrated dose rates. The equipment integrated radiation doses are below the equipment qualification dose rates of 2.0E+8 R and 2.0E+9 R for gamma and beta radiation, respectively, as set forth in Table 3I-16.

(14) Containment Atmospheric Monitoring System (CAMS)

The CAMS monitors hydrogen concentration, oxygen concentration and radiation level in both the drywell and wetwell. The hydrogen and oxygen concentration sensors are located outside of primary containment. The sensors monitor containment gas concentrations via taps located within the containment. Therefore, the condition of the sampled process steam provides the only significant impact on this system. Because long pipe runs connect the primary containment to the sensing devices, heat transfer between the process steam and the pipe walls will prevent degradation of the sensors due to severe thermal conditions. The pressure in the sensed gas will be approximately that of the primary containment. The sensors will be selected to survive these pressures. Finally, the sensors are subjected to a radiation environment provided by the process fluid. However, the integrated dose will be below the design basis values.

The wetwell radiation sensors are not exposed to the temperature and pressure environment of the primary containment. The sensors are located in shafts embedded in the primary containment wall and are isolated from the primary containment environment by a substantial amount of concrete. Both wetwell and drywell radiation sensors are qualified to 595 K and 0.65 MPa. Therefore, there will be no threat to the performance of the wetwell radiation sensor.

The radiation sensors for the drywell are located inside the drywell. Therefore they will be exposed to the drywell environment. The qualification temperature of 595 K is not exceeded in the drywell. The COPS limits the drywell pressure to 0.72 MPa. This is only slightly over the qualification pressure and should not damage the sensors. Therefore the sensors are expected to survive. Also, the wetwell sensors can be used as a backup to the drywell sensors in the unlikely event that the drywell sensors are degraded.

19E.2.1.2.3.4 Summary

Three categories of events — 10CFR50.34(f), in-vessel core melt scenarios, and ex-vessel core melt scenarios — were analyzed to determine equipment and instrumentation required to survive the accident environments resulting from these events. Table 19E.2-29 contains a list of the required equipment and instrumentation for each category. The bounding environments are shown in Figures 19E.2-26a through 19E.2-26e, 19E.2-27a through 19E.2-27f, 19E.2-28a through 19E.2-28f, and 19E.2-29a through 19E.2-29f. The equipment and instrumentation has been shown to survive these environments.

19E.2.1.3 Phenomenological Assumptions

This subsection contains a summary of those phenomena which are not considered in an integral fashion using MAAP. These phenomena fall into two categories: those which are ruled out as being incredible for the ABWR and others which are neglected because they produce an insignificant change to the overall performance of the ABWR under severe accident conditions. A more detailed explanation of some of these phenomena is given in Subsection 19E.2.3.

19E.2.1.3.1 Steam Explosions

Large scale steam explosions are deemed incredible. The geometry of the ABWR will prevent a sufficiently large contiguous mass of corium from falling into water in either the vessel or lower drywell regions. A more detailed description of this phenomenon as well as the justification for its neglect is provided in Subsection 19E.2.3.1. Small steam explosions which do not in themselves threaten the integrity of the vessel or containment are calculated by MAAP. Additionally, a scoping calculation is performed in Subsection 19E.2.6.7 to determine the mass of core material which could participate in a steam explosion without damaging the containment.

19E.2.1.3.2 Degree of Metal-Water Reaction

The metal-water reaction rate used in the integrated analysis is that calculated by the MAAP models. One limit on the generation of hydrogen occurs when all of the zirconium in the cladding is assumed to react with steam to form zirconium oxide and hydrogen gas. The separate effects calculation in Subsection 19E.2.3.2 shows that the containment is capable of withstanding the static pressure that would be generated were this maximum hydrogen production to occur, as required by 10 CFR 50.34(f).

19E.2.1.3.3 Suppression Pool Bypass Due to Additional Failures

This assumption covers one of the potential types of suppression pool bypass. Subsection 19E.2.3.3 shows that the total increased risk due to suppression pool bypass caused by additional failures is within the uncertainty of the PRA ,with the exception of failure of a wetwell/drywell vacuum breaker. Therefore, only the failure of a wetwell/drywell vacuum breaker needs to be considered explicitly. A sensitivity study was performed in Subsection 19E.2.6.11 to examine the impact of vacuum breaker leakage and failure on fission product release. Subsection 19E.2.7.3 presents an uncertainty analysis which determines the impact of bypass on risk.

19E.2.1.3.4 Effect of RHR Heat Exchanger Failure in a Seismic Event

During a seismic event it is possible for the RHR heat exchangers to fail by shear of their anchor bolts. This could potentially lead to drainage of the suppression pool if the RHR suction lines are not isolated. Calculations were performed which show that the operator has about half an hour to isolate the heat exchanger.

If the heat exchanger is not isolated then the RHR pump rooms will be subjected to additional loading caused by the static head of the water, and potentially by chugging loads as steam discharges from the broken pipe. It is seen that the RHR pump room integrity will not be breached by these loads.

Additional details about the pool drainage and structural loading may be found in Subsection 19E.2.3.4. The impact of suppression pool drainage on fission product release, should this event occur is found in Subsection 19E.2.4.5.

19E.2.1.3.5 Radiative Heating of the Equipment Tunnel

A potential concern for the ABWR during severe accidents is radiative heating of the equipment tunnel. After vessel failure, the corium in the lower drywell could radiate energy directly to the walls of the equipment tunnel. This could potentially reduce the structural material strength, eventually resulting in the tunnel buckling under its own weight.

The adoption of the passive flooder (Subsection 9.5.12) precludes this occurrence since the flooder opens when the temperature of the lower drywell airspace reaches 533 K (500°F). Upon opening, water from the suppression pool would flood the lower drywell, covering the corium. This stops any radiative heat transfer from the corium to the tunnel walls. Therefore, no significant material strength reduction of the equipment tunnel caused by increased temperature is possible.

19E.2.1.3.6 Basemat Penetration

Basemat penetration by the core debris will not lead to containment failure. In each of the sequences considered, the debris will be quenched and cooled before basemat penetration can occur. The passive flooder opens when the lower drywell temperature reaches 533 K (500°F). Even were this to fail, when the sideways penetration of the pedestal walls reach 24 mm (8 inches), water from the suppression pool would flood the lower drywell.

The lower drywell design meets the 0.02 square meters/MWt specification of the EPRI Debris Coolability Requirements for Advanced Light Water Reactors (Reference 19E.2-2). A conservative analysis was performed following the methods of the ARSAP Debris Coolability Requirement (Reference 19E.2-3) and utilizing the concrete ablation rate from CORCON (Reference 19E.2-4). Assuming a 10-hour delay in adding water to the drywell, this resulted in an ablation depth of 0.9 m (3 ft) before the corium is completely quenched and cooled by the water from the suppression pool.

Additionally, uncertainty analysis was performed in Subsection 19E.2.7.2 to assess the potential for continued core-concrete attack. The study concluded that debris cooling is highly probable for the ABWR design and that there is little impact of contained core-concrete interaction on containment performance.

19E.2.1.3.7 Hydrogen Burning and Explosions

The ABWR containment is inerted. Hydrogen burning and explosions are not possible in an inerted containment. An explicit consideration of the short periods of time when the containment is not inerted is not necessary as discussed in Subsection 19D.5.6.4.

19E.2.1.3.8 Mode of Vessel Failure

In the unlikely event of a core melt sequence with substantial relocation of debris which leads to vessel failure, the vessel failure location is expected to be in the bottom head. A failure of the RIPs has been proposed; however, as discussed below, this is not a credible mechanism for the ABWR. Figure 5.4-2 gives a pictoral description of the location of the RIPs in the RPV. Figure 5.4-1 shows more RIP detail.

Since the core melt progression is expected to contain the corium inside the core shroud, debris would not approach the RIP impellers or RPV RIP nozzles which are located outside the shroud. However, if the shroud is perforated by the corium, the corium might then enter the top of RIP impellers and possibly enter the stretch tube/shaft annulus. This is extremely unlikely since this annulus thickness decreases in the downward direction to 1.5mm (the variance between the 215mm diameter RIP shaft and the 218mm inside diameter of the stationary stretch tube). Any molten material relocating through the RIP would quickly freeze or flow through the pump rather than flowing along the pump shaft.

In the event the corium did flow down the stretch tube/shaft annulus, the motor housing to RPV nozzle weld might fail allowing the RIP/motor to drop. Figure 1.2-3b shows the two RIP vertical restraints which connect the bottom of each RIP motor housing to the RPV bottom head. These restraints prevent the RIP/motor from dropping out of the RPV in case the motor housing weld fails for any reason. Therefore, in the exceedingly unlikely event of RIP failure, the pump will not fall from the vessel, and the penetration through the vessel would be small.

Nevertheless, the corium is expected to freeze and, consequently, not flow down the annulus into the motor housing. Therefore, the RPV RIP nozzle motor housing reactor coolant pressure boundary would not be breached. Failure of the vessel in the lower head region is the expected mechanism for the release of core debris from the vessel.

19E.2.1.3.9 Impact of Suppression Pool Flashing

In the event of Containment Overpressure Protection System (COPS) operation, the wetwell will depressurize fairly quickly. This in turn could cause flashing in the wetwell. The impact of flashing on the pool level was evaluated, and it was determined that the pool would not rise to the elevation of the COPS penetration. The potential for entrainment of contaminated water was also evaluated. It was found that entrainment would not lead to increased offsite risk. The details of this analysis may be found in Subsection 19E.2.3.5.

19E.2.1.4 Definition of Base Case Assumptions

In the context of this study the phrase "base case" is used to describe those studies which determine the nominal response of the ABWR to severe accident conditions using best estimate phenomenological models and no credit for system recovery. Several accident sequences were considered using the base case assumptions. The effects of the base case assumptions on the results of the analysis are determined by means of sensitivity studies and uncertainty analyses as necessary.

19E.2.1.4.1 Core Melt Progression and Hydrogen Generation

Critical to the melt progression of the fuel is the question of blockage in the core. In the base cases it was assumed that blockage occurs as predicted by MAAP-ABWR using the default core melt progression input parameters. This decreases the generation of hydrogen in the core, since there will not be steam flow past the hot zirconium during the later stages of the melt process.

The effect of this assumption on the overall response of the plant is determined by turning off the core blockage model in MAAP. This is done with the sensitivity study in Subsection 19E.2.6.1. For this case steam continues flowing past the fuel rods as they melt. The production of hydrogen continues until there is no more water available for reaction. This leads to a somewhat higher partial pressure of hydrogen, and higher containment pressure.

19E.2.1.4.2 In-Vessel Recovery

For sequences in which there is no core cooling available at the onset of the accident it may be possible to recover core cooling at some later time. It is important to know the time which allows for in-vessel recovery in order to determine the probability of system recovery in the containment event trees, Subsection 19D.5.11.3.3. Recovery is of particular interest for the study of Loss of Offsite Power and Station Blackout sequences.

The base sequences do not model in-vessel recovery. This possibility is considered using a sensitivity study. The MAAP code calculates in-vessel recovery only if a core cooling injection source is recovered before channel blockage occurs. However, the effects of in-vessel recovery can be simulated by the use of a wetwell failure as discussed in Subsection 19E.2.4.2.

19E.2.1.4.3 System Recovery After Vessel Failure and Normal Containment Leakage

All of the base analyses assume that any failed system will remain inoperable throughout the duration of the accident. However, in order to determine the appropriate accident management strategy, it is necessary to understand the behavior of the system if a system were to recover. The recovery of any ECC system would be like the use of the firewater system. Only the recovery of the RHR system will prevent containment overpressurization. If the containment pressure is maintained below the rupture disk setpoint, the only fission product release mechanism is normal containment leakage. This mechanism is discussed in Subsection 19E.2.4.3.

19E.2.1.4.4 Early Drywell Head Failure

One type of loss of containment structural integrity in the containment event trees is early drywell head failure following a high pressure melt sequence. The consequences associated with this event are discussed in Subsection 19E.2.4.4.

19E.2.1.4.5 Consequences of Suppression Pool Drain

In a seismic event, a mode of RHR heat exchanger failure was identified which could potentially result in the draining of the suppression pool into the RHR pump rooms. An analysis was performed to examine the impact of this on pump room integrity (Subsection 19E.2.3.4) which showed that the room would remain intact.

Therefore, the suppression pool may be viewed as having moved into the pump rooms. The pump room will have no ability to withstand the increase in pressure due to decay heat. Rather the room will leak and the pressure will remain near atmospheric pressure. Thus, there will be no holdup of noble gasses. However, since all of the fission products will pass through the pool in the pump room, significant fission product scrubbing of the volatile fission products will occur. Subsection 19E.2.4.5 examines the resulting dose from this type of sequence.

19E.2.1.5 Resolution of Phenomenological Uncertainties

The ABWR is designed to limit the sensitivity to various phenomenological uncertainties. Nevertheless, an uncertainty study was performed. Severe accident phenomenological uncertainties were addressed in an engineering sense. This means that only those parameters that have a major impact on the timing and magnitude of fission product release from the containment were investigated in detail. Each parameter was considered individually, although interactions between some key phenomena were considered.

The uncertainty analysis is a four step process. The first step is a literature survey which identifies all severe accident issues. Second, these issues are screened for their applicability to the ABWR. These two steps are combined in this study. Next sensitivity studies have been performed over a credible range of key parameter values to determine the potential for a significant impact on fission product release and timing. If such impact is demonstrated, then the issue is carried forward into the final step, a detailed uncertainty analysis. The propagation of uncertainty distributions was not carried out as done in NUREG-1150 (Reference 19E.2-19).

19E.2.1.5.1 Identification and Screening of Phenomenological Issues

The first step in performing an uncertainty analysis is to identify the key phenomena and their associated uncertainties. To do this, the available literature was surveyed as discussed in Subsection 19E.2.5. Some of the severe accident issues were screened out, as they are not applicable to the ABWR design. For example, hydrogen combustion phenomena are not important in the ABWR since the containment is inerted. Issues identified which could have impact on the severe accident performance were included in the sensitivity studies which follow.

19E.2.1.5.2 Sensitivity Studies

Sensitivity studies were performed for the ABWR response to severe accident phenomena in order to determine those issues which may have significant impact on the offsite risk associated with the ABWR design. Given this goal, the ultimate measurement of sensitivity is the offsite dose. At a given site the primary factors which influence the dose are the magnitude and time of release. Therefore, changes in these parameters were used to determine the need for detailed uncertainty analyses.

19E.2.1.5.2.1 Core Melt Progression and Hydrogen Generation

Critical to the melt progression of the fuel is the question of blockage in the core. In the base cases it was assumed that blockage occurs as predicted by MAAP-ABWR using the default core melt progression input parameters. This decreases the generation of hydrogen in the core, since there will not be steam flow past the hot zirconium during the later stages of the melt process.

The effect of this assumption on the overall response of the plant is determined by turning off the core blockage model in MAAP-ABWR. This is done with the sensitivity study in Subsection 19E.2.6.1. For this case steam continues flowing past the fuel rods as they melt. The production of hydrogen continues until there is no more water available for reaction. This leads to a somewhat higher partial pressure of hydrogen, and higher containment pressure. There is virtually no impact on source term, and the time of fission product release is not substantially altered. Therefore, it is judged that the ABWR severe accident performance is not sensitive to in-vessel hydrogen production.

19E.2.1.5.2.2 Fission Product Release from the Core

The base sequences use the Cubicciotti model for fission product release from the fuel. If the release from the fuel occurs at a different rate, any potential release from the containment could be affected through the containment residence time and suppression pool scrubbing. The effect of the release rate on source term is examined in Subsection 19E.2.6.2. The study indicates that there are modest differences in the location of the fission products within the containment. However, because of the depth and subcooling of the suppression pool and the presence of the COPS, there is no appreciable variation in the release from the containment. Therefore, no further investigation of the impact of fission product release from the fuel is required.

19E.2.1.5.2.3 CsI Revaporization

An important aspect of fission product behavior is the propensity of the aerosols to adhere to the relatively cooler surfaces of the vessel and containment. While the deposition process is fairly well understood, there is considerable uncertainty in the revaporization of the fission products, particularly that of CsI. A sensitivity study was conducted, as reported in Subsection 19E.2.6.3, to examine the impact of delayed revaporization. A variation of fission product behavior inside the containment was observable. However, there is not a substantial difference in the release fraction from the containment. Therefore, no further consideration of CsI revaporization is needed.

19E.2.1.5.2.4 Time of Vessel Failure

The detailed melt progression of a severe accident is subject to considerable uncertainty. The melt progression assumed in MAAP retains the molten core material above the core plate until a local failure of the core plate occurs which results in a large pour of core debris into the lower plenum of the vessel. As a result of this model, the lower head of the vessel fails almost immediately, even though there is water in the lower plenum at the time. In other melt progression models, the molten fuel drips down the fuel rods in a process called candling. Under this assumption, it is possible for molten corium to be relocated in the lower plenum slowly, where it is quenched. This results in a delayed vessel failure after the water in the lower plenum has boiled off.

A sensitivity study was performed in Subsection 19E.2.6.4 to determine the impact of the time and mode of vessel failure on containment performance. It was observed that there is little impact on the base scenarios. However, it was noted that the mode of vessel failure could impact other phenomena such as direct containment heating and core concrete interaction. Discussion of these relationships may be found in Subsections 19E.2.7.1 and 19E.2.7.2, respectively.

19E.2.1.5.2.5 Recriticality During In-Vessel Recovery

A potential challenge to the containment has been identified for accidents in which the core melt is arrested in the vessel. Experiments have indicated the potential for the boron carbide in the control blades to form a eutectic with steel at 1500 K and relocate before the fuel relocates. Thus, if core cooling is recovered after the control material has relocated, there is a potential for the core to return to a critical condition. A sensitivity study was performed in Subsection 19E.2.6.5 to examine the potential for recriticality and the implications of its occurrence for the ABWR design. The study concluded that there was a very short time window during which a return to criticality was possible. Further, even if it should occur, recriticality is not likely to lead to containment failure. Thus, recriticality does not pose a significant threat to the ABWR design and need not be considered in an uncertainty analysis.

19E.2.1.5.2.6 Debris Entrainment and Direct Containment Heating

If a core melt accident occurs in which the reactor pressure vessel is at high pressure at the time of vessel failure, the debris may be entrained out of the lower drywell. If the debris is finely fragmented as it is dispersed, the pressure in the containment can rise rapidly. This process is known as direct containment heating. If the magnitude of the pressure rise is high enough, the containment may be challenged. This would lead to an early failure of the containment structure and large releases of fission products. Therefore, uncertainty analysis was performed. The conclusions of this study are given in Subsection 19E.2.1.5.3.1.

19E.2.1.5.2.7 Fuel Coolant Interactions

Containment challenges from fuel coolant interactions may occur when molten debris reacts rapidly, perhaps explosively, with water. Fuel coolant interactions are most likely to challenge the containment when molten debris falls into water. Examination of the containment event trees indicates that only a very small fraction of all sequences have water in the lower drywell before vessel failure. Despite this low probability, scoping studies were conducted considering both the impulse and static loads. As discussed in Subsection 19E.2.6.7, the shock wave transmitted to the structure provides the limiting loads. Using conservative estimates for the impulse load capability of the pedestal, the structure can withstand the loads associated with a steam explosion involving 9.5% of the core mass. This is three times the mass of a credible fuel coolant interaction in the ABWR. Therefore, the ABWR is very resistant to fuel coolant interactions. This failure mechanism need not be considered further in the containment event trees or the uncertainty analysis.

19E.2.1.5.2.8 Core Concrete Interaction and Debris Coolability

The issue of debris coolability has long been an area of considerable uncertainty in the progression of a core melt accident. If core concrete attack continues, the timing and magnitude of potential fission product release can be affected: the pedestal could be eroded which could threaten containment structure, non-condensable gasses could pressurize the containment leading to early rupture disk opening, and additional fission products could be released from the molten core. The ABWR design has a large drywell floor area and redundant systems which can flood the lower drywell. However, experiments performed to date have been unable to provide conclusive evidence that these features cool the debris sufficiently to prevent core concrete interaction. Therefore, uncertainty analysis was performed as discussed in Subsection 19E.2.1.5.3.2.

19E.2.1.5.2.9 Fission Product Release Location

The adoption of the containment overpressure protection system (COPS) in the ABWR containment design serves to significantly reduce the uncertainties in the timing, location and area of any fission product release. The setpoint of the rupture disk was selected such that there is a small probability of containment failure before the rupture disk opens. The probabilities for containment failure depend on the accident progression. They were calculated as described in Subsection 19E.2.8.1.1. These values were used, along with the appropriate source terms, in the containment event trees.

19E.2.1.5.2.10 Fission Product Release Flow Area

The presence of the COPS serves to substantially reduce the uncertainties associated with the flow area for the release of fission products from the containment. The limiting flow area was chosen such that any slight variation would not affect the ability of the system to relieve the containment pressure. However, if the drywell head fails before the COPS opens, there is a great deal of uncertainty in the size of the opening. A sensitivity study was performed, as reported in Subsection 19E.2.6.10, which concluded that there is a small impact on the fission product release. In addition, only a small fraction of all releases occur as a results of drywell head failure. Therefore, no further consideration of containment failure area is necessary.

19E.2.1.5.2.11 Suppression Pool Bypass

The suppression pool bypass study of Subsection 19E.2.3.3 was not able to show conclusively that a stuck open vacuum breaker would not lead to an increase in risk. Subsection 19E.2.6.11 considers the potential impact on fission product release of a fully or partially stuck open vacuum breaker. The study concludes that there may be a substantial increase in offsite dose if a vacuum breaker sticks open. Therefore, this issue is examined using a detailed uncertainty analysis. The results of this examination are summarized in Subsection 19E.2.1.5.3.3.

19E.2.1.5.2.12 High Temperature Failure of the Drywell

One of the failure modes identified for the containment was the degradation of the seals for the moveable penetrations in the drywell due to high temperature. In the base analyses discussed in Subsection 19E.2.2, the only sequences which exceeded the threshold temperature of 533 K (500°F) were those in which debris was entrained into the upper drywell and sprays were not available. Sensitivity studies were performed, as reported in Subsection 19E.2.6.12, to determine the potential for other sequences to exceed the threshold temperature which could lead to early fission product release. The largest increase in drywell temperature was only 5 K, which left ample margin to a high temperature failure. Therefore, no further study of this area is necessary.

19E.2.1.5.2.13 Suppression Pool Decontamination Factor

The pressure suppression pool is a very effective means of removing fission products from the gas space in a severe accident. The efficiency of the scrubbing process is typically characterized in terms of a decontamination factor (DF) defined by the mass of debris which enters the pool divided by the mass of debris which leaves the pool. MAAP-ABWR uses correlations based on the SUPRA code to calculate the DF. In order to investigate the sensitivity of the offsite consequences of a severe accident to the suppression pool decontamination factor, a simple sensitivity study was performed as described in Subsection 19E.2.6.13. The MAAP-ABWR code was modified to allow a constant DF of 100, a very conservative value for the ABWR configuration, to be used for all species (except noble gasses, for which the DF is 1.0). This resulted in an increase in fission product release of about four orders of magnitude. Nonetheless, there was no notable increase in offsite dose above a small conditional probability assuming COPS operation. Thus, there is not a significant impact on dose, even for a DF of 100. Thus, no further consideration of suppression pool decontamination factor is required in an uncertainty analysis.

19E.2.1.5.2.14 Suppression Pool pH

The pH of the suppression pool can affect the chemical form of iodine. This, in turn, has an impact on the release of iodine from the containment. A study was performed, as documented in Subsection 19E.2.6.14, to examine the potential for the suppression pool to become acidic. It was concluded that the pool would remain basic for longer than one day. Therefore, the iodine will remain in the pool and the fission product release will not be affected. Therefore, no further consideration of this phenomenon is required.

19E.2.1.5.3 Uncertainty Analyses

A systematic examination of severe accident challenges was performed as part of the ABWR PRA development. After screening the challenges for their applicability to the ABWR, a sensitivity study was performed to examine their potential impact on the ABWR severe accident performance. As a result of this screening, three issues were identified for more detailed examination as being potentially risk significant. The following provides a discussion of the impact of direct containment heating (DCH), pool bypass, and Core Concrete Interaction (CCI) on containment failure probability and risk profile.

19E.2.1.5.3.1 Direct Containment Heating

A large number of calculations were performed to determine the impact of DCH on the probability of containment failure and offsite risk. The analysis investigated uncertainties in a variety of areas:

- Mode of vessel failure

- Mass of molten core debris at the time of vessel failure
- Potential for high pressure melt ejection
- Fragmentation of debris in the containment

Additional sensitivity studies were performed to examine other phenomena which could affect DCH. The study concluded that a deterministic best estimate for the peak pressure from DCH would not lead to containment failure. Consideration of the uncertainties in the phenomena lead to an estimated CCFP of an extermely small fraction of all core damage events. Since the probability of containment failure due to DCH is very low, there is no measurable impact on offsite dose.

19E.2.1.5.3.2 Core-Concrete Interactions

A large number of calculations were performed as part of the investigation into core-concrete interactions in the ABWR. These calculations addressed uncertainties in the following parameters:

- Amount of core debris
- Debris-to-water heat transfer
- Amount of additional steel in the debris
- Delayed flooding of the lower drywell
- Fire water injection instead of passive flooder operation

The conclusion from all of these uncertainty calculations were:

(1) For the dominant core melt sequences that release core material into the containment, a vast majority result in no significant CCI. An insignificant number of sequences are expected to experience dry CCI.

(2) Even for those low frequency cases with significant CCI, radial erosion remains below the structural limit of the pedestal. After consideration of uncertainties a very small percentage of the sequences with significant CCI will suffer pedestal failure. Combining this conclusion with the first, an even smaller percentage of all core melt sequences with vessel failure will lead to additional drywell failures as a result of CCI.

(3) The time of fission product release is not significantly affected by continued CCI.

(4) The fission product release is dominated by the noble gasses when the containment overpressure protection system operates. This conclusion is unaffected by assumptions on debris coolability. Therefore, the offsite dose for sequences with rupture disk operation is not impacted by core concrete attack.

These conclusions would indicate that the uncertainties associated with CCI have an insignificant influence on the containment failure probability and risk.

19E.2.1.5.3.3 Pool Bypass

Analyses performed in Subsection 19E.2.3.3.3(4) indicate that the only significant mode of suppression pool bypass occurs via the vacuum breakers. Uncertainty analyses and sensitivity studies were performed to assess the effect of pool bypass on risk. Some of the key conclusions of these studies are summarized below:

(1) The probability of a large leakage path between the wetwell and drywell is very small.

(2) There is a small probability that there is a small leakage path between the drywell and wetwell. Based on the Morowitz plugging model, most of these sequences are expected to plug before the rupture disk setpoint is reached. In sequences with plugging, there is no significant increase in the time of fission product release or in offsite dose.

(3) Use of the firewater spray system can prevent early opening of the rupture disk for a bypass path of any size.

The sum of the frequency of pool bypass sequences as a result of vacuum breaker failure with no drywell spray available is an extremely small percentage of all core damage events. Since this value is extremely low there is no impact on offsite dose.

19E.2.2 Accident Sequences

The accident sequences are chosen such that both the core damage accident classes and the containment event tree classes are well represented. Using an early version of the PRA, the more probable classes of accidents were considered in selecting the accident sequences to be studied. Subsequent to the initial review, sequences were added to provide a good estimate of risk.

A complete accident sequence is designated by an eight digit character. The first four characters indicate the general conditions of the accident. The next two digits are used to identify any mitigating systems used. The seventh digit indicates the mode of release, and the eighth character indicates the magnitude of the release. A summary of the accident sequence codes is given in Table 19E.2-3.

The first consideration in selecting accident sequences for analysis was to represent the core damage event trees. To accomplish this each accident class was examined to determine the most severe sequence. The frequency of the event was then considered. If the frequency of the most severe sequence was below the cutoff frequency and if it was significantly smaller than the overall frequency of the class then the next most severe case was examined. Note that all sequences in the final PRA with extremely small probabilities were not completely dismissed. They were retained in the sum of the event class frequencies.

Eight accident sequences were selected for analysis with MAAP. Table 19E.2-4 shows how each accident class relates to the accident sequences analyzed. Each of the eight accident sequences is described below:

- LCLP

  Loss of all core cooling with vessel failure occurring at low pressure represents accident class ID and some IB-1 and IB-3 sequences.

- LCHP

  Loss of all core cooling with vessel failure occurring at high pressure models accident classes IA and IIIA as well as some IB-1 and IB-3 sequences. The results are somewhat non-conservative for some of the Class IIIA sequences because the rate of water loss from the vessel may be somewhat faster for medium break LOCAs. Small break LOCAs will be accurately modeled by this case. Even for the case of the medium break LOCA the results should be reasonably accurate, because the definition of a medium break LOCA is that which does not depressurize the vessel quickly enough to allow the low pressure systems to operate without ADS. Furthermore, the low frequency of Class IIIA events allows their consideration here.

- SBRC

  Station blackout with RCIC operating for 8 hours is class IB-2.

- LHRC

  Loss of heat removal in the containment sequences are characterized by a cooled core but the containment structure fails due to loss of containment heat removal. This sequence embodies class II.

- LBLC

  Large break LOCA with loss of all core cooling represents class IIID.

- NSCL

  Transient with no scram or core cooling; vessel fails at low pressure models class IC.

- NSCH

  Transient with no scram or core cooling; vessel fails at high pressure represents class IE.

- NSRC

  The station blackout with no scram or boron injection sequence assumes that the RCIC system is available for core cooling. The reduced flow to the core reduces the reactor power. Also modeled by this sequence are other loss of offsite power sequences where the operator manually reduces flow to the reactor in order to reduce power. This sequence portrays class IV with successful flow control.

For each accident sequence, there are a variety of mitigating systems which could be used to prevent or reduce the release of fission products to the environment. The fifth and sixth digits of the accident sequence indicator describe the mitigating features which were assumed to operate.

- 00

  This symbol is used when none of the mitigative features are operated, due to failure of the system or the operator, or the absence of the initiating condition for the system.

- IV

  There are several means by which the operator may arrest the core melt in the vessel. If any ECC system is recovered or if the firewater system started before vessel failure occurs, it may be possible to prevent vessel failure, assuring that any fission products generated are scrubbed through the suppression pool via the SRV lines. In-vessel recovery is treated as a sensitivity study in Subsection 19E.2.4.2.

- PF

  The passive flooder system is described in Subsection 9.5.12. This system automatically opens a connection between the suppression pool and the lower drywell region when the temperature of the lower drywell airspace reaches 533 K (500°F). This serves to keep the corium temperature low, preventing core-concrete interaction, and preventing radiative heat transfer from the corium to the containment structures and atmosphere.

The passive flooder system is designed to cause the lower drywell to be flooded when there is no water overlying core debris in the lower drywell. If there is no overlying water pool the fusible material in the valve will heat up and melt the fusible plug. If there is water overlying the debris pool, the lower drywell will not heat up sufficiently to cause the passive flooder to open. Examination of the Containment Event Trees (Subsection 19D.5.11) shows that the firewater addition system is expected to operate in most of the accident sequences. Therefore, the passive flooder is not needed in the majority of accidents. Rather, the lower drywell flooder is viewed as a passive backup system which floods the lower drywell, in order to keep the temperature in the drywell low, and in order to allow quenching of the core debris.

- FA

The firewater addition system (also referred to as the AC-independent water addition mode of the RHR System), described in Subsection 5.4.7, allows the operator to manually tie the fire protection system into the residual heat removal (RHR) injection line. If this action is performed within about 15 minutes after the water level reaches Level 1, this will prevent core damage, as described in Subsection 19.3.1.3.1. The firewater system also acts as a mitigating feature after core damage. Under these circumstances, the water from the firewater system pours through the vessel and onto the corium on the floor of the lower drywell. This stops core-concrete attack and radiative heating in the same manner as the passive flooder. In addition, the firewater system adds water to the containment increasing the thermal mass. This reduces the rate of containment pressurization and delays or prevents significant fission product release. The operator is instructed to turn off the firewater system when the water level in the suppression pool is at the vessel bottom elevation, unless firewater is the only means available for core cooling and the vessel is still intact. Operator actions governing use of the firewater addition system is specified in the emergency procedure guidelines in Appendix 18A.

Information about the hardware connections are supplied in the description of the RHR system in Subsection 5.4.7.1.1.10. In particular, Figure 5.4-10 shows the connections from either the diesel-driven pumps or the fire truck to the RHR system. The connection to the diesel-driven pump are in the RHR valve room. Opening valves F101 and F102 allows water to flow from the fire protection system into the RHR piping. Periodic stroke testing of these valves is required by Table 3.9-8 to ensure valve operability. The fire truck connection is located outside the reactor building at grade level. Both connections to the RHR system are protected by a check valve (F100 and F104, respectively) to insure that RCS pressurization does not result in a breach of the injection path. The nominal flow rate for the firewater addition system is between 0.06 m^3/s with no containment backpressure and 0.04 m^3/s at the COPS setpoint.

- HR

  Containment heat removal is provided by the RHR system. For the base analyses the RHR system is conservatively assumed to be unavailable.

- PS

  Passive flooder and drywell spray both operate. The drywell sprays are one function of the RHR system. During severe accidents, especially those which cause vessel failure to occur at high pressure, the drywell sprays keep the upper drywell cool. This prevents degradation of penetration seals which could result in leakage through the movable penetrations and the release of fission products below the pressure capacity of the containment. The upper drywell drains into the suppression pool. Therefore, the use of the drywell sprays will not prevent the temperature in the lower drywell from increasing. Therefore, the passive flooder will open when the lower drywell becomes sufficiently hot.

- FS

  A firewater addition spray function was added to the firewater system as a backup to the RHR drywell spray. Used in spray mode the firewater system adds external water to the containment increasing the thermal mass of the system. The spray also provides cooling of the upper drywell region. The operator will operate the spray system if the temperature of the drywell rises to a level which could threaten the seals and water level in the vessel cannot be maintained. The firewater spray causes the pressure and the temperature of the upper drywell to decrease rapidly. When the water level in the suppression pool reaches the suppression pool to lower drywell vent the operator is instructed to turn the firewater system off. If drywell head failure occurs the firewater spray system may be restarted. This causes any fission product aerosols to agglomerate on the spray droplets, reducing the fission product release to the environment. No credit is taken for this action.

There are several mechanisms whereby fission products may be released from the containment to the environment. The mode of release is designated by the seventh character in the accident sequence indicator.

- N

  Normal containment leakage does not allow significant release to the environment as discussed in Subsection 19E.2.4.3.

- P

  Leakage through movable penetrations in the drywell is assumed to occur when the gas temperature exceeds 533 K (500°F) and the pressure exceeds 0.46 MPa. Further discussion of this type of leakage is given in Appendix 19F. If containment heat removal is not recovered drywell head failure or rupture disk opening could follow the onset of leakage.

- R

  An overpressure protection relief rupture disk is described in Subsection 6.2.5.2.6 and in Subsection 19E.2.8.1.

- D

  The drywell head is assumed to fail before the rupture disk opens. The median failure pressure of the drywell head is 1.025 MPa if the temperature in the upper drywell is below 533 K (500°F). However, as discussed in Attachment 19FA, there is a small probability the drywell fails at lower pressure. At higher temperatures, the drywell head is assumed to fail at a lower pressure as described in Appendix 19F.

- E

  Early structural failure of the containment has been proposed for cases which result in the failure of the vessel at high pressure. The effect of an early containment structural failure is examined in Subsection 19E.2.4.4.

- S

  Suppression pool drainage into the RHR pump rooms may be possible following an unisolated RHR suction line break. For these cases the release will be scrubbed but the release of fission products will begin with the onset of fuel damage. These cases are considered in a sensitivity study in Subsection 19E.2.4.5.

The final character in the accident sequence designator is assigned after the sequence has been simulated with MAAP-ABWR. This eighth character indicates the magnitude of the release predicted by MAAP-ABWR. Negligible, low, medium, and high categories were established as follows according to the amount of noble gasses and volatile fission products released:

	Noble Gas	Volatiles
N	<100%	<0.1%
L	<100%	<1%
M	<100%	<10%
H	<100%	>10%

Additionally, the character 0 indicates that no core damage occurred, therefore there is no release of radioactivity.

In the following subsections each of the accident classes is considered in turn. For each general accident condition several possible mitigating actions are considered as suggested by the accident progression.

19E.2.2.1 Loss of All Core Cooling With Vessel Failure at Low Pressure (LCLP)

The initiating event selected for this sequence is a Main Steam Isolation Valve (MSIV) Closure, followed by reactor scram. The feedwater is conservatively assumed to trip, with a coastdown of 5 seconds. Four of the Reactor Internal Pumps (RIPs) trip on high vessel pressure. The SRVs cycle open and closed to relieve the steam pressure. As the water level falls, the remainder of the RIPs trip on low level. The ECC injection systems are assumed to fail.

The sequence of events which includes passive flooder and rupture disk opening for this accident is shown in Table 19E.2-5. Figures 19E.2-2a through 19E.2-2h show the system behavior throughout the accident sequence.

About one half hour after accident initiation, sufficient decay heat has been generated to lower the water level to two thirds core height, and the operator opens one SRV to provide steam cooling. The vessel blows down (Figure 19E.2-2a), while the fuel heats up (Figure 19E.2-2d) and begins to melt. There is little generation of hydrogen gas due to the metal-water reaction during the in-vessel portion of the accident (Figure 19E.2-2f) because the vessel blowdown limits the available steam when the cladding is hot. About two hours after the initiation of the transient, the lower vessel head fails.

The corium falls into the lower drywell along with any remaining water in the lower plenum of the vessel. Rapid corium to water heat transfer quenches the corium (Figure 19E.2-2d) and results in non-equilibrium steam generation causing a pressure increase in the drywell (Figure 19E.2-2b). Then the pressure decreases slightly as the containment temperature and pressure equilibrate with the pool conditions. Just under one hour is then required to boil away the water in the lower drywell (Figure 19E.2-2e) before the corium begins to heat up (Figure 19E.2-2d). After the water in the lower drywell boils off the drywell pressure decreases because steam is condensed on the containment heat sinks but there is no steam generated.

(1) Passive Flooder Operation (PF)

After the corium in the lower drywell is uncovered, the corium and the gas above it begin to heat up. When the lower drywell atmosphere reaches 533 K (500°F) at about 5 hours (Figure 19E.2-2c), the passive flooder opens. Water then pours from the wetwell into the drywell (Figure 19E.2-2e) to the level of the upper horizontal vent. This covers the corium, quenching it. This generates a small pressure spike (Figure 19E.2-2b). Following this there is again a slight decrease in pressure as the drywell returns to equilibrium with the pool.

Since the peak corium temperature during this process is 1600 K (2400°F) no significant core concrete attack occurs during the heatup of the corium, therefore no additional non-condensable gasses are generated. When the corium is quenched the generation of additional non-condensable gasses is prevented (Figure 19E.2-2f).

After the passive flooder opens the corium is covered by an overlying water pool, causing the temperature of the lower drywell gas to decrease (Figure 19E.2-2c). The small, periodic oscillations seen in the lower drywell water level after the passive flooder opens (Figure 19E.2-2e) are due to a physical instability caused by the small pressure and density differences between the lower drywell and the wetwell.

The oscillations begin when there is a small pressure differential between the wetwell and the lower drywell. The pressure differential causes relatively cool water from the suppression pool to flow into the lower drywell. This reduces the bulk temperature of the lower drywell pool. Since MAAP assumes the pool is well mixed, the surface temperature also decreases, resulting in a decrease in the partial pressure of steam in the lower drywell gas space. This pressure decrease (Figure 19E.2-2b) draws additional water into the lower drywell pool from the suppression pool.

When the elevation of water in the lower drywell is sufficient to eliminate the pressure differential, the flow from the wetwell stops. The cooled water in the lower drywell then begins to heat back up to saturation due to heat loss from the debris bed. Once saturated pool conditions are reached, steaming begins and the lower drywell pressure increases. This could cause reverse flow through the flooder line. The subsequent loss of mass in the lower drywell would cause the region to heat up more quickly, exacerbating the amplitude and period of the oscillations. Therefore, the MAAP-ABWR flooder line model includes a check valve which prevents flow from the lower drywell into the wetwell.

While this instability is based on physical phenomena, MAAP-ABWR overpredicts its severity. MAAP-ABWR models this system as two perfectly mixed pools, with overlying gas spaces at potentially different pressures. In the large scale of the plant, the cool water enters the lower drywell pool underneath the surface boundary layer of the pool. Since the density is slightly higher than that of the bulk pool, it will tend to sink. This will tend to damp the oscillation.

The size of the oscillation is dependent, in part on the time step because the decrease in the bulk pool temperature is a function of the amount of cool water added to the lower drywell. To determine the sensitivity of the containment response to the time step used by MAAP-ABWR, a representative sequence was run using very small time steps. While the results showed a slight decrease in the magnitude and period of the oscillations, no significant effect on the overall transient response was observed.

The upper drywell temperature continues to increase since the remaining fuel in the vessel loses its decay heat energy to the vessel walls and drywell via radiative and convective heat transfer. The pressurization of the containment continues (Figure 19E.2-2b) because the corium is now transferring heat directly to the water which results in steaming.

The containment continues to pressurize until the wetwell pressure reaches 0.72 MPa at 20.2 hours (Figure 19E.2-2b), when the rupture disk opens. No penetration leakage (Appendix 19F) is predicted since the temperature in the upper drywell remains below 533 K (500°F), until well after the rupture disk opens (Figure 19E.2-2c).

Figures 19E.2-2g and 19E.2-2h give the release fractions of the noble gases, cesium iodide, and cesium hydroxide as functions of time. The release of noble gases is nearly complete one hour after the rupture disk opens. The release of the volatile species, CsI and CsOH, occurs over a much longer period of time and is nearly complete at 100 hours. The release fraction of CsI at 72 hours is less than 1E-7.

There is a small probability that the drywell head will fail prior to the rupture disk opening for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 20.2 hours, drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. Noble gas release is nearly complete at 32 hours, and the volatile fission product release continues until 120 hours. The duration of the release is significantly longer for the drywell head failure sequence since the heat source in the drywell allows only a slow depressurization of the wetwell which contains the noble gases. The CsI release fraction at 72 hours is 7.5E-2, which is much greater than the release for the corresponding rupture disk case.

(2) Firewater Spray (FS)

If the operator fails to initiate the firewater addition system in the first 20 minutes of the accident to prevent core damage, there is still potential for significant benefit from its use after vessel failure is assumed to occur. The results of a sequence using the firewater addition system are given in Table 19E.2-6 and Figures 19E.2-3a through 19E.2-3e.

The firewater system adds water to the containment through the RHR-C injection lines. When trying to prevent vessel failure the operator is instructed to inject water to the vessel via the LPFL line. If very high temperatures are observed in the drywell, the valves are realigned to the drywell spray. The water then pours from the upper drywell into the wetwell via the wetwell/ drywell connecting vents, and eventually overflows into the lower drywell. This cools the corium, preventing core-concrete attack and additional metal-water reaction. Since external water is used, the effective heat capacity of the containment is increased. Furthermore, since the decay heat in the corium is delivered by convection to the water, no significant radiation heat transfer takes place, and the lower and upper drywell atmospheres remain cool. Therefore, no degradation of the movable penetration seals is expected, and no leakage through these penetrations will occur.

In this case it was assumed that the operator starts the firewater system four hours after the initiation of the event. The first four hours of the transient are identical to the LCLP-PF-R-N sequence discussed above. When the firewater system starts a pressure spike (Figure 19E.2-3b) is observed in the drywell which is caused by the evaporation of droplets in a superheated atmosphere. After the containment atmosphere is cooled (Figure 19E.2-3a), the pressure drops fairly rapidly to match the droplet temperature. This causes some water to spill over from the wetwell to the lower drywell (Figure 19E.2-3c).

The firewater addition system continues to add water, first filling the wetwell to the level of the suppression pool return path. At 7 hours, the pressure oscillations in the drywell cause the water level in the wetwell-drywell connecting vents to rise, and overflow into the lower drywell. This causes the drywell pressure to decrease further, in much the same manner as the oscillation discussed in (1) for the passive flooder. After this pressure-induced transient, the water level in the suppression pool continues to rise until, at 16 hours, it overflows. Then water begins to spill into the lower drywell and the mass of water in both the wetwell and the lower drywell increase in proportion to their surface areas (Figure 19E.2-3c). During this time the increase in pressure (Figure 19E.2-3a) is due to the slow compression of the non-condensable gases above the water.

Supplementary calculations have shown that the pressure in the containment is minimized when the water level is near the bottom of the vessel, assuming that the drywell and wetwell are at the same pressure. For this reason, the operator is directed to turn off the firewater system when the water level in the suppression pool reaches the elevation of the bottom of the vessel, which occurs at 23.6 hours (Figure 19E.2-3c).

After the firewater spray is turned off the pressures in the drywell and wetwell increase (Figure 19E.2-3a) to values consistent with the temperature of the suppression pool and non-condensable gas pressure. The pressure in the drywell regions continues to increase as steam is generated by the corium in the lower drywell. This forces water to be displaced from the lower drywell to the suppression pool via the wetwell / drywell connecting vents. When water can no longer flow directly from the drywell into the wetwell the drywell region begins steaming. This steam flows to the suppression pool where it is quenched. During this period the pressure in the wetwell stays nearly constant while that in the drywell region increases (Figure 19E.2-3a).

At 26 hours the wetwell becomes nearly saturated and the pressure in the wetwell begins to increase along with that in the drywell. At 31.1 hours the pressure in the wetwell has reached 0.72 MPa and the rupture disk opens. After the rupture disk opens the pressure decreases rapidly (Figure 19E.2-3a) and fission product release begins. At about 57 hours the water in the lower drywell boils away leaving the corium uncovered. The gas temperature in the lower drywell increases to 533 K (500°F) (Figure 19E.2-3b) and the passive flooder opens at 61 hours, allowing water to flow from the suppression pool to the drywell (Figure 19E.2-3c). The noble gas release is nearly complete at 35 hours, and the volatile fission product release is nearly complete at 76 hours. The release fraction of CsI at 72 hours is about 1E-7, as shown in Figures 19E.2-3d and 19E.2-3e, respectively.

There is a small probability that the drywell head will fail before the rupture disk opens for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 31.1 hours, drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. Noble gas release is nearly complete at 69 hours, and the volatile fission product release continues until 90 hours. The CsI release fraction at 72 hours is 5.3E-2, which is much greater than the release fraction for the corresponding rupture disk case.

19E.2.2.2 Loss of All Core Cooling with Vessel Failure at High Pressure (LCHP)

The initiator used for this analysis is a station blackout with loss of all core cooling. For this sequence the operator is assumed to fail to depressurize the vessel. The complete sequence of events for this accident with the passive flooder and drywell spray operating is shown in Table 19E.2-7. Figures 19E.2-4a to 19E.2-4i show the system response to the presumed accident.

The early stages of this transient are identical to those of a LCLP accident. The MSIVs close, the reactor scrams and the feedwater coasts down. The core becomes uncovered at 17 minutes, and metal-water reaction begins generating hydrogen (Figure 19E.2-4g) as the core heats up. The vessel continues to cycle on the SRV setpoints (Figure 19E.2-4a) as the water in the core boils away, and the core melts. Since the suppression pool temperature is below the suppression pool heat capacity temperature limit at the time of vessel failure, SRV loads are not a concern.

At 2.0 hours the vessel fails. The initial discharge of corium and water from the lower plenum is entrained by the steam from the vessel into the upper drywell and wetwell because the vessel fails at high pressure (Figure 19E.2-4a). As the steam is driven from the lower drywell the corium is carried into the upper drywell and wetwell (Figure 19E.2-4e). That portion of the corium which is blown into the wetwell is immediately quenched. It heats up only very slowly, as the suppression pool heats (Figure 19E.2-4c). The corium which is transferred into the upper drywell is initially cooled (Figure 19E.2-4c) by the atmosphere and by contact with the floor of the upper drywell.

(1) Passive Flooder and Drywell Spray Operation (PS)

The passive flooder opens 30 seconds after the vessel fails as the temperature in the lower drywell reaches 533 K (500°F) (Figure 19E.2-4d). This allows water from the suppression pool to flood the lower drywell, cooling the corium in the lower drywell. This does not, however, ensure that the upper drywell remains cool, since there is corium in this region. In order to prevent leakage through the movable penetrations in the upper drywell, the sprays must be initiated within the first 4 hours of the transient.

When the drywell spray is turned on the temperatures of both the corium and the gas in the upper drywell drop sharply (Figures 19E.2-4c and 19E.2-4d). The containment pressure also drops as steam is condensed by the spray droplets (Figure 19E.2-4b). The rapid depressurization of the lower drywell also causes water to flow from the suppression pool to the lower drywell through the open passive flooder (Figure 19E.2-4f).

After the drywell sprays are turned on, the containment slowly repressurizes (Figure 19E.2-4b). The pressure difference between the wetwell and the drywell is very small because the recirculation of water from the suppression pool to the drywell keeps the steam near the saturation pressure of the suppression pool water. If at any time during this sequence the RHR heat exchangers begin to operate, the containment would depressurize. Containment failure and fission product release would be averted.

If the heat exchangers are not recovered, the rupture disk will open when the wetwell pressure reached 0.72 MPa at 25.0 hours. Upon rupture disk opening, fission products leave the containment. The release of noble gases continues for about 8 hours after the rupture disk opens (Figure 19E.2-4h). The volatile fission product release continues for about 25 hours. The release fraction of CsI at 72 hours is less than 1E-7 (Figure 19E.2-4i).

There is a small probability that the drywell head will fail prior to rupture disk opening for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 25.0 hours, drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. Noble gas release is nearly complete at 35 hours, and the volatile fission product release continues beyond 5 days. The CsI release fraction at 72 hours is 2E-4.

(2) Firewater Spray Operation (FS)

It is possible for the operator to delay the time of containment structural failure and reduce the fission product release by adding water to the containment after a loss of core cooling with vessel failure at high pressure. Consider a case which begins identically to LCHP-PS-R-N, a loss of all core cooling occurs and the reactor scrams. The operator is assumed to fail to blowdown the reactor, vessel failure occurs at high pressure and corium is entrained into the upper drywell and wetwell.

It is assumed that the operator turns on the firewater addition spray system 1.9 hours after the start of the accident, just before the passive flooder would operate. The pressure and the upper drywell temperature decrease rapidly. The additional water from the spray is initially directed to the suppression pool. Since the flow from the sprays does not initially enter the lower drywell, the passive flooder opens at 2.0 hours. This begins to flood the lower drywell. The containment then remains in a stable condition for several hours with the containment pressure and suppression pool mass increasing. Water is present in the lower drywell for the remainder of the sequence since the passive flooder is open.

When the suppression pool water level reaches the suppression pool to lower drywell vent, the operator is instructed to turn off the firewater system. The corium in the upper drywell then causes the temperature in the upper drywell to increase.

The continued use of sprays which add water to the containment is prohibited by the EPGs. There is, however, a high probability that the RHR System would have been recovered in this interval. The operator could use this system to maintain low drywell temperatures. To model the potential impact of continued containment pressurization, the heat exchanger is assumed inoperable. The effect of this change in spray water source is an increased rate of suppression pool heating and containment pressurization, leading to an earlier containment failure than would be predicted if the spray continued to be supplied by firewater addition. The wetwell pressure reaches 0.72 MPa at about 50 hours and the rupture disk opens. The volatile fission product release continues for the next 75 hours and the CsI release fraction at 72 hours is less than 1E-7.

There is a small probability that the drywell head will fail before the rupture disk opens for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 50 hours, drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. The noble gas release is nearly complete at 55 hours and the volatile fission product release continues for more than 5 days. The release fraction of CsI at 72 hours is 1.5E-4.

(3) Passive Flooder Operation

If the operator takes no actions after a high pressure core melt, high temperatures will ensue in the upper drywell and leakage will occur through the large movable penetrations as discussed in Appendix 19F. The sequence of events for this case is summarized in Table 19E.2-8 and is depicted in Figure 19E.2-5a through 19E.2-5e.

The passive flooder opens when the temperature in the lower drywell reaches 533 K (500°F) at 2.0 hours (Figure 19E.2-5b). Water then flows from the wetwell into the lower drywell (Figure 19E.2-5d), quenching the corium in the lower drywell (Figure 19E.2-5c). In contrast, the corium in the upper drywell heats up, after an initial heat loss to the upper drywell atmosphere and structures (Figure 19E.2-5c). This heats the upper drywell atmosphere. The seal degradation temperature of 533 K (500°F) determined in Appendix 19F is reached about in 2.1 hours (Figure 19E.2-5b), but leakage does not start at this time because the pressure is still relatively low (Figure 19E.2-5a).

The containment continues to pressurize, and leakage through the movable penetrations begins at 18.1 hours. This initiates the release of fission products (Figure 19E.2-5e). However, since the leakage is not sufficient to pass all of the decay heat energy, the containment continues to pressurize (Figure 19E.2-5a).

At about 40 hours, the pressure in the drywell dips by about 0.04 MPa. This dip is caused by the flow of water from the suppression pool into the lower drywell which reduces the average temperature of the water in the lower drywell (Figure 19E.2-5d). The temperature decrease results in a decrease in pressure because the drywell is filled with saturated steam. The initial flow of water from the suppression pool causes the pressure of the lower drywell to drop, which in turn causes more water to flow from the suppression pool. The flow stops when enough water has been added to the lower drywell such that the static head above the flooder balances the pressure decrease. While this may be a mathematical artifact of the calculation, it has no serious impact on the analysis.

At about 69 hours into the accident the drywell gas temperature has reached a steady value of 830 K (1035°F) (Figure 19E.2-5b) and the drywell pressure has reached a steady value of 0.66 MPa (Figure 19E.2-5a). The containment does not reach the wetwell rupture disk setpoint pressure of 0.72 MPa, nor does it reach the pressure necessary to fail the drywell head. The drywell head failure pressure at 830 K is reduced to 0.75 MPa because high temperatures in the drywell weaken the drywell head seal as discussed in Appendix 19F.

The fission product release begins at 18.1 hours (Figure 19E.2-5e). The noble gas release continues well beyond 5 days, while the volatile fission product release is nearly complete at 70 hours. The release fraction of CsI at 72 hours is 8.8E-2.

19E.2.2.3 Station Blackout with RCIC Available (SBRC)

This accident initiator, SBRC, represents a station blackout sequence with failure of the combustion turbine generator (CTG). These are characterized by the unavailability of all AC Power except that obtained from the batteries through inverters. Therefore, the RCIC system and firewater are the only systems available for core cooling. This sequence assumes RCIC operates for approximately 8 hours, providing core cooling (Subsection 19E.2.1.2.2). After the RCIC fails, the operator depressurizes the vessel and begins injection with the firewater addition system which can maintain core cooling indefinitely. However, no containment cooling system is available since all the diesel generators and the CTG were assumed to fail.

Two types of station blackout sequences are considered. In the first, the operator successfully initiates the firewater addition system. This sequence is then similar to class II events. There is no core damage unless the containment structural failure leads to core damage. The sequence of events for the case in which core cooling is maintained is summarized in Table 19E.2-9 and is depicted in Figures 19E.2-6a through 19E.2-6e. The more serious sequence of events is that in which the operator fails to inject with the firewater system. This case is summarized in Table 19E.2-10 and is shown in Figures 19E.2-7a through 19E.2-7f.

A reactor scram occurs immediately upon loss of power. The MSIVs close and the RIPs coast down. Feedwater pumps also coast down and the water level begins to fall. When the water level reaches Level 2, the RCIC system initiates. The steam boiled off in the core is routed to the suppression pool through the SRVs.

Initially, the RCIC suction is taken from the condensate storage tank (CST). After 1.3 hours, the suppression pool level high-high alarm is reached, and RCIC suction switches to the suppression pool. Later, at 4.4 hours, the high suppression pool temperature alarm occurs, and the operator manually switches RCIC suction back to the CST. The reactor is maintained in this quasi-steady condition, with the suppression pool heating up, and the containment pressurizing, for approximately 8 hours. After the RCIC system is presumed to fail, the water in the vessel continues to boil off to the suppression pool. The pool begins to overflow the drywell at about 9 hours.

The data in Figures 19E.2-6a through 19E.2-6e, in which core cooling is maintained by the addition of firewater, begins at 6 hours. The sequence in which core cooling is maintained is identical to the sequence in which it is not until the addition of firewater at about 10 hours. Thus, Figures 19E.2-7a through 19E.2-7f can be substituted for Figures 19E.2-6a through 19E.2-6e for the first 6 hours.

(1) Firewater Addition Prevents Core Damage

It is possible for the operator to prevent core damage during an SBRC sequence by using the firewater system to inject water into the vessel after the RCIC is assumed to fail. To do this, operator must depressurize the reactor and align the valves to begin injecting with the firewater addition system.

Before RCIC failure, the containment pressure increases slowly while the RCIC operates (Figure 19E.2-7b). After RCIC failure the water level in the vessel drops quickly. At 9.8 hours the water level reaches 2/3 core height and the operator depressurizes the vessel. As the vessel pressure falls, the containment pressure increases quickly. During the blowdown the water level in the suppression pool has become sufficient to cause the water to begin to overflow from the wetwell into the lower drywell region (Figure 19E.2-6e). After the blowdown, when the firewater system is injecting, the pressure rises more slowly since only decay heat is being added to the suppression pool (Figure 19E.2-6a). The decay heat addition causes a slight volumetric expansion of water in the suppression pool. Since the water level in the suppression pool is already at the overflow point, the expansion results in flow to the lower drywell and causes a slight decrease in suppression pool mass.

The depressurization causes the water to flash to steam, lowering the water level in the vessel (Figure 19E.2-6d). MAAP-ABWR predicts that the core heats up to about 1150 K (1610°F) during this time (Figure 19E.2-6c). Therefore, core damage will not occur. When the pressure reaches the shutoff head of the firewater addition system, 1.96 MPa, water injection begins and the core cools rapidly. The water level in the vessel then rises until it reaches level 8 (Figure 19E.2-6d). The operator then maintains water level between level 8 and level 3.

When the wetwell pressure reaches 0.72 MPa after 32.3 hours, the drywell head is presumed to fail. However, because no core damage has occurred there is no release of fission products.

(2) Passive Flooder Operation

If the operator fails to use the firewater addition system after the RCIC fails, then core damage will occur. The sequence of events for this case is shown in Table 19E.2-10. The system response to this accident is shown in Figures 19E.2-7a to 19E.2-7f.

Eight hours after the loss of offsite power, RCIC is assumed to fail. The water level begins to fall, although the rate of the water level decrease is slower than that for the LCLP sequence because the decay heat is lower. The operator depressurizes the vessel when the water level reaches two-thirds core height by opening one SRV (Figure 19E.2-7a) at 9.7 hours (SRV operability is discussed in Subsection 19E.2.1.2.2). If the operator fails to begin injection using the firewater system then the fuel melts slowly, and the vessel fails at 12.3 hours.

The corium and the lower plenum water then fall to the lower drywell floor. The containment continues to pressurize as this water boils (Figure 19E.2-7b). At 21.1 hours the lower drywell dries out (Figure 19E.2-7e) and the corium begins to heat up (Figure 19E.2-7d). The corium radiates energy to the lower drywell gas (Figure 19E.2-7c). When the gas temperature reaches 533 K (500°F) at 23.5 hours, the passive flooder opens.

When the passive flooder opens water pours from the wetwell into the lower drywell (Figure 19E.2-7e). This quenches the corium and causes the drywell pressure to increase rapidly to the rupture disk rupture pressure, 0.72 MPa in about 4 minutes.

The fission product release for this sequence (Figure 19E.2-7f) begins at 23.5 hours, the time of rupture disk opening. The noble gas release lasts about 3 hours. The volatile fission products are released slowly over the next 75 hours. The CsI release fraction at 72 hours is less than 1E-7.

There is a small probability that the drywell head will fail prior to rupture disk opening for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 23.5 hours, drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. Noble gas release is nearly complete at 38 hours, and the volatile fission product release continues until 105 hours. The CsI release fraction at 72 hours is 3.4E-1, which is much greater than the release for the corresponding rupture disk case.

19E.2.2.4 Loss of Containment Heat Removal (LHRC)

This case, LHRC, was simulated using an MSIV closure event with loss of the drywell coolers, since this event isolates the reactor immediately, and will therefore direct the most heat to the suppression pool of any Class II event. The sequence of events is shown in Table 19E.2-11. Figures 19E.2-8a through 19E.2-8c show the system response to this sequence.

For most of these sequences ECCS suction is initially drawn from the CST. When the high suppression pool level is reached the suction is switched to the suppression pool. However, for simplicity, no credit was taken for the CST inventory. The effect of this assumption is to underestimate the mass of water in the suppression pool, thus overpredicting the increase in suppression pool temperature and containment pressure. Additionally, in the later stages of this transient the operator could switch the suction for the ECCS back to the condensate storage pool or use the firewater addition system, either of which provides a source of makeup water to the suppression pool.

MSIV closure causes scram and feedwater trip. As the water level falls core cooling (RCIC) initiates. Since the reactor is isolated all of the decay heat is directed to the pool, causing the pool temperature to increase (Figure 19E.2-8b). When the suppression pool temperature reaches suppression pool temperature limit, the operator blows down the reactor in accordance with the EPGs. As the vessel pressure falls, RCIC trips due to insufficient turbine pressure. The water level falls, and the HPCF system initiates.

The containment pressurizes very slowly. At 21.7 hours, the pressure reaches 0.72 MPa, (Figure 19E.2-8a) and the rupture disk opens. After the rupture disk opens, the suppression pool begins to boil off (Figure 19E.2-8c). The system will remain in this quasi-steady state for a very long time.

If at any time during this transient a source of makeup water to the containment can be used, the reactor can be maintained indefinitely in this state. As mentioned above, either the firewater addition system or the water in the CST could provide a source of makeup water to the containment.

If makeup water is not supplied, the water level in the suppression pool will eventually become so low that the core cooling pumps are unable to draw sufficient suction, and core cooling could be lost. The transient was simulated for 72 hours in this analysis and that condition was not reached. When there is insufficient suppression pool suction the operator could still maintain core cooling by switching the ECCS suction back to the CST. The CST has at least 8-hour capacity for core cooling based on the station blackout performance assessment (Subsection 19E.2.1.2.2).

If core cooling is lost, the water in the vessel will begin to boil off slowly, and eventually, core melt will occur, no earlier than three hours after the loss of core cooling. The analysis of this transient was not carried any further because there is a very long time for the operator to take the necessary action to terminate the event.

19E.2.2.5 Large LOCA with Failure of All Core Cooling (LBLC)

A main steamline break is assumed to represent the LBLC case, since it has the largest flow area and will cause the most rapid loss of coolant from the vessel. The sequence of events for this case is similar to that for LCLP (loss of core cooling with vessel failure at low pressure), however, the core melt will occur earlier for the LBLC case. The sequence of events for the LBLC case with the passive flooder and rupture disk opening is shown in Table 19E.2-12. The system response to this event is given in Figures 19E.2-9a through 19E.2-9d.

The feedwater system is conservatively assumed to trip at the initiation of the event for this analysis. The reactor scrams on a high drywell pressure signal, and the MSIVs close as the vessel pressure drops. The core uncovers in 2.8 minutes and the fuel begins to heat up. Vessel failure occurs at 1.4 hours.

At the time of vessel failure, the corium and water from the lower plenum fall into the lower drywell. The corium is quenched by this lower drywell water. The water in the lower drywell then begins to boil away (Figure 19E.2-9c), pressurizing the containment (Figure 19E.2-9a).

(1) Passive Flooder Operation

After the water in the lower drywell is boiled away by the decay heat energy in the corium, the corium begins to heat up, raising the lower drywell temperature (Figure 19E.2-9b). When the gas temperature in the lower drywell reaches 533 K (500°F) at 5.7 hours the passive flooder opens automatically. Water flows into the lower drywell (Figure 19E.2-9c) and the temperature drops as steam is generated (Figure 19E.2-9b).

After the passive flooder opens the containment pressurizes slowly (Figure 19E.2-9a) as steam is generated in the lower drywell. The entire containment remains cool (Figure 19E.2-9b) since the corium is covered.

When the wetwell pressure reaches 0.72 MPa, at 19.1 hours (Figure 19E.2-9a), the rupture disk opens. The fission product release occurs over the next 105 hours (Figure 19E.2-9d). The CsI release fraction at 72 hours is less than 1E-7.

There is a small probability that the drywell head will fail prior to rupture disk opening for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 19.1 hours, drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. Noble gas release is nearly complete at 31 hours, and the volatile fission product release continues until 100 hours. The CsI release fraction at 72 hours is 2.2E-2, which is much greater than the release for the corresponding rupture disk case.

(2) Firewater Spray

If the operator initiates the firewater addition system to add water to the containment through the RHR line then the time to containment structural failure will be delayed. For this analysis it is assumed that the operator begins injection 4 hours after the start of the accident. The sequence of events after vessel failure for this sequence is similar to that for the LCLP-FS-R-N sequence shown in Figures 19E.2-3a to 19E.2-3e.

When the firewater system is initiated there is some splashing of water into the lower drywell. This prevents the code from predicting operation of the passive flooder.

Eventually, at about 11 hours, the suppression pool overflows into the lower drywell. Water is added to the containment via the firewater system until the water level in the suppression pool reaches the level of the vessel bottom. During this time there is no boiling in the lower drywell. The containment pressurizes slowly due to the compression of the non-condensable gasses. At 23.4 hours the firewater system is shut off. As in the LCLP-FS-R-N case [Subsection 19E.2.2.1(2)], the containment pressure first increases very slowly as the water in the lower drywell heats to saturation. Then after boiling begins, the pressure rises more rapidly.

The wetwell pressure reaches 0.72 MPa at 29.5 hours, the rupture disk opens, and fission product release begins. At about 62 hours the lower drywell has dried out leaving the corium uncovered. This causes the gas temperature in the lower drywell to increase to 533 K (500°F) causing the passive flooder to open. The release of volatile fission products is nearly complete at 67 hours. The release fraction of CsI at 72 hours is less than 1E-7.

There is a small probability that the drywell head will fail before the rupture disk opens for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 29.5 hours, drywell head failure precludes rupture disk opening. Fission product release begins directly from the drywell. Noble gas release is nearly complete at 60 hours, and the volatile fission product release continues until 95 hours. The CsI release fraction at 72 hours is 2.4E-2, which is much greater than the release fraction for the corresponding rupture disk case.

19E.2.2.6 Concurrent Loss of All Core Cooling and ATWS with Vessel Failure at Low Pressure (NSCL)

The sequence chosen to represent the NSCL case is a station blackout case with failure to scram. This sequence is analogous to the LCLP case, with the additional failure of reactivity control. The sequence of events for this case, if the operator does not initiate the firewater addition system is given in Table 19E.2-13. Some of the important parameters are depicted in Figures 19E.2-10a through 19E.2-10d.

Upon loss of power, the MSIVs close and the feedwater and recirculation pumps trip. All automatic and manual attempts to insert control rods are assumed to fail. The SRVs open to relieve the vessel pressure. Furthermore, all injection pumps, including the RCIC and SLC pumps fail to inject water into the vessel. Because of the increased power level the water level in the vessel falls rapidly and the core is uncovered in 3.7 minutes.

The temperature of the uncovered core now begins to rise (Figure 19E.2-10b), and core damage begins. At 30 minutes the operator is assumed to initiate ADS and the vessel blows down. When the vessel fails at 1.3 hours, the pressure is sufficiently low to prevent entrainment. The corium, together with any water in the lower plenum, falls into the lower drywell (Figure 19E.2-10c).

The corium is quenched in the lower drywell by the water from the lower plenum. The water then boils, causing the drywell pressure to rise (Figure 19E.2-10a). All of the water is boiled off at 1.9 hours (Figure 19E.2-10c).

(1) Passive Flooder

If no actions are taken by the operator to initiate the firewater system, the passive flooder will open when the temperature of the lower drywell reaches 533 K (500°F) at 4.4 hours. At that time, water from the wetwell will pour into the lower drywell, covering the corium. This prevents core concrete attack and metal-water reaction from occurring because the corium is not sufficiently hot for either reaction to occur (Figure 19E.2-10b).

The containment pressure then begins to rise slowly as steam is generated (Figure 19E.2-10a). The rupture disk opens at 18.7 hours, and the fission products are released (Figure 19E.2-10d). The noble gas release lasts about 2 hours. The volatile fission product release lasts about 85 hours. The CsI release fraction at 72 hours is less than 1E-7.

There is a small probability that the drywell head will fail prior to rupture disk opening for this case. Assuming the drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. Noble gas release in nearly complete at 33 hours, and the volatile fission product release continues until 100 hours. The CsI release fraction at 72 hours is

8.5E-2, which is much greater than the release for the corresponding rupture disk case.

(2) Firewater Spray

If the operator begins injection using the firewater addition system after vessel failure has occurred, then the time of drywell head failure can be delayed. The sequence of events for this case is similar to the LCLP-FS-R-N case shown in Figures 19E.2-3a through 19E.2-3e.

For this sequence, where neither scram or core cooling was successful, the operator is assumed to initiate the firewater system within 4 hours. When the firewater system is initiated, there is some splashing of water into the lower drywell. This prevents the passive flooder from opening. The firewater addition serves to keep the drywell cool, and increases the thermal mass of the suppression pool, slowing the containment pressurization rate. The water level in the suppression pool reaches the spillover height at about 15 hours. When the water level of the suppression pool reaches the bottom of the vessel, at 23.7 hours, the operator is assumed to turn off the system.

The containment pressurization rate then increases, and the rupture disk opens at 30.7 hours. At about 57 hours the water over the corium boils away leaving the corium uncovered. The gas temperature in the lower drywell increases to 533 K (500°F) and the passive flooder opens at 61 hours. The volatile fission product release continues for the next 8 hours. The release fraction of CsI at 72 hours is less than 1E-7.

There is a small probability that the drywell head will fail before the rupture disk opens for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 30.7 hours, drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. Noble gas release is nearly complete at 62 hours, and the volatile fission product release continues until 85 hours. The CsI release fraction at 72 hours is 6.4E-2, which is much greater than the release fraction for the corresponding rupture disk case.

19E.2.2.7 Concurrent Loss of All Core Cooling and ATWS with Vessel Failure at High Pressure (NSCH)

The NSCH sequence is analogous to the LCHP sequence described in 19E.2.2.2 with the additional failure of reactivity control. The main effect of failure to scram or inject boron is to decrease the time of vessel failure, since the reactor stays at power for the first few minutes of the transient. However, the power level soon drops due to additional voiding in the core. The sequence of events for the NSCH sequence where the passive flooder is the only mitigating system is given in Table 19E.2-14. Figures 19E.2-11a through 19E.2-11d illustrate the key parameters.

Following an isolation event the water in the vessel boils rapidly, and the core becomes uncovered in 3.6 minutes. If the operator fails to blow down to low pressure, a high pressure vessel melt occurs in 1.3 hours. Since the suppression pool temperature is below the suppression pool heat capacity temperature limit at the time of vessel failure, SRV loads are not a concern. As with an LCHP event, corium is entrained into the wetwell and upper drywell (Figure 19E.2-11c).

(1) Passive Flooder (PF)

If the operator does not initiate the firewater system then the passive flooder will open at 1.4 hours when the temperature of the gas in the lower drywell reaches 533 K (500°F) (Figure 19E.2-11b). This immediately cools the corium in the lower drywell and the gas temperature in this region drops to near the saturation temperature.

The only heat sinks available to remove the decay heat generated by the corium in the upper drywell region are the concrete walls and the atmosphere. Since the heat transfer to the concrete is not very effective, the gas temperature in the upper drywell increases steadily. Shortly after the passive flooder opens the temperature in the upper drywell exceeds the penetration leakage temperature threshold (Figure 19E.2-11a). However, since the pressure is only 0.25 MPa, leakage does not occur at this time but is delayed until 17.8 hours when the drywell pressure reaches 0.46 MPa as shown in Figure 19E.2-11a.

At 47.7 hours, the pressure in the drywell dips sharply by about 0.06 MPa. This dip is caused when the pressure difference between the wetwell and lower drywell sides of the passive flooder allows water to flow from the suppression pool into the lower drywell, which is now filled with water. The initial flow of water from the suppression pool causes the temperature of the lower drywell pool to decrease which in turn results in depressurization of the lower drywell. This induces more water to flow from the suppression pool. The flow stops when enough water has been added to the lower drywell so the static head above the flooder balances the pressure decrease. While this may be only a mathematical artifact of the calculation, it has no serious impact on the analysis.

At about 67 hours into the accident the drywell gas temperature has reached a steady value of 850 K (1070°F). At the same time the drywell pressure has reached a steady value of 0.67 MPa (Figure 19E.2-11a). The containment does not reach the rupture disk setpoint pressure of 0.72 MPa, nor does it reach the pressure necessary to fail the drywell head. The drywell head failure pressure at 850 K is reduced to 0.71 MPa because high temperatures in the drywell weaken the drywell head seal as discussed in Appendix 19F.

Fission product release begins when drywell penetration leakage starts, at 17.8 hours. The initial release rate is very small (Figure 19E.2-11d) because of the small penetration leakage. The noble gas release continues well beyond 5 days, while the volatile fission product release is nearly complete at 65 hours. The release fraction of CsI at 72 hours is 7.3E-2.

(2) Firewater Spray Addition (FS)

The scenario in which the operator begins the firewater spray after vessel failure has occurred is the analog to the LCHP-FS-R-N case considered in Subsection 19E.2.2.2(2). The only major differences after the spray is initiated are the temperature of the pool, and consequently the pressure in the containment.

Comparisons of the LCHP-PF-P-M and NSCH-PF-P-M pressure histories (Figures 19E.2-5a and 19E.2-11a, respectively) shows that the additional power generated in the ATWS sequence causes the pressure for this case to be about 0.02 MPa higher than the non-ATWS sequence. This increase in pressure represents the additional power generated in the first hours of the ATWS transient. After this time the power level will drop to decay heat levels because of a strong negative void coefficient in the core.

Therefore, since the difference in the pressures of the two cases is small, the transient considered here, a simultaneous loss of all core cooling and failure of reactivity control with vessel failure at high pressure, in which the operator start the firewater spray system after vessel failure, will behave like the LCHP-FS-R-N case considered in 19E.2.2.2(2). No further analysis of this sequence was performed.

19E.2.2.8 Concurrent Station Blackout with ATWS (NSRC)

The final sequence considered here, NSRC, is the case where a station blackout occurs with failure of the combustible gas turbine and all reactivity control fails. In this sequence the RCIC is the only system available to provide core cooling. The sequence of events for this case is given in Table 19E.2-15 and some of the key parameters are shown in Figure 19E.2-12a through 19E.2-12f.

Upon the loss of power, the reactor isolates immediately. The vessel pressure increases and SRVs cycle to control pressure (Figure 19E.2-12a). The water level falls rapidly and at 1.1 minutes, the RCIC system begins injecting. The water level continues to fall and at 2.2 minutes the top of the core becomes uncovered.

Although the top few nodes heat up to about 850 K (1070°F), the core does not melt at this time due to steam cooling (Figure 19E.2-12d). The power level during this time is about 4% (Figure 19E.2-12c). This amount is that required to boil the water injected by

RCIC. During this time the containment pressurizes fairly rapidly due to the relatively high rate of steam generation (Figure 19E.2-12b).

All of the water added by the RCIC System is converted to steam in the core. The steam flows through the SRVs to the suppression pool where it is quenched, adding to the mass of the pool. At 1.9 hours the suppression pool begins to overflow into the lower drywell.

If the operator is unable to shutdown the reactor by means of either the rods or boron injection then the containment pressure will reach the RCIC turbine exhaust pressure limit in 3.6 hours (Figure 19E.2-12b). This causes the RCIC to trip. As there is no other source of vessel injection available, the water level in the vessel will drop and the core will begin to melt, as seen by the increasing fuel temperature in Figure 19E.2-12d. At the same time the power will drop to the decay heat level because of increasing voids (Figure 19E.2-12c).

(1) Passive Flooder

The operator depressurizes the reactor 10 minutes after the RCIC is tripped. Vessel failure ensues at 5.6 hours. Corium and water fall into the lower drywell (Figure 19E.2-12e). A short time later, at 8.6 hours, the wetwell pressure reaches 0.72 MPa and the rupture disk opens. The containment begins to depressurize (Figure 19E.2-12b) and fission product release begins. The lower drywell dries out at about 30 hours and the passive flooder opens soon after. The noble gas release occurs within the first 5 hours after the rupture disk opens, while the volatile fission product release continues for 100 hours. The release fraction of CsI at 72 hours is less than 1E-7.

There is a small probability that the drywell head will fail prior to rupture disk opening for this case. Assuming the drywell head fails when the wetwell pressure reaches 0.72 MPa at 8.6 hours, drywell head failure will preclude rupture disk opening. Fission product release begins directly from the drywell. Noble gas release is nearly complete at 19 hours, and the volatile fission product release continues until 50 hours. The CsI release fraction is 4.8E-1 at 72 hours, which is much greater than the release for the corresponding rupture disk case.

(2) Firewater Sprays Operated

The operator can delay the release of fission products by initiating the firewater spray before the rupture disk opens. If the firewater spray begins at 6.1 hours, 30 minutes after vessel failure, the fission product release does not begin until 26.4 hours. Upon firewater spray initiation the containment pressure and temperature decrease. At 22 hours the level has reached the bottom of the vessel and the operator is instructed to turn off the spray. The containment begins to pressurize until, at 26.4 hours, the wetwell pressure reaches 0.72 MPa and the rupture disk opens. The containment rapidly depressurizes and fission product release begins. At 49 hours the lower drywell dries out, leaving the corium uncovered and the passive flooder opens at 52 hours. The noble gas release is nearly complete at 33 hours, while the volatile fission product release continues until about 120 hours. The CsI release fraction at 72 hours is 1E-7.

There is a small probability that the drywell head will fail before the rupture disk opens for this case. Assuming the drywell head fails as the wetwell pressure reaches 0.72 MPa at 26.4 hours, drywell head failure will preclude rupture disk opening. Fission product release is nearly complete at 38 hours, and the volatile fission product release continues until 85 hours. The CsI release fraction at 72 hours id 2.0E-1, which is much greater than the release fraction for the corresponding rupture disk case.

19E.2.2.9 Summary

Table 19E.2-16 gives a summary of the critical parameters for the accident sequences discussed above. For each sequence considered in the analysis which results in fission product release, the time of vessel failure, the start of fission product release and the time of rupture disk opening are given. Also shown are the duration of the release and the release fraction for CsI after 72 hours.

19E.2.3 Justification of Phenomenological Assumptions

Several separate effects studies were performed to supplement the MAAP analyses of severe accident sequences. These studies were performed to address the technical issues which could potentially have impact on the ABWR response to postulated severe accidents. They were selected for consideration based on the results of past PRA experience within the industry.

19E.2.3.1 Steam Explosions

A steam explosion is caused by thermal energy release to water, which causes rapid steam formation, expansion, and substantial pressure or impact loads on structures. It is possible that the high thermal energy content of molten core debris can cause a steam explosion if it enters water under conditions favorable to rapid heat transfer.

The potential for an ex-vessel steam explosion for a postulated severe accident in the ABWR plant is evaluated in this subsection, and is found to be extremely low.

19E.2.3.1.1 The Steam Explosion Process

Figure 19E.2-13 helps explain the process of steam explosions. It is postulated that a loss of cooling mechanism causes the reactor core to melt, followed by vessel breach and discharge of molten core debris with high thermal energy into the lower drywell, which is assumed to contain a stagnant pool of water. The energy transfer rate to water depends on the volume of submerged debris and available surface area for heat transfer. If many small particles of molten debris enter or form in the water, heat transfer will be rapid. Larger particles have less surface area per unit volume, and correspondingly slower heat transfer. Moreover, internal heat diffusion in large particles can limit the heat transfer rate to the water.

High velocity discharge of liquid debris into air can form spray-size droplets before they enter a water pool. However, for most cases debris discharge in the ABWR is expected to occur by gravity draining from a depressurized vessel, which could form larger droplets in air, about 24 mm. Smaller droplets would be formed by a stream of molten debris falling through water. An event called triggering can occur if the energy transfer from droplets forms additional droplets from the debris stream and rapidly mixes them with surrounding water. Both external triggering and self-triggering can cause steam explosions. External triggering has been employed in experiments by the use of submerged explosive devices, which include exploding wires, primicord, and blasting caps. The corresponding energy release of external triggers promotes the rapid breakup of molten debris into small particles.

Self-triggering sometimes is caused by debris stream impingement and shattering on submerged structures. If triggering suddenly creates increased surface area for heat transfer, the rapid formation of the steam causes water acceleration, which can create substantial pressure and impact forces.

19E.2.3.1.2 Previous Studies

Analytical and experimental studies of steam explosion phenomena are summarized in a 1983 IDCOR study (References 19E.2-5 and 19E.2-6). Analytical models and experimental studies reported in the literature are discussed from the standpoint of necessary conditions required to produce large scale steam explosions. It was determined that the following specific conditions had to be satisfied for steam explosions to occur:

(1) Many tonnes of molten core debris must enter the water.

(2) The debris must be coarsely fragmented into about one centimeter diameter or smaller particles and thoroughly mixed with water.

(3) A trigger must initiate a localized explosion which subsequently fragments adjacent particles into submillimeter size, and rapidly mixes them with the surrounding water in less than a millisecond, promoting rapid vaporization.

(4) A continuous liquid slug must cover the vaporization zone so that it can be propelled upward like a missile by the explosive interaction.

It was concluded in the IDCOR study that for both in-vessel and ex-vessel steam explosions, the formation of tonnes of coarsely fragmented molten core debris dispersed in water, with the associated large steam generation rates, is fundamentally inconsistent with a continuous overlying liquid slug required for efficient energy transfer. That is, steam explosions do not provide a set of credible physical processes leading to failure of either the primary system or the reactor containment building. The IDCOR conclusion and the conclusion of this analysis differ from the earlier WASH-1400 report (Reference 19E.2-7), in which energetic steam explosions were believed possible, leading to early containment failure.

Molten debris discharge from a reactor vessel at high pressure is more likely to be atomized and enter the pool as small droplets, which can rapidly transfer thermal energy and increase the potential for a steam explosion. However, the major conclusion from data and analytical models discussed in the IDCOR study (Reference 19E.2-5) imply that low vessel pressure and gravity discharge of molten core debris in ABWR has an extremely low potential for generating a steam explosion.

The IDCOR study (Reference 19E.2-5) reports that in various experiments it was possible to cause a steam explosion with an external trigger, which broke the molten metal into small drops and mixed them with surrounding water. Several experiments were reported in which iron thermite was observed to undergo self-triggering prior to a steam explosion. However, the thermite at a temperature of 3000 K apparently remained liquid during the triggering process, offering only surface tension resistance to molten droplet formation. Molten core debris is expected to be discharged at the liquidus temperature of 2600 K. The outer surface of small droplets freezes rapidly after entering water, perhaps even while falling a long distance through air, so that further droplet division requires more energy to fracture the outer crust formed than it does to overcome liquid surface tension. This helps explain why self-triggering can be observed with some highly superheated metals, but is much less likely with molten core debris.

Experimental work reported in the IDCOR study was performed in small scale test facilities, which leads to questions about how accurately the experiments represent full size severe accident steam explosion response. Theofanous (Reference 19E.2-8) addressed the scaling concern by formulating the basic phenomena of steam explosions, and comparing computer solutions for different scales. Calculated pressure and volume fractions of steam, melt, and coolant, were compared on a normalized time scale, and show that for properly scaled molten debris pours, comparable behavior can be expected in scales as low as 1/8 of full size. Most of the experiments discussed in the IDCOR report (Reference 19E.2-5) were at smaller scales, which leaves the scaling question short of full resolution. However, it is expected that the basic theoretical formulations, which are consistent with experimental phenomena at small scales, can be extrapolated to evaluate the potential for steam explosion in full size applications.

That is largely the IDCOR approach which leads to the conclusion of low steam explosion potential during a severe accident in a full scale reactor.

19E.2.3.1.3 Theoretical Considerations

The theoretical considerations of this study are based on simplified, bounding analyses which tend to be conservative in the promotion of steam explosions. These considerations are used to evaluate the geometric conditions expected for a molten debris pour into water, the heat transfer and steam formation rates, pressure rise and water hydrodynamic response time. It is concluded from these considerations that the potential of an ex-vessel steam explosion in ABWR is extremely low.

(1) Estimated Debris Droplet Formation Size

Hydrodynamic instability causes droplet formation when two parallel, adjacent liquid streams with different densities travel at different velocities. Figure 19E.2-14a shows the heavier liquid, of density ρ_h, on the bottom, flowing horizontally with velocity v_h, underneath the lighter liquid, of density ρ_L, flowing with velocity v_L. The condition for unstable interface waves can be obtained from Lamb (Reference 19E.2-9) in the form

$$\frac{8\pi^3\sigma}{\lambda^3} + \frac{2\pi g(\rho_h - \rho_L)}{\lambda} - \frac{4\pi^2\rho_h\rho_L(v_h - v_L)^2}{\lambda^2(\rho_h + \rho_L)} < 0 \qquad (19E.2\text{-}1)$$

where σ is the surface tension of the heavier liquid and λ is the wave length. An unstable wave can grow to the amplitude at which it detaches from the heavier liquid and forms a droplet of approximate diameter λ, or radius $r \approx \lambda / 2$.

Figure 19E.2-14b shows a corium stream of density ρ falling vertically at velocity V through stationary fluid of density ρ_∞. Here, the gravity term does not play a role in wave growth, and Equation 19E.2-1 gives the approximate minimum stable radius of droplets formed as

$$r_{min} > \frac{\pi\sigma(\rho + \rho_\infty)}{\rho\rho_\infty V^2} \qquad (19E.2\text{-}2)$$

If the debris stream discharge is determined by gravity draining, its downward velocity at distance z from the debris surface in the reactor is $V = \sqrt{(2gz)}$. Droplet sizes formed in air and water would be different because density ρ_∞ plays a strong role in Equation 19E.2-2.

When the debris stream first enters the water pool, it undergoes deceleration, a, due to the drag force. Under these conditions, the forces on the debris stream resemble those of Figure 19E.2-14a, except that the term g in Equation 19E.2-1 must be replaced by –(a + g). Stability of the frontal surface of liquid debris in contact with water can be evaluated by setting $v_L = v_h = 0$ in Equation 19E.2-1. It follows that the expected stable droplet size formed is

$$r^2_{min} \approx \frac{\pi^2 \sigma}{(a + g)\ (\rho - \rho_\infty)} \qquad (19E.2\text{-}3)$$

If a mass of debris M enters the pool at velocity V_0, the drag force causes a deceleration

$$a = \frac{C_d A \rho_\infty V_0^2}{2M} \qquad (19E.2\text{-}4)$$

where

A = the projected area of M, and

C_d = the drag coefficient.

(2) Debris Stream Broadening in Water

Equation 19E.2-4 gives an approximate deceleration of a debris mass entering the pool. If an average debris mass

$$M \approx A_0 L \rho$$

decelerates at

$$a = dV/dt = (dV/dy)\ (dy/dt) = VdV/dy$$

in the pool, its velocity at depth y below the water surface is obtained by integrating Equation 19E.2-4 with $V_0 = V$, which gives

$$\frac{A}{A_0} = \exp\frac{C_d \rho_\infty y}{\rho L} \qquad (19E.2\text{-}5)$$

The stream area broadens about 11% for an approximate C_d of 1.0. A debris stream which broadens in the pool would re-absorb small interface droplets formed by instability in the water. This action would tend to reduce the formation of many small interface droplets for high heat transfer into the water. It follows that substantial droplet formation in the water pool would have to occur by self-triggering. The dynamics of steam formation and the triggering process are discussed after a consideration of steam formation from a single debris droplet.

(3) Steam Formation, Single Debris Droplet

The amount of steam formed if all debris droplet thermal energy is transferred to an associated water mass M_w at P_∞ is given by

$$M_w = \frac{E'}{h_{fg}(P_\infty)} \qquad (19E.2\text{-}6)$$

where

E' = the energy remaining for steam formation after heating M_w from a subcooled state to saturation.

That is, if E is the droplet total thermal energy,

$$E' = E - E_{sc} \qquad (19E.2\text{-}7)$$

where

E_{sc} = the energy required to saturate the water mass,

$$E_{sc} = M_w c_v (T_{sat} - T_{sc}) \qquad (19E.2\text{-}8)$$

The maximum volume of steam formed at ambient pressure is

$$V_g = \frac{E' v_{fg}(P_\infty)}{h_{fg}(P_\infty)} \qquad (19E.2\text{-}9)$$

If the steam volume is spherical, its radius is

$$R_\infty^3 = 3V_\infty / 4\pi \qquad (19E.2\text{-}10)$$

(4) Thermal Response Time of Corium Droplet

An idealized spherical debris droplet of radius r at temperature T undergoes convection cooling to the ambient fluid at a heat transfer rate,

$$q = 4\pi h r^2 (T - T_\infty)$$

Assuming uniform droplet internal temperature, the droplet internal thermal energy relative to its surroundings,

$$E = (4/3)\rho\pi c_v r^3 (T - T_\infty) \quad (19E.2\text{-}11)$$

is diminished at a rate q, for which

$$T - T_\infty = (T_i - T_\infty)\exp(-t/\tau_h)$$

where the time constant τ_h gives the convective time response as

$$\tau_h = \rho c_v r/3h \quad (19E.2\text{-}12)$$

The internal conduction of heat occurs with an approximate time constant (Reference 19E.2-10),

$$\tau_c = 2r^2/\alpha \quad (19E.2\text{-}13)$$

Either τ_c or τ_h may control the heat transfer rate to surrounding water. Figure 19E.2-14a gives the conduction and convection response times in terms of droplet radius for convection bounds defined by an enhanced film boiling coefficient of 3.0 times the Berenson horizontal flat plate value (Reference 19E.2-11), and with a nucleate boiling coefficient. Debris droplets at the liquidus temperature of 2600 K with surface waviness are expected to undergo enhanced film boiling heat transfer. Enhancement factors between 3.0 and 6.0 have been observed at liquid surfaces disturbed by gas bubbling (Reference 19E.2-12). The convective response time is seen to be proportional to the droplet radius in Figure 19E.2-14a. It is seen that internal conduction could become limiting for droplet sizes above 0.2 mm radius if nucleate boiling occurred, and above 10 mm radius if enhanced film boiling dominated the surface heat transfer.

(5) Hydrodynamic Response Time

A steam bubble formed by a single debris droplet grows to an equilibrium radius R_∞ at ambient pressure. The growth time depends on its rate of expansion, which can be estimated from the Rayleigh equation (Reference 19E.2-9) for a spherical bubble,

$$RR'' + \left(\frac{3}{2}\right)(R')^2 = (P_b - P_\infty)/\rho_1 \quad (19E.2\text{-}14)$$

where primes indicate derivatives with respect to time.

Since the pressure of the gas inside the bubble is not known it is necessary to introduce additional equations for the growth rate of the bubble. The rate at which mass enters the bubble may be approximated by

$$h_{fg}m'_g = hA_b(T_i - T_\infty)\exp(-t/\tau_h) \tag{19E.2-15}$$

where τ_h is given by Equation 19E.2-12.

An energy balance for the bubble growth may also be written:

$$P_bV' - h_{fg}m'_g + U' = 0 \tag{19E.2-16}$$

Assuming an ideal gas

$$U = P_bV_b/(k-1) \tag{19E.2-17}$$

If the bubble is further assumed to be spherical, one may combine Equations 19E.2-16 and 19E.2-17 to yield

$$12\pi kP_bR^2R' + 4\pi P'_bR^3 = 3m'_gh_{fg}(k-1) \tag{19E.2-18}$$

Combining this equation with the mass rate Equation 19E.2-15 and the Rayleigh bubble Equation 19E.2-14 forms a system of three differential equations in the two dependent variables P and R. These equations were solved numerically assuming values of the constants which are typical of a corium-steam system. The initial conditions and other assumed parameter values are shown in Table 19E.2-17. A hydrodynamic time constant

$$\tau_L = R/R' \tag{19E.2-19}$$

was obtained which is plotted in Figure 19E.2-14a.

Figure 19E.2-14a shows that τ_L is less than the convective heat transfer response times for either nucleate or enhanced film boiling. Therefore, in cases where the heat transfer from the debris droplets is controlled by convection, the surrounding water with a shorter dynamic time response gently expands with the steam bubble without permitting a high pressure difference to form. It follows that steam volume formation for the range of debris droplets shown in Figure 19E.2-14a is primarily determined by the droplet surface heat transfer rate.

(6) Conditions for Self-Triggering

Self-triggering could occur if the mechanical energy ΔW released from a molten debris droplet was sufficient to form additional droplets and mix them with surrounding water. The process of self-triggering is shown in Figure 19E.2-16. A debris droplet of radius r rapidly transfers its thermal energy to an associated water region from which steam is formed. The expanding steam performs a net amount of work on its surroundings. If part of the expansion work is sufficient to form one or more debris droplets and mix them with surrounding water, a propagating event could occur, creating the potential for a steam explosion. The work required to form a debris droplet of radius r is approximately

$$\Delta W_{\sigma} = 4\pi\sigma r^2 \tag{19E.2-20}$$

where

σ = the surface tension of the liquid.

If the work required for mixing a new droplet with surrounding water is conservatively neglected, the condition for triggering is

$$\Delta W = \Delta W_{\sigma} \tag{19E.2-21}$$

An estimate of the expansion work done by a steam bubble which expands to volume V_{∞} is given by

$$\Delta W = (P - P_{\infty}) V_{\infty} \tag{19E.2-22}$$

where P is an average pressure during expansion. The term $(P - P_{\infty})$ can be approximated from the Rayleigh bubble equation, written as

$$P - P_{\infty} = \rho_L (RR'' + 3(R')^2/2) \tag{19E.2-23}$$

The bubble wall acceleration, R", is negative during the expansion. This can be shown from a large amplitude solution to the Rayleigh equation for the sudden appearance of a high pressure bubble which expands adiabatically (Reference 19E.2-13). If the term RR" is neglected, Equation 19E.2-23 yields a higher $P - P_{\infty}$, resulting in a conservatively high estimate of expansion work. The bubble wall velocity is estimated from the maximum size given by Equation 19E.2-10 and the convection response time of Equation 19E.2-12, that is, R_{∞}/τ_h. It follows that the debris droplet radius which could promote self-triggering can be estimated from

$$r_{self\text{-}trig} > \frac{2\sigma}{9\rho_L h^2}(\rho c_v)^{1/3}\frac{(h_{fg})^{5/3}}{[v_{fg}(T_i - T_{\infty})]^{5/3}} \tag{19E.2-24}$$

(7) Conditions for a Steam Explosion

It is assumed that many droplets of molten debris have formed and are in the process of forming a submerged volume of steam, as shown in

Figure 19E.2-17. The steam formation time corresponds to the convection response time τ_h of Equation 19E.2-12, since all droplets transfer heat simultaneously. The total involved water mass M_L provides an equivalent inertia during steam expansion. The equation of motion for M_L can be written as

$$(P - P_\infty) A_L = M_L y'' \qquad (19E.2\text{-}25)$$

where

M_L = $\rho_L A_L L$

The solution for y, based on an average pressure, is

$$y = (P - P_\infty) t^2 / 2\rho_L L$$

for which the approximate hydrodynamic response time for the overlying pool is

$$\tau_p^2 = \left.\frac{2y\rho_L L}{(P - P_\infty)}\right|_{y = L} \qquad (19E.2\text{-}26)$$

The average pressure is estimated from

$$P \approx P_\infty (2V_{total} / A_L L) \qquad (19E.2\text{-}27)$$

The total steam volume V_{total}, if formed at ambient pressure, can be obtained from Equation 19E.2-9 with E' replaced by

$$E'_{total} = NE' \qquad (19E.2\text{-}28)$$

where N is the total number of debris droplets participating in the steam formation process. It is possible for a steam explosion to occur if the condition

$$\tau_p \gg \tau_h \qquad (19E.2\text{-}29)$$

is satisfied. That is, if water motion is sluggish relative to the submerged steam formation, then it is possible to accelerate M_L to high velocity, accompanied by high pressure and impact.

19E.2.3.1.4 Application to ABWR

Table 19E.2-17 gives approximate values of the important parameters, partially explained in Figure 19E.2-18, which were used in evaluating the potential for an ex-vessel steam explosion in the ABWR.

First, the expected corium droplet sizes were found. The debris stream velocity and radius entering the water pool were obtained as V = 11 m/s, and R_0 = 3.7 cm. This then allowed the computation of the stable droplet sizes formed by the debris stream falling through air and water:

r_{air} = 24mm

r_{water} = 0.03mm

However, debris stream broadening in the water will prevent small droplets from forming at the interface. The stream deceleration when entering the water was about 178 m/s^2, based on a cylindrical debris mass of approximately 3.7 cm radius and an equal length. This yielded droplet sizes of

r_{decel} = 2.5 mm

The expected average debris droplet size in the water corresponds to the instability of deceleration. With this information, the important response times for bubble growth were determined:

τ_h = 9.2s convection

τ_c = 1.8s internal conduction

τ_L = 0.006s bubble growth, single droplet

That is, steam bubble growth from debris particle energy is limited by the convective heat transfer rate.

Equation 19E.2-24 shows that self-triggering could occur if a debris droplet radius is greater than 8.3 mm, and therefore is unlikely in the ABWR for the expected droplet size of 2.5 mm.

A conservative, bounding analysis was considered in which it was assumed that a debris mass in the pool was broken up into small droplets of the expected 2.5 mm radius. The resulting heat transfer and hydrodynamic response times were evaluated according to the conditions for a steam explosion given in Equation 19E.2-29.

It was assumed that a debris stream which extended throughout the pool depth was the participating mass, corresponding to

$$M_d = \rho \pi R_0^2 L = 213\text{kg}$$

with a total thermal energy of

$$E = M_d c_v (T_{di} - T_\infty) = 256\text{MJ}$$

The mechanical work of steam expansion is

$$\Delta W = P_\infty E \frac{v_{fg}(P_\infty)}{h_{fg}(P_\infty)} = 19.3\text{MJ}$$

This indicates that about 7% of the total thermal energy is converted into mechanical energy. This is far higher than the 1% to 3% range reported in experiments (Reference 19E.2-5), and is therefore highly conservative for assessing the potential for a steam explosion.

If the participating liquid mass is equivalent to the total in the lower drywell if the passive flooder were somehow to fail open before vessel failure,

$$M_L = \rho_L A_L L = 485,000\text{kg}$$

Then the corresponding hydrodynamic response time is

$$\tau_\rho = 0.38\text{s}$$

The convection heat transfer response time, found previously is

$$\tau_h = 9.2\text{ s}$$

It follows that the condition for a steam explosion given in Equation 19E.2-29 is not satisfied, even for this bounding case, which employs the highly conservative 7% thermal energy conversion. Therefore, the steam explosion potential in ABWR is extremely low.

19E.2.3.2 100% Metal-Water Reaction

An analysis of the capability of the ABWR to withstand 100% fuel-clad metal-water reaction was performed in accordance with 10 CFR 50.34(f). Since the system is inerted, the containment atmosphere will not support hydrogen combustion. Therefore, it is necessary only to consider static loads on the containment.

A simple analysis was performed to determine the effect of the added hydrogen mass and heat energy associated with 100% fuel-clad metal-water reaction. Since the design basis accident for peak containment pressure is a large break LOCA, this accident was chosen as the basis for the analysis.

In order to simplify the analysis several conservative assumptions were made. Since it is not possible to release the hydrogen before the first pressure peak, only the second peak is considered. The hydrogen is distributed in the same manner as the nitrogen. All of the metal-water reaction heat energy is assumed to be absorbed by the suppression pool water. Finally, no credit was taken for the drywell and wetwell heat sinks.

Consideration of 100% fuel clad metal-water reaction results in a peak pressure of about 0.618 MPa. The governing service level C (for steel portions not backed by concrete) / factored load category (for concrete portions including steel liner) pressure capability of the containment structure is 0.770 MPa which is the internal pressure required to cause the maximum stress intensity in the steel drywell head to reach general membrane yielding according to service level C limits of ASME-III, Division 1, Subarticle NE-3220. Therefore, the ABWR is able to withstand 100% fuel clad metal-water reaction as required by 10 CFR 50.34(f).

19E.2.3.3 Suppression Pool Bypass

19E.2.3.3.1 Introduction

This subsection reviews the potential risk of certain suppression pool bypass paths and demonstrates that, with the exception of the wetwell drywell vacuum breakers, they present no significant risk following severe accidents. Because of this insignificance, only the vacuum breakers require further consideration in the ABWR PRA. The approach used in this evaluation is similar to that submitted to the NRC in support of the GESSAR (Reference 19E.2-14) review.

The result of this evaluation is that the remainder of these potential bypass paths contribute a small percentage of the total plant risk and therefore do not need to be specifically evaluated further in the PRA.

(1) Definition of Suppression Pool Bypass

Suppression Pool Bypass is defined as the transport of fission products through pathways which do not include the suppression pool. In such cases, the scrubbing action for fission product retention is lost and the potential consequences of the release are higher.

The potential for suppression pool bypass has been a subject of analysis since the early days of WASH-1400 (Reference 19E.2-7). The “V” sequence which represented a break of the low pressure line outside of the primary containment was one of the more dominant release sequences in WASH 1400.

The IDCOR analysis and BMI-2104 also reviewed sequences in which the suppression pool scrubbing action was not obtained in the release pathway.

In order to review the importance of suppression pool bypass pathways, the potential mechanisms, probabilities and source locations were reviewed to identify where fission products might be released outside of the containment. The analysis has conservatively focused on the station blackout event because it leads to a higher likelihood of suppression pool bypass and because it is considered one of the more probable initiating events for core damage sequences.

The principle conclusion of the review is that, with the exception of certain lines addressed in containment event trees of the PRA, suppression pool bypass pathways do not contribute significantly to risk. Consequently, the probabilistic risk assessment does not require a separate evaluation of bypass sequences, unless the bypass develops during the course of an event, for example, as a result of low suppression pool water level. Such cases are considered in Subsection 19.5.7.

Nevertheless, certain bypass lines which result from piping failures outside of the primary containment are included in this review in order to assess their significance.

(2) Mechanisms for Suppression Pool Bypass

All lines which originate in the reactor vessel or the primary containment are required by sections of 10 CFR 50 to meet certain requirements for containment isolation. Lines which originate in the reactor vessel or the containment are required by General Design Criteria 55 and 56 to have dual barrier protection which is generally obtained by redundant isolation valves. Lines which are considered non-essential in mitigating an accident are also required to automatically isolate in response to diverse isolation signals. Other lines which may be useful in mitigating an accident are considered exceptions to the General Design Criteria (NUREG 0800, Section 6.2.4) and are permitted to have remote manual isolation valves, provided that a means is available to detect leakage or breaks in these lines outside of the primary containment.

A potential mechanism for suppression pool bypass is the “Ex-containment LOCA” which results from the combined failure of a line outside of the primary containment along with the failure of its redundant isolation valves to close. If this combination of events occurs, the operator is made aware of the situation through leakage detection alarms and is instructed by plant procedures to manually isolate the lines, if possible, when the sump water level in areas outside containment exceeds a predetermined point.

Because of these provisions, the probability of suppression pool bypass occurring from the “Ex-containment LOCA” is extremely small since it requires the simultaneous failures of a piping system, redundant and electrically separate isolation valves and the failure of the operator to take action. Subsection 19E.2.3.3.4 summarizes an evaluation of the core damage frequency from ex-containment LOCAs.

The plant design criteria ensure a highly reliable system for containment isolation. Nevertheless, even though there is diversity in the types of valves, all types have experienced failures at operating nuclear plants and certain events, such as station blackout events, may make the early isolation of some lines impossible. This subsection evaluates the significance of bypass paths in order to justify that no additional treatment in the PRA is necessary.

(3) Methodology for Evaluation of Suppression Pool Bypass

The evaluation of suppression pool bypass pathways is based on a methodology which evaluates the potential relative increase in offsite consequence from bypass events over those events with suppression pool scrubbing. Then, knowing this amount of increase, if it can be shown that the probability of bypass is sufficiently low as to offset the increased consequence, the added risk from these pathways will be insignificant.

The justification for this approach is as follows:

Risk = Total [Event core damage Frequency x Consequence] (19E.2-30)

$$= F_{nbp} \times C_{nbp} + F_{bp} \times C_{bp} \qquad (19E.2\text{-}31)$$

Where:

F_{nbp} = The total core damage frequency of non-bypass events

C_{nbp} = The consequence of a non-bypass event

F_{bp} = The total core damage frequency by bypass events which are equivalent to a complete bypass of the suppression pool

C_{bp} = The consequence of a complete bypass event

If the total bypass risk is to be insignificant, the last term in Equation 19E.2-31 must be much less than the first, or:

$$\frac{F_{bp}}{F_{nbp}} \ll \frac{C_{nbp}}{C_{bp}} \qquad (19E.2\text{-}32)$$

The total bypass and non-bypass event frequencies (F) noted above are the total core damage frequencies for these events assuming that all events have the same consequence. Since this is seldom the case, the bypass frequency must be defined such that the proper consequence is applied. This is accomplished through evaluation of flow split fractions (f) as discussed below.

The total bypass frequency can be expressed as:

$$F_{bp} = F_{cd} \times \sum_{i} P_{cbpi} \qquad (19E.2\text{-}33)$$

where:

F_{cd} = The total core damage frequency,

P_{cbpi} = The total conditional probability of full suppression pool bypass path i, given a core damage event

The conditional probability of full bypass can be further refined by the expression:

$$P_{cbpi} = P_{bpi} \times f_i \qquad (19E.2\text{-}34)$$

where:

f_i = The fraction of fission products generated during a core damage event which pass through line i [Subsection 19E.2.3.3.3(1) discusses this term in more detail].

The flow split fraction (f) is defined as the ratio of the flow rate which passes out of the bypass pathway to the total flow rate of aerosols generated during the core melt process. The line flow split reduces the consequence associated with smaller lines due to inherent flow restrictions in those lines as compared with the consequence of larger lines. The flow split fraction accounts for this consequence reduction by reducing the equivalent bypass probability.

P_{bpi} = The conditional probability of bypass in line i [Subsection 19E.2.3.3.3(2) discusses this term in more detail].

The conditional probability of bypass is established through a detailed evaluation of each potential bypass pathway, establishing the failure which must occur for a bypass path to develop and assigning a probability to that failure.

Core damage events result in essentially two types of release: releases which bypass the suppression pool and those that do not. With this simplification, the total non-bypass frequency can also be defined as:

$$F_{nbp} = F_{cd} - F_{bp} \qquad (19E.2\text{-}35)$$

Inserting Equations 19E.2-33, 19E.2-34 and 19E.2-35 into Equation 19E.2-32 yields:

$$P_{bpi} \times f_i \ll C_{nbp} / C_{bp} \qquad (19E.2\text{-}36)$$

Assuming F_{bp} is much less than F_{cd} which would be consistent with the basis for containment isolation.

If Equation 19E.2-36 is satisfied, then the total bypass risk is insignificant.

(4) Criteria for Exclusion of Bypass Sequences in the PRA

As noted previously, if it can be shown that the probability of bypass is sufficiently low as to offset the increased consequence, the risk resulting from release through bypass pathways will be insignificant.

To establish a threshold for this frequency, the consequence ratio (right side of Equation 19E.2-36) was evaluated using the MAAP-ABWR and CRAC codes to establish the approximate order of magnitude for evaluation purposes.

For non-bypass case, the offsite dose from normal containment leakage following core damage was used as a basis. “NCL”, “Case 0” of Section 19.3, is the consequence from normal containment leakage; “Case 7” of Section 19.3 may be used as an approximation of the full suppression pool bypass consequence. There is no credit for plate-out or holdup in the reactor building. Therefore, phenomena such as hydrogen burning in the reactor building will have no impact on this analysis.

The corresponding ratio is based on values in Table 19P-1 and can be used in the evaluation of pool bypass significance. Further evaluation of “Ex-containment LOCA” suppression pool bypass paths in the PRA is not necessary if it can be shown that the total bypass probability is significantly less than this consequence ratio.

19E.2.3.3.2 Identification and Description of Suppression Pool Bypass Pathways

Identification of the potential suppression pool bypass pathways was based on information in the ABWR Standard Safety Analysis Report and supporting piping and instrument diagrams. The potential pathways are shown in matrix form in Table 19E.2-18.

Table 19E.2-1 summarizes the results of reviewing the ABWR design for lines which are potential pathways. For each line the table provides the line sizes, pathways and type of isolation up to the second isolation valve. The bypass lines identified in Table 19E.2-1 were derived from a systematic review of the ABWR P&IDs and other drawings.

Several lines in Table 19E.2-1 were excluded from further consideration on the basis of a variety of judgements discussed in the table notes. In general, the exclusion was based on deterministic rather than probabilistic arguments. For instance, the CUW return line to feedwater and LPFL Loop A were included in Table 19E.2-1 and excluded from further analysis because the bypass path is protected by the feedwater check valves.

The remaining lines are considered potential sources for significant fission product release following severe accidents. Although the probability that these lines could release a significant amount of fission products is extremely small, they are reviewed further in Subsection 19E.2.3.3.3 to assess the importance of these releases.

19E.2.3.3.3 Evaluation of Bypass Probability

Equation 19E.2-36 of Subsection 19E.2.3.3.1 establishes the need for evaluation of the flow splits and failure probability for each line not excluded in Table 19E.2-1. This subsection provides the basis for the evaluation of each of these factors.

(1) Evaluation of Bypass Flow Split Fraction (f_i)

To assess the fraction of aerosol release which bypass the suppression pool a flow split fraction is needed, the flow split fraction (f) is defined as the ratio of the flow rate which passes out of a bypass pathway to the total flow rate of aerosols generated during the core melt process. Two generalized bypass paths have been evaluated:

(a) a path from the RPV which passes to the reactor building with the remainder passing to the suppression pool through the SRVs, and

(b) a path from the drywell to the reactor building with the remainder passing to the suppression pool through the drywell vents.

The flow split fraction may be defined as:

$$f = \frac{W_j}{W_j + nW_k} \qquad (19E.2\text{-}37)$$

where

W_j = the flow rate which passes through the bypass pathway

W_k = the vent flow rate in a single line (SRV or drywell vent) which passes to the suppression pool

n = the number of flow paths to the suppression pool

This can be simplified into the form:

$$f = \frac{f'}{1 + f'} \tag{19E.2-38}$$

where

f' = W_j / nW_k

From the formula for turbulent compressible fluid flow (Reference 19E.2-15)

$$W = 1891\ Yd^2 [(dP)/KV]^{1/2} \tag{19E.2-39}$$

where

W = Flow rate(lb/h) (1 lb = 0.454 kg)

Y = Expansion factor

d = Internal diameter (in) (1 in = 25.4 mm)

(dP) = Differential pressure (psid) (1 psid = 6.89×10^3 Pa)

K = Resistance coefficient = f ''L/D + K'

f '' = Friction factor

L/D = Pipe length to diameter ratio, including corrections for valves, bends

K' = Additional factors for entrance and exit effects

V = Specific volume of fluid (f^3/lb) (1 f^3/lb = 0.0623 m^3/kg)

Solving for f ',

$$f' = \frac{1891 Y_j d_j^2 [dP/KV]^{1/2}}{1891 n Y_k d_k^2 [dP/KV]^{1/2}} = \frac{Y_j d_j^2 [dp/K]^{1/2}}{n Y_k d_k^2 [dp/K]^{1/2}} \tag{19E.2-40}$$

Equation 19E.2-40 may be rearranged to show:

$$f' = (1/n)\,[Y_j/Y_k]\,[d_j/d_k]^2\,[dP_j/dP_k]^{1/2}\,[K_k/K_j]^{1/2} \qquad (19E.2\text{-}41)$$

The expressions in Equation 19E.2-41 were evaluated numerically for the actual line configurations to arrive at the flow split fractions used. The following assumptions were made in this analysis:

(a) Containment pressure following the core melt is assumed to be at an average of 0.411 MPa during the post core melt period. Although the containment pressure could eventually increase to a higher level, the average is used to assess the total amount of release since a release would be occurring throughout this period. This pressure is typical of those calculated in severe accident analyses (Figures 19E.2-2a through 19E.2-12a).

(b) Prior to RPV melt-through, the reactor pressure vessel (RPV) is maintained at a relatively low pressure [0.790 MPa] by the automatic depressurization system or equivalent manual operator action. Four ten inch safety relief valves (ADS valves) are conservatively assumed to be open to release RPV effluent to the suppression pool. This is consistent with the minimum instructions in the EPGs. Ten 0.7 m horizontal vent paths are assumed to be uncovered consistent with the ABWR design configuration. For conservatism the vents are assumed to be one-quarter uncovered.

(c) The pressure drop in the bypass path between the fission product source and the release point is a function of whether the line produces sonic or sub-sonic velocities. For RPV sources, an average 0.790 MPa internal RPV pressure is assumed during the core melt process. This is based on an average 0.411 MPa drywell pressure and an assumed SRV design which closes the SRV when a differential pressure of about 0.345 MPa exists between the main steamline and the SRV discharge line.

Depressurization of the RPV or containment throughout the bypass path is not considered. The assumption is made that pressure is continuously generated during the severe accident in sufficient quantity to uncover the SRV discharge or drywell vents.

(d) The pressure from the non-bypass path between the fission product source and the suppression pool release point depends on the suppression pool level. The suppression pool level is assumed to be higher than normal because of the depressurization of the RPV to the Suppression pool through the SRVs. For RPV sources, the SRVs experience about a 6.0 m (20 ft) elevation head over the SRVs during the core melt process. For drywell sources a 4.5 m (15 ft) elevation head is experienced over the upper horizontal vent. For the station blackout sequence, the effect of ECCS system operation on suppression pool level has been ignored.

(e) The length of lines discharging to the suppression pool and through the bypass paths affects the resistance coefficient in Equation 19E.2-39. Based on the ABWR arrangement drawings this length is estimated to be approximately 25 m (85 ft.). For the drywell sources, the path to the suppression pool is estimated to be 1.5 m (5 ft.).

Other values used in the calculation are listed below:

Parameter	Assumed Value	Basis
Resistance Coefficient	(K=f”L / D)	
Friction Factor	0.011 to 0.018	Reference 19E.2-14 (pg A-25) (Size dependent)
Line Diameter (D)	Various	Line size (Table 19E.2-1)
Other Resistances (K)		Reference 19E.2-14 (pg A-30)
Gate valve	13	
Check valve	135	
Globe valve	340	
Entrance effects	0.5	
Exit effects	1.0	
Expansion Factor (Y)	0.6 to 0.9	Reference 19E.2-14 (pg A-22) (dP, K dep.)

Table 19E.2-19 shows samples results (f ' from Equation 19E.2-41) for a line with two motor-operated valves. In the evaluation of individual bypass lines the actual configuration is used. The evaluation of flow split fractions is considered to be conservative for several reasons:

(i) Bypass release paths would normally be expected to be more restricted than evaluated due to smaller lines, more valves and pipe bends, valves being partially closed or pipe breaks being smaller than the piping diameter.

(ii) No credit is taken for additional retention of fission products in the reactor building, in piping or through radioactive decay.

(iii) For drywell sources, a higher than analyzed differential pressure should exist between the drywell and wetwell. This will lead to lower flows through the bypass path.

(2) Evaluation of Failure Probabilities (P_{bpi})

The failure probabilities used for the detailed calculation of the bypass probabilities are summarized in Table 19E.2-20. The bases for these probabilities are provided below:

(a) Failure to close probability with a common mode failure probability (P_1) is assumed for failure of both valves in a single line to close.

(b) Current operating plants evaluate MSIV leakage against a leakage requirement of 0.33 m^3/h per valve :

		Probability*	
Group	**Leakage**	**Per Valve**	**Per Line**
G1	<0.33 m^3/h		
G2	0.33 m^3/h to 18 m^3/h		
G3	>18 m^3/h		

* Probability is not part of DCD (Refer to SSAR)

The MSIV leakage probability (P2) is assigned a value to correspond to the total line leakage probability. Flow split fractions were determined and a weighted average flow split fraction (weighted by the line leakage probabilities) was determined for use in the evaluation.

(c) The probability of flow passing to the main condenser is judged to be governed by the failure of the bypass valve to close. This probability (P3) is taken from Reference 19E.2-16. Once flow passes to the main condenser, the condenser is assumed to fail (P4) via the relatively low positive pressure rupture disks.

(d) The main steamline break probability (P5) was line break probability (P15).

(e) Normally open pneumatic (P6) and DC motor operated valves (P7) have failed to close. Causes include improper setting of torque switches leading to valve stem failure, undetected valve operator failures and improper packing materials or lubricants. GE has issued several service information letters on valve problems and recommended actions to prevent recurrence of the failures. These failure rates in general are not significantly affected by the valve environments. A common-cause failure among air-operated valves was considered for lines containing redundant series valves.

(f) Normally open AC solenoid and motor-operated valves are subject to a common mode failure (P8) if motive power is unavailable such as during a Station Blackout event. For station blackout events these valves will have a conditional failure probability to close of 1.0.

However, since a loss of power is not expected to result from a break outside containment, an industry failure rate may be used. For those lines with redundant valves, a common cause failure probability was assumed.

(g) Check valves have been observed to fail in such a way as to permit full reverse flow, a condition necessary to permit suppression pool bypass for some lines. Maintenance errors associated with testable check valves have also been observed. The failure rate~~s~~ for check valves allowing complete reverse flow (P9) was based on 7000 hours of operation per operating cycle. A common-cause failure among check valves was considered for lines containing redundant series check valves. Only Feedwater and the SLC paths contain more than one check valve.

(h) When power is available, some normally closed valves open during an event in response to an injection signal, even though the actual injection fails (a requirement for a core damage to occur).

The probability that ECCS valves are not closed by an operator (P10) is considered remote during a severe accident. For station blackout events, since the valves do not open, these lines do not contribute to potential bypass risk.

(i) Some normally closed valves may be open at the beginning of the event. The failure probability (P11) for these valves assumes they are open 4 hours during a 7000-hour operating cycle and that the operator fails to recognize the open path and close the valve.

(j) Some valves may be opened by the operator during the course of the event. Such action may be in compliance with written procedures or it may occur due to confusion in following a procedure. The probability that valves are inadvertently opened (P12) is considered a violation of planned procedures.

(k) Pipe rupture is extremely rare in stainless steel piping. However, carbon steel piping has been observed to fail under certain conditions. The probabilities of line rupture as a function of line size (P13, P14, P15) are taken from Reference 19E.2-14. Except for the CUW break, four line segments outside of the containment are assumed for each bypass line (CUW system estimated to have 50 segments). The intermediate line size [80A to 150A (3 to 6 inches)] break probability is assumed to be twice that of the large line size [greater than 150A (6 inches)].

For pipe failures in an individual bypass line, it was presumed that an undetected break in an unpressurized line could occur at any time. Therefore, the conditional probability of a bypass path was then taken to be the same as the failure rate during a one-year period (which was estimated to be 7,000 hours). This approach of estimating pipe failure probability is judged to be conservative.

Whether the bypass path is the initiator or occurs simultaneously with the event is inconsequential in the evaluation based on the following discussion. The approach taken in the bypass study is to consider the presence of a bypass path as an independent event from the events which caused the core damage in a specific sequence. This approach is acceptable because for large breaks the associated systems are not in general relied upon to prevent core damage and no consequence of these failures have been identified which would affect the systems preventing core damage. Therefore whether the break is an initiator or consequential does not affect the final evaluation. Similarly, none of the systems associated with the smaller bypass lines are associated with preventing core damage. Therefore, they too are not associated with the cause of the core melt.

The ACRS has expressed concern regarding the failure of the CUW suction in combination with failure of the isolation valves to close. The concern is that the isolation valves must close under high differential pressure conditions and the entire secondary containment may be subjected to high temperature and humidity conditions that may fail the ECCS systems. In addition, there may be a flooding situation that could have a high consequence if it leads to an eventual loss of suppression pool and CST inventory or flooding of other ECCS rooms. Such an event would not be consistent with this presumed independence of the assumed conditional probabilities.

If a break in the CUW suction line were the postulated LOCA, the containment isolation valves would be expected to close, terminating the event. NRC concerns over Motor Operated Valve (MOV) closure capability are being addressed as an industry activity. In this evaluation it was assumed that the valves fail to close due to a common cause failure. Should the isolation valves fail to close, the operator can close the CUW remote manual shutoff valve. If all three valves should fail to close, the system arrangement assures that the core is not uncovered and EPGs require depressurization and controlling water level below the break elevation which both slows the break flow and terminates any long-term release from the break. Therefore, if the EPG actions are taken, no additional consequence of the event occur.

The system arrangement routes the CUW lines above the core to avoid a potential siphon of the core inventory. In the event of an unisolated CUW line break, lowering the RPV level to below the shutdown cooling suction and depressurizing the RPV would be sufficient to terminate the break flow without causing core damage. These actions are included in Subsection 19D.7.

(3) Evaluation of Bypass Probability

Table 19E.2-21 summarizes the results of these evaluations. For each potential bypass pathway, it shows the flow split fraction based on the line size and valve configuration, the equation to calculate the bypass probability, the results of the probability calculations using the data from Table 19E.2-20 and the bypass fraction for the line. The table also includes reference to the sketch (Figure 19E.2-19a to 19E.2-19k) which illustrates the potential pathways.

(4) Evaluation of Results

Subsection 19E.2.3.3.1(4) provides a conservative justification that certain bypass paths do not substantially increase the offsite risk. The bypass fraction is shown in Table 19E.2-21, for all potential paths not addressed in the containment event trees.

Potential bypass through the Wetwell-Drywell Vacuum Breakers are included in the containment event trees. (Subsection 19D.5).

Based on the above discussion, it can be concluded that suppression pool bypass paths and Ex-Containment LOCAs not addressed by the containment event trees do not contribute a significant offsite risk and do not need further evaluation in the PRA.

19E.2.3.3.4 Evaluation of Ex-Containment LOCA Core Damage Frequency

(1) Introduction

To provide a separate assessment of the importance of bypass paths, a more comprehensive analysis of the frequency of core damage from LOCAs outside containment was conducted using event tree and fault tree techniques.

Conservative and simplified event trees of LOCA outside containment events were developed and included as Figures 19E.2-20a through 19E.2-20c. The end-point for these trees is core damage with or without bypass of the containment.

(2) Assumptions

The following definitions and considerations were applied in development of the trees.

- V_1—Line Break Outside

  The frequency of piping breaks in small, medium or large breaks outside of containment and which communicate directly with the reactor vessel. The lines are grouped by type of isolation. The basis for each event initiation frequency is the line size and the total number of lines considered. The basis for the pipe break frequency is provided in Subsection 19E.2.3.3.3 (2)(k).

- X_1—Line Isolation

  The conditional probability of automatic isolation valves failing to close given the ex-containment LOCA. Values used and the manner in which probabilities were combined are shown on Figures 19E.2-20a through 19E.2-20c.

- P_1—Operator Action

  The conditional probability that operator fails to act to manually isolate the ex-containment LOCA. Such a failure to act could be due to a lack of instrumentation availability or mechanical failure. For most bypass paths considered, the very conservative assumption was made that no operator action is taken. For ECCS discharge lines and warm-up lines the operator is assumed to act to close an open valve, if needed. The basis for the value chosen [Subsection 19E.2.3.3.3(2)] is based on general operator awareness of the potential for these paths to be unisolated. Although the leak detection system is adequate to alert the operator of a break in the system, instrumentation failure is not considered to provide a strong contribution to the failure probability. For CUW and RCIC line breaks, operator action within one hour was assumed.

- Q_1—Other Divisions not Affected

  For most lines it is conservatively assumed that the LOCA affects the division in which the break occurs. This factor represents the conditional probability that the LOCA also affects the required makeup for core cooling from other electrical divisions. It is assumed that such failure results from environmental effects from flooding or pressurization / steam effects.

  A systematic evaluation of potential cold flooding due to ex-containment breaks was summarized in Appendix 19R, "Probabilistic Flooding Analysis." Flooding in the reactor building is assumed to disable the system affected and potentially flood the Reactor Building corridor, but not disable other makeup equipment due to the water-tight doors contained in the design. The analysis of an unisolated CUW break in Subsection 19R.4.5 shows that no cooling systems will be damaged due to flooding.

  Compartment pressurization and environmental effects of high pressure LOCAs in secondary containment were considered in the development of Figures 19E.2-20a through 19E.2-20c. Equipment in the ABWR design is arranged with consideration of divisional separation. A high energy line break in a division would cause the blowout panels from the division to relieve the initial pressure spike to the steam tunnel. Subsequent pressurization of the room could eventually cause a release of the energy into the adjacent divisions.

  As doors from the corridor and penetrations are forced open, the environment of the adjacent divisions could be affected by the presence of steam. However, the equipment is qualified to 373 K (212°F) and 100% humidity. Where a LOCA could occur in an area adjacent to a separate division, a value was assumed for Q_1, based on engineering judgment, to represent the possibility for failure of these adjacent systems. For the CUW line break outside containment, Q_1, was assumed to be 1.0 because equipment in all three safety divisions will experience a high temperature and steam environment. The impact of this environment is reflected in the coolant makeup unavailability (Q_o).

  For line breaks in the turbine building the effect of the break would not impact the divisional power distribution and, for these sequences, the Q_1 value was judged to be negligible.

  Although line routing are not specified, the analysis assumes that breaks inside reactor building equipment rooms affect the division in which the breaks occur; LOCAs outside of the secondary containment are not assumed to fail a division of equipment.

- Q_o—Coolant Makeup

This factor represents the conditional probability of core cooling failure by all sources of cooling with consideration to those affected by the ex-containment LOCA. The values used are derived from an evaluation of the PRA fault trees.

The conditional probability when one or more electrical divisions are affected were derived by disabling the most limiting division in the LOCA event trees and then calculating the resulting conditional probability. Only the medium LOCA CUW or RCIC line breaks could potentially affect all divisions since larger lines are in containment or the steam tunnel and smaller lines do not contain sufficient energy to affect all divisions.

For LOCAs which occur in the reactor building, the event is assumed to fail the division in which the break occurs. For other LOCAs, such as LOCAs in the turbine building, no divisional impact is assumed.

Consideration of inventory depletion due to the LOCA outside containment is addressed by EPGs which specify that coolant makeup sources using inventory sources outside of containment be used as the preferred source. In the ABWR design small breaks can be accommodated by any of the high pressure coolant makeup systems (RCIC, HPCF B and HPCF C) which are in separate divisions and which draw water from the condensate storage tank. Since condensate is effectively an unlimited supply for small breaks and makeup capability exists, no additional concern is necessary for the small break LOCAs outside of containment.

Medium and large breaks outside of containment can be accommodated by any of the three divisions in the short term following a break without concern for inventory loss in the RPV. All penetrations, except the RPV/CUW bottom head drain (a unique situation addressed separately in Subsection 19.9.1 by an event specific procedure), are above the top of active fuel so that core uncovery due to inventory depletion is not a concern. In the longer term, the break will depressurize the RPV which effectively reduces the loss of inventory from the break to a level well within the makeup capacity of other available systems which makeup from sources outside of containment, such as condensate. Due to the reduction in loss rate through the break, significant time is available for operators to compensate for the usage of water and flooding in the affected area. Furthermore, operators are assumed to follow plant procedures in isolating the break or controlling RPV level to a level below the affected penetration, if necessary. Adequate instrumentation and long term makeup from condensate sources would normally be available.

(3) Conclusion

For each of the event trees shown in Figures 19E.2-20a through 19E.2-20c the total non-bypass and bypass core damage frequencies were evaluated.

Ex-containment LOCA events without bypass represent a small fraction of the total core damage frequency . Therefore, they are justified as not being further evaluated in the PRA.

Although the consequence from bypass events is greater than for non-bypass events, the total frequency of bypass events concurrent with core damage is extremely small. The core damage frequency of ex-containment LOCAs with bypass are an extremely small percentage of the total evaluated core damage frequency. Large LOCAs can be excluded from further consideration on the basis of low probability. Exclusion of Intermediate and Small bypass sequences is based on the additional consideration of the reductions in consequences of the ex-containment LOCAs due to the flow splits provided by restrictions due to line sizing. This is discussed in Subsection 19E.2.3.3.3.

In addition, since significant margin exists between the current PRA results and the safety goals, it can be concluded that the bypass events do not significantly contribute to the offsite exposure risk.

19E.2.3.3.5 Suppression Pool Bypass Resulting from External Event Analysis

The effect of external events on the Suppression Pool Bypass evaluation is discussed in Appendix 19I to determine if a significant potential for bypassing the suppression pool results from component failures induced by a seismic event. Only seismic events were considered to provide a significant challenge to the creation of bypass paths beyond that already considered in the PRA.

19E.2.3.4 Effect of RHR Heat Exchanger Failure in a Seismic Event

A failure of the RHR heat exchanger mounting can conservatively be postulated to shear the pipe between the RHR pump discharge and the RHR heat exchanger. About 30 minutes is available for the operator to close the RHR suction valve to the suppression pool. If no power is available, or if the operator failed to close the suction valve(s), the suppression pool will drain to the RHR equipment rooms.

This subsection describes the analysis of these sequences which concludes that structural integrity of the RHR equipment room will be maintained and that, in effect, the suppression pool scrubbing is transferred from the wetwell to the RHR equipment rooms.

19E.2.3.4.1 RHR Equipment Room Flooding

The RHR equipment rooms drain to a sump. This sump also receives drains from the HPCF equipment room (in two cases) and from the RCIC room (in one case).

The analysis of the resulting loads in the RHR equipment and the basis for concluding that the room will remain intact is described in the following paragraphs.

19E.2.3.4.2 Dynamic Loads Induced by Chugging

The dynamic loads on the RHR equipment room wall resulting from a postulated break of the RHR pump discharge pipe were estimated using applicable test data. The most limiting wall is assumed to run parallel with the discharge pipe at a distance of approximately 1.22 m (4 ft) from the piping. The length of this wall is 13.1 m (43 ft), the height is 6.1 m (20 ft). The RHR pump discharge piping is assumed to run 0.61 m (2 ft) above the equipment room, with the rupture located exactly opposite to the middle of the wall (worst case).

The dynamic loads result from the discharge of the containment atmosphere through the broken pipe into the water pool in the RHR equipment room. It was conservatively assumed that the entire volume of the equipment room was flooded with the suppression pool water.

The gas discharged from the broken pipe will be initially almost pure nitrogen, later a mixture of nitrogen and steam with decreasing nitrogen content, and finally, after all the nitrogen is purged out of the containment, pure steam. The mean flow rates through the broken pipe will be a function of pressure in the containment, which in turn will initially depend on the accident scenario. In the long term, however, the mass flow rate will be driven by the steam generated from the decay heat. It is assumed that there will be no pressurization of any airspace remaining in the RHR equipment room.

This situation is similar to the discharge of the drywell atmosphere through the drywell vents into the suppression pool during a LOCA. The test results from LOCA tests conducted by GE for a wide range of break sizes demonstrate that the highest wetwell pressure loads due to this discharge are experienced late in the event during the "chugging" regime characterized by low mass fluxes <48.9 kg/s•m^2(<10 lbm/s•ft^2) and high steam/air ratios (<1% air). At higher mass fluxes the "condensation oscillation regime" and higher air contents, the loads were substantially lower.

To estimate the chugging loads on the RHR room wall, the Mark III PSTF test data were used. The Mark III data were chosen because of the horizontal orientation of the vents and because no pressurization of the airspace above suppression pool which approximates the situation in the ABWR RHR room. The highest chugging loads on the wall seen during the Mark III experiments were 0.790 MPa. These pressures were observed on the drywell wall adjacent to the vent exit into the pool. Because of the close proximity of the pressure sensor to the source of the pressure disturbance (the collapsing steam bubble) this pressure can be considered to be the actual bubble pressure.

The period between the pressure spikes was typically 1 to 5 seconds or more. Following the peak pressure spike, a series of lower amplitude pressure oscillations were observed, with frequencies that were in the range of the natural frequencies of the vents and water pool. The maximum amplitude of these oscillations was typically less than 10% of the maximum pressure spike.

Given the RHR equipment room geometry, and using a conservative pressure attenuation model (supported by the Mark III experimental data), it was calculated that the peak, spatially averaged, dynamic wall pressure will be below 0.028 MPa, if the maximum bubble pressure of 0.790 MPa is assumed. With higher flowrates and higher non-condensable contents in the discharge, the loads are expected to be lower. Therefore, this conclusion should also cover a range of severe accidents during which non-condensable gases (e.g., H_2, CO_2) are generated from metal-water reaction and / or corium-concrete interaction.

19E.2.3.4.3 RHR Equipment Room Structural Integrity

The structural integrity of the RHR equipment room structure was evaluated for the loads resulting from the seismic-induced flood. The RHR room is located at the reactor building basemat level in each of the three divisions. The wall is approximately 13 m (43.64 ft) wide, 6.5 m (21.32 ft) tall, and 0.5 m (1.64 ft) thick.

The compartment walls were examined for their abilities to withstand a 2g earthquake which is the median peak ground acceleration required to fail the heat exchanger mounting causing the postulated room flooding. No structural damage is predicted, although some concrete cracking is inevitable. After the earthquake, the wall would be structurally sound to withstand the loads imposed by flooding as described below.

The seismic-induced flood imposes loadings to the room in the form of hydrostatic and hydrodynamic pressures. It is assumed that no damaging aftershocks would occur during flood. From the above discussion the most significant hydrodynamic load is caused by chugging. The pressure transient on the wall is idealized by a sharp pressure spike with a maximum amplitude of about 0.028 MPa preceded by a half cycle sinusoidal and followed by a decay sinusoidal with much smaller amplitudes.

To find the dynamic effect on the wall response, the pressure transient described above is approximated by an isosceles triangular pulse with a peak value of 0.028 MPa. For this type of loading, the maximum dynamic amplification factor is about 1.5 regardless of structural frequencies. For conservatism, the equivalent static chugging pressure is taken to be 0.143 MPa.

Under the combined hydrostatic pressures of a fully flooded condition and equivalent static chugging pressure uniformly distributed over the entire wall, the stress analysis was performed and the resulting maximum moment is found to be about 44% of the ultimate moment capacity in accordance with the ultimate strength design method for reinforced concrete. The maximum shear stress is within the ACI-349 code allowable. The leaktight RHR room access door was also evaluated and is found to be structurally sound against flood loadings.

In summary, the structural integrity of the RHR room can be maintained for the seismically-induced flood.

19E.2.3.5 Impact of Flashing During Venting

The adoption of the Containment Overpressure Protection System (COPS) in the ABWR design limits the potential release from the containment in the unlikely event that containment failure is immanent. In the absence of significant suppression pool bypass, the fission products will be scrubbed as they pass through the suppression pool. The predominant conditions in the suppression pool yield very high decontamination factors for all fission products except the noble gasses. Given the extremely low releases from the gas space which result from suppression pool scrubbing before the rupture disk opens, the potential release resulting from the rapid depressurization at the time the rupture disk opens must be considered.

It is shown that the initial decompression wave generated by the opening of the COPS rupture disk is not large enough to lower the pool surface to its saturation pressure and therefore no initial swell due to vapor flashing occurs. The suppression pool does not start to flash until the wetwell has depressurized to the pool saturation pressure.

Comparison of the time constant for blowdown with the time constant for the pressure wave propagation around the wetwell demonstrates that the suppression pool acts as a one-dimensional body for the purpose of this analysis. This allows the calculation of the pool swell height. Comparison of this level to the location of the containment penetration indicates that there is no potential for water to enter the COPS piping. This eliminates the need for consideration of both water loads on the COPS piping and of fission product transport with water. It is also necessary to consider the potential for water droplets to be entrained from the pool surface and carried into the COPS piping. Calculation of entrainment at the surface of the suppression pool is considered using the work of Rozen, et. al. (Reference 19E.2-17) and is found to have an insignificant impact on fission product release.

19E.2.3.5.1 Response of Suppression Pool Surface to Decompression Wave

19E.2.3.5.1.1 Summary

Sudden opening of the containment overpressure protection system (COPS) rupture disk causes a gas discharge from the ABWR pool airspace. The associated decompression wave which enters the airspace spreads to the pool surface. It is necessary to determine how the pool surface responds to the arriving decompression. If the decompression wave causes pool pressure to fall below the saturation pressure, rapid vapor formation would cause the pool to swell as a flashing steam / water mixture. However, if the arriving decompression does not cause the pool pressure to fall below its saturation value, flashing would not occur, and the pool would respond as a compressed liquid.

The theoretical modeling used to determine pool response from operation of the COPS includes prediction of:

- The gas discharge rate
- The velocity and decompression disturbances originating where the COPS enters the airspace
- Expansion of the decompression into the airspace, and its attenuation with distance
- Decompression transmission from the airspace into the pool at the water surface
- The pool water dynamic and thermodynamic response

It was found that the originating decompression wave entering the containment airspace was 38.8 kPa, dropping below the initial 721 kPa air pressure. The decompression wave leaving the COPS pipe of 0.275 m (0.9 ft) radius would reach the pool surface a distance of 4 m (13.12 ft) away, attenuating from 38.8 kPa to 2.67 kPa. Since sound speed and density of water are much higher than corresponding values in air, a decompression wave entering the water is nearly twice that arriving in the air, or about 5.34 kPa. The decompression is not large enough to cause pool pressure to drop below its saturation pressure of 330 kPa at its initial temperature of 410 K, or 137°C (738 R or 278°F). The pool surface would move upward at only 0.0044 m / s (0.014 fps) for the transmitted decompression.

19E.2.3.5.1.2 The Gas Discharge Rate

The COPS pipe has a radius R and area A. The open COPS rupture disk has a flow area a. Since the airspace pressure P_o is 721 kPa and discharge is into the atmosphere at 101 kPa, the initial air flow is expected to be choked in the valve throat at a choked mass flux of (Reference 19E.2-37)

$$G_{gc} = \left(\frac{2}{k+1}\right)^{(k+1)/2(k+1)} \sqrt{kg_0 P_0 \rho_{g0}} \qquad \text{(19E.2-41a)}$$

The quasi-steady mass flow rate through the pipe and valve is expressed as

$$m = G_{gc} a \qquad \text{(19E.2-41b)}$$

Assuming isentropic flow from the airspace to the throat, and expressing the airspace sound speed as:

$$C_{g0} = \sqrt{(kg_0 P_0)/\rho_{g0}} \qquad \text{(19E.2-41c)}$$

the discharging mass flow rate is obtained in the form,

$$\frac{m}{AC_{g0}\rho_{g0}} = \left(\frac{2}{k+1}\right)^{(k+1)/2(k+1)} \frac{a}{A} \qquad \text{(19E.2-41d)}$$

19E.2.3.5.1.3 Disturbance Entering the Airspace

It is assumed that the COPS valve opens instantly, causing an instantaneous quasi-steady flow in the attachment pipe. This assumption gives the maximum pipe velocity, which corresponds to a maximum initial decompression wave.

Acoustic theory can be applied if pressure disturbances do not create Mach numbers much greater than 0.2. An area ratio of a / A = 0.132 (diameter ratio of d / D = 0.364) with an airspace state described by

P_0 = 721 kPa

T_0 = 410 K (278°F)

g_0 = 6.16 kg / m^3 (0.384 lbm / ft^3)

C_{g0} = 406 m / s (1332 fps)

yields a gas velocity in the pipe of 31 m / s (102 fps). The corresponding mach number is 31 / 406 = 0.076, which justifies treating the decompression as an acoustic wave.

It is further assumed that the discharge begins suddenly, imposing the pipe flow velocity of 31 m / s at its entrance. In order to employ spherical propagation of the acoustic wave, an imaginary hemisphere of pipe radius R = D / 2 = 0.55 / 2 m = 0.275 m (0.902 ft) has twice the pipe flow area, reducing the entrance velocity on the hemisphere to 31 / 2 = 15.5 m / s (50.8 fps). The acoustic equation,

$$\delta P_0 = \frac{C_\rho \delta V}{g_0} \qquad (19E.2\text{-}41e)$$

can be employed to show that the corresponding decompression disturbance is P_0 = 38.8 kPa (5.6 psid).

19E.2.3.5.1.4 Expansion Into Airspace

The acoustic decompression wave propagation is governed by the spherical wave Equation19E.2–41b,

$$\frac{\partial^2 P}{\partial t^2} - \frac{C^2 \partial}{r^2 \partial r}\left(r^2 \frac{\partial P}{\partial r} \right) = 0 \qquad (19E.2\text{-}41f)$$

with the boundary and initial conditions at r = R of

$$P = P_0 - \delta P_0 \qquad (19E.2\text{-}41g)$$

a boundary condition as r approaches infinity of

$$P = P_0 \qquad (19E.2\text{-}41h)$$

and initial conditions at t = 0 of

$$P = P_0 \qquad (19E.2\text{-}41i)$$

$$\frac{\partial P}{\partial r} = 0 \qquad (19E.2\text{-}41j)$$

A solution for the outgoing decompression wave is given by

$$\frac{\delta P}{\delta P_0} = \frac{R}{r} e^{-(Ct/r - r/R + 1)} H_s\left(t - \frac{r - R}{C} \right) \qquad (19E.2\text{-}41k)$$

where H_s is the Heaviside step function, which is zero for negative arguments, and 1.0 for positive arguments. A pressure disturbance in the airspace will travel from r = R to another r at the acoustic speed C, which requires a time (r - R)/C. When it does arrive, H_s is 1.0, and the arriving magnitude is

$$\delta P = \frac{R}{r} \delta P_0$$

It is seen from Equation (19E.2-41k) that even after the decompression arrives at r, its amplitude decays exponentially with time. This feature is excluded from the analysis for conservatism.

If the water surface is a distance r = 4 m away from the COPS pipe, the arriving decompression wave will have an amplitude of only 2.67 kPa.

19E.2.3.5.1.5 Transmission into the Pool

The arriving decompression wave undergoes both simultaneous transmission and reflection at the pool surface interface. Acoustic theory for a plane wave arriving at a flat surface discontinuity of density and sound speed gives the ratio of transmitted to oncoming pressure disturbances as

$$\frac{\delta P_{transmitted}}{\delta P_{oncoming}} = \frac{2}{1 + \rho_1 C_1 / \rho_2 C_2} \qquad (19E.2\text{-}41l)$$

where subscripts 1 and 2 refer to the airspace and water in this case. A water density and sound speed of 1000 kg/m3 and 1220 m/s yields a transmitted/oncoming pressure of

$$\frac{\delta P_{transmitted}}{\delta P_{oncoming}} = 1.99$$

That is, the decompression wave arriving at the pool surface nearly doubles from the oncoming value to 5.34 kPa. The plane wave analysis employed here is based on left and right traveling waves which add to satisfy continuity and energy conservation at the interface (Reference 19E.2-38). A similar analysis for spherical waves is obtained from the method of images to provide a plane surface of symmetry. The local pressure transmission and reflection amplitudes are the same as those obtained from the plane wave analysis (Reference 19E.2-38).

19E.2.3.5.1.6 Water Dynamic and Thermodynamic Response

The 5.34 kPa decompression wave transmitted into the water pool does not lower the initial 721 kPa pressure anywhere near the 330 kPa saturation pressure. Therefore, the arriving decompression cannot cause rapid pool flashing and swelling. Steam formation will occur in the pool later when continued decompression of the airspace lowers the pressure below saturation.

The water is expected to respond acoustically to the arriving decompression, taking on a velocity obtained from Equation 19E.2-46, written for the liquid as

$$\delta V_L = \frac{g_0 \delta P}{\rho_L C_L} \qquad (19E.2\text{-}41m)$$

where subscript L refers to the water, and δP is the transmitted pressure disturbance. The resulting pool velocity is only 0.0044 m/s (0.014 fps).

19E.2.3.5.2 Critical Time Constants for Blowdown Response

The time constant for the depressurization of the wetwell airspace is calculated from critical flow considerations. Comparing this value to the time constant for propagation of a pressure wave around the wetwell annulus allows one to determine if non-uniform effects in the suppression need to be considered in calculating the suppression pool response.

The depressurization time constant for the wetwell airspace is estimated based on the critical flow through the rupture disk opening and the ideal gas law. There are two sources of steam to the wetwell airspace: the blowdown through the vent system of steam and non-condensable gas from the drywell, and the boiling or steaming of the suppression pool which results from the pressure decrease. If both of these sources are neglected, the time constant for the depressurization of the wetwell will conservatively be underestimated. If one further neglects the effects of any temperature change which results from the blowdown (a second order effect), the rate of depressurization is:

$$\frac{dP}{dt} = \frac{0.665ART\sqrt{P\rho_g}}{V_w M_{a,W}} \qquad (19E.2\text{-}42)$$

where:

P	=	pressure
A	=	rupture disk flow area
R	=	universal gas constant
ρ_g	=	density of gas
V_w	=	volume of wetwell airspace
$M_{a,w}$	=	molecular weight of gas species in wetwell.

Conservatively assuming the wetwell vapor space has only steam, for a blowdown from 0.65 MPa to atmospheric conditions, the assumptions above yield a time constant on the order of 9 minutes. A typical time constant for a pressure wave going around the torus which comprises the wetwell is about 0.5 seconds. Comparison of these two numbers indicates clearly that the entire suppression pool will participate in the blowdown. Thus, two dimensional effects may be neglected.

19E.2.3.5.3 Pool Swell

In order to maximize the potential level in the suppression pool, the analysis assumes that the firewater system has added enough water to fill the pool. The level in the suppression pool rises above the bottom of the vessel because water is transferred from the drywell to the suppression pool. Two sources of steam which may lead to level swell are included in this discussion. The first steam source is the flow from the drywell through the connecting vents into the suppression pool. The second source of steam which could lead to level swell is the flashing of the pool itself as the system depressurizes.

19E.2.3.5.3.1 Pool Swell Due to Suppression Pool Flashing

Pool swell due to the flashing of the suppression pool may be estimated by use of a drift flux model. Vapor flashing during the wetwell depressurization will be slow and vapor formation will not occur explosively. The suppression pool depressurization rate is much slower than LOCA depressurization rates so the effect of bubble acceleration is less severe. The drift flux model is regularly applied in licensing calculations for LOCA depressurizations occurring over approximately 100 seconds. The suppression pool depressurization occurs over approximately 500 seconds, and begins at a much lower pressure than typical LOCA situations. The drift flux model is appropriate for this scenario because the suppression pool depressurization rate is slow.

The drift flux calculation neglects the contribution of the vapor in the wetwell air space to the flow out of the rupture disk. All vapor flowing out of the rupture disk is assumed to flow from the surface of the pool. This assumption is conservative because it maximizes vapor generation in the suppression pool.

A uniform void generation rate is assumed at each point in the liquid. The average void fraction is then given by:

$$\bar{\alpha}_p = \frac{j_g / U_\infty}{2 + C_0 j_g / U_\infty} \qquad (19E.2\text{-}43)$$

(Reference 19E.2-1) where the mass flow rate, W_p, at the top of the pool determines the superficial gas velocity:

$$j_g = W_p / A\rho_g \qquad (19E.2\text{-}44)$$

and the drift velocity, U_∞, is given by:

$$U_\infty = 1.53\left[\sigma g\left(\frac{\rho_1 - \rho_g}{\rho_1^2}\right)\right]^{1/4} \tag{19E.2-45}$$

where:

σ	=	Surface tension of liquid
g	=	Acceleration due to gravity
ρ_l	=	Density of liquid

Then, by assuming the mass of the pool is approximately equal to the initial pool mass, the average void fraction is used to calculate the average pool height:

$$h = \frac{h_0\rho_1}{\bar{\alpha}_p\rho_g + (1-\bar{\alpha}_p)\rho_1} \tag{19E.2-46}$$

where:

h_0	=	initial pool height.

19E.2.3.5.3.2 Pool Swell Due to Flow From Drywell

A drift flux model is also used to determine the void fraction in the region of the pool above the horizontal vents due to flow from the drywell. The horizontal vents are located at the inner wall of the suppression pool annulus. If quenching of steam in the suppression pool (which is subcooled at the onset of the blowdown) is neglected, the void fraction in the region above the vents is a constant:

$$\alpha = \frac{j_g/U_\infty}{1 + C_0 j_g/U_\infty} \tag{19E.2-47}$$

(Reference 19E.2-1) where the terms are analogous to those defined for Equations 19E.2-43 through 19E.2-45, but now refer to drywell conditions. Comparison of Equation 19E.2-47 to Equation 19E.2-43 indicates that the pool swell elevation is much more sensitive to through flow from the drywell than it is to flashing of the suppression pool.

After the void fraction has been determined, the pool level can be calculated using the relationship in Equation 19E.2-46. However, the difficulty in applying these equations to the case with flow from the drywell is the determination of the appropriate area which participates in the pool swell. Therefore, in order to determine if pool swell is a concern, the problem is considered in reverse. That is, the increase in pool height needed to raise water to the elevation of the vents is assumed to be present. This allows the calculation of a void fraction and effective area for flow. If one then assumes that there is a semi-circular region of influence around each of the vents, the critical radius may be determined:

$$r = \sqrt{\frac{2A}{10\pi}} \qquad (19E.2\text{-}48)$$

If this value is less than the distance between the inner and outer walls of the suppression pool, then pool swell is not expected to lead to carryover of water into the COPS.

19E.2.3.5.3.3 Steam Source

The gas flow through the rupture disk comes from three possible sources: the wetwell vapor space, the drywell vapor space and flashing of the suppression pool. In this calculation of pool swell, the wetwell vapor source is neglected. This results in a somewhat conservative estimate of the pool swell. In order to determine the fraction of flow from each of the sources, the response of the suppression pool and the drywell to a change in wetwell pressure is calculated. Comparison of these values allows the ratio of the flow rates from suppression pool flashing and drywell throughflow to be determined.

The pool flashing rate is determined by consideration of the conservation of energy equation in the suppression pool:

$$\frac{d}{dt}(m_p h_f) = W_p h_g \qquad (19E.2\text{-}49)$$

where:

m_p = mass of water in the suppression pool,

h_f = specific enthalpy of saturated liquid,

h_g = specific enthalpy of saturated vapor.

Taking the derivative on the left hand side of the equation and introducing the derivative of enthalpy along the saturation curve, one concludes that:

$$W_p = \frac{m_p \frac{dh_f}{dP}}{h_{fg}} \dot{P} \tag{19E.2-50}$$

The ideal gas law is used in the drywell to derive the relationship:

$$W_D = \frac{\dot{P} V_D M_{a,D}}{RT_D} \tag{19E.2-51}$$

where all terms were defined previously and the subscript D refers to the conditions in the drywell.

The ratio of the flow rates from the drywell to pool flashing is found by combining Equations 19E.2-50 and 19E.2-51:

$$\frac{W_D}{W_P} = \frac{V_D M_{a,D} h_{fg}}{RT_D m_P \frac{dh_f}{dP}} \tag{19E.2-52}$$

Pool swell is of chief concern for cases in which the firewater addition system has been used to add water to the containment. The suppression pool mass for this case is about 7.0E6 kg. An upper bound estimate of the mass flow ratio assumes that the drywell contains nitrogen at relatively low temperature 373 K (100°C) and that the suppression pool is hot 410 K (137°C). Under these conditions the flow rate ratio is 0.043. These conditions will not occur in the ABWR, since the drywell cannot be cool when the containment pressure is high. However, this value is useful to gain an understanding of the range of Equation 19E.2-52. The bounding calculation shows that less than 5% of the flow through the COPS is being drawn through the horizontal connecting vents. Therefore, the primary contributor to pool swell is flashing of the suppression pool.

19E.2.3.5.3.4 Application to ABWR

The scenarios used in the suppression pool level swell calculations are identical to the accident sequences described in Subsection 19E.2.2.1, Loss of All Core Cooling With Vessel Failure at Low Pressure (LCLP), leading to the opening of the Containment Overpressure Protection System Rupture Disk (R). These results are typical of all initiating events leading to the opening of the rupture disk. The passive flooder actuation scenarios will lead to the highest pool water temperature; thus the passive flooder cases are limiting for the onset of flashing. The firewater addition scenarios will lead to a higher water level swell for given thermal hydraulic conditions because the initial water height is higher.

Figures 19E.2-2 a-j show the drywell and wetwell conditions during a passive flooder actuation scenario. This scenario occurs when the passive flooder (PF) opens to cover the corium. This scenario leads to the maximum suppression pool water temperature. Figure 19E.2-2j shows that the maximum suppression pool temperature is 410 K.

Figures 19E.2-3 a-g show the drywell and wetwell conditions during a firewater addition scenario. This scenario occurs when the firewater system (FS) is actuated four hours after the initiation of the event. This scenario leads to the maximum suppression pool water level. Figure 19E.2-3f shows that the maximum suppression pool level is 14.5 m.

Pool swell is maximized at high temperature (410 K, 137 °C) and high water level (14.5 meters, corresponding to an elevation of 1.35 m). The geometry of the containment and the bounding conditions are shown in Figure 19E.2-25. It is presumed that the rupture disk has just opened. Since the pool swell elevation is more sensitive to flow from the drywell, the upper bound value for the mass flow ratio found above is used. For the limiting suppression pool conditions, the average void fraction due to pool flashing is about 4%. This results in a pool swell of 0.65 meters, corresponding to an elevation of 2.0 meters. Since the bottom of the COPS penetration is at an elevation of 4.25 meters, this mechanism alone will not lead to flooding of the COPS penetration.

If the pool level were to rise an additional 2.25 meters near the outer wall of the suppression pool due to flow from the drywell, the COPS penetration could be flooded. A void fraction of 13% due to through flow from the drywell is required for this additional pool swell. Applying Equations 19E.2-47, 19E.2-48 and the upper bound value from Equation 19E.2-52, one arrives at an radius of 0.84 meters for the region affected by flow from the drywell. This area would be located near to the horizontal connecting vents at the inner wall of the suppression pool. Since the distance between the inner and outer walls of the suppression pool is 7.5 meters, one may safely conclude that pool swell will not threaten the COPS under these conditions.

TRAC calculations have been performed regarding suppression pool swelling during depressurization. TRAC uses two-fluid modeling instead of the drift flux model.

The TRAC level swell model has been qualified against test data. The PSTF experimental blowdown facility was used to provide information on liquid flashing due to a depressurization, and the subsequent swell of the liquid level. When compared with the TRAC model of the PSTF test, it was found that “the two-phase level comparisons show close agreement (± 10%)” (Reference 19E.2-39) The TRAC PSTF qualification validates the TRAC suppression pool swelling results.

A TRAC study of a typical Mark II containment (Reference 19E.2-36) showed a maximum pool level swell height of 0.79 m above the initial pool level. When the Mark II suppression pool level swell is calculated with the drift flux model used for the ABWR calculations, a maximum pool level swell of 2.33 m is obtained. This is almost three times as high as the TRAC two-fluid modeling results. This result demonstrates that the ABWR pool swell calculation is conservative.

19E.2.3.5.4 Carryover Due to Entrainment

The entrainment of water droplets by the steam flow through the suppression pool is potentially a concern since the water could carry fission products through the COPS to the environment. A very simple estimate analysis based on the work by Kutateladze (Reference 19E.2-18) indicates the potential entrainment for a pool of water sparged from below. The threshold for the entrainment of a droplet is based on the velocity of the steam from the surface of the suppression pool:

$$U_{threshold} = 2.7\left[\sigma g\left(\frac{\rho_1 - \rho_g}{\rho_g^2}\right)\right]^{1/4} \qquad (19E.2\text{-}53)$$

Assuming the properties of steam at the rupture disk setpoint, the threshold velocity is about 6 m/s. The superficial velocity from the surface of the suppression pool is 0.02 m/s, assuming all of the flow through the COPS was passed through the suppression pool. Thus, there is more than two orders of magnitude between the superficial velocity which would be observed under the conditions of interest and the threshold for entrainment. This indicates there will be no significant entrainment from the surface of the pool.

A more sophisticated analysis is possible using the work of Rozen, et. al. (Reference 19E.2-17) to estimate even very low amounts of entrainment. This method uses the superficial velocity of steam rising from the pool and the pressure of the system to determine the typical droplet size and the ratio of liquid mass to vapor mass which is entrained from the surface of the pool.

For cases in which the firewater system has been used to add water to the suppression pool, the distance between the bottom of the COPS penetration (elevation 4.25 meters) and the pool surface (elevation 1.35 meters) is 2.9 meters. Assuming the maximum pool swell of 0.65 meters, discussed above, the height between the COPS and the pool surface is 2.25 meters. The correlation selected to calculate carryover is conservative for cases in which the water pool is at least two meters below the COPS penetration.

Using this correlation, the ratio of liquid mass to vapor mass is about 4E-6. If one considers an energy balance on the suppression pool before and after the rupture disk opens, it can be determined that just over one tenth of the suppression pool flashes to steam during the blowdown. Thus, the fraction of suppression pool liquid which might be transported from the suppression pool as a liquid is 4E-7.

The fission products in the suppression pool will exist as a dissolved salt and as sediment on the bottom of the pool. Therefore, the fraction of the fission products which can be carried out the COPS by entrainment will be some fraction less than the ratio of the liquid entrained from the pool surface. However, a release fraction of 4E-7 will not lead to significant offsite dose.

19E.2.3.6 Behavior of Access Tunnels

If core debris is entrained out of the lower drywell and into the access tunnels, it is possible that the integrity of the tunnels could be compromised. This depends on several key factors, such as the amount of debris entrained into the tunnels, whether the debris remains in the tunnels, the heat transfer characteristics between the debris and the tunnel walls, and the strength and loading of the tunnel material.

19E.2.3.6.1 Potential for Debris to Enter Tunnel

Based on the configuration of the lower drywell and the equipment contained therein, it is highly unlikely that debris will be carried into the tunnels unless there is significant debris entrainment. Based on work at INEL, (Reference 19E.2-35) local failure of the lower head is expected. In fact, the drain plug located at center of the bottom head appears to be the dominant failure location. A localized failure should result in a concentrated discharge from the center of the lower head. Immediately below the reactor vessel are the CRD mechanisms. Splashing off of the CRDs is not judged to result in a significant amount of debris transport to the tunnels. Since the debris is likely to be discharged from the center of the CRD array, radial movement through a forest of vertical structures is not expected and transport of the debris outside of the CRD array is not judged to be likely. In fact, the CRDs will tend to columnate the flow, since they are long, vertically oriented and have little change in cross section along their length.

Approximately 6 meters below the bottom of the vessel is the equipment platform constructed from thin steel grating material. This grating is located at about the elevation of the tunnel bottom. No other structures exist at or above this elevation to divert the discharging debris into the tunnel. The grating surface area is small compared to the overall cross sectional area of the lower drywell and the thermal properties of the debris would result in immediate melting of the grate. Further, the center of the equipment platform, where the debris is likely to flow, does not have any grating to allow movement of the CRDs during refueling. Thus, the presence of the equipment platform is not expected to result in significant splashing of the debris into the tunnels.

Using Ishii's methodology, debris entrainment thresholds were only reached for high pressure melt ejection events with very large vessel failure areas (Subsection 19EA.3.6.2). Based on work done at INEL and contained in the expert elicitations in NUREG 1150, and consistent with the DCH analysis in Attachment 19EA, a very small probability is assigned to a large (> 2 m^2) vessel failure area. Combining this with the probability of a high pressure core melt with melt ejection, this scenario constitutes only a small percentage of all core damage events. Thus, the potential for debris entrainment and the transport of debris to the access tunnels is judged to be quite low for the ABWR.

19E.2.3.6.2 Bounding Calculation Assuming Debris Enters Tunnel

Bounding calculations are performed to address those very low probability scenarios in which debris is transported into the tunnels.

Each access tunnel is a circular steel tube, approximately 11 meters long and 4.6 meters in diameter. The thickness of the steel is 2 cm. The bottom of the tunnel is located 7.4 meters below the bottom of the RPV. Since the low water level in the suppression pool is 6.2 meters below the RPV, the portion of the tunnel that will be in contact with core debris will be submerged. The opening from the tunnel to the lower drywell is a rectangle centered on the tunnel axis. The height of the personnel access is 3.55 meters and the height of the equipment access is 3.8 meters; both are 2.2 meters wide. Due to the reduction in area at the lower drywell wall intersection (i.e. the curb that is formed at the entrance to the tunnel), it will be assumed that any debris transported into the tunnel will remain there. The material properties used in this analysis are provided in Table 19E.2-30.

19E.2.3.6.2.1 Amount of Debris Entrained into Access Tunnels

As noted above, debris can only be entrained during a high pressure core melt scenario. NUREG-1150 indicates an upper bound of 40% for the core debris that exits the vessel at the time of vessel failure, albeit at a low probability . It is judged more likely that only 10% will exit during the initial blowdown. Therefore, 10% debris mass is used for this analysis. From previous estimates of debris entrainment discussed above and the fact that the access tunnels are dead end volumes, it is likely that the debris will not be entrained into the tunnels. However, it is assumed that all of the debris exiting the vessel will be entrained along the lower drywell walls, and since each tunnel occupies 6.5% of the perimeter of the lower drywell, 0.65% of the core debris, 1675 kg, will enter each access tunnel during the blowdown. This debris is assumed to be instantaneously carried into the tunnel and will quickly spread to a depth of 3.5 cm. In addition, it is assumed to be molten with little superheat. This is consistent with the direct containment heating analysis contained in Attachment 19EA.

19E.2.3.6.2.2 Heat Transfer to the Tunnel Wall

As soon as the debris comes into contact with the tunnel shell, the interface will immediately assume a temperature that is between the steel and debris temperatures. This initial contact temperature can be calculated by assuming that both the steel and debris are semi-infinite slabs and equating the heat flux at the interface.

$$q''_i = \frac{k_C(T_{C,0} - T_i)}{\sqrt{\pi\alpha_C t}} = \frac{k_{St}(T_i - T_{St,0})}{\sqrt{\pi\alpha_{St} t}} \tag{19E.2-54}$$

For $T_{C,0}$ = 2500 K, and $T_{St,0}$ = 373 K, the initial contact temperature is 1087 K. The interface will remain at this temperature until the thermal boundary reaches the outside of the shell; it will then increase (as will be discussed below) because the heat transfer to the water can not keep up with the heat supplied by the debris. The thickness of the thermal boundary can be expressed as

$$\delta_{thermal} \cong \sqrt{\pi\alpha t} \tag{19E.2-55}$$

It takes approximately 19 seconds for the thermal boundary to reach the water side of the tunnel shell. The thermal boundary in the debris, on the other hand, takes more than 200 seconds to reach the upper surface of the debris.

In order for the steel shell to achieve steady state, the water on the outside of the steel shell must be able to remove heat as fast as it is supplied by the debris. Steady state conduction through steel 2 cm thick with surface temperatures of 373 K and 1087 K is $1.07\text{x}10^6$ W/m^2. The critical heat flux for a downward oriented horizontal plate at one atmosphere is only $4.5\text{x}10^5$ W/m^2 (Reference 19E.2-34). Although the critical heat flux increases with pressure, steady state can not be achieved in this situation.

19E.2.3.6.2.3 Tunnel Wall Integrity

In order to simplify the effect of transient behavior, bounding calculations are performed to estimate the times associated with the transient. As the thermal boundary layers penetrate the materials, the magnitude of the heat flux at the interface is falling. Assuming that the debris behaves as a semi-infinate slab, the time at which the debris begins to supply less than CHF to the interface can be calculated by

$$q''_{CHF} = \frac{k_C \Delta T_C}{\sqrt{\pi\alpha_C t}} \tag{19E.2-56}$$

Approximating the effect of the heat of fusion by assuming the bulk corium temperature to be 3000 K and assuming the interface temperature remains 1087 K, approximately 194 seconds are required for the heat flux at the interface to drop to CHF. If the steel shell can maintain its integrity for longer than this time, the tunnel could remain intact.

The time for the entire thickness of steel to heat to 1200 K (the temperature at which the steel is assumed to lose all strength) can be compared to the time to supply less than CHF. If it is long compared to 194 seconds, the tunnel may remain intact. This time is computed by equating the total heat supplied by the debris to the heat transferred to the steel while the thermal boundary layer is growing plus the heat necessary to raise the steel from its steady state temperature profile to a uniform 1200 K. (Note that the heat transferred during the boundary layer growth takes into account, in a crude way, the heat that is given up to the water prior to dryout. It is assumed that after 19 seconds, critical heat flux has been reached, and there is essentially no heat transfer to the water.)

$$\int_0^t q''_{C,0}\,dt = \int_0^{19} q''_{St,0}\,dt + \rho_{St} C_{P,St} \int_0^{\delta_{St}} (1200 - T_{St}(x))\,dx \qquad (19E.2\text{-}57)$$

The debris is assumed to be infinite during this time, and the steel is assumed to behave as an infinite slab during the first 19 seconds. It is also assumed that the contact temperature is constant, at 1087 K, during the entire time. Solution of this indicates that the tunnel shell will reach 1200 K at approximately 46 seconds. Thus, this rather crude analysis indicates that the tunnel may fail in the unlikely event that debris is entrained.

19E.2.3.6.3 Impact of Tunnel Failure

Failure of the tunnel wall would occur at the lowest point. This would result in a flow path from the lower drywell vapor space into the suppression pool. As indicated earlier, there will initially be at least 1 meter of water above the bottom of the tunnel. Thus, no fission product bypass of the pool would occur. Since the event being considered is a high pressure melt scenario with entrainment of debris, the operator must initiate the firewater addition system in drywell spray mode to prevent high temperature failure of the drywell. This action will indirectly result in additional water being added to the suppression pool as it spills from the upper drywell, through the connecting vent system to the wetwell. Thus, several meters of water would be present above the tunnel failure elevation to provide scrubbing of fission products.

19E.2.3.6.4 Conclusion

It is unlikely for core debris to be entrained or splashed into the access tunnels. A small percentage of all core damage sequences could lead to debris entering the tunnels.

However, in the event that it does, the tunnel steel will reach temperatures that may compromise its integrity. The heat transfer through the thin steel wall is so high that the water on the outside of the tunnel quickly goes into dryout, and the heat can no longer be removed at a rate sufficient to maintain the tunnel integrity.

Failure of the tunnel wall would occur at the lowest point and would result in a fission product release path into the suppression pool. However, since several meters of water will be present above the tunnel failure site, fission products would be scrubbed and no containment bypass would result.

19E.2.4 Supplemental Accident Sequences

In order to quantify the PRA, sequences were analyzed using MAAP-ABWR to assess the effects of recovery. Additionally, some sequences with unusual characteristics, such as those having early containment structural failure, are considered in this subsection.

19E.2.4.1 Time of Firewater System Initiation

The firewater spray initiation times used in the base analyses are simply assumptions used for the purpose of the study. This subsection examines the possible variation in accident progression which would result if the time of spray initiation is varied from that assumed in the base studies.

For example, in some cases the firewater system is not initiated for four hours. As a consequence of the accident progression, as modeled in the CETs, it is known that the operator failed to initiate the firewater injection system. Thus, it is logical to assume that the operator does not initiate the system immediately after vessel failure. If the system were operated immediately, the containment water level would reach the level of the bottom of the vessel somewhat sooner (a maximum of four hours earlier in this example). At this time the operator would be directed to terminate injection. As seen in Figure 19E.2-3a, the containment pressure rises at this time eventually leading to opening of the rupture disk. The change in time of rupture disk opening in this case would be about four hours earlier than that in the base analysis.

On the other hand, if the operator did not initiate the firewater addition system in the assumed four hour period, more of the water initially in the lower drywell would boil off. Eventually, the debris in the lower drywell will begin to heat up. This would lead to actuation of the passive flooder in the lower drywell. This would quench the debris and keep the drywell cool. If at some later time the firewater system is initiated, the thermal mass of the suppression pool would be increased as in other sequences with firewater addition. Since the containment water level would reach the bottom of the vessel later than in the nominal case, the firewater injection would be terminated later, leading to later opening of the rupture disk. The effect on the magnitude of fission product release would be negligible. Although the later time of release might argue for delaying the initiation of the firewater system, the effect on risk is judged to be outweighed by the simplicity of telling the operator to initiate the firewater system as soon as possible in all circumstances.

The operator is instructed to initiate the firewater addition system as soon as it is determined that the water level in the vessel cannot be maintained using other systems. However, if the firewater system is not initialized quickly, the passive flooder will open allowing the lower drywell to be flooded from the suppression pool. Thus, the assumed time for initiation of the firewater addition system does not have a significant impact on the accident progression or on any eventual fission product release.

19E.2.4.2 In-Vessel Recovery

This subsection examines the in-vessel recovery sequence to determine how fission product scrubbing should be modeled for these sequences.

The potential for recovery of vessel injection systems before vessel failure occurs is believed to be an important feature in the mitigation of severe accidents. The sequences with fifth and sixth characters IV in the accident sequence designator in the containment event trees have core melt arrest in the vessel. For the ABWR any of the ECC Systems or the firewater addition system is capable of adding sufficient water to the vessel to prevent core damage, and in theory, to halt the core melt progression once it has begun. It is expected that the ECC Systems can prevent core damage if injection is delayed for as much as half an hour after accident initiation. The firewater system prevents core damage if injection is begun within 20 minutes after the loss of injection.

In MAAP, it is not possible to halt core damage once the first channel region has blocked. It is expected the in-vessel recovery would be possible for at least one hour from the initial loss of injection. Since this occurs very shortly after the onset of core damage, it is very difficult to determine the effects of in-vessel recovery on fission product release directly.

However, the salient feature of core melt arrest in the vessel is suppression pool scrubbing. If the core melt is arrested in the vessel then all of the fission products which leave the vessel must do so via the SRVs. These discharge through quenchers at the bottom of the suppression pool, ensuring fission product scrubbing. Although LOCA events may allow an unscrubbed release into the containment, the probability of a LOCA with failure of the COPS is a very low frequency event and may be neglected.

Fission product scrubbing is also provided if the release is from the wetwell airspace, as would occur for cases with COPS operation. The release fractions associated with this type of release are examined in the base analyses of Subsection 19E.2.2. The results of that study are applied to in-vessel recovery in the effects analysis of Subsection 19E.3.

19E.2.4.3 System Recovery After Vessel Failure and Normal Containment Leakage

This subsection describes the determination of containment leakage when pressures are below the ultimate pressure capability of the containment.

The majority of accidents for the ABWR do not lead to COPS operation or containment structural failure. In these accidents the RHR system is recovered to cool the containment following core damage. These sequences are indicated by the characters HR in the fifth and sixth digits of the accident sequence designator in the containment event trees. Although COPS does not open and there is no COPS operation or structural failure of the containment in these cases, there will still be a small release of fission products due to normal containment leakage. These sequences are binned as NCL in the containment event trees.

To estimate the fission product release associated with normal containment leakage following core damage a sensitivity study was performed using MAAP-ABWR. A loss of all core cooling with vessel failure at low pressure case was chosen for the analysis. The transient was run for three days. The RHR System was assumed to be initiated in suppression pool cooling mode just before the wetwell pressure reached the COPS setpoint. The containment leakage area was chosen such that the leak rate was equal to the technical specification limit of 0.5% per day at rated pressure.

Two cases were run for this sensitivity study, one with leakage from the drywell and the other with leakage from the wetwell. In both cases the first appreciable fission product release occurs at about three hours. The noble gas release fraction at 72 hours is 0.052% for the cases with drywell leakage and 4.4% for the case with wetwell leakage. The magnitude of the noble gas release for the wetwell leakage case is larger than that for the drywell leakage case because the noble gases are forced through the SRVs and wetwell / drywell connecting vents and into the wetwell as the steaming rate in the drywell increases. Thus, the amount of noble gases available to escape through the leak is greater in the wetwell than in the drywell. The CsI release fraction at 72 hours is 2.3E-5 for the case with drywell leakage and less than 1E-7 for the case with wetwell leakage. The volatile fission product release is much less for the case with wetwell leakage because of the benefit of fission product scrubbing provided by the suppression pool.

In quantifying the offsite dose associated with containment leakage, a conservative approach has been adopted. The larger release will be used for each species. That is, the noble gas release of 4.4% will be used with the CsI release fraction of 2.3E-5. While this is somewhat conservative there will not be a large impact on risk.

19E.2.4.4 Early Drywell Head Failure

This subsection describes the modeling of fission product release for cases with early drywell head failure resulting from a high pressure core melt.

In Subsection 19D.5 the frequency of the vessel failing at high pressure leading directly to loss of containment integrity was estimated. These sequences are indicated by the character E in the seventh digit of the accident sequence code. This sensitivity study examines the potential fission product release associated with such an event. Only two types of sequences can lead to this occurrence: a loss of all core cooling with vessel failure at high pressure (LCHP), or a concurrent ATWS and loss of all core cooling with vessel failure at high pressure (NSCH). The LCHP event was chosen to represent this case as it has a higher probability of occurrence.

The history of this event is identical to the LCHP events described in Subsection 19E.2.2.2 until the time of vessel failure. It is assumed that the drywell head fails at vessel failure. There is no significant effect of the drywell failure on the entrainment of corium into the upper drywell, or on the opening of the passive flooder.

The pressure in the containment remains low, usually less than 0.2 MPa. Just before the three hour mark MAAP-ABWR predicts that the drywell tear becomes plugged by aerosols using the Morowitz plugging model. The pressure rises to a peak value of 0.3 MPa before the aerosols are blown out and the containment pressure falls to about 0.2 MPa.

The fission product release for this sequence is much higher than that for the base case (LCHP-PF-P-H). Fission product release begins at the time of vessel failure (2.0 hours). The noble gas release is very slow since most of the noble gasses are trapped in the wetwell. After 61 hours the noble gas release is essentially complete. The volatile fission product release is predominantly governed by the revaporization of the fission products from the vessel internals. After 72 hours this revaporization is nearly complete. The CsI and CsOH release fractions are about 24% and 16%, respectively.

Since the fission product release is significantly higher than that for the base case, this information will be included in the consequence analysis of Subsection 19E.3.

19E.2.4.5 Suppression Pool Drain

This subsection describes the modeling of sequences in which the suppression pool water drains to the RHR pump rooms.

The draining of the suppression pool has been proposed as a potential mechanism for the loss of containment integrity following a seismic event. These sequences are designated with the seventh digit S in the accident sequence code. The water from the suppression pool would flood the pump rooms as discussed in Subsection 19E.2.3.4. This analysis indicates that the pump room integrity will not be lost.

However, there is a pipe chase that leads up from the top of the pump room which has no capacity to withstand high pressure. There is no effective fission product holdup if heat exchanger failure and suppression pool drain occur. This sensitivity study evaluates the fission product release associated with this structural failure mode.

Since this failure is caused by a seismic event it is assumed that if one heat exchanger fails, causing the suppression pool to drain into the RHR pump rooms then all three heat exchangers fail. A comparison of the total floor area of the pump rooms to that of the suppression pool shows that the water level in the pump rooms would rise to more than three meters, assuming equal gas space pressures in the wetwell and the pump rooms. As the wetwell pressurizes due to the accident, the water level would increase further. Therefore, the suppression pool may be envisioned as being displaced to the pump rooms rather than being lost. Any release of fission products to the atmosphere must pass through the RHR suction line, into the pump room, and are then scrubbed in the pool now located in the pump room.

The pump rooms will leak to the corridor through a total of six doors at an assumed rate of 1.14 m^3/h (5 gpm) per door. This is less than 5% of the suppression pool volume per day. This will have negligible impact on the water level in the pump rooms.

For simplicity, the fission product release following heat exchanger failure and suppression pool drain was modeled by assuming a large opening in the wetwell above the normal water level. No significant pressure head was allowed to develop in the wetwell. A loss of all core cooling event with vessel failure at low pressure and passive flooder operation was chosen (LCLP-PF) to model the transient. Dryout of the lower drywell, which could occur if no water was added to that region, was not modeled since the suppression pool elevation in this analysis was sufficient to prevent this occurrence.

The fission product release occurs as fission products are released from the fuel. The fission products exit the vessel through the SRVs. Scrubbing occurs as the fission products are blown through the RHR suction line into the pool in the RHR pump rooms. The only delay associated with any release of fission products which are not trapped in the pool is the dilution effect brought about by a large wetwell gas volume. This effect is analogous to that which would occur in the reactor building.

The release of fission products begins as the fuel begins to melt at about 0.5 hours. The noble gas release was essentially complete at 8 hours. The release of volatile fission products was very small due to scrubbing. The final release fraction of CsI after 84 hours was less than 1.E-5.

19E.2.5 Identification and Screening of Phenomenological Issues

The first step in performing an uncertainty analysis is to identify the key phenomena and their associated uncertainties. To do this, GE has surveyed various sources (References 19E.2-19 through 19E.2-27).

The following provides a summary of the key literature reviews. Some of the severe accident issues are screened out as not being applicable to the ABWR design. At the end, a list of sensitivity issues will be presented for investigation in the ABWR PRA.

19E.2.5.1 Review of NUREG/CR-4551 Grand Gulf and Peach Bottom Analyses

The ABWR containment shares some similarities in design to the Mark III BWR containment. The NUREG-1150 study of Grand Gulf was used to identify phenomena and issues which may need to be addressed in the ABWR uncertainty analysis. In addition, the Peach Bottom (Mark I) analysis was also reviewed for insights. The results of the NUREG-1150 Grand Gulf and Peach Bottom containment analyses are presented below.

19E.2.5.1.1 Grand Gulf

The Grand Gulf accident progression event tree (APET) consists of 125 event headings. The events treated in the Grand Gulf (GG) APET can be grouped into ten categories based on similar accident progression phenomena or characteristics. This grouping is summarized on Table 19E.2-22 along with the Grand Gulf APET events which fall into each group. A summary of the phenomena and issues addressed by each event group are discussed below:

(1) Damage State Grouping Events

The first fifteen events in the GG APET and Event 20 were sorting type events which summarized the plant damage state for a sequence based on the availability of various core injection and containment systems, the timing of core damage, the availability of AC and DC power and the vessel pressure.

(2) Structural Capacity/Initial Containment Status

Four events (Events 16 - 19) summarized the early status of containment integrity and pool bypass and defined the structural capacities of the containment and drywell to quasi-static and impulse loading.

(3) Systems Behavior/Operator Actions

Twelve events defined operator actions and systems availability during the course of the accident progression including whether hydrogen ignitors were available, the status of containment sprays and whether the containment was vented. These event questions were generally asked prior to core damage, during core damage, at vessel failure and late after vessel failure. Other events considered were reactor vessel pressure during core damage, upper pool dump, SRVs sticking open, and restoration of in-vessel injection during core damage.

(4) AC/DC Power Availability

Six events were related to AC and DC power availability/recovery during core damage, following vessel failure and late in the accident progression.

(5) Criticality

One event assessed whether the debris would be in a critical configuration after core injection recovery.

(6) Hydrogen Related Phenomena/Issues

Forty-eight events in the GG APET were related to assessing the impact of hydrogen production and combustion on containment and drywell integrity. These hydrogen event questions were asked at numerous time periods throughout the accident progression: during core damage, at vessel failure, following vessel failure and late in the accident sequence.

The hydrogen production event questions considered hydrogen production in-vessel during core damage and that released at vessel failure and during core concrete interactions (CCI). Several events were included to assess the transient concentrations of hydrogen, oxygen and steam in the drywell and containment throughout the accident progression and to determine if regions were inert (or non-inert) to deflagrations or detonations during various time periods.

For distinct time periods throughout the accident progression the probability of ignition of hydrogen diffusion flames, uncontrolled deflagrations, and detonations were considered along with the efficiencies of the burns and the peak burn pressures (and detonation impulse loads). Additional events compared these loads with the containment and drywell structural capacities and determined if failure or leakage would result.

(7) Containment/Drywell Pressurization and Failure

Twenty-two events assessed containment and drywell pressure and level of leakage resulting from a combination of loads (gradual overpressurization from steam and non-condensable gases) not directly associated with hydrogen combustion. This set of events also assessed the response of the reactor pedestal and drywell to the pressure loads resulting from energetic events which may occur at vessel failure including steam explosions and rapid steam generation in the reactor cavity, blowdown of the reactor vessel from high pressure and high pressure melt ejection.

(8) Core-Concrete Interactions/Pedestal Failure

Seven events were directed at assessing the behavior of debris in the reactor cavity following vessel failure. These events determined whether there was a water supply to the debris, whether the debris was coolable, (and if not) the nature of the resulting CCI and whether the CCI would result in pedestal failure.

(9) Steam Explosion Related

Five events assessed the likelihood and consequences of steam explosions occurring in-vessel or ex-vessel in the reactor cavity. In-vessel steam explosions which failed the upper reactor vessel head, drywell and containment (alpha mode failure) or which failed the lower head of the vessel were considered. The probability of large ex-vessel steam explosions occurring and failing the pedestal (by impulse loading) were also evaluated.

(10) Core Damage Progression and Vessel Breach

Four events were related to assessing the general in-vessel accident progression and vessel failure characteristics. These events evaluated the amount of core debris in the initial core slump, the amount of debris mobile in the lower head at vessel failure, the mode of vessel failure and whether an HPME occurred.

19E.2.5.1.2 Peach Bottom

The major phenomena considered in the Peach Bottom APET which were not addressed in the GG APET were liner melt-through and over temperature failure of the containment (drywell) penetrations.

19E.2.5.1.3 Application of NUREG/ CR-4551 Results to ABWR

Since the ABWR containment is inerted, the GG APET events associated with details of hydrogen production and combustion are not relevant.

The remaining GG APET areas are generally considered applicable to the ABWR. Insights from the GG APET have been factored into the ABWR containment event tree analysis considering differences between the two designs.

The design of the ABWR lower drywell is very different than the Peach Bottom pedestal cavity. The manway used to gain access to the lower drywell is about 5 meters above the floor. The liner, which represents the containment boundary, in the lower drywell is protected by a layer of sacrificial concrete at least one meter thick. Therefore, the debris will not come in contact with the liner in a manner which could lead to liner melt-through. Therefore, liner melt-through is not addressed in the ABWR analysis.

However, in the unlikely event of vessel breach with the vessel at high pressure, it is considered possible that debris transported into the upper drywell may threaten containment integrity as a result of a general heatup of the upper drywell atmosphere if the drywell sprays are not available. As discussed in Attachment 19EA, the debris will not be transported as a contiguous mass. Therefore, the formation of a debris pool in the upper drywell is not a credible event. However, there may be some debris in the upper drywell which could lead to long-term high temperature failure of the containment. The effects of high upper drywell temperature are considered in the CET in assessing the probability of drywell failure.

19E.2.5.2 Review of NUREG-1335

Table A.5 from NUREG-1335 is included here as Table 19E.2-23. This table includes a list of the parameters identified by the NRC to be addressed in an Individual Plant Examination (IPE). All of these will be addressed in the final list of sensitivity analyses to be carried out for the ABWR except for those discussed below:

(1) Combustion in Containment

As noted above, the ABWR containment is inerted and, therefore, combustion will not result in a challenge to containment.

(2) Induced Failure of the Reactor Coolant System

This is mainly an issue for PWRs. The thin walls of the reactor coolant system outside of the vessel may fail to due extended exposure to elevated temperature and pressure. For typical conditions in a BWR during an accident, induced failures are judged to not occur.

(3) Direct Contact of Debris on Containment

Due to the configuration of the ABWR cavity, under a low pressure vessel failure scenario, core debris will be retained in the cavity and will not come in direct contact with the containment boundary. For a high pressure melt scenario, debris that is entrained into the upper drywell will be dispersed and will not result in the coherent flow of debris to the containment shell needed to cause containment failure.

19E.2.5.3 Review of Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B (EPRI).

This document was reviewed to ensure that there were no new issues that had not previously been identified in the above documents. In this document, the following key issues are highlighted for BWR sensitivity analyses:

(1) Hydrogen Generation In-vessel,

(2) Mass of Molten Core released at vessel failure,

(3) CsI re-vaporization,

(4) Debris Coolability,

(5) Containment Failure Mode,

(6) Chemical form of Iodine.

All of these issues are being addressed in the ABWR sensitivity and uncertainty analysis. Some issues are being considered indirectly in the framework of the phenomenological issues they affect. For example, the mass of molten core released at vessel failure is considered in terms of the impact on high pressure melt ejection, direct containment heating and core debris coolability. The chemical form of Iodine is dominated by the effect of suppression pool pH, and is discussed in that context.

19E.2.5.4 Review of ALWR Requirements Document

The EPRI ALWR Requirements Document includes a top-level section referred to as the Key Assumptions and Guidelines (KAG) which defines the manner in which a probabilistic risk assessment is to be performed for advanced plants. Paragraphs 6.2 and 6.3 address those parameters which could be important for the containment response:

(1) Parameters related to hydrogen burns,

(2) Core Debris Coolability,

(3) Pressure capacity of the containment failure location and failure size,

(4) High Pressure Melt Ejection,

(5) Ex-vessel combustible gas generation,

(6) Operator Actions,

(7) Suppression pool scrubbing,

(8) Iodine composition and revaporization.

As stated previously, hydrogen burning is precluded in the ABWR design by use of an inerted containment. Operator actions are being considered in a separate study. The remainder of these issues are included in the ABWR sensitivity and uncertainty analyses.

19E.2.5.5 Summary and Conclusions

Table 19E.2-24 is the list of issues to be investigated in an ABWR sensitivity analysis and has been derived from the documents described above.

19E.2.6 Sensitivity Analysis and Scoping Studies for Phenomenological Issues

Sensitivity studies are performed for the ABWR response to severe accident phenomena in order to determine those issues which may have significant impact on the offsite risk associated with the ABWR design. Given this goal, the ultimate measurement of sensitivity is the offsite dose. At a given site the primary factors which influence the dose are the magnitude and time of release. Therefore, changes in these parameters will be used to determine the need for detailed uncertainty analyses. The issues to be investigated in the ABWR sensitivity analysis is given in Table 19E.2-24.

19E.2.6.1 Core Melt Progression and Hydrogen Generation

This subsection examines the effect of the MAAP core melt progression modeling on the hydrogen generation due to metal-water reaction.

The progression of a severe accident during the period when the core is melting is important in predicting the amount of hydrogen produced during the core melt. The standard melt progression using MAAP is characterized by molten corium forming blockages in the channels which prevent steam from flowing in the channels. This model has two major effects on the melt progression. First, once a region has been blocked, it is impossible for that region to be cooled since no water can flow into the channel to arrest the core melt. Therefore, a core melt can not be arrested in the vessel after the onset of core damage. Secondly, the blockage of the channel prevents steam from flowing past the hot, uncovered portion of the fuel. This serves to limit the metal-water reaction which can occur in the vessel.

Metal-water reaction in a BWR is dominated by the oxidation of zirconium. This reaction has two important consequences in a severe accident. First, the reaction is exothermic, that is it adds energy to the containment. Second, as oxygen from the steam is consumed in the oxidation reaction, hydrogen gas is generated which adds to the partial pressure of the non-condensable gasses in containment. Both of these effects tend to increase the pressurization rate of the containment and shorten the time to fission product release.

A sensitivity study was performed to determine the effects of the blockage model on hydrogen generation. Four cases were examined, two at low pressure (corresponding to LCLP-FS-R-N and LCLP-PF-R-N) and two at high pressure (LCHP-PF-P-M and LCHP-PS-R-N). These cases were identical to their respective base cases, described in Subsection 19E.2.2, except that the model parameter for blockage and hydrogen generation (FCRBLK, Reference 19E.2-1) was set to prevent blockage and to cause the metal-water reaction to continue past the eutectic temperature of the corium.

For the cases at low pressure, the amount of zirconium oxidation increased from 6.3% of the active clad to 15.8%. The time of vessel breach decreased from 1.8 hours to 1.1 hours. For the dominant case with the firewater system operating, the rupture disk opens at 30.6 hours as compared to 31.1 hours for the base case. The CsI release to the environment increases slightly to about 1.E-6; however, the release is still negligible and will not affect the offsite dose. For the case with passive flooder operation, the time of rupture disk opening decreased from 20.2 hours to 16.7 hours. The change in the magnitude of fission product release was negligible.

The blockage model had a more pronounced effect on the amount of zirconium oxidized for the high pressure cases. The fraction of zirconium oxidized for the no blockage case was 35.9%, increased from 5.1% for the case which included the blockage model. For the LCHP-PS-R-N case, the time to rupture disk opening is decreased from 25.0 to 20.0 hours. The impact on the magnitude of fission product release is negligible.

However, for the LCHP-PF-P-H case the effect of an increase in pressure is more significant because leakage through the movable penetrations is assumed to occur at 0.46 MPa. The time fission product release begins for this case is reduced from 18.1 hours in the base case to 7.1 hours with increased hydrogen production. Additionally, the magnitude of the CsI release fraction at 72 hours is increased from 8.7% to 12.5%.

The difference in the effects of blockage on hydrogen production can be best explained by considering the steam flow past the hot fuel cladding. For cases with vessel failure at low pressure, the reactor is blown down before significant heatup of the cladding has occurred. Although the blockage model does not predict complete blockage until shortly before vessel failure, the loss of water in the core region which occurs during the blowdown effectively terminates the metal-water reaction after only 6.3% of the active cladding has been oxidized. The conditions found in the high-pressure vessel failure cases are more conducive to hydrogen generation for three reasons:

(1) Higher steam temperature in the vessel prior to vessel failure,

(2) A greater mass of water in the core region, and

(3) A longer time before vessel failure.

Despite these conditions, the blockage model causes slightly less of the zirconium to be oxidized by MAAP-ABWR for base cases with vessel failure at high pressure than for cases with vessel failure at low pressure. The blockage model used in the base cases presumes that molten material forms blockages in the core which prevent steam flow past the fuel cladding. This terminates zirconium oxidation and limits hydrogen production. The core is fully blocked in the high-pressure melt sequence at 1.2 hours, while in the low-pressure sequence full blockage is delayed until 1.8 hours.

When the blockage model is disabled, the effect of the blowdown becomes more apparent. The lower water level in the low-pressure core melt sequence results in less steam generation from decay heat and less hydrogen generation. Therefore, much more hydrogen is generated in the high-pressure case which has more steam available for metal-water reaction.

In summary, the blockage and eutectic cutoff models used in MAAP reduces the hydrogen generation by a factor of 2 to 7 compared to the cases where these models are not used. For the more dominant LCLP-FS-R-N, LCLP-PF-R-N and LCHP-PS-R-N sequences there is very little change in release and time to rupture disk operation. The only case which resulted in a significant impact on the timing and magnitude of fission product release is the LCHP-PF-P-H sequence. However, examination of the containment event trees in Subsection 19D.5 indicates the probability of this event is very small. Therefore, it is judged that the ABWR severe accident performance is not sensitive to in-vessel hydrogen production.

19E.2.6.2 Fission Product Release From Core

The base sequences shown in Subsection 19E.2.2 use the Cubicciotti model for fission product release from the fuel. If the release from the fuel occurs later than the time predicted by the MAAP model then there could be more airborne fission products available for release from the containment. Also, as the accident progresses, the decontamination factor associated with the suppression pool will decrease as the pool heats up. Conversely, if the release is more rapid, the fission products will pass through the SRVs or the drywell to the suppression pool earlier. This will result in more efficient scrubbing of the fission products.

The effect of the release rate can be modeled in MAAP-ABWR by use of the variable SCALFP (Reference 19E.2-1) which decreases the release rate. Since early releases will result in lower releases from containment, this possibility will not be examined. In order to investigate the sensitivity of the dose to the release rate from the fuel, the LCLP-PF-R-N sequence was run with SCALFP changed from its nominal value of 1.0 to 10.0. This reduces the rate of release by an order of magnitude.

The behavior of the noble gases is not noticeably altered by the slower release. Some variation of the volatile release is observed. The most risk significant of the volatile fission products, CsI, is used as the measure of the behavior of the fission products. In the nominal case approximately 65% of the fission products are carried into the suppression pool shortly after vessel failure. A small percentage of the CsI is found in the drywell at this time, but the majority of the remaining fission products remain in the vessel where they are slowly revaporized. Finally, after the rupture disk opens, the flow through the vessel is sufficient to cause vaporization of the remaining 25% CsI in the vessel. The final release fraction of CsI through the rupture disk to the environment is less than 1.E-7.

The same basic trends may be observed in the behavior of the sequence with SCALFP equal to 10. However, the amount of material in each location varies substantially during the progression of the accident. At the time of vessel failure only 25% of the CsI has been swept to the suppression pool. About 20% of the CsI is still present in the corium which relocates to the lower drywell. The remaining 55% of the material remains in the vessel, either in the fuel itself or on the various cool surfaces of the vessel. Slow release of CsI from the vessel then occurs until the time of the rupture disk opening when the fraction of CsI in the vessel and that in the suppression pool are both about 40%. The amount of fission products in the drywell remains relatively unchanged during this period. As in the nominal case, the remaining CsI leaves the vessel soon after the rupture disk opens. The final release fraction of CsI to the environment is also 1.E-7 for this case.

Despite the large variations in the location of the fission products within the containment during the accident, there is no appreciable variation in release from the containment due to the presence of the containment overpressure protection system in the design. Therefore, no further investigation of the impact of fission product release from the fuel is required.

19E.2.6.3 CsI Revaporization

An important aspect of fission product behavior is the propensity of the aerosols to adhere to the relatively cooler surfaces of the vessel and containment. While the deposition process is fairly well understood, there is considerable uncertainty in the revaporization of the fission products. MAAP assumes that the fission products are revaporized such that the local vapor pressure is consistent with the temperature of the surface. However, it has been proposed that chemical reactions may occur on the deposition surfaces which bind the fission products. This could result in delayed revaporization as the heat sink temperature slowly rises due to the decay heat of the fission products.

In the vessel of a BWR, most of the fission product deposition occurs on the steam dryers. After the fission products are deposited, they slowly begin to heat the dryers due to the decay heat they carry. As the temperature of the dryer increases, the fission products are revaporized. Thus, the impact of chemical binding of fission products to the dryers may be simulated by assuming a larger dryer mass. This causes the dryer temperature to rise more slowly, which in turn slows the re-evolution process. For this study, the dryer mass was doubled and the base sequence LCLP-PF-R-N was recalculated.

As in the discussion of fission product release in Subsection 19E.2.6.2, the CsI will be used as the representative fission product compound. There is no real difference in the timing of the key events. However, comparison of the results of this calculation to the base sequence described in Subsection 19E.2.2.1 shows that there is 2% to 5% more CsI in the vessel at any time during the transient. Nonetheless, there is not a substantial difference in the release fraction from the containment. In both cases the release fraction of CsI at 72 hours is about 1.E-7. Based on this small release fraction, no further consideration of CsI revaporization is necessary.

19E.2.6.4 Time of Vessel Failure

The detailed progression of a core melt during a severe accident is subject to considerable uncertainty. The core melt progression assumed in MAAP retains the corium above the core plate until local core plate failure occurs, resulting in a large pour of core debris into the lower plenum of the vessel. Before this time, water in the lower plenum has very little impact on the accident progression because heat transfer to the lower plenum water pool is very small. Consequently, the lower plenum is nearly full of water at the time of core plate failure.

Due to the large amount of core debris poured into the vessel head at the time of core plate failure, local failure of the instrument tubes is predicted very soon after debris enters the lower plenum. Therefore, there is insufficient heat transfer to the corium to quench it in the vessel; and, molten corium and water are relocated to the lower drywell. Figure 19E.2-2e shows that approximately 85,000 kg of water falls into the lower drywell at the time of vessel failure for a low pressure core melt scenario (LCLP).

In other melt progression models the molten fuel drips down the fuel rods in a process called candling. Under this assumption, it is possible for molten corium to be relocated in the lower plenum slowly, where it is quenched. Vessel failure could then be delayed until all water in the lower plenum is boiled off and the corium is reheated. This delay allows more time for operator action which could prevent vessel failure from occurring.

During the time when the water in the lower plenum is boiling, steam would continue to flow past the fuel rods which could result in increased hydrogen production. The impact of hydrogen production on the containment response is discussed in Subsections 19E.2.3.2 and 19E.2.6.1 which conclude that increased metal-water reaction will not have a significant impact on the offsite risk.

More important than the hydrogen generation is the behavior of the fission products assuming this type of core melt progression. As modeled in the MAAP program, a significant fraction of the volatile fission products are not swept into the suppression pool as they are released from the melting fuel. Rather, they are retained on the relatively cool surfaces in the vessel such as the steam dryers. Later, as these structures heat up, the fission products are revaporized. If the vessel is still intact, the fission products will be swept directly into the suppression pool via the safety relief valves where most of the volatile species will be retained.

For typical sequences using MAAP-ABWR, up to 80% of the volatile fission products are deposited on vessel surfaces just prior to vessel failure. These fission products would be released to the drywell atmosphere very slowly and would only be swept into the suppression pool gradually as steam is generated in the drywell and the containment pressurizes. A low pressure core melt sequence (LCLP-PF-R-N) was rerun with a modified version of MAAP-ABWR in which vessel failure was delayed until the water in the lower plenum had been boiled dry. In this sequence, only about 30% of fission products remained in the vessel at the time of vessel failure. This indicates that MAAP-ABWR base analysis may overpredict the airborne fission products in the drywell. This could result in a significant conservatism for sequences in which the drywell head is presumed to fail. Therefore, the base analysis is conservative as regards the in-vessel effects of debris coolability in the lower head and time of vessel failure.

The assumed core melt progression model will have minor impact on the long-term ex-vessel portion of a severe accident. In the base analyses shown in Subsection 19E.2.2, there is an initial quenching of the corium in the lower plenum followed by a period of time in which the water in the lower drywell boils off. The corium then reheats and the passive flooders open. The influx of water through the flooders quench the corium. If the corium is retained in the vessel until the water from the lower plenum was boiled off, then the initial quenching of debris in the lower drywell will not occur and the passive flooder will open earlier relative to the time of vessel failure. However, this will not have a significant effect on the overall plant response to a severe accident.

The potential for the debris to be cooled in the lower plenum may have an important effect on some of the phenomena which are important immediately after vessel failure. If the debris is not coolable, as was assumed in the base analyses, there may be a large amount of molten debris at the time of vessel failure. If, on the other hand, the debris is cooled in the lower plenum, the penetrations may be expected to fail when only a small fraction of the material is molten. Both of these possibilities are considered in the direct containment heating and debris coolability uncertainty studies contained in Subsection 19E.2.7.

19E.2.6.5 Recriticality During In-Vessel Recovery

A potential challenge to the containment has been identified for accidents in which the core melt is arrested in the vessel. Experiments have indicated the potential for the boron carbide in the control blades to form a eutectic with steel at 1500 K and relocate (Reference 19E.2-28). This is considerably less than the temperature at which the fuel relocates (2500 K). Thus, as the core heats up and begins to melt, there may be regions of the core which are uncontrolled. If the vessel were reflooded after the onset of control blade relocation there is a potential for regions of the core to become critical raising the power level. This could increase the rate of containment pressurization and could potentially lead to operation of the rupture disk or to containment failure.

There are several mechanisms which tend to reduce the potential that the core becomes critical. First, when the cold water is injected into the hot core, it is likely that the any fuel which had been at very high temperature will shatter and form a rubble bed. Analyses performed by PNL (Reference 19E.2-29) indicate that the rubble bed geometry is subcritical since it is undermoderated. Similarly, if there has been substantial relocation of fuel from the upper part of the core, the lower portion of the core will be undermoderated and will probably be subcritical. Finally, if recriticality occurs, boron can be injected via the SLC system to return the core to a subcritical state.

Presuming that the core recriticality occurs as a result of in-vessel recovery, the power level would rise to a level determined largely by the rate of injection. Thus, in effect, a partial ATWS condition develops. As with any ATWS condition, voiding in the core tends to limit the power generation. Thus, the more injection available to the core, the higher the power level. Depending on the precise configuration of the core and control material, it is possible that some of the fuel is damaged locally. However, since coolant is necessary for power generation above the decay heat level, widespread melting of the fuel is inconsistent with the increased power level associated with recriticality.

The steam generated in the core would flow through the SRVs to the suppression pool which would begin to heat up, pressurizing the containment. The emergency operating procedures direct the operator to inject boron via the SLC system and to reduce the water level. Boron injection terminates the recriticality event. Lowering the water level reduces the power generation to a level which can be removed from the containment via the containment heat removal system. If no steps are taken to reduce the power level or to terminate the event, the containment will continue to pressurize leading to opening of the rupture disk and possibly to containment failure. In either case, any fission products released from the fuel in the period in which the core was melting and not retained in the suppression pool could be released from containment.

In order to examine the potential for recriticality to the ABWR containment a low-pressure core melt sequence was examined in detail to estimate the length of time in which recriticality is possible. Qualitative judgements are made about the potential for fuel shattering and the effects of fuel relocation. Additionally, a transient was run using a modified version of MAAP-ABWR to provide a conservative estimate of the minimum time available for the injection of boron should recriticality occur.

19E.2.6.5.1 Potential for Recriticality

In examining the potential for recriticality it is important to recognize that the heating and relocation of the core does not occur uniformly. Variations in the time of uncovery, heat transfer to other structures and the decay power cause the core heatup to progress from the top central portion of the core to the outer and lower regions. In general, once a portion of the core begins to heat up, it heats quickly until it reaches its melt point and begins to relocate.

A MAAP-ABWR calculation of a low-pressure core melt sequence was examined in detail to investigate the heatup and melting behavior of the core. The ABWR core has been modeled using a mesh of ten axial and five radial nodes such that each cell has equal volume. Each node is assumed to have a single temperature. The relocation of the boron carbide is not modeled in MAAP. However, judicious examination of the MAAP analysis can give useful insights.

Before looking at the MAAP-ABWR analysis, consider the possibility for temperature differences between the fuel and the control blades. The source of energy for the heating of the core is the decay heat in the fuel. This leads to the observation that the temperature of the control blades should be less than that of the bulk of the core. Any temperature difference between the control material and the fuel would tend to decrease the time window for recriticality. In order to estimate the temperature difference, a simple radiation calculation is performed which neglects heat transfer to the steam and assumes that the heat transfer between the fuel and the control blade will cause both to heat up at the same rate. Thus,

$$\frac{\dot{Q}_{blade}}{\dot{Q}_{decay}} = \frac{(mc_p)_{blade}}{(mc_p)_{blade} + (mc_p)_{fuel} + (mc_p)_{can}} \tag{19E.2-58}$$

where:

$\dot{Q}$ = Rate of heat addition,

mc_p = Thermal mass.

Using approximate values for the thermal masses, only about 10% of the decay energy will go the control blade.

For an indication of the temperature difference between the blade and the fuel when the blade begins to melt, a simplified radiation heat transfer calculation is performed. The channel box walls are neglected and black body radiation is assumed.

$$\dot{Q}_{blade} = (FA)_{blade}\sigma(T_{fuel}^4 - T_{blade}^4) \qquad (19E.2\text{-}59)$$

where:

FA	=	Effective area for radiation heat transfer,
σ	=	Stefan-Boltzman coefficient,
T	=	Temperature.

The view factor from the control blade to the fuel is taken to be one which neglects the effects near the center of the cross. These assumptions tend to minimize the temperature difference. Assuming a decay heat level of 2% rated power and incipient melting of the control blades, the lower bound on the temperature difference between the fuel and the control blades is about 15 K. Even if different assumptions were made, maximizing the temperature difference, the fuel and control material temperatures would be very close to each other at these high temperatures. Therefore, the use of a single temperature for the fuel and the control blade is a reasonable assumption.

A MAAP-ABWR calculation of a low-pressure core melt sequence was examined to determine the core heatup and relocation characteristics. The core was nodalized using 5 radial rings and 10 axial levels. About 48 minutes after the start of the accident, the temperature in the inner rings of levels 8 and 9 exceeded 1500 K, the temperature at which relocation of the control material might begin. Within a minute levels 6 and 7 also exceed 1500. At 52 minutes, the fuel exceeds the temperature for zirconium melting (2100 K); and, by 55 minutes, there is widespread melting of the core in this region.

After the fuel exceeds the melt point for zirconium, any remaining cladding will be highly oxidic. It is judged to be highly likely that the rapid addition of cold water to the vessel would result in local shattering and relocation of the fuel. Thus, one would not expect a region which has exceeded the zirconium melt point to become recritical.

As time progresses, the region which might be devoid of control material moves downwards. At the same time, fuel from the upper regions of the core also relocates filling these regions with fuel. This reduces the mass ratio of the moderator to the fuel reducing the potential for recriticality. Therefore, it is judged that the critical interval for recriticality is a period of about 7 minutes.

The probability of recovering core cooling in this interval is fairly small. In order for recriticality to occur, there must be a system (or operator) failure that deprives the core of all cooling for about 50 minutes, then injection must be recovered in a time window of about seven minutes. Based on the standard models for recovery of systems and operator error, it is concluded that the probability of this occurrence is small. Therefore, the probability of a recriticality event occurring is small.

19E.2.6.5.2 Implications of Recriticality

Despite the judgement of a low potential for recriticality, an assessment of the effects of a recriticality event are examined. If vessel injection is recovered and some portion of the core becomes critical, the power level would rise above the decay heat level. As long as core injection continues, the fuel would be cooled, thus, no significant fuel damage would occur. However, the additional power generation could increase the rate of containment pressurization. The operator could terminate the recriticality event by initiation of the SLC system or mitigate the event by controlling the vessel injection flow rate.

To bound the impact on the containment, a calculation was performed to determine the earliest time at which the rupture disk could open given a recriticality event. This time indicates the time available to terminate recriticality via the stand-by liquid control system or, as a minimum, to reduce the power level via flow control and slow the rate of pressurization.

MAAP was not designed to analyze recovery scenarios. The model does not contain criticality models, nor can it assess power associated with a degraded core configuration. However, with one minor modification, it is possible to force an ATWS to occur late in an accident which, in effect, is a recriticality event with the entire core uncontrolled. MAAP-ABWR includes the Chexal-Layman correlation for power during an ATWS. This result will bound the power generation in a recriticality event.

The low-pressure core melt scenario discussed above was used to estimate the time to rupture disk opening during a recriticality event. It was assumed that recovery of injection occurred at approximately 50 minutes. In order to determine the minimum time to rupture disk operation, all of the LPFL and HPCF systems were presumed to be available. A full ATWS condition was forced at the time of injection recovery. Based on the Chexal-Layman correlation, MAAP-ABWR predicts a power level of 15%. The containment pressurizes to the rupture disk setpoint about 55 minutes after recovery of injection. During this time the operator has ample indication that the reactor is critical since the containment pressure is rising very rapidly.

This estimate overpredicts the power level and, thus, underpredicts the time until the rupture disk might open for several reasons:

(1) As discussed above, it is expected that only a small region in the core will become critical. Most of the core will be shut down. Thus, the bulk of the core will generate power at decay heat level. The Chexal-Layman correlation represents the condition where the entire core is uncontrolled. Thus, the power level associated with recriticality will be a fraction of the ATWS power predicted by Chexal-Layman.

(2) The Chexal-Layman correlation is based on conditions at rated reactor pressure. At low pressure, the void fraction will be considerably higher. This causes the power level to be substantially reduced at low pressure. Many of the recovery scenarios will occur with the vessel at low pressure. For these cases, the use of Chexal-Layman is conservative. If the vessel is at high pressure, the LPFL systems will not have sufficient head to inject and the power level will be lower than that calculated here.

(3) It is highly unlikely that the all of the ECC systems will be recovered at the same time. As shown in Subsection 19D.5, the dominant core damage event in the ABWR is initiated by a transient with failure of all core cooling (Classes IA and ID). These sequences represent a majority of all core damage events. The simultaneous recovery of all ECC systems is not credible for these scenarios. At a given pressure the power level is directly proportional to the flow rate. Thus, the power should be about one fifth that given here since it is highly likely that only one ECC system is recovered.

(4) Even if all injection systems were to inject, the operator is instructed to reduce the core flow if the power rises above decay heat level. Studies of ATWS at high pressure have shown that the use of flow control will reduce the power to about 4%. Analyses performed for the success criteria in Subsection 19.3.1.3.1(2) show that the containment can be maintained below Service Level C by use of flow control and the containment heat removal system (Reference 19E.2-30).

Thus, one hour is a very conservative estimate of the time until the opening of the rupture disk. It is expected that the actual time until the containment pressure reached the rupture disk setpoint would be several hours. If the operator initiates SLC injection as directed in the Emergency Procedures, the recriticality event would be terminated. Therefore, the risk associated with a recriticality event is not judged to be significant.

19E.2.6.5.3 Conclusions

The potential for recriticality, as well as the implications of its occurrence, was examined. It was concluded that the time window in which recriticality could occur is very small and that only a small portion of the core could become critical at any time of recovery of injection. A very conservative calculation was performed which assumed that the entire core was uncontrolled and all ECC systems were used. This bounding calculation indicates the containment does not exceed the rupture disk setpoint for at least one hour after recovery. It is expected that the actual time until rupture disk operation would be several hours. This allows ample time for the operator to terminate the event by use of the stand-by liquid control system or to mitigate the event by reducing the rate of injection to the vessel and initiating containment heat removal. Thus, it is concluded that recriticality does not pose a significant threat to the ABWR design.

19E.2.6.6 Debris Entrainment and Direct Containment Heating

If a core melt accident occurs in which the reactor pressure vessel is at high pressure at the time of vessel failure, the debris may be entrained out of the lower drywell. If the debris is finely fragmented as it is dispersed, the pressure in the containment can rise rapidly. This process is called direct containment heating (DCH). The magnitude of the pressure rise is dependent on the amount of debris involved in the event. If a large fraction of debris participates in the DCH event the containment may be challenged. Since this would lead to an early failure of the containment structure in the drywell, the fission product releases from this type of scenario are judged to be high. Therefore, it was decided to bypass the performance of a sensitivity study for this case and perform a detailed uncertainty analysis. The results of this uncertainty analysis can be found in Subsection 19E.2.7.1.

19E.2.6.7 Fuel Coolant Interactions

Challenges of the containment during a severe accident may result from fuel coolant interactions. Fuel coolant interactions are most likely to challenge the containment when molten debris falls into water. Examination of the containment event trees indicates that a very small percentage of all sequences have water in the lower drywell before vessel failure. Both the impulse and static loads are considered. Fuel coolant interactions (FCI) may occur either at the time of vessel failure when corium and water fall from the lower plenum of the vessel, or when the lower drywell flooder opens after vessel failure has occurred.

Fuel coolant interactions were addressed in the early assessment for the ABWR response to a severe accident. Subsection 19E.2.3.1 examined the hydrodynamic limitations for steam explosions and concluded that there was no potential for a large scale steam explosion. The pressurization of the containment from non-explosive steam generation was calculated in the analyses for the accident scenarios. Attachment 19EB examines the available experimental database for its relevance to the ABWR configuration, and provide a simple, scoping calculation to estimate the ability of the ABWR containment to withstand a large, energetic fuel coolant interaction.

Four potential failure modes are considered. The transmission of a shock wave through water to the structure may damage the pedestal. Similarly, a shock wave through the airspace can cause an impulse load. However, since the gas is compressible, the shock wave transmitted through the gas will be much smaller than that which can be transmitted through the water. Therefore this mechanism is not considered here. Third, loading is caused by slugs of water propelled into containment structures as a result of explosive steam generation. Finally, the rapid steam generation may lead to overpressurization of the drywell.

The details of the analysis are presented in Attachment 19EB. The studies show that the limiting loading mechanism is the shock wave transmitted to the structure. Using a conservative bound for the impulse load capability of the pedestal, the structure can withstand the loads associated with a steam explosion involving 9.5% of the core mass. This is three times the mass of a credible fuel coolant interaction in the ABWR. Therefore, the ABWR pedestal is very resistant to fuel coolant interactions. This failure mechanism need not be considered further in the containment event trees or the uncertainty analysis.

19E.2.6.8 Core-Concrete Interaction and Debris Coolability

The issue of debris coolability has long been an area of considerable uncertainty in the progression of a core melt accident. In the ABWR design the lower drywell floor area is large in order to facilitate the spreading of the core debris. The firewater addition system, as well as the passive flooder design, ensure that debris will always be covered by water in the event of a severe accident.

However, experiments performed to date have been unable to provide conclusive evidence that these features cool the debris sufficiently to prevent core concrete interaction from occurring. If core concrete interaction were to continue unabated, there are two possible challenges for the ABWR containment design. First, the generation of non-condensable gas would contribute to the slow pressurization of the containment, even if containment heat removal is available. Second, if the concrete were eroded to a sufficient depth, the pedestal walls could be weakened to the point that the vessel was no longer sufficiently supported. If the vessel then tipped or fell, the piping attached to the vessel could cause the containment penetrations to tear, most likely in the drywell region of the containment.

The time of fission product release from the containment for either of these mechanisms would be fairly late but is dependent on the heat transfer from the corium to the overlying pool of water. Additionally, continued core concrete interaction can lead to an increase in the amount of fission product release. Since core concrete interaction can lead to a mode of drywell failure and because of the high visibility of this issue, it was decided to bypass the sensitivity study and to perform detailed uncertainty analysis for the dual issues of debris coolability and core concrete interaction.

19E.2.6.9 Fission Product Release Location

The adoption of the rupture disk in the ABWR containment design serves to significantly reduce the uncertainties in the timing, location and area of any fission product release. As discussed in Subsection 19E.2.8.1, the Containment Overpressure Protection System (COPS) is highly reliable. The setpoint of the rupture disk, 0.72 MPa, was selected such that there is a very small probability that the containment structure fails. As shown in Subsection 19F.3.1.2, the weakest portion of the ABWR containment is the drywell head. The median failure pressure of the drywell head is estimated to be 1.03 MPa abs. The other portions of the containment have an estimated failure pressure of 1.34 MPa. Thus, it is expected that most fission product releases will be via the rupture disk.

A fragility curve for the drywell head, Figure 19FA-1, shows the uncertainty in the failure pressure for the drywell head. The uncertainty of the rupture pressure for the COPS is very small as discussed in Subsections 19E.2.8.1.1 and 19E.2.8.1.2. Integrating over these two distributions, one can determine the probability that the drywell head fails before the COPS actuates. Because of the pressure difference between the wetwell and the drywell, three cases must be considered. For sequences in which the firewater system is used and water is added to the containment, as described in Subsection 19E.2.2, there is a small chance that the drywell head will fail. For sequences without water addition to the containment, the drywell head failure probability is even smaller. These probabilities are used in the quantification of the containment event trees in Subsection 19D.5. The third case applies to sequences with no pressure difference between wetwell and drywell. In these cases the drywell head failure probability is smaller yet.

19E.2.6.10 Fission Product Release Flow Area

The presence of the COPS serves to substantially reduce the uncertainties associated with the flow area for release of fission products from the containment. In the unlikely event that fission products are released from the containment, the release will almost always be via the COPS. Since this is an engineered feature of the plant, the uncertainties associated with the available flow area are very small. The COPS is designed to allow steam flow equivalent to 2.4% rated power. Since the decay heat level will be less than 1% at the time COPS operation is required, it is judged that the containment response is not sensitive to any small variation in the COPS effective flow area.

However, for the few cases discussed in Subsection 19E.2.6.9, the pressurization of the containment leads to failure of the drywell head. For these cases there is substantial uncertainty in the failure area. Therefore, two sensitivity cases were analyzed. In the first case the nominal failure area of 0.0129 m^2 (20 sq in) was increased by a factor of two. In the second case the failure area was divided by two. This broad range should bound any possible variations in the failure flow area.

The results for the three cases are identical until the time of drywell head failure. After drywell head failure the basic trends of the data are unchanged. The containment pressure is larger for cases with the smaller failure area than for those with larger areas. There is also a small variation in source term for the three cases. In the nominal case the release fraction of CsI is 9.7%. For the larger flow area, the release fraction increases to 12.6%; while, for the smaller flow area, the release drops 4.2%. Considering the upper bound, doubling the flow area increases the release by only 30%. Since a small percentage of all releases are a result of drywell head failure, the change in offsite consequences will be small. Therefore, no further consideration of containment failure area is necessary.

19E.2.6.11 Suppression Pool Bypass

The BWR containment is designed such that all gas generated in the vessel and the drywell passes through the suppression pool. This serves to quench the steam in the gas stream, which substantially decreases the pressurization rate of the containment. In addition, any fission products carried in the gas stream are scrubbed and retained in the suppression pool. Since the ABWR is designed such that any fission product release is from the wetwell airspace, this substantially reduces the risk in the unlikely event of a severe accident. Subsection 19E.2.3.3.3(4) examined mechanisms which could result in suppression pool bypass, and determined that the only pathway which could significantly increase risk is vacuum breaker failure or leakage. The results of a sensitivity study performed to examine the impact of vacuum breaker performance is summarized in this subsection. Details of the analysis can be found in Subsection 19EE.3.

The dominant severe accident sequence [Loss of all core Cooling with vessel failure occurring at Low Pressure (LCLP)] was chosen to evaluate plant performance. MAAP-ABWR runs were made with effective vacuum breaker area, $A/\sqrt{K}$, varying from 0 to 2030 cm^2 (315 in^2). The upper bound corresponds to one fully open vacuum breaker. Five variations were analyzed. In each case the overpressure relief rupture disk opened when the wetwell pressure reached 0.72 MPa. The five scenarios were:

(1) Bypass leakage begins after passive flooder activation; aerosol plugging is neglected.

(2) Bypass leakage is present from the beginning of the accident; aerosol plugging is neglected.

(3) Bypass leakage begins after passive flooder activation; aerosol plugging of the vacuum breaker opening is considered.

(4) Bypass leakage is present from the beginning of the accident; aerosol plugging of the vacuum breaker opening is considered.

(5) Bypass leakage is present from the beginning of the accident and the operator initiates the firewater spray system.

Suppression pool bypass can lead to a significant increases in fission product release. Releases can be on the order of 10 to 20% for a fully stuck open vacuum breaker. For sequences in which the firewater addition system is used in spray mode, the time to release is not significantly affected. However, for sequences without sprays, the time from the beginning of the accident until the onset of the release can be significantly reduced. The use of the Morowitz blockage model results in a significant improvement in the calculated risk associated with suppression pool bypass. Nonetheless, there is a substantial increase in consequences associated with large bypass areas. Therefore, suppression pool bypass is examined with a detailed uncertainty analysis in Subsection 19E.2.7.3.

19E.2.6.12 High Temperature Failure of Drywell

One of the failure modes identified for the containment was the degradation of the seals for the moveable penetrations in the drywell due to high temperature (Subsection 19F.3.2.2). In the base analyses discussed in Subsection 19E.2.2, the only sequences which exceeded the threshold temperature of 533 K (500°F) were those in which debris was entrained into the upper drywell and sprays were not available. In these cases the debris can radiate directly to the upper drywell structures. For the other sequences, the debris is covered by water so elevated temperature in the upper drywell is dependent on heat transfer from remaining fuel in the vessel to the upper drywell.

To ascertain the sensitivity of the drywell temperature to parameters which could affect it, several sensitivity studies were performed. All of the studies were performed using a low-pressure core melt sequence. The LCLP-PF-R-N sequence, with passive flooder operation, was selected since cases with firewater spray available are not expected to result in high drywell temperatures.

In the first calculation performed, the mass of equipment in the drywell was decreased to reduce the thermal mass in the upper drywell. The mass was arbitrarily decreased to half of the nominal value used in the base analyses. The temperature in the upper drywell at the time the rupture disk opened decreased from its nominal value of 500 K (441°F) to 487 K (418°F). While this result is somewhat counterintuitive, it can be easily explained. In the early stages of the accident, the temperature in the drywell is higher in the sensitivity case. This results in a small increase in the amount of fuel which melts and relocates into the lower drywell. Consequently, there is less heat generation in the vessel and less radiative heat transfer to the upper drywell. The overall containment performance is not affected by the slight decrease in temperature.

A second analysis was then performed in which the mass of equipment in the upper drywell was increased by a factor of two. In this case the upper drywell temperature at the time of rupture disk opening is virtually unchanged from the nominal case. In the very long term, well after the rupture disk opens, there is a slight increase in temperature compared to the nominal case as one would expect based on the previous result. However, there is no significant impact on containment performance.

The final sensitivity case performed considered the impact of increasing the convective heat losses from the vessel to the drywell 50% above its nominal value. A slight increase in the upper drywell gas temperature was observed in this case. At the time the rupture disk opened, the upper drywell temperature was 505 K (450°F) as compared to 500 K (441°F) in the nominal case. The overall containment performance is not affected by this slight change.

In summary, the three sensitivity studies performed to assess the sensitivity of the drywell temperature to the detailed modeling assumptions indicate that the ABWR is not sensitive to those parameters which affect drywell temperature. Therefore, no further study of this area is necessary.

19E.2.6.13 Suppression Pool Decontamination Factor

From the standpoint of severe accidents, one of the most important features of a pressure suppression containment is the suppression pool. The suppression pool not only quenches any steam which enters it, reducing the rate of containment pressurization, it also traps the fission products carried with the gas flow. This process, known as scrubbing, significantly reduces the amount of fission product aerosols available for release from the containment.

The efficiency of the scrubbing process is typically characterized in terms of a decontamination factor (DF) defined by the mass of debris which enters the pool divided by the mass of debris which leaves the pool. MAAP-ABWR uses correlations based on the SUPRA code to calculate the DF. These correlations typically result in very high retention of fission products in the pool for all species of interest except the noble gasses which have a DF of 1.0.

In order to investigate the sensitivity of the offsite consequences of a severe accident to the suppression pool decontamination factor, a simple sensitivity study was performed. The MAAP-ABWR code was modified to allow a constant DF to be input for all species except the noble gasses. Two calculations were then repeated assuming a conservative DF of 100. None of the other fission product removal mechanisms were affected by the change.

The two cases selected for study were both low-pressure core melt sequences. In the first sequence, LCLP-FS-R-N, the firewater system is assumed to be available, while in the second case, LCLP-PF-R-N, the passive flooder operated to cool the debris. Both cases indicated a significant increase in the fission product release. For the case with the firewater system available, the fraction of CsI release increased from 1.5E-7 to 1.2E-3. For the case with the passive flooder the results were similar, the CsI release increased from 1.2E-7 to 1.6E-3.

CRAC cases were run in order to determine the effect of these changes on the consequences of release. The results of this calculation are shown in Figure 19E.2-21. Case 1 is the nominal case and Case 4 uses the release fractions from this sensitivity study. The conditional probability of exceeding the offsite dose indicated on the x-axis is shown. The probability of the dose is dependent on the weather. The curve shows that there is virtually no impact until a conditional probability of 0.04. Thus, there will not be a significant impact on offsite dose, even for this very conservative DF of 100. Thus, it has been shown that the consequences of a severe accident are not very sensitive to variation in the suppression pool decontamination factor. No further consideration of this phenomena is required in uncertainty analysis.

19E.2.6.14 Suppression Pool pH Control

The chemical form of iodine may be affected by the acidity of the suppression pool. If the pool becomes acidic (pH <7) the formation of volatile and organic forms of iodine may be enhanced. Experiments have indicated the potential for the radiolytic formation of nitric acid (HNO_3) in the suppression pool. This can then lead to the conversion of I^- to I_2 in the pool. The gas species remain in equilibrium with the I_2 in the pool, with the relative amounts governed by a partition fraction between the water- and gas-borne species. Reference 19E.2-31 states, “If the pH is controlled so that it stays above 7, a reasonable value for the I^- converted to I_2 is 3.E-4 ...[Calculation] indicates a small production of volatiles for PWRs but virtually none for BWRs”.

Calculations were performed following the methods of Reference 19E.2-31 to determine the potential for the formation of nitric acid to lead to an acidic suppression pool. These sequences differed by consideration of varying initial suppression pool pH, caused by the transport of CsOH to the pool as a result of the accident. In each calculation, the pH of the pool is monitored over time as nitric acid is formed radiolytically. The results of two calculations which bound the expected transfer of CsOH to the pool are given below. In both cases, the transfer of CsI to the pool is assumed to be the same as that for CsOH.

Initial CsOH fraction in pool	**10%**	**80%**
Time (h)	**pH**	**pH**
0	9.65	10.56
1	9.65	10.56
10	9.63	10.53
24	9.59	10.49

The results of these calculations indicate that the pH of the suppression pool will not drop to the acidic range within 24 hours of accident initiation. Therefore, nitric acid formation due to radiolysis will not have a significant impact on the source term. No further consideration of this phenomenon is necessary.

19E.2.7 Detailed Phenomenological Uncertainty Studies

19E.2.7.1 Direct Containment Heating

Direct Containment Heating (DCH) is the sudden heatup and pressurization of the containment resulting from the fragmentation and dispersal of core material in the containment atmosphere. DCH is a concern for sequences in which the vessel fails at high pressure since the steam flow from the vessel provides the motive force for entrainment. In the event of a sufficiently large DCH event, the containment could fail at the time of vessel failure. This would lead to very high releases to the environment. In the past DCH has been addressed for Pressurized Water Reactors. BWRs have very reliable vessel depressurization systems. Thus, the frequency of accidents with the vessel remaining at high pressure is extremely low. However, with the many sources of low-pressure injection available to the ABWR to prevent core damage, the frequency of all core damage sequences is very low. Therefore, high pressure core melts appear as contributors to the total core damage frequency, albeit with a very low probability.

A detailed uncertainty analysis utilizing decomposition event trees (DETs) was performed to assess the peak drywell pressure resulting from a DCH event. This analysis is given in Attachment 19EA. A large number of calculations were performed to determine the impact of DCH on the probability of containment failure and offsite risk. The analysis investigated uncertainties in a variety of phenomena:

(1) Mode of vessel failure,

(2) Mass of molten core debris at the time of vessel failure,

(3) Potential for high pressure melt ejection,

(4) Fragmentation of debris in the containment.

Additional sensitivity studies were performed to examine other phenomena which could affect DCH. The study concluded that a deterministic best estimate for the peak pressure from DCH would not lead to containment failure. Consideration of the uncertainties in the phenomena lead to a very small CCFP for all core damage events. Additional sensitivity analyses were considered which indicate that an upper bound on the impact of DCH is a small percentage. Even in this limiting case, the probability of DCH failing containment is well below the goal of 10%. Furthermore, since the probability of containment failure due to DCH is very low, there is no measurable impact on offsite dose.

19E.2.7.2 Debris Coolability

The issue of debris coolability has long been an area of considerable uncertainty in the progression of a core melt accident. In the ABWR design, the lower drywell floor area is large in order to facilitate the spreading of the core debris. The firewater addition system, as well as the passive flooder design, ensure that debris will always be covered by water in the event of a severe accident.

However, experiments performed to date have been unable to provide conclusive evidence that these features cool the debris sufficiently to prevent core concrete interaction from occurring. If core concrete interaction were to continue unabated, there are two possible challenges for the ABWR containment design. First, the generation of non-condensable gas would contribute to the slow pressurization, even if containment heat removal is available. Second, if the concrete were eroded to a sufficient depth, the pedestal walls could be weakened to the point that the vessel would no longer sufficiently supported. If the vessel then tipped or fell, the piping attached to the vessel could cause the containment penetrations to tear, most likely in the drywell region of the containment. Additionally, continued core concrete interaction can lead to an increase in the amount of fission product release.

A detailed uncertainty analysis utilizing decomposition event trees (DETs) was performed to determine the potential for continued core concrete interaction and its impact on the containment response. This analysis is given in Attachment 19EC. A large number of calculations were performed. These calculations addressed uncertainties in the following parameters:

(1) Amount of core debris,

(2) Debris-to-water heat transfer,

(3) Amount of additional steel in the debris,

(4) Delayed flooding of the lower drywell,

(5) Fire water injection instead of passive flooder operation.

The conclusions from all of these uncertainty calculations were:

(1) For the dominant core melt sequences that release core material into the containment, a large percentage result in no significant CCI. An insignificant number of sequences are expected to experience dry CCI.

(2) Even for those low frequency cases with significant CCI, radial erosion remains below the structural limit of the pedestal. After consideration of uncertainties, only a small percentage of the sequences with significant CCI will suffer pedestal failure. Combining this conclusion with the first, an extremely small percentage of all core melt sequences with vessel failure will lead to additional drywell failures as a result of CCI.

(3) The time of fission product release is not significantly affected by continued CCI.

(4) The fission product release is dominated by the noble gasses when the containment overpressure protection system operates. This conclusion is unaffected by assumptions on debris coolability. Therefore, the offsite dose for sequences with rupture disk operation is not impacted by core concrete attack.

These conclusions would indicate that the uncertainties associated with CCI have an insignificant influence on the containment failure probability and risk.

19E.2.7.3 Suppression Pool Bypass

Suppression pool bypass (the passage of gas and vapor from the drywell directly into the wetwell airspace) can lead to increased fission product releases. As shown in

Subsection 19E.2.3.3.3(4), the only mode of suppression pool bypass that has the possibility of significantly increasing risk is vacuum breaker leakage. Attachment 19EE determined the probabilities and consequences for vacuum breaker leakage areas from zero to that corresponding to one vacuum breaker stuck fully open.

Fission product release fractions were determined with MAAP-ABWR using the dominate accident sequence [Loss of all core Cooling with vessel failure occurring a Low Pressure (LCLP)] modified to include a path between the drywell and the wetwell airspace. Plugging of leakage paths by fission products was considered for small pathways. Leakage probabilities were determined by reviewing recent operating experience of wetwell to drywell vacuum breakers in BWRs with Mark I, II and III containments.

Suppression pool bypass does not significantly add to the risk associated with the ABWR because the bypass areas resulting in increased releases are offset by low probabilities of occurrence. No leakage and, correspondingly, no impact on plant risk is expected to occur for almost all of the accident demands. Small amounts of leakage have a small probability per event, and can result in medium volatile fission product releases (1 to 10% of initial inventory). Volatile fission product releases on the order of 10 to 20% of initial inventory can result when large amounts of suppression pool bypass are present. However, the impact on plant risk is still negligible because the probability of large leakage is very small.

19E.2.8 Severe Accident Design Feature Considerations

Although the frequency of core damage is very low in the ABWR design, features were added to the design to ensure a robust response of the containment to a severe accident. This subsection discusses the important considerations for the severe accident design features.

19E.2.8.1 Containment Overpressure Protection System

ABWR has a very low core damage frequency. Furthermore, in the unlikely event of an accident resulting in core damage, the fission products are typically trapped in the containment and there is no release to the environment. Nonetheless, in order to mitigate the consequences of a severe accident which results in the release of fission products and to limit the effects of uncertainties in severe accident phenomena, ABWR is equipped with a Containment Overpressure Protection System (COPS). This system is intended to provide protection against the rare sequences in which structural integrity of the containment is challenged by overpressurization. It has been determined that these rare sequences comprise a small percentage of the hypothesized severe accident sequences.

The COPS is part of the atmospheric control system and consists of two 200A (8-inch) diameter overpressure relief rupture disks mounted in series on a 250A (10-inch) line which connects the wetwell airspace to the stack. The second rupture disk, located at the inlet to the plant stack, has a very low setpoint, less than 0.03 MPa differential pressure. The setpoint of the inner rupture disk, located near the containment boundary, will be selected such that the COPS opens when the wetwell pressure is 0.72 MPa. The COPS provides a fission product release point at a time prior to containment structural failure. Thus, the containment structure will not fail. By engineering the release point in the wetwell airspace, the escaping fission products are forced through the suppression pool. In a core damage event initiated by a transient in which the vessel does not fail, fission products are directed to the suppression pool via the SRVs, scrubbing any potential release. In a severe accident with core damage and vessel failure or in a LOCA which leads to core damage, the fission products will be directed from the vessel and drywell through the drywell connecting vents and into the suppression pool again ensuring any release is scrubbed. Eventually, if the containment pressure cannot be controlled, the rupture disk opens. Any fission product release to the environment is greatly reduced by the scrubbing provided by the suppression pool.

In the absence of the COPS, unmitigated overpressurization of the containment will result in failure of the drywell head for most severe accident scenarios (Some high-pressure core melt sequences result in fission product leakage through the moveable penetrations in the drywell rather than drywell head failure.). To compare the consequences of severe accidents resulting in fission product releases via drywell head failure to those with releases through the COPS, MAAP-ABWR was used to simulate a series of severe accident sequences for both release mechanisms. These severe accident sequences are described in Subsection 19E.2.2. Failure pressure of the drywell head was assumed to be equal to its median ultimate strength, 1.025 MPaG. The results of these runs show releases of volatile fission products, after 72 hours, for the COPS cases to be several orders of magnitude less than for the corresponding drywell head failure cases. The CsI release fractions are compared in Table 19E.2-25. Most accident sequences show this large difference in releases between drywell head failure and COPS cases.

19E.2.8.1.1 Pressure Setpoint Determination

Several factors were considered in determining the optimum pressure setpoint for the rupture disk. The results of the previous analysis show that it is desirable to avoid drywell head failure. This can be assured by providing a rupture disk pressure setpoint below the pressure that would begin to challenge the structural integrity of the containment. However, as the pressure setpoint is reduced, the time to containment failure and fission product release is also reduced. Thus, the setpoint of the rupture disk must optimize these competing factors: minimizing the probability of drywell head failure while maximizing time before fission product release to the environment.

The service level C capability of the containment serves as one indication of a lower bound for the structural integrity of the containment. As shown in Appendix 19F, the service level C for the ABWR is 0.77 MPa, limited by the drywell head. Thus, it is desirable to set the rupture disk setpoint below this value.

The distribution of drywell head failure pressure and the distribution of rupture disk burst pressure were also considered in determining the burst pressure. As stated in Attachment 19FA, the drywell head failure pressure is assumed to have a lognormal distribution with a median failure pressure equal to its ultimate strength of 1.025 MPaG. The variability of rupture disk opening pressures is best modeled with a normal or Gaussian distribution. Typical high quality rupture disks exhibit a tolerance of ±5% of the mean opening pressure. Tests have shown that this ±5% tolerance spans ±2 to ±2.5 standard deviations of the rupture disk population. This analysis of the Containment Overpressure Protection System conservatively assumes that only ±2 standard deviations are included within the ±5% tolerance. Because the setpoint of the outer rupture disk is very low, the variability of the pressure is neglected in comparison to the variability of the inner, high pressure disk.

A critical parameter in determining the risk of drywell head failure before rupture disk opening is the pressure difference between the drywell and wetwell. Late in an accident the drywell is at higher pressure than the wetwell. For a given rupture disk setpoint, the probability of drywell head failure increases as the pressure difference increases. The maximum drywell to wetwell pressure difference is 0.1 MPa. This pressure difference occurs for cases in which firewater spray was activated after vessel failure but terminated before containment failure. Cases without firewater spray have pressure differences of no more than 0.05 MPa.

A COPS setpoint of 0.72 MPa at 366 K (200°F) was chosen. The residual risk of drywell head failure may be calculated by combining the two distributions with an offset corresponding to the pressure difference between the wetwell and the drywell. A 0.72 MPa setpoint results in a small probability of drywell head failure prior to rupture disk opening for a 0.1 MPa drywell to wetwell pressure difference. For a drywell to wetwell pressure difference of 0.05 MPa, the drywell head failure probability prior to rupture disk opening is smaller. This is judged to be an acceptable level of risk.

19E.2.8.1.2 Variability in Rupture Disk Setpoint

Nickel was chosen as the material for the rupture disk for evaluation purposes due to its relative insensitivity to changes in temperature. At temperatures above room temperature the opening pressure of a typical nickel rupture disk will decrease by about 2% for a 56 K (100°F) increase in temperature. Thus, in order to estimate the uncertainty due to variations in the temperature of the ABWR rupture disk, a sensitivity study was performed in which the pressure setpoint of the rupture disk was varied.

The nominal pressure setpoint of the rupture disk is 0.72 MPa at 366 K (200°F). Two cases were examined using MAAP-ABWR in this sensitivity study. For both cases the LCLP-PF-R sequence was used as the base case. First, the rupture disk pressure setpoint was reduced to 0.708 MPa which corresponds to a rupture disk temperature of 422 K (300°F); and, second, the pressure setpoint was increased to 0.735 MPa which corresponds to a temperature of 311 K (100°F). This temperature range, from 311 to 422 K (100 to 300°F), bounds all anticipated rupture disk temperatures.

The elapsed time to rupture disk opening was within 0.8 hours of the base case value of 20.2 hours for both cases tested. Higher rupture disk temperatures (i.e. lower pressure setpoints) reduce the time to rupture disk opening and lower rupture disk temperatures (i.e. higher pressure setpoints) increase the time to rupture disk opening. There were no significant changes in fission product release. For both cases the CsI release fraction at 72 hours remained less than 1E-7.

Another parameter affected by the variation in the rupture disk temperature is the probability of drywell head failure prior to rupture disk opening in a severe accident. Using the rupture disk and drywell head failure distributions, it was determined that the probability of drywell head failure prior to rupture disk opening increased slightly for the case with the rupture disk temperature of 311 K (100°F). With a rupture disk temperature of 422 K (300°F), the probability decreased slightly. The rupture disk temperature variation has a similar effect on the severe accident sequences in which the firewater spray system is activated. The probability of drywell head failure prior to rupture disk opening increases slightly for the case with the rupture disk temperature of 311 K (100°F) and decreases slightly for the case with the rupture disk temperature of 422 K (300°F).

The results of this sensitivity study show that variations in rupture disk temperature, which cause small variations in rupture disk opening pressure, have a minor effect on the performance of the ABWR Containment Overpressure Protection System.

19E.2.8.1.3 Sizing of Rupture Disk

The size of the rupture disk has also been optimized. If the rupture disk is too small, it could be incapable of venting enough steam to prevent further containment pressurization. On the other hand, if the rupture disk is too large, level swell in the suppression pool could introduce water into the COPS piping. If this were to occur, the piping could be damaged or there could be carryover of waterborne fission products from the containment.

A 200A (8-inch) rupture disk was selected. This is sufficient to allow 35 kg/s of steam flow at the opening pressure of 0.72 MPaA and corresponds to a energy flow of about 2.4% rated power. The minimum acceptable flow rate is 28 kg/s of steam flow at the same pressure. For virtually all severe accident sequences, the rupture disk would not be called upon until about 20 hours after scram. The decay heat level at this time is about 0.5%. Thus, there is ample margin in the sizing of the rupture disk for severe accidents.

An additional accident was considered in the selection of the rupture disk size. In the event of an ATWS with the additional failure of the standby liquid control system, the operator is directed to lower water level to control power. Analysis has shown that the RHR system is capable of removing the energy generated by the ATWS from the containment (Subsection 19.3.1.3.1). If the additional failure of containment heat removal is assumed, a simple calculation indicates that an the rupture disk area is just sufficient to limit the containment pressure below service level C.

Calculations were also performed to investigate the potential effects of pool swell and fission product carryover at the time of COPS operation. These analyses (Subsection 19E.2.3.5) indicate that pool swell does not threaten the integrity of the COPS piping and that no significant entrainment of fission products will occur due to carryover.

19E.2.8.1.4 Comparison of ABWR Performance With and Without COPS

The results of the MAAP-ABWR calculations for the various accident scenarios were investigated in Subsection 19E.2.2. The releases are summarized in Table 19E.2-25. Comparisons of CsI release fraction at 72 hours show large differences between the COPS and drywell head failure cases. CsI release fraction at 72 hours for drywell head failures is on the order of 0.1% to 37%. For all cases with release via the COPS, MAAP-ABWR predicts release fractions of about 1E-7. Table 19E.2-26 summarizes several critical parameters for the dominant low pressure core melt scenario.

There is, of course, some reduction in the elapsed time to fission product release for the COPS cases when compared to the drywell head failure cases. For the dominant accident sequences in which the operator initiates the firewater spray system prior to overpressurization, the time difference between rupture disk opening and drywell head failure is only 3 to 4 hours. A typical example is the Loss of All Core Coolant with Vessel Failure at Low Pressure with Firewater Spray addition sequence (LCLP-FS), as described in Subsection 19E.2.2.1. For this sequence the wetwell pressure will reach 0.72 MPa and the rupture disk will open at 31.1 hours. Without the rupture disk, the drywell will reach 1.025 MPa at 35.0 hours.

The potential for increased risk due to the rupture disk opening early has been considered. It is assumed that recovery of RHR capability is sufficient to terminate containment pressurization and prevent drywell head failure. In the 3.9 hours between rupture disk opening and hypothetical drywell head failure for the LCLP-FS sequence, the probability of recovering RHR capability is very small (Subsection 19.3.2.7). This represents the probability that the COPS was opened unnecessarily since RHR would have been recovered in this time period.

For cases with passive flooder operation, the fission product release occurs about 6 to 8 hours sooner than it would have if the drywell head was allowed to pressurize to 1.025 MPa. For the range of severe accident sequences described in Subsection 19E.2.2, the probability of RHR recovery in a similarly defined time window is small.

For both cases, there is a small probability that RHR will be recovered before the time at which containment would fail if the rupture disk setpoint has been surpassed. In light of this fact and given the difference in magnitude of the fission product release, it is clearly preferable to direct the fission products through the rupture disk.

19E.2.8.1.5 Suppression Pool Bypass

A comparison of performance for cases with suppression pool bypass flow through an open vacuum breaker valve was also considered. Cases were run with bypass effective area varying from 5 to 2030 cm^2 (0.0054 to 2.19 ft^2). A fully open vacuum breaker has a flow area of 2030 cm^2. The dominant the Loss of All Core Coolant with Vessel Failure at Low Pressure sequence was considered with Passive Flooder Operation since previous analysis has shown that the firewater system is capable of mitigating bypass.

No credit was taken for aerosol plugging of the bypass leakage in this analysis; and, therefore, the results are conservative. Also, it was assumed that the bypass leakage was present from the beginning of the accident sequence. As the bypass area increases, the fraction of fission product aerosols which pass through the suppression pool decreases. Thus, the benefit of a wetwell release of fission products is significantly reduced as the bypass area increases.

For effective bypass areas less than 50 cm^2 (0.054 ft^2), CsI releases at 72 hours from the COPS cases were smaller than for the corresponding drywell head failure cases. However, the differences in CsI releases at 72 hours were only factors of 2 to 4 rather than several orders of magnitude. The time difference between drywell head failure and rupture disk opening was 4 to 8 hours for these small bypass areas. For bypass effective areas greater than 50 cm^2 (0.054 ft^2), CsI release fractions at 72 hours are on the order of 10% for both the drywell head failure cases and the COPS cases. On the other hand, the time difference between rupture disk opening and drywell head failure is only 2 to 4 hours for these larger bypass areas. These relatively small time differences will not significantly affect the magnitude of the offsite dose. Attachment 19EE has a complete discussion of suppression pool bypass flow through vacuum breaker valves.

19E.2.8.1.6 Potential Impact of Hydrogen Burning and Detonation

Hydrogen burning and detonation are not a concern for the ABWR containment because the containment is inerted with hydrogen. There could be a potential for burning in the COPS system and the stack after the rupture disk opens. However, due to the design and operation of the COPS system, this issue does not have an impact on risk.

Hydrogen burning and detonation will be precluded in the piping associated with the COPS system. The piping will be inerted during operation with rupture disk located at the inlet of the stack. This, combined with initial purging of the piping, will ensure that the inertion of the containment will extend out to the stack, and prevent burning of hydrogen in the portion of the COPS system which is within the reactor building. Therefore, there will be no concern of the leading edge of the containment atmosphere mixing with the gas in the piping and causing a burn. After passing of the leading edge of the gas flow, the mixture in the piping will be identical to that in the containment. The gas flow through the system will prevent the backflow of air into the COPS piping.

Hydrogen burning could occur in the plant stack as the gas flow enters the stack. The stack is a non-seismic structure located on top of the reactor building. Because of this configuration, the reactor building has been designed to withstand the loads associated with the collapse of the plant stack. Furthermore, no credit is taken in the analysis for the plant stack to reduce the offsite dose by providing for an elevated release. All releases were presumed to occur at the elevation of the top of the reactor building. Therefore, hydrogen burning or detonation in the stack will have no impact on the consequences of a severe accident as modeled in this analysis.

No burning will occur within the COPS piping. Furthermore, no credit was taken for the plant stack to reduce the source term to the environment and the reactor building can withstand the collapse of the plant stack. Therefore, hydrogen burn or detonation in the COPS system will have no impact on risk and no further consideration of this phenomenon is required.

19E.2.8.1.7 Summary

A wetwell pressure setpoint of 0.72 MPa for the overpressure relief rupture disk meets the design goal. The probability of containment structural failure is minimized while maximizing the time to fission product release in a severe accident. The small probability of containment structural failure if the pressure reaches the rupture disk setpoint in a severe accident, combined with the already low core damage frequency and reliable containment heat removal, produces an extremely low probability of significant fission product release. In addition, the elapsed time to rupture disk opening is greater than 24 hours for most severe accident sequences.

The net risk reduction associated with the implementation of the COPS system in the design of the ABWR is summarized in Table 19E.2-27 and Figure 19E.2-22. All sequences which would result in COPS operation were assumed to lead to failure of the drywell head. This may slightly overpredict the probability of drywell head failure since there will be somewhat more time available for the recovery of containment heat removal if the COPS system were not present. Table 19E.2-26 indicates a low probability of RHR recovery in the interval between the time of COPS initiation and the time of drywell head failure if COPS were not present. For the case with firewater addition to the containment, the probability of RHR recovery during the period of interest is small. Therefore, no significant error is introduced into the calculation.

Table 19E.2-27 indicates that the probability of drywell head failure increases many times for sequences with core damage (Classes I and III) if the COPS system is not present. For Class II sequences, the loss of containment heat removal may lead to core damage for those sequences which have drywell head failure. Since the probability of drywell head failure increases by a very large factor without the COPS system, the core damage probability associated with Class II events also increases by the same amount. Figure 19E.2-22 shows the probability of exceedence versus whole body dose at 0.81 km (0.5 mile) for the ABWR and for the ABWR without the COPS system. The offsite dose is reduced as a result of the COPS implementation into the design.

19E.2.8.2 Lower Drywell Flooder

19E.2.8.2.1 Introduction

This subsection provides the bases for sizing the lower drywell flooder system. The system is described in detail in Subsection 9.5.12.

The lower drywell flooder provides an alternate source of water to the lower drywell once it contains core debris. The primary water source is the firewater addition system. Water present in the lower drywell cools the core debris and establishes a water pool above the debris. Water absorbs heat by first heating up to saturation conditions and then boiling away. Debris cooling requires that the water absorb the heat generated in the debris bed and the latent and sensible heat released by the debris as its temperature decreases. Quenching prevents or mitigates core concrete interaction (CCI). An overlying water pool also scrubs fission products which may be released from the debris bed.

The flooder system is comprised of ten piping lines. Each line originates in one of the ten vertical pipes which are part of the drywell to wetwell connecting vent system. The vents are arranged symmetrically around the perimeter of the lower drywell. The flow through each flooder line will be initiated by melting a fusible plug at the line exit (lower drywell side). Since 10.2-cm (4-inch) diameter fusible disks may be commercially available, the flooder line diameter was chosen as 10.2 cm (4 inches).

The teflon disk resides between the stainless steel disk and the fusible plug in the flooder valve (Figure 19E.2-24). Its purpose is to insulate the fusible plug from the relatively cold suppression pool water. If insulation was not provided, melting of the plug might not be uniform and operation of the flooder valve might be impaired. The disk will not melt or stick in the valve because teflon has a softening temperature of approximately 673 K (400°C) and a maximum continuous operating temperature of 561 K (288°C) both of which are above the plug melting temperature of 533 K (260°C). Furthermore, teflon has high chemical resistance and will not adhere to the stainless steel plug nor the fusible plug.

The minimum acceptable flow rate for the flooder system corresponds to the flow rate which can just absorb the heat generated in the debris bed. Minimum acceptable flow is calculated in Subsection 19E.2.8.2.2. The expected flow rate in the flooder system can be obtained by applying Bernoulli's equation to the flooder geometry. This calculation is presented in Subsection 19E.2.8.2.3.

19E.2.8.2.2 Minimum Acceptable Flow Rate

Heat is generated in the debris bed by fission product decay and zirconium oxidation. Any flooder flow in excess of the amount required to remove generated heat will participate in quenching the debris and establishing a water pool above the debris bed. As shown in Attachment 19EC, the time required to quench the debris is not a critical parameter in determining containment performance. Therefore, the minimum acceptable flow rate for the lower drywell flooder system is the rate which will completely absorb all the heat generated in the debris bed.

The decay heat generation rate at the time when debris is expected to first enter the lower drywell during credible accident scenarios is approximately 1% of rated power (39 MW). Thirty-nine megawatts can be used as a first approximation of the decay heat generation rate of the debris bed in the lower drywell. This assumption is highly conservative because the entire core mass will never completely relocate into the lower drywell. Furthermore, noble gasses and volatiles will escape from the molten debris, carrying away the decay heat associated with these two constituents (approximately 20% of the total).

Heat can also be generated in the bed by exothermic reactions of the debris constituents. The most energetic reactions involve oxidation of zirconium by water vapor and carbon-dioxide. The only source of significant amounts of oxidizing agents is the concrete beneath the debris bed. The water above the bed will not contribute significantly to oxidation because the surface of the bed will form a crust which will quickly be depleted of zirconium. NUREG-5565 indicates that a typical ablation rate for concrete is 5.08 cm (2 inches) per hour. The generation rate, assuming that the H_2O

and CO_2 released during ablation completely react with zirconium, is 3.6 MW. Combining these two sources of heat yields a debris bed heat generation rate of 43 MW.

The heat absorption capability of the suppression pool water is 2,350 MJ/m^3. Therefore, the minimum acceptable flow rate for the lower drywell flooder system is 0.018 m^3/s (18 liters/s). Assuming a four-inch throat, as discussed in Subsection 19E.2.8.2.3, this flow can be provided by two lines of the lower drywell flooding system. Alternatively, if nine flooder lines are active, this system flow corresponds to a minimum individual line flow of 2 liters/s.

19E.2.8.2.3 Expected Flooder Flow Rate

The flow rate through the flooder system will be governed by the flow area, the hydrostatic driving head and head losses in the lines.

The flow area depends on the diameter of the flooder lines and the number of lines that are participating. Assuming that one flooder fails to operate, the flow area is

$$\begin{aligned} A_f &= \frac{\pi}{4} d_f^2 n_f \\ &= 0.073 m^2 \end{aligned} \qquad (19E.2\text{-}60)$$

where:

d_f = diameter of lines (0.1016 m, 4 in), and

n_f = number of lines (9, assuming one fails).

The elevation of the flooder line exit below the water level in the drywell-to-wetwell connecting vents determines the hydrostatic head, Figure 19E.2-23. Due to steaming in the drywell, the drywell pressure is greater than the wetwell pressure and the water level in the drywell-to-wetwell connecting vents is assumed to be depressed to the bottom of the first row of horizontal vents. This leaves a hydrostatic head, Δz, of 0.375 meters to the inlet of the flooder lines.

Form and frictional head losses decrease the flow through the flooder lines. Form losses are due to entrance and exit effects as well as the 90° elbow and valve. A loss coefficient, k, of 3 conservatively accounts for all the head losses in the flooder system.

Applying Bernoulli's equation to steady, irrotational flow and assuming that the level of the suppression pool does not change (since the surface area of suppression pool is much greater than the flooder flow area) yields a flooder flow rate of

$$\dot{v}_{fl} = A_f\sqrt{\frac{2g\Delta z}{1+k}} = 0.099\,m^3/s \quad (19E.2\text{-}61)$$

where:

$\dot{v}_{fl}$ = the total volumetric flow rate through nine lines, and

g = the acceleration of gravity.

For a liquid density of 980 kg/m^3, this corresponds to a system flow rate of 97 kg/s and an individual line flow rate of 10.8 kg/s. This is the expected flow rate through the flooder system assuming complete expulsion of the fusible plug and minimum hydrostatic driving head.

19E.2.8.2.4 Time to Fill Lower Drywell

Water that enters the lower drywell provides cooling to the debris bed. It also establishes an overlying liquid layer. Neglecting the subcooling of the flooder water, heat transfer from the debris bed to the water will result in vaporization. The amount of flooder flow which is vaporized is

$$\dot{V}_{vap} = \frac{\dot{Q}}{h_{fg}\rho_{liq}} \quad (19E.2\text{-}62)$$

where:

$\dot{V}_{vap}$ = volume rate at which flooder water is vaporized,

$\dot{Q}$ = heat transfer from the debris bed to the flooder water,

h_{fg} = latent heat of vaporization of water,

ρ_{liq} = density of water.

The amount of flooder flow which can contribute to filling the lower drywell is

$$\dot{v}_{fill} = \dot{v}_{fl} - \dot{v}_{vap} \quad (19E.2\text{-}63)$$

The time to fill the lower drywell to the exit of the flooder is

$$t_{fill} = \frac{V_{fill}}{\dot{v}_{fill}} \tag{19E.2-64}$$

where

V_{fill} = the volume of the lower drywell below the flooder exit.

The flooder exit will be 1.15 meters above the lower drywell floor. The surface area of the lower drywell floor is 88.25 m^2. Thus,

$$V_{fill} = 101.5 m^3$$

Flooder actuation is expected to occur approximately five hours after reactor scram during most severe accident scenarios. The decay heat level at this time is approximately 1% of the rated power. Assuming the entire core relocates to the lower drywell, the debris bed will have a decay heat generation rate, Q_d, of 39 MW. If all of this heat is transferred to the flooder water, the rate and time to fill the lower drywell are

$$\dot{v}_{fill,\ d} = 0.080 m^3/s$$

$$t_{fill,\ d} = 21 \text{minutes}$$

The maximum heat flux from the surface of a debris bed that has been experimentally observed (Subsection 19EB.2.2) is 2 MW/m^2. The lower drywell has a surface area of 88.25 m^2. Thus, the maximum cooling rate of the debris bed, Q_{max}, is 177 MW. For this heat transfer rate, the rate and time to fill the lower drywell are

$$\dot{v}_{fill,\ max} = 0.016 m^3/s$$

$$t_{fill,\ max} = 1.8 \text{hours}$$

In practice, this high heat flux is not expected to be maintained as the debris is quenched. Nonetheless, the time to fill the lower drywell to the elevation of the flooder exit will be bounded by these two values, 21 minutes and 1.8 hours. This difference in timing will not have a significant impact on the fission product release from the containment since the steam produced during debris quenching will carry any fission products released during this time into the suppression pool.

19E.2.8.2.5 Consequences of One Flooder Line Opening First

Core debris that enters the lower drywell will be distributed fairly uniformly. The lower drywell floor was designed so that debris spreading would not be hindered. The temperature of the lower drywell air space and structures should be even more uniform because of convective and radiative heat transfer from debris material. Cooler regions will tend to absorb more heat than warmer ones resulting in temperature equalization.

However, if highly non-uniform debris dispersal occurs, it has been postulated that one flooder line could open and its operation could delay or even prevent the other lines from activating. In the worst physical case, the initiation of one flooder line causes crust formation without completely quenching the debris. The crust limits heat transfer from the surface of the debris bed. Core-concrete interaction (CCI) will occur if surface heat transfer is reduced enough.

CCI results in large quantities of gases being formed under the surface of the crust. The gases will increase in pressure due to continued generation until the crust ruptures or they escape from the edges of the bed. In either case, the gases will pass from the debris bed into the lower drywell airspace. The passage either will be unobstructed with gasses exiting the debris above the water elevation or through an overlying layer of water. Since only one flooder line is presumed active, the water layer, if it exists, will be thin and no significant amount of heat will be transferred from the gas to the liquid.

Concrete has an ablation temperature of approximately 1500 K. The released gases from core concrete interaction will be at least at this temperature. Higher temperatures may be reached by the gases as they interact with debris material in their exit. Thus, gases enter the lower drywell air space at very high temperature. The CCI gases will increase the temperature of the lower drywell air space. More flooder lines will become active as the lower drywell temperature increases. For this reason, the activation of a single flooder line is transient condition at worst and is not expected to adversely affect the operation of the other lines.

19E.2.8.2.6 Valve Opening Time

The fusible plug valve is designed to open when the lower drywell temperature reaches 533 K. The fusible material is made up of an alloy mixture of two or more of the following metals: tin, silver, bismuth, antimony, tellurium, zinc and copper. Alloy contents are chosen so that the plug melts when its temperature reaches 533 K.

The melting points of the individual metals are as follows:

Metal	Melting Point (K)
Antimony (Sb)	903

Metal	Melting Point (K)
Bismuth (Bi)	544
Copper (Cu)	1356
Silver (Ag)	1233
Tellurium (Te)	722
Tin (Sn)	505
Zinc (Zn)	692

The basic configuration of the fusible plug valve is shown in Figure 19E.2-24. The plastic cap has a melting point much lower than that of the fusible plug. Flow initiation occurs when the small annular groove, 2.0 mm in depth, melts. Hydrostatic pressure then expels the remainder of the plug, the stainless steel disk and the teflon disk.

The valve opening time is the time required to melt the fusible metal in the annular groove. To estimate the opening time, a calculation has been made for a pure bismuth plug. Bismuth was used because it has the closest melting point to 533 K.

Heat transfer from the surrounding stainless steel pipe to the plug is by conduction. Heat transfer from steam in the lower drywell to the stainless steel pipe is by convection. The pipe also receives radiative heat from the debris on the lower drywell floor. Heat transfer to the bottom of the valve was neglected. The debris bed surface temperature and lower drywell gas temperature were estimated using a representative MAAP-ABWR sequence. Using these assumptions, the valve opening time was calculated to be less than approximately 10 minutes depending on the steam absorbtivity. This is a representative time from when the lower drywell gas space reaches 533 K until the flooder line becomes active.

19E.2.8.2.7 Estimation of Net Risk

In order to assess the net risk of the passive flooder system, a sensitivity study was performed using three failure probabilities for the passive flooder node, P, in the containment event trees. In these cases, the failure probability of the passive flooder was incrementally increased from its base case value.

As indicated in Table 19E.2-28, the overall results are not sensitive to this parameter. Failure of the passive flooder leads to an increase in the probability of Dry CCI. Thus, the probability of Dry CCI increases by the same increments for the three sensitivity cases. However, the base case results for Dry CCI are so small that a very large magnitude increase does not impact other results significantly.

The principal conclusions of the sensitivity studies are:

(1) Pedestal failure does not increase since it is dominated by the Wet CCI sequences.

(2) The only probabilistic output which shows any significant variation is drywell head seal overtemperature leakage (Pen OT) which exhibits an increase of magnitude with an increase in the passive flooder failure probability. The change in seal leakage is much less than the change in passive flooder failure probability since high RPV pressure sequences with entrainment of debris to the upper drywell and failure of the upper drywell sprays dominate the seal leakage sequences in the base analysis.

(3) Even for the case where the passive flooder is assumed to be unavailable, the frequency associated with the Dry CCI is extremely small. Since only the Dry CCI cases have failure of the passive flooder, this frequency represents an upper bound for the impact of passive flooder failure on offsite dose.

Thus, it is seen that the lower drywell flooder has negligible impact on net risk. Therefore, no chart of the impact on risk was created. The value of the lower drywell flooder system is not measured as a direct impact on risk. Rather, it should be viewed as a passive system which serves to limit the impact of uncertainty in operator actions and allows the ABWR design to mitigate a severe accident in a purely passive manner.

19E.2.8.2.8 Summary

The passive flooder meets its design goal of preventing or, at least, mitigating core concrete interaction in the lower drywell. The flow rate required to remove the heat generated in the debris bed is 0.018 m^3/s which can be provided by two of the ten flooder lines. The expected flow rate is 0.099 m^3/s (nine of the ten lines active). If the expected flow rate is achieved, a one-meter layer of water will be established above the bed in a time between 21 minutes and 1.8 hours after flow initiation. One flooder line opening first is not expected to prevent the other lines from opening during a severe accident in which significant amounts of core debris is present in the drywell. The flooder lines will become active within ten minutes of the lower drywell gas space reaching 533 K. The passive flooder has negligible impact on the net risk of the plant since it provides a redundant function to the firewater addition system.

19E.2.8.3 Corium Shield

During a hypothetical severe accident in the ABWR, molten core debris may be present on the lower drywell floor. The EPRI ALWR Requirements Document specifies a floor area of at least 0.02 m^2/MWt to promote debris coolability. This has been interpreted in the ABWR design as a requirement for an unrestricted lower drywell floor area of 79 m^2.

The ABWR has two drain sumps in the periphery of the lower drywell floor which could collect core debris during a severe accident if ingression is not prevented. If ingression occurs, a debris bed will form in the sump which has the potential to be deeper than the bed on the lower drywell floor. Debris coolability becomes more uncertain as the depth of a debris bed increases. Therefore, debris should be kept out of the sumps.

The two drain sumps have different design objectives. One, the floor drain (HCW) sump, collects water which falls on the lower drywell floor. The other, the equipment drain (LCW) sump, collects water leaking from valves and piping. Both sumps have pumps and instrumentation which allow the plant operators to determine water leakage rates from various sources. Plant shutdown is required when leakage rate limits are exceeded for a certain amount of time. A more complete discussion on the water collection system can be found in Subsection 5.2.5.

Debris will be prevented from entering into the lower drywell sumps by shield walls (corium shields) built around their periphery. The shields will be constructed from material which will prevent or minimize interactions with the core debris. The shield for the floor drain sump will have channels at floor level that allow nearly unrestricted water flow at rates on the order of and somewhat greater than the leakage limits. The channels will be sized so that they plug with core debris during a severe accident; thus preventing debris ingression into the sump. The equipment drain sump will be solid. A complete description of corium shields can be found in Attachment 19ED.

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Table 19E.2-1 Potential Suppression Pool Bypass Lines

	Pathway					Basis For
Description	**Number of Lines**	**From**	**To**	**Size (mm) (1 in. = 25.4 mm)**	**Isolation Valves**	**Exclusion (See Notes)**
Main Steam	4	RPV	ST	700	(AO, AO)	-
Main Steam Line Drain	1	RPV	ST	80	MO, MO	3
Feedwater	2	RPV	ST	550	CK, CK	-
Reactor Inst. Lines	37	RPV	RB	6	CK	-
CRD Insert	205	RPV	RB	1	CK, MA	1
HPCF Discharge	2	RPV	RB	200	CK, MO	-
HPCF Equalizing	2	RPV	RB	20	MO, MO	-
HPCF Suction	2	SP	RB	400	MO	2
Supp Pool Instrumentation	6	SP	RB	6	CK	2
SLC Injection	1	RPV	RB	40	CK, CK	-
RCIC Steam Supply	1	RPV	RB	150	(MO, MO)	-
RCIC Discharge	1	RPV	RB	150	CK, MO	5
RCIC Min. Flow	1	SP	RB	150	MO	2
RCIC Suction	1	SP	RB	200	MO	2
RCIC Turbine Exhaust	1	SP	RB	350	MO, CK	2
RCIC Turb. Exh Vac Bkr	1	SP	RB	40	CK, CK	2
RCIC Vac Pump Discharge	1	SP	RB	50	MO, CK	2
RHR LPFL Discharge	2	RPV	RB	250	CK, MO	-
RHR Equalizing Lines	2	RPV	RB	20	MO, MO	-
RHR Wetwell Spray	2	WW	RB	100	MO	2,4
RHR Drywell Spray	2	DW	RB	200	MO, MO	4
RHR SDC Suction	3	RPV	RB	350	MO, MO	3
CUW Suction	1	RPV	RB	200	(MO, MO, MO)	-
CUW Return	1	RPV	RB	200	MO, MO	5
CUW Head Spray Line	1	RPV	RB	150	CK, MO, MO	3
CUW Instrument Lines	4	RPV	RB	6	CK	-
Post Accident Sampling	4	RPV	RB	25	(MO, MO)	-
RIP Motor Purge	10	RPV	RB	<1	CK, CK	1
RIP Cooling Water	4	RPV	RB	200	MO, MO	1

Table 19E.2-1 Potential Suppression Pool Bypass Lines (Continued)

Description	Number of Lines	Pathway From	Pathway To	Size (mm) (1 in. = 25.4 mm)	Isolation Valves	Basis For Exclusion (See Notes)
LDS Instruments	9	RPV	RB	6	CK	-
SPCU Suction	1	SP	RB	200	MO, MO	2
SPCU Return	1	SP	RB	250	MO, CK	2
Cont. Atmosphere Monitor	6	DW	RB	20	MO	-
LDS Samples	2	DW	RB	30	(SO, SO)	-
Drywell Sump Drains	2	DW	RB	100	MO, MO	-
HVCW/RBCW Supply	4	DW	RB	125	CK, MO	1
HVCW/DWCW Return	4	DW	RB	125	MO, MO	1
DW Exhaust/SGTS	1	DW	RB	550	AO, AO	7
Wetwell Vent to SGTS	1	WW	RB	550	AO, AO	2
DW Purge	1	DW	RB	350	AO	-
Inerting Makeup	1	DW	WW	50	AO, AO	-
WW Inerting/Purge	1	WW	RB	550	AO, AO	2
Instrument Air (and Service Air)	2	DW	RB	50	CK, MO	1
SRV Pneumatic Supply	3	DW	RB	50	CK, MO	1
Flammability Control	2	DW	RB	100	(AO, MO)	3
ADS/SRV Discharge	8	RPV	WW	300	RV	-
ACS Supply	2	DW	WW	550	AO, AO	-
WW/DW Vacuum Breaker	8	DW	WW	500	CK	-
Miscellaneous Leakage	1	DW	RB	---	NONE	6
Access Tunnels	2	DW	RB	---	NONE	6

NOTES:

Legends and Acronyms

Pathway

Source (From)		Termination (To)	
RPV	Reactor Pressure Vessel	WW	Wetwell
DW	Drywell	RB	Reactor Building
SP	Suppression Pool	WW	Wetwell
		ST	Steam Tunnel

Isolation Valve Types

AO	Air Operated
MO	Motor Operated
RV	Relief Valve
SP	Suppression Pool
CK	Check Valve
MA	Manually Actuated
SO	Solenoid Operated
( )	Common Mode Failure Potential [Subsection 19E.2.3.3.3(2)]

Bases for Exclusion

(1) Closed system such as closed cooling water systems which do not directly connect to the RPV or containment atmosphere require two failures to become a bypass pathway: a leak or break within the cooled component and a line break outside of containment. Very low flow is expected out of the break or leak at the cooled component is likely due to the high degree of restriction. These pathways are not considered further on the basis of this very low flow rate. Similarly, internal restrictions within the CRD and the ball check valve in the drive flange provides the basis for excluding these lines.

(2) Pathways which originate in the primary containment wetwell airspace or the suppression pool are excluded because fission product aerosols would first be trapped in the suppression pool and would thus not be available for release through the bypass path.

(3) Some lines are closed during normal plant operation and would not be expected to be opened in the short term following a plant accident. These lines are excluded on the basis of low frequency of use. Furthermore, should a bypass pathway develop later when the line is used, the fission product source term would be expected to have been already significantly reduced due to decay and other removal mechanisms.

(4) Some lines which originate in the primary containment are designed for operating pressures higher than would be expected in the containment during a severe accident. These lines [with design pressures greater than about 0.790 MPa] were excluded since the probability of a break under less than normal operating pressures and coincident with the severe accident is extremely small.

(5) Some lines return to the feedwater line. These pathways (such as LPFL loop A and CUW) are excluded since they are bounded by the evaluation of feedwater.

(6) Acceptable long term leakage from the containment to the reactor building following a design basis accident is specified at 0.5% of containment volume per 24 hours. During severe accident conditions this leakage could be somewhat greater due to higher than design basis containment pressure. The contribution of this leakage to overall risk is considered in Subsection 19E.2.3.4. A discussion of the drywell access tunnels is included in Appendix 19F.

(7) Drywell purge lines are normally closed a fail closed. The potential for inadvertent opening is considered remote and is addressed by Emergency Procedure guidelines.

Table 19E.2-2 ABWR Plant Ability to Cope with Station Blackout for up to 8 Hours

Plant Parameter		Design Basis Value	Station Blackout Basis
a)	RPV Level	Core covered	Core covered
	RPV Pressure	0.446 MPa RCIC trip	>1.1356 MPa
		1.1356 MPa RCIC rated flow	
b)	D.C. Battery Capacity	11,400 amp-h	Sufficient with load shedding
		Div. 1, 2, 3 & 4	
c)	RCIC Water Source	1) CST - 566 m^3 (20 x 10^3 ft^3) 2) Suppression pool - 3566 m^3 (126 x 10^3 ft^3)	CST sufficient with RPV pressurized
d)	RCIC Room Temperature	339 K (66°C)	<339 K (66°C)
e)	Drywell Temperature	444 K (171°C)	<444 K (171°C)
f)	Drywell Pressure	0.41 MPa	<0.41 MPa
g)	Wetwell Temperature	377 K (104°C)	<377 K (104°C)
h)	Wetwell Pressure	0.41 MPa	<0.41 MPa
i)	Control Rooms		
	- Main	331 K (58°C)	331 K (58°C)
	- Lower	331 K (58°C)	331 K (58°C)
	- Computer	331 K (58°C)	331 K (58°C)

Table 19E.2-3 Definition of Accident Sequence Codes

Characters 1 to 4: General Condition Indicator	
LCLP	Loss of All Core Cooling with Vessel Failure occurring at Low Pressure
LCHP	Loss of All Core Cooling with Vessel Failure occurring at High Pressure
SBRC	Station Blackout with RCIC operating for 8 hours
LHRC	Loss of Heat Removal in the Containment
LBLC	Large Break LOCA with Loss of All Core Cooling
NSCL	Transient without Scram and with Failure of All Core Cooling, Vessel Failure occurs at Low Pressure
NSCH	Transient without Scram and with Failure of All Core Cooling, Vessel Failure occurs at High Pressure
NSRC	Station Blackout without Scram, RCIC operates
Characters 5 and 6: Mitigating Features	
00	No mitigating features operated
IV	In-Vessel Recovery
PF	Passive Flooder
FA	Firewater Addition System Injects into the Vessel
HR	Containment Heat Removal
PS	Passive Flooder and Drywell Spray
FS	Firewater Addition System switched to Drywell Spray Mode
PB	Passive Flooder with Suppression Pool Bypass
Character 7: Mode of Release	
N	Normal Containment Leakage
P	Leakage through Moveable Penetrations
R	Containment Overpressure Protection System Rupture Disk Opening
D	Drywell Head Failure
E	Early Containment Structural Failure
S	Suppression Pool Failure
Character 8: Magnitude of Release	
0	No core damage, no fission product release
N	Negligible: Less than 0.1% volatile fission products
L	Low: 0.1% to 1% volatile fission products
M	Medium: 1% to 10% volatile fission products
H	High: More than 10% volatile fission products

Table 19E.2-4 Grouping of Accident Classes into Base Sequences

Accident Class	Initiator Code	Base Sequence Subsection Number
IA	LCHP	19E.2.2.2
IB-1	LCLP	19E.2.2.1
	LCHP	19E.2.2.2
IB-2	SBRC	19E.2.2.3
IB-3	LCLP	19E.2.2.1
	LCHP	19E.2.2.2
IC	NSCL	19E.2.2.6
ID	LCLP	19E.2.2.1
IE	NSCH	19E.2.2.7
II	LHRC	19E.2.2.4
IIIA	LCHP	19E.2.2.2
IIID	LBLC	19E.2.2.5
IV-1	NSRC	19E.2.2.8

Table 19E.2-5 Sequence of Events for LCLP-PF-R-N

Loss of All Core Cooling with Vessel Failure at Low Pressure
Passive Flooder Operates and Drywell Head Fails

Time	Event
0.0	MSIV Closure
4.2 s	Reactor Scrammed
0.4 h	Indicated Water Level at 2/3 Core Height One SRV Opened by Operator
1.8 h	Vessel Failed
2.7 h	Water in Lower Drywell Boiled Off Corium Heatup Begins
5.4 h	Passive Flooder Opens
20.2 h	Rupture Disk Opens

Table 19E.2-6 Sequence of Events for LCLP-FS-R-N

Loss of All Core Cooling with Vessel Failure at Low Pressure
Firewater Addition System Injects and Rupture Disk Opens

Time	Event
0.0	MSIV Closure
4.2 s	Reactor Scrammed
0.4 h	Indicated Water Level at 2/3 Core Height One SRV Opened by Operator
1.8 h	Vessel Failed
2.7 h	Water in Lower Drywell Boiled Off Corium Heatup Begins
4.0 h	Firewater Spray Started
7.0 h	Suppression Pool Overflows to the Lower Drywell
23.6 h	Firewater Spray Stopped
31.1 h	Rupture Disk Opened
56.6 h	Water in Lower Drywell Boiled Off
61.1 h	Passive Flooder Opened

Table 19E.2-7 Sequence of Events for LCHP-PS-R-N

Loss of All Core Cooling with Vessel Failure at High Pressure
Passive Flooder and Drywell Spray Operates, Rupture Disk Opens

Time	Event
0.0	MSIV Closure
4.2 s	Reactor Scrammed
0.3 h	Core Uncovered
2.0 h	Vessel Fails Corium and Water Entrained into Upper Drywell
2.0 h	Passive Flooder Opens
4.0 h	Drywell Spray Initiated
25.0 h	Rupture Disk Opens

Table 19E.2-8 Sequence of Events for LCHP-PF-P-M

Loss of All Core Cooling with Vessel Failure at High Pressure
Passive Flooder Operates, Penetration Leakage Occurs

Time	Event
0.0	MSIV Closure
4.2 s	Reactor Scrammed
0.3 h	Core Uncovered
2.0 h	Vessel Fails Corium and Water Entrained into Upper Drywell
2.0 h	Passive Flooder Opens
2.1 h	Seal Degradation Temperature Reached
18.1 h	Leakage Begins through Moveable Penetrations Fission Product Release Begins

Table 19E.2-9 Sequence of Events for SBRC-FA-R-0

Station Blackout with RCIC Operational for 8 Hours
Firewater Addition to Vessel Used to Prevent Core Damage, Rupture Disk Opens

Time	Event
0.0	MSIV Closure
4.2 s	Reactor Scrammed
52.0 s	RCIC Injection, Suction from CST
1.3 h	RCIC Suction Switched to Suppression Pool
4.4 h	RCIC Suction Switched to CST
8.0 h	RCIC Failure
9.0 h	Suppression Pool began to overflow to Lower Drywell
9.8 h	Manual ADS
9.9 h	Collapsed Water Level Falls below Top of Active Fuel Firewater Addition System Injection Begins
32.3 h	Rupture Disk Opened

Table 19E.2-10 Sequence of Events for SBRC-PF-R-N

Station Blackout with RCIC Operational for 8 Hours
Passive Flooder Operates and Rupture Disk Opens

Time	Event
0.0	MSIV Closure
4.2 s	Reactor Scrammed
52.0 s	RCIC Injection, Suction from CSP
1.3 h	RCIC Suction Switched to Suppression Pool
4.4 h	RCIC Suction Switched to CSP
8.0 h	RCIC Failure
9.3 h	Core Uncovered
9.7 h	One SRV opened by operator
12.3 h	Vessel Fails
21.1 h	Lower Drywell Water Boils Away
23.5 h	Passive Flooder Opens Rupture Disk Opens

Table 19E.2-11 Sequence of Events for LHRC-00-R-0

Isolation with Loss of Containment Heat Removal
Rupture Disk Opens

Time	Event
0.0	MSIV Closure
4.2 s	Reactor Scrammed
1.1 min	RCIC Injection
2.9 h	Manual Open 1 SRV
3.0 h	HPCF Injection
3.1 h	RCIC Trip on Low Turbine Pressure
21.7 h	Rupture Disk Opens
>72 h	Potential Loss of Core Cooling

Table 19E.2-12 Sequence of Events for LBLC-PF-R-N

Large Break LOCA With Loss of Core Cooling
Passive Flooder Operates and Rupture Disk Opens

Time	Event
0.0	Main Steam Line Break
0.2 s	High Drywell Pressure Signal
4.4 s	Reactor Scrammed
14.9 s	MSIV Closed
2.8 min	Core Uncovered
1.4 h	Vessel Failed
5.7 h	Passive Flooder Opened
19.1 h	Rupture Disk Opens

Table 19E.2-13 Sequence of Events for NSCL-PF-R-N

Concurrent Loss of All Core Cooling and ATWS with Vessel Failure at Low Pressure
Passive Flooder Operates and Rupture Disk Opens

Time	Event
0.0	MSIV Closure
3.7 s	Core Uncovered
0.5 h	One SRV Opened by Operator
1.3 h	Vessel Failed
1.9 h	Water in Lower Drywell Boiled Off Corium Heatup Begins
4.4 h	Passive Flooder Opens
18.7 h	Rupture Disk Opens

Table 19E.2-14 Sequence of Events for NSCH-PF-P-M

Concurrent Loss of All Core Cooling and ATWS with Vessel Failure at High Pressure
Passive Flooder Operates, Penetration Leakage Occurs

Time	Event
0.0	MSIV Closure
3.6 min	Core Uncovered
1.3 h	Vessel Fails Corium and Water Entrained into Upper Drywell
1.4 h	Passive Flooder Opens
1.4 h	Seal Degradation Temperature Reached
17.8 h	Leakage Begins Through Moveable Penetrations Fission Product Release Begins

Table 19E.2-15 Sequence of Events for NSRC-PF-R-N

Concurrent Station Blackout and ATWS
Passive Flooder Operates, Rupture Disk Opens

Time	Event
0.0	MSIV Closure
4.1min	Core Uncovered
1.9 h	Suppression Pool Began to Overflow to Lower Drywell
3.6 h	RCIC Tripped
3.8 h	SRV Opened
5.6 h	Vessel Failed
8.6 h	Rupture Disk Opened Fission Product Release Begins

Table 19E.2-16 Summary of Critical Parameters for Severe Accident Sequences

Accident	Time of Vessel Failure	Fission Product Release Time	Time of Rupture Disk Failure	End of CsI Release	Release Fraction of CsI @ 72 hours
LCLP-PF-R-N	1.8 h	20.2 h	20.2 h	100 h	<1E–7
LCLP-FS-R-N	1.8 h	31.1 h	31.1 h	76 h	1E–7
LCHP-PS-R-N	2.0 h	25.0 h	25.0 h	50 h	<1E–7
LCHP-FS-R-N	2.0 h	50 h*	50 h*	125 h*	<1E–7*
LCHP-PF-P-M	2.0 h	18.1 h	N/A	70 h	8.8E–2
SBRC-PF-R-N	12.3 h	23.5 h	23.5 h	100 h	<1E–7
LBLC-PF-R-N	1.4 h	19.1 h	19.1 h	125 h	<1E–7
LBLC-FS-R-N	1.4 h	29.5 h	29.5 h	67 h	<1E–7
NSCL-PF-R-N	1.3 h	18.7 h	18.7 h	105 h	<1E–7
NSCL-FS-R-N	1.3 h	30.7 h	30.7 h	69 h	<1E–7
NSCH-PF-P-M	1.3 h	17.8 h	N/A	65 h	7E–2
NSCH-FS-R-N	1.3 h	50 h*	50 h*	125 h*	<1E–7*
NSRC-PF-R-N	5.6 h	8.6 h	8.6 h	110 h	<1E–7
NSRC-FS-R-N	5.6 h	26.4 h	26.4 h	120 h	<1E–7

* Release parameters are approximate. See sequence discussion for more detail.

Table 19E.2-17 Important Parameters for Steam Explosion Analysis

Symbol	Value	Description
m'	500 kg/s	Mass Flow Rate of Corium from Vessel
Q	0.056 m^3/s	Volumetric Flow of Corium from Vessel
α	7.E-6 m^2/s	Thermal Diffusivity of Corium
c_v	480 J/kg-K	Specific Heat of Corium
ρ	9000 kg/m^3	Density of Corium
σ	1.0 N/m	Surface Tension of Molten Corium
h	390 W/m^2-K	Heat Transfer Coefficient for Corium Droplet
T_i	2600 K	Initial Temperature of Corium Droplet
ρ_a	1.1 kg/m^3	Density of Air
ρ_L	1000 kg/m^3	Density of Water
$v_{fg}(P_\infty)$	1.7 m^3/kg	Specific Volume of Evaporation for Water
$h_{fg}(P_\infty)$	2257 kJ/kg	Specific Enthalpy of Evaporation for Water
L	5.5 m	Height of Water in Lower Drywell
A_L	88.2 m^2	Area of Lower Drywell
H	6 m	Distance from Bottom of Vessel to Surface of Water in Lower Drywell

Table 19E.2-18 Potential Bypass Pathway Matrix*

	From			
To	RPV	Drywell	Wetwell Airspace	Suppression Pool
Drywell	No	NA	NA	NA
Wetwell Airspace	Yes†	Yes†	NA	NA
Reactor Building	Yes	Yes	Yes	Yes
Turbine Building	Yes	Yes	Yes	Yes

* This matrix shows the paths that potentially bypass the suppression pool.

† Pathways which originate in the drywell and potentially release into the wetwell are potential bypass paths if the containment is vented or the wetwell fails during the severe accident.

Table 19E.2-19 Flow Split Fractions

Line Size		Flow Split Fraction	
mm	in	RPV Source	Drywell Source
6	0.25	1.5E–05	5.4E–05
12	0.5	9.4E–05	3.4E–04
25	1	5.7E–04	2.0E–03
50	2	3.3E–03	1.2E–02
100	4	1.8E–02	6.2E–02
150	6	4.8E–02	1.5E–01
200	8	8.9E–02	2.5E–01
250	10	1.4E–01	3.6E–01
300	12	2.0E–01	4.6E–01
350	14	2.6E–01	5.4E–01
400	16	3.2E–01	6.2E–01
450	18	3.8E–01	6.7E–01
500	20	4.3E–01	7.2E–01
700	28	6.1E–01	8.4E–01
1000	40	7.7E–01	9.2E–01

Table 19E.2-20 Failure Probabilities

Symbol	Description	Prob/Event*	Basis
P1	Main Steam Isolation Valve Common Mode Failure		a
P2	MSIV leakage probability		b
P3	Turbine Bypass Isolation		c
P4	Main condenser failure		c
P5	MSL break outside containment		d
P6	Air operated valve (NO)		e
P7	Motor -operated valve (NO)		e
P8	Motor-operated valve (NO)		f
P9	Check Valve		g
P10	Motor-operated valves (NC)		h
P11	Motor-operated valves (NC)		i
P12	Inadvertent opening		j
P13	Small line break		k
P14	Medium line break		k
P15	Large line break		k
P16	CUW line break		k

* Probabilites not part of DCD (Refer to SSAR)

Table 19E.2-21 Summary of Bypass Probabilities

Pathway	Flow Split Fraction	Bypass Probability Equation	Bypass Probability*	Bypass Fraction*	Figures 19E.2-19a to 19E.2-19k
Lines from the RPV					
Main Steam	6.7E–1	4*P1*(P3*P4+P5)			A
Main Steam Leakage	2.2E–5	4*P2*(P3*P4+P5)			A
Feedwater	5.2E–1	2*P9*P9*P15			B
Reactor Inst. Lines	3.1E–5	30*P13*P9			D
HPCF Discharge†	1.1E–1	2*P9*P10*P14			C
HPCF Equalizing Line†	1.0E–3	2*P10*P11*P13			C
SLC Injection	3.0E–3	1*P9*P13			B
RCIC Steam Supply	6.9E–2	1*P8*P14			E
LPFL Discharge†	1.7E–1	2*P9*P10*P15			C
LPFL Equalizing Line†	1.0E–3	2*P10*P11*P13			C
CUW Suction	1.2E–1	1*P8*P14			E
CUW Inst Lines	3.1E–5	4*P13*P9			D
Post Acc Sampling	1.0E–3	4*P8*P11			J
LDS Instruments	3.1E–5	9*P13*P9			D
SRV Discharge	6.9E–2	8*P14			K
Lines from the Drywell					
Cont Atmos Monitor	8.9E-4	6*P8*P13			J
LDS Samples	1.7E-3	2*P8*P11			J
Drywell Sump Drain	3.0E-2	2*P8*P13			J
DW Purge	5.4E-1	1*P6*P11			I
Inerting Makeup	1.2E-2	1*P6			I
ACS Supply	7.5E-1	2*P12*P6			H
WW-DW Vac Bkr‡	2.6E-1	8*P9			G
		Grand Total excluding vacuum breaker			
			Goal		

* Probabilites not part of DCD (refer to SSAR).
† These lines may be excluded for station blackout events.
‡ Addressed on Containment Event Trees.

Table 19E.2-22 NUREG/CR-4551 Grand Gulf APET Events by Category

Event Number	Description
Plant Damage State Grouping Events	
1	Initiating Event Type
2	Station Blackout
3	DC Power Availability
4	S/RV Fails to Reclose
5	HPCI Failure
6	RCIC Failure Initially
7	CRD Injection Failure
8	Condensate System Failure
9	LPCS/LPCI Systems Failure
10	RHR Failure
11	Service Water/LPCI Crosstie Failure
12	Fire Protection Crosstie Failure
13	Containment Spray Failure
14	Vessel Depressurization
15	Time Core Damage
20	Plant Damage State Summary
Structural Capacity/Initial Containment Status	
16	Containment Isolation (Pre-existing Leakage)
17	Extent of Pool Bypass Initially
18	Containment Capacity (Quasi-static/Dynamic Loading)
19	Drywell Capacity (Quasi-static/Dynamic Loading)
Systems Behavior/Operator Actions	
21	Ignitors Turned On Before Core Damage
22	Containment Vented Before Core Damage
23	SRV Vacuum Breakers Stick Open
26	RV Pressure During Core Damage
27	Status of Hydrogen Ignitors Before Vessel Breach
28	RV Injection Restored During Core Damage
30	Containment Spray Status
53	Upper Pool Dump

Table 19E.2-22 NUREG/CR-4551 Grand Gulf APET Events by Category (Continued)

Event Number	Description
81	Containment Spray Status Following Vessel Breach
103	Containment Vented Following Vessel Breach
106	Containment Spray Status Late
119	Containment Vented Late
AC/DC Power Availability	
24	AC Power Recovered During Core Damage
25	DC Power Available During Core Damage
79	AC Power Recovered Following Vessel Breach
80	DC Power Available Following Vessel Breach
104	AC Power Recovered Late
105	DC Power Available Late
Criticality	
29	Core in Critical Configuration Following Injection Recovery
Hydrogen Related Phenomena/Issues	
31	Amount Oxygen in Wetwell During Core Damage
32	Amount Oxygen in Drywell During Core Damage
33	Amount Steam in Containment During Core Damage
34	Amount Steam in Drywell During Core Damage
35	Amount Hydrogen Released In-vessel During Core Damage
36	Level In-vessel Zirconium Oxidation
39	Max. Hydrogen Concentration in Wetwell Before Vessel Failure
40	Extent Wetwell Inert During Core Damage
41	Diffusion Flames Consume Hydrogen Before Vessel Breach
42	Max. Hydrogen Concentration in Drywell Before Vessel Failure
43	Deflagrations in Wetwell Before Vessel Breach
44	Detonation in Wetwell Before Vessel Breach
45	Containment Impulse Load Before Vessel Breach
46	Hydrogen Burn Efficiency Before Vessel Breach
47	Peak Hydrogen Burn Containment Pressure
48	Extent of Drywell Leakage Due to Early Detonation in Containment
49	Extent of Containment Leakage Due to Early Detonation in Containment

Table 19E.2-22 NUREG/CR-4551 Grand Gulf APET Events by Category (Continued)

Event Number	Description
56	Extent Drywell Inert at Vessel Breach
57	Sufficient Hydrogen in Drywell for Combustion/Detonation Before Vessel Breach
65	Detonation in Drywell at Vessel Breach
66	Deflagration in Drywell at Vessel Breach
68	Amount Hydrogen Released at Vessel Breach
69	How Much Hydrogen Released at Vessel Breach
78	Hydrogen Concentration in Containment Immediately After Vessel Breach
82	Extent Wetwell Inert After Vessel Breach
83	Sufficient Oxygen in Containment for Combustion
84	Hydrogen Ignition in Containment at Vessel Breach
85	Hydrogen Ignition in Containment Following Vessel Breach
86	Hydrogen Detonation in Wetwell Following Vessel Breach
87	Impulse Loading to Containment Following Vessel Breach
88	Hydrogen Burn Efficiency Following Vessel Breach
89	Peak Containment Pressure From Hydrogen Burn at Vessel Breach
91	Extent of Drywell Leakage Due to Detonation in Containment at Vessel Breach
92	Extent of Containment Leakage Due to Detonation at Vessel Breach
101	Hydrogen (and CO) Produced During CCI
102	Level Zirconium Oxidation in Pedestal Before CCI
107	Late Concentration Combustible Gases in Containment
108	Level Wetwell Inert After Vessel Breach
109	Sufficient Oxygen in Containment for Late Combustion
110	Hydrogen Ignition in Containment Late
111	Detonation in Wetwell Following Vessel Breach
112	Containment Impulse Load Late
113	Hydrogen Burn Efficiency Late
114	Peak Containment Pressure From Late Hydrogen Burn
115	Extent of Drywell Leakage Due to Detonation in Containment Late
116	Extent of Containment Leakage Due to Late Detonation
117	Level of Containment Leakage Due to Late Combustion
118	Level of Drywell Leakage Due to Late Combustion

Table 19E.2-22 NUREG/CR-4551 Grand Gulf APET Events by Category (Continued)

Event Number	Description
Containment/Drywell Pressurization/Failure	
37	Containment Pressure During Core Damage
38	Extent of Containment Leakage Due to Slow Pressurization Before Vessel Breach
50	Level Containment Leakage Before Vessel Breach
51	Level of Drywell Leakage Due to Containment Pressurization
52	Level Pool Bypass Following Early Combustion Events
55	Containment Pressure Before Vessel Breach
70	Drywell/Wetwell Pressure Differential Resulting from Vessel Breach
71	Peak Pedestal Pressure at Vessel Breach
72	Drywell Impulse Load at Vessel Breach Sufficient to Cause Failure
73	Drywell Pressurization at Vessel Breach Sufficient to Cause Failure
74	Pedestal Failure Due to Pressurization at Vessel Breach
76	Pedestal Failure Causes Drywell Failure
77	Containment Pressure at Vessel Breach Prior to Hydrogen Burn
90	Level Containment Pressurization At Vessel Breach
93	Level Containment Leakage Following Vessel Breach
94	Level of Drywell Leakage Due to Containment Pressurization
95	Level Pool Bypass Following Vessel Breach
96	Containment Pressure After Vessel Breach
122	Level Late Pool Bypass
123	Late Containment Pressure Due to Non-condensables or Steam
124	Late Containment Failure Due to Non-condensables or Steam
125	Long Term Level Containment Leakage
Core Concrete Interactions/Pedestal Failure	
54	Water in Reactor Cavity
97	Water Supplied to Debris Late
98	Water in Cavity After Vessel Breach
99	Nature of Core Concrete Interactions (CCI)
100	Fraction of Core Not Participating in HPME Participates in CCI
120	Amount Concrete Erosion to Fail Pedestal
121	Time of Pedestal Failure

Table 19E.2-22 NUREG/CR-4551 Grand Gulf APET Events by Category (Continued)

Event Number	Description
Steam Explosion Related	
58	Alpha Mode Event Fails Vessel and Containment
60	Large In-vessel Steam Explosion
62	In-vessel Steam Explosion Fails Vessel
67	Large Ex-vessel Steam Explosion
75	Pedestal Failure From Ex-vessel Steam Explosion
Core Damage Progression and Vessel Breach	
59	Fraction of Core Participating in Core Slump
61	Fraction Core Debris Mobile at Vessel Breach
63	Mode of Vessel Breach
64	High Pressure Melt Ejection

Table 19E.2-23 NRC Identified Parameters for Sensitivity Study from NUREG-1335

- Performance of containment heat removal systems during core meltdown accidents
  - In-vessel phenomena (primary system at high pressure)
  - H_2 production and combustion in containment
  - Induced failure of the reactor coolant system pressure boundary
  - Core relocation characteristics
  - Mode of reactor vessel melt-through
- In-vessel phenomena (primary system at low pressure)
  - H_2 production and combustion in containment
  - Core relocation characteristics
  - Fuel/coolant interactions
  - Mode of reactor vessel melt-through
- Ex-vessel phenomena (primary system at high pressure)
  - Direct containment heating concerns
  - Potential for early containment failure due to pressure load
  - Long-term disposition of core debris (coolable or not coolable)
- Ex-vessel phenomena (primary system at low pressure)
  - Potential for early containment failure due to direct contact by core debris
  - Water availability in cases with long-term core-concrete interactions
  - Long-term disposition of core debris (Coolable or not coolable)

Table 19E.2-24 Issues to be investigated in ABWR Sensitivity Analysis

In-vessel
Hydrogen generation
Core Blockage and Melt Progression
Fission Product release from core
CsI revaporization
Time of vessel failure
Recriticality following in-vessel recovery
Ex-vessel
Debris entrainment and direct containment heating
Mass of molten material at time of vessel failure
Mode of vessel breach
Potential for pedestal failure
Steam explosions
Mass of molten material at time of vessel failure
Presence of water in lower drywell at vessel failure
Potential for pedestal failure
Core concrete interaction and debris coolability
Debris-to-water heat transfer
Debris-to-crust heat transfer
Mass of molten material at time of vessel failure
Presence of water in lower drywell at vessel failure
Potential for pedestal failure
Non-condensable gas generation
Containment failure location and pressure
Containment failure area
Pool bypass
High temperature failure of drywell
Suppression Pool decontamination factor

Table 19E.2-25 Comparison of Volatile Fission Product Releases

Accident Sequence	CsI Release Fraction at 72 hours w/ COPS	CsI Release Fraction at 72 hours w/o COPS
LCLP-PF	< 1E–7	4.8%
LCLP-FS	1.5E–7	3.7%
LCHP-PF	8.8%[*]	8.8%[*]
LBLC-PF	< 1E–7	0.3%
LBLC-FS	< 1E–7	0.6%
NSCL-PF	< 1E–7	5.4%
NSCL-FS	< 1E–7	4.2%
NSCH-PF	7.3%	7.3%[*]
NSRC-PF	< 1E–7	37.0%
NSRC-FS	< 1E–7	14.5%

* Leakage through the moveable penetrations maintains containment pressure below the COPS setpoint.

Table 19E.2-26 Comparison of Low Pressure Core Melt Performance with and without Containment Overpressure Protection System

	w/ COPS	w/o COPS
Without Water Addition to Containment		
ΔP (Drywell-Wetwell)	0.1495 MPa	0.1495 MPa
Time of fission product release	20.2 h	27.5 h
CsI release fraction @ 72 hours	<1.E–7	4.8%
Probability of RHR recovery in time window*		
Probability of eventual DW head failure w/o CHR*		
With Water Addition to Containment		
ΔP (Drywell-Wetwell)	0.198 MPa	0.198 MPa
Time of fission product release	31.1 h	35.0 h
CsI release fraction @ 72 hours	1.5E–7	3.7%
Probability of RHR recovery in time window*		
Probability of eventual DW head failure w/o CHR*		

* Probabilities not part of DCD (Refer to SSAR).

Table 19E.2-27 Probability of Release Mode With and Without COPS

	Class I/III		Class II		
	RD Opens	DW Head Failure	RD Opens	DW Head Failure	Core Damage
Base Case (with COPS)*					
Without COPS*					

* Probabilities not part of DCD (Refer to SSAR).

Table 19E.2-28 Sensitivity Studies for Passive Flooder Reliability Frequencies of Important CET Results

	Failure Rate of Passive Flooder on Demand*			
	0.001	0.01	0.1	1.0
Type of CCI				
No CCI				
Wet CCI				
Dry CCI				
Pedestal Condition				
No Ped Failure				
Ped Failure				
FP Release Mode				
COPS				
DW Head				
Pen. Overtemperature				

* Probabilities not part of DCD (refer to SSAR).

Table 19E.2-29 Equipment and Instrumentation Required to Survive Severe Accident Scenarios

Equipment and Instrumentation	10CFR50.34(f)	In-Vessel Severe Accident	Ex-Vessel Severe Accident
Equipment			
RHR	+	+	+
ADS	+	+	-
ACIWA	+	+	+
Containment Structure	+	+	+
Pedestal	+	+	+
CIVs - Inboard	+	+	+
CIVs - Outboard	+	+	+
Electrical Penetrations	+	+	+
Mechanical Penetrations	+	+	+
Hatches	+	+	+
Passive Flooder	-	-	+
COPS	-	+	+
Vacuum Breakers	+	+	+
RIP Vertical Restraints	+	+	+
Recombiners	+	+	+
Instrumentation			
RPV Water Level	+	+	-
RPV Pressure	+	+	-
Suppression Pool Water Temperature	+	+	+
DW/WW Radiation Monitor	+	+	+
DW/WW H_2 Concentration	+	+	+
DW/WW O_2 Concentration	+	+	+
DW Temperature	+	+	+
DW Pressure	+	+	+
WW Pressure	+	+	+

Table 19E.2-29 Equipment and Instrumentation Required to Survive Severe Accident Scenarios (Continued)

Equipment and Instrumentation	10CFR50.34(f)	In-Vessel Severe Accident	Ex-Vessel Severe Accident
DW Water Level	+	+	+
WW Water Level	+	+	+

\+ Indicates that the equipment/instrumentation is required for the event,

\- Indicates that the equipment/instrumentation is not required for the event.

Table 19E.2-30 Material Properties Used in Tunnel Integrity Analysis

Steel	
K_{St}	30 W/mK
ρ_{St}	8000 kg/m^3
Cp_{St}	550 J/kgK
α_{St}	6.8 x 10^{-6} m^2/s
Debris	
k_C	8 W/mK
ρ_C	8000 kg/m^3
Cp_c	500 J/kgK
α_C	1.9 x 10^{-6} m^2/s

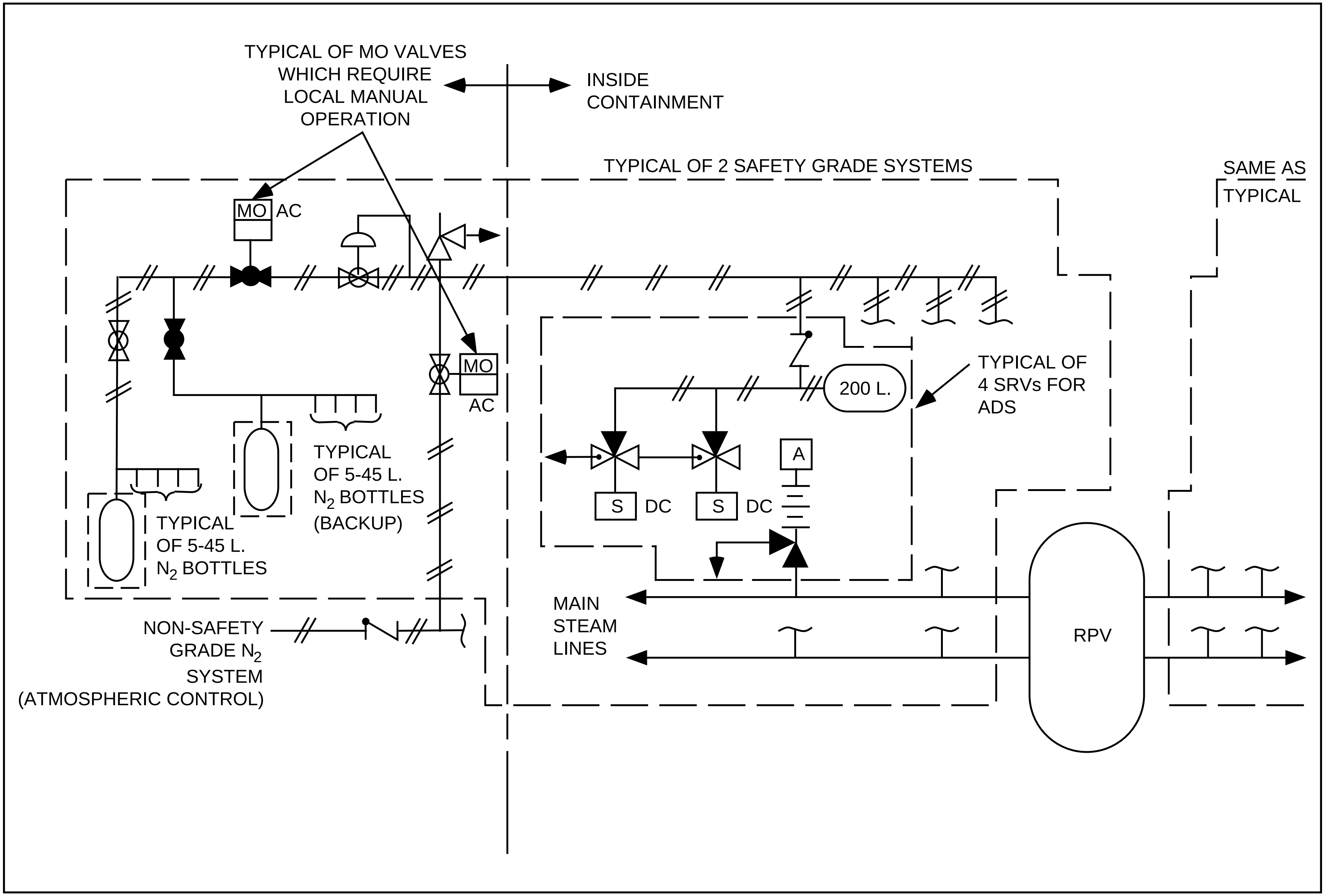


Figure 19E.2-1 Simplified Sketch of N_2 Supplies to Safety Grade ADS Valves

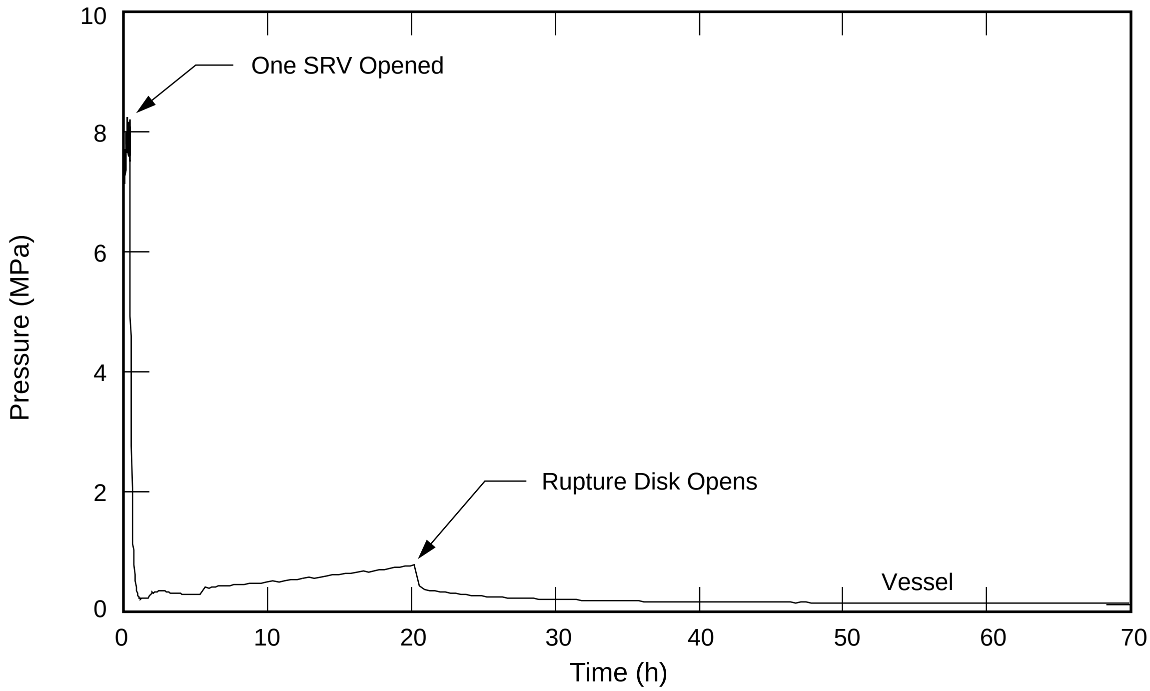


Figure 19E.2-2a LCLP-PF-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Vessel Pressure

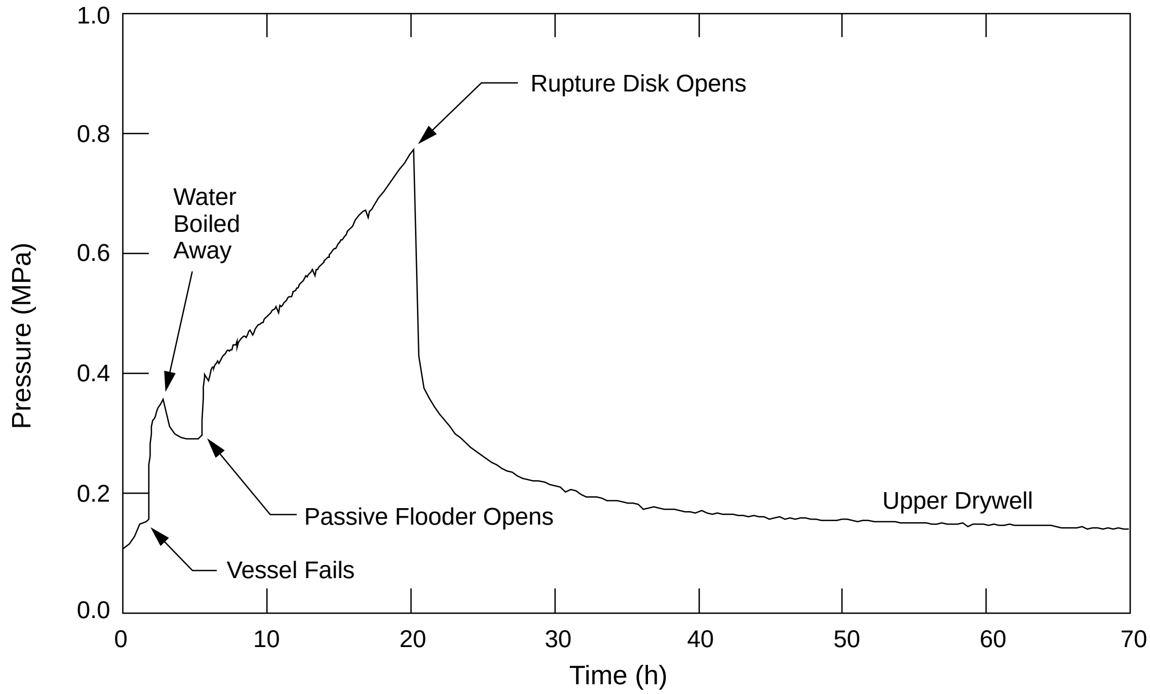


Figure 19E.2-2b LCLP-PF-R-N: Loss of all core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Drywell Pressure

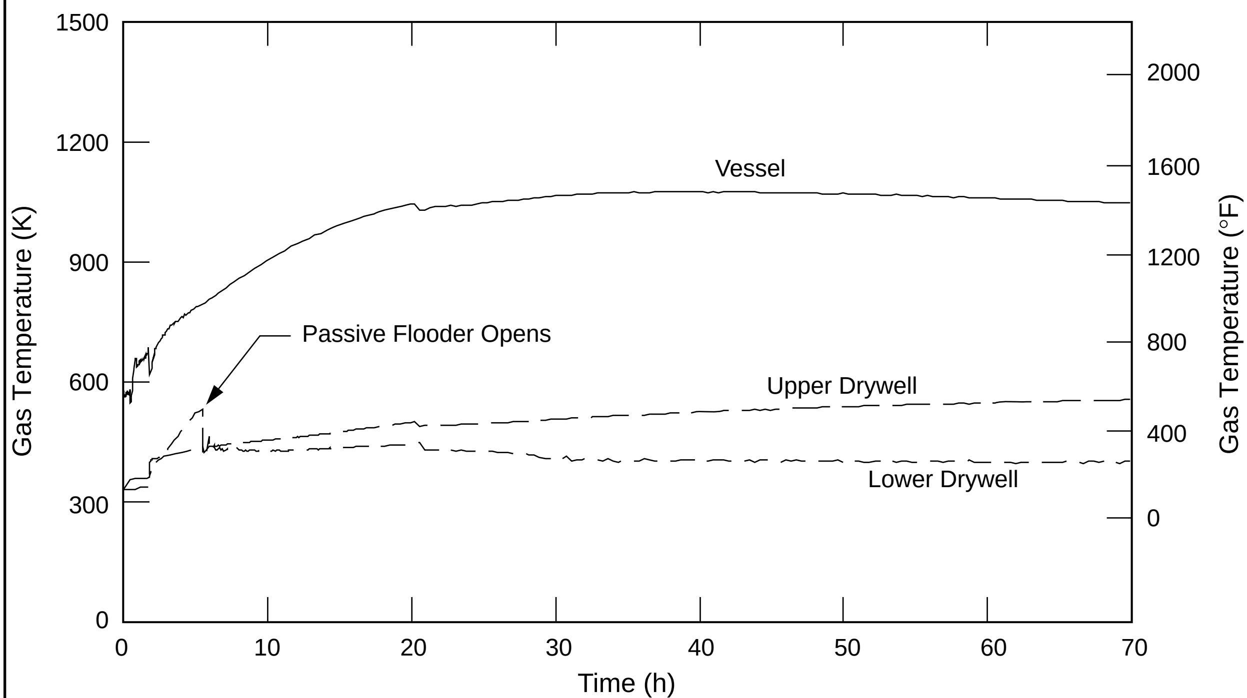


Figure 19E.2-2c LCLP-PF-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Gas Temperature

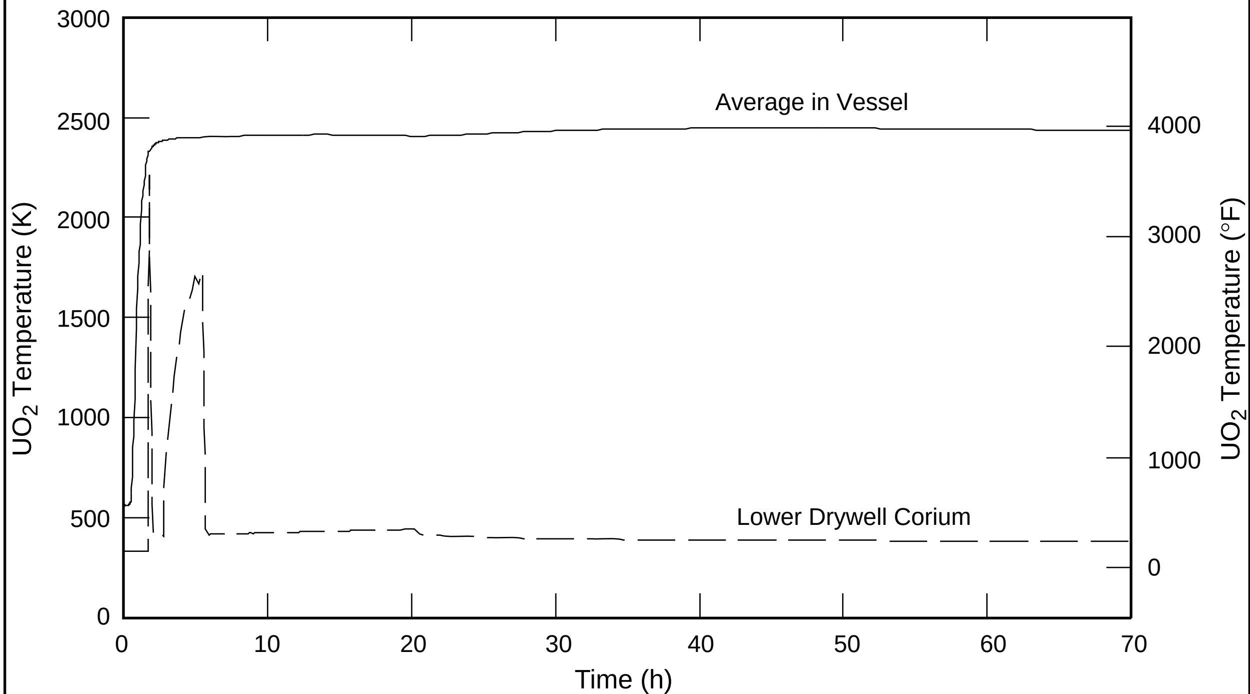


Figure 19E.2-2d LCLP-PF-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: UO_2 Temperature

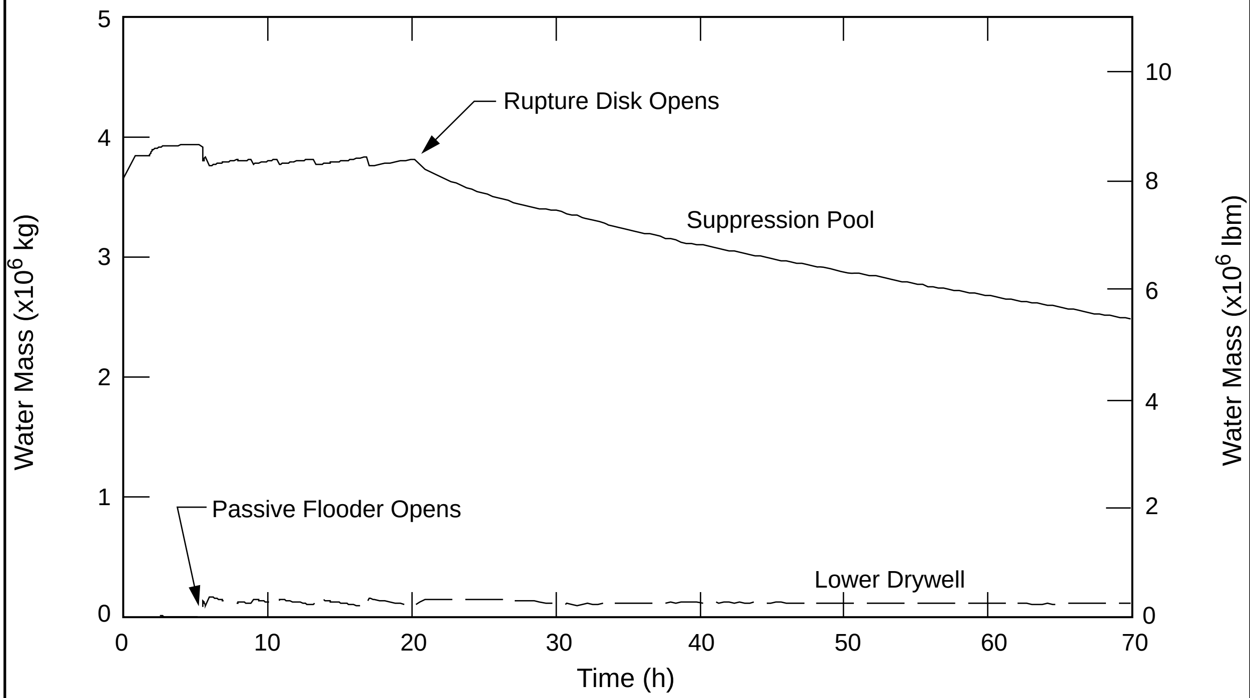


Figure 19E.2-2e LCLP-PF-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Water Mass

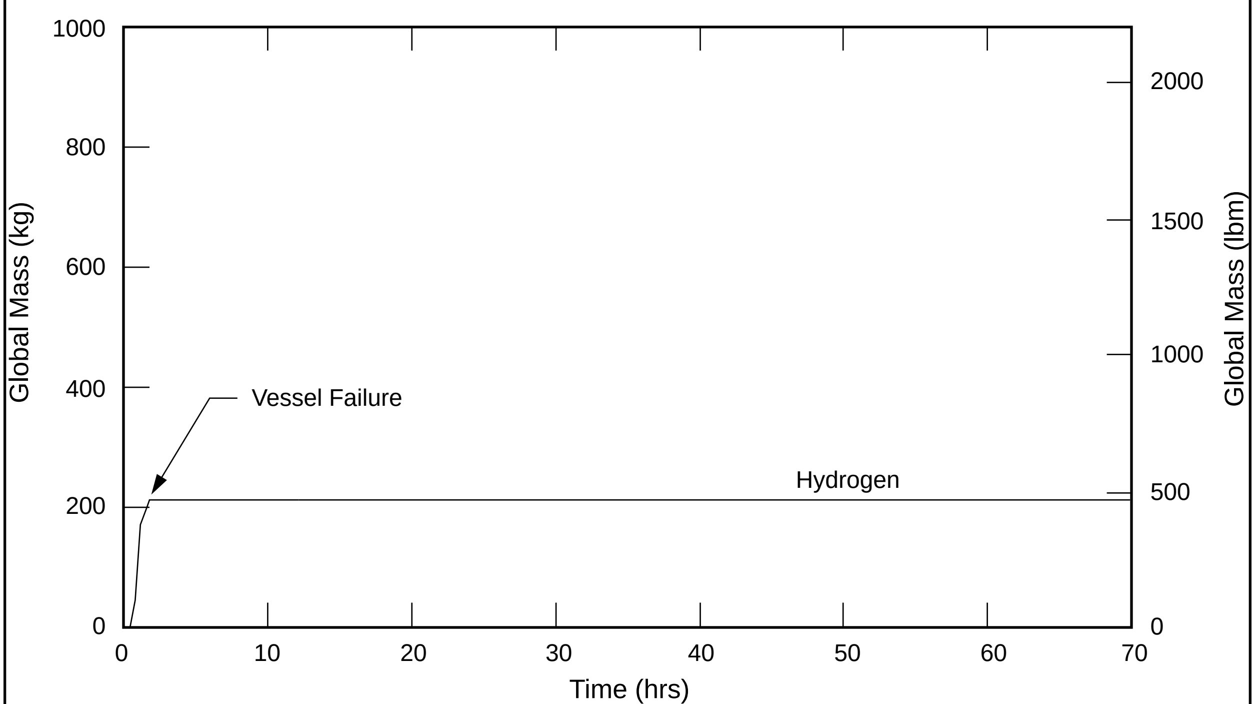


Figure 19E.2-2f LCLP-PF-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Mass of Non-Condensables

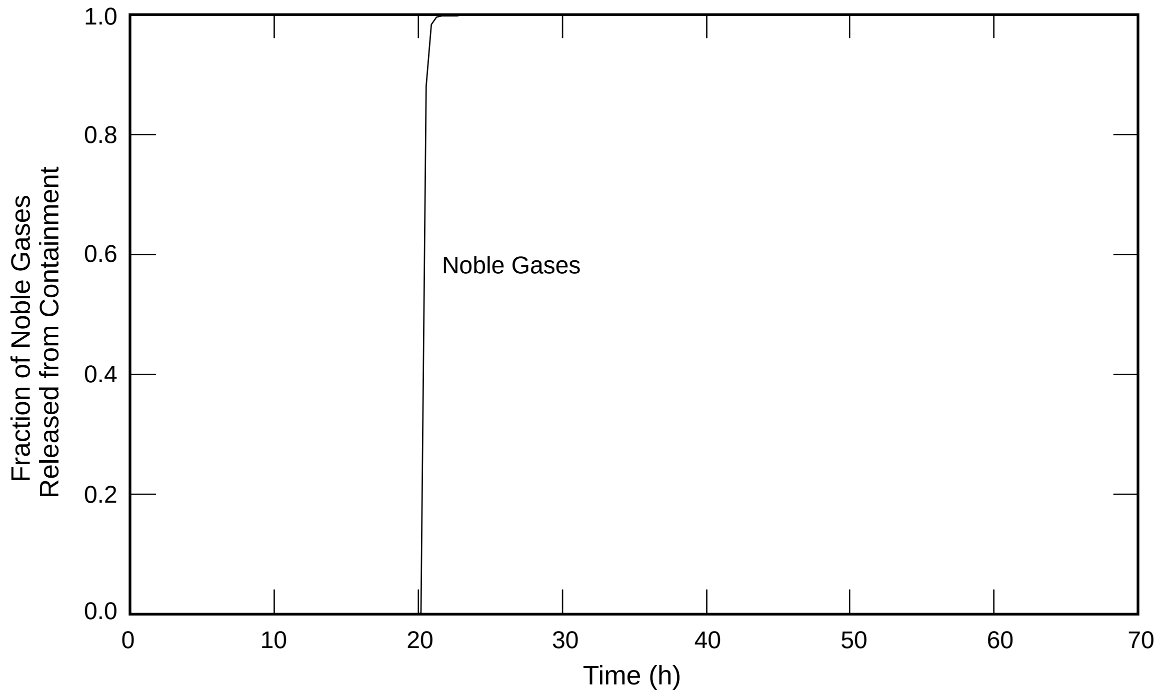


Figure 19E.2-2g LCLP-PF-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Noble Gases

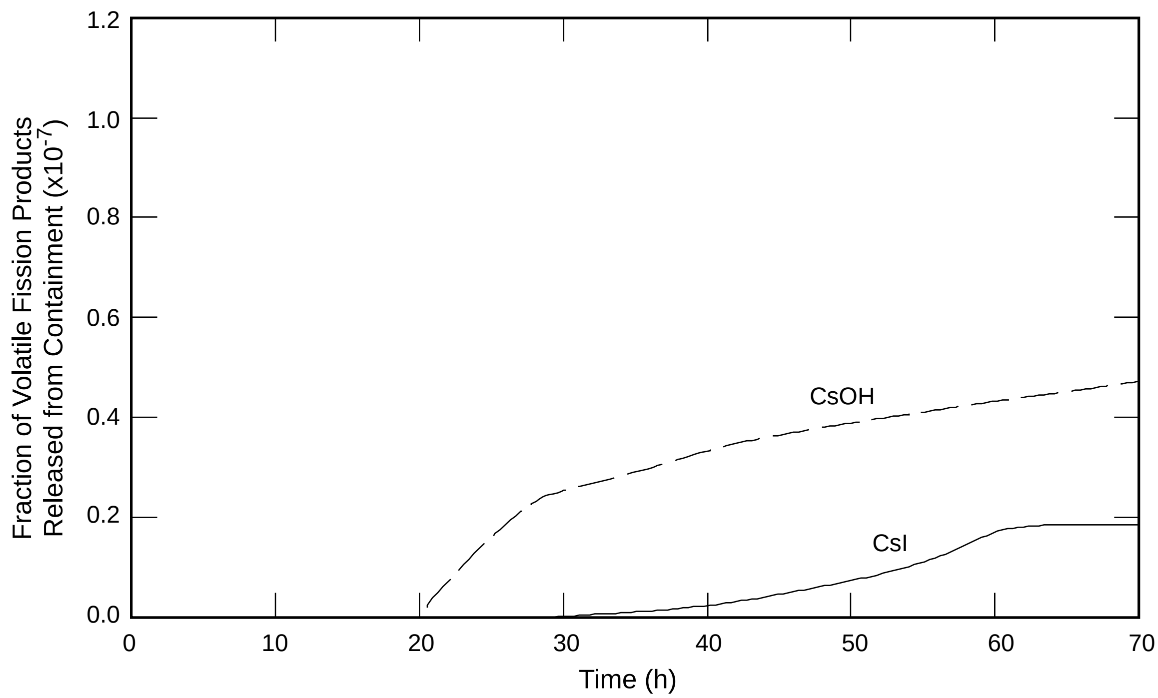


Figure 19E.2-2h LCLP-PF-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Volatile Fission Products

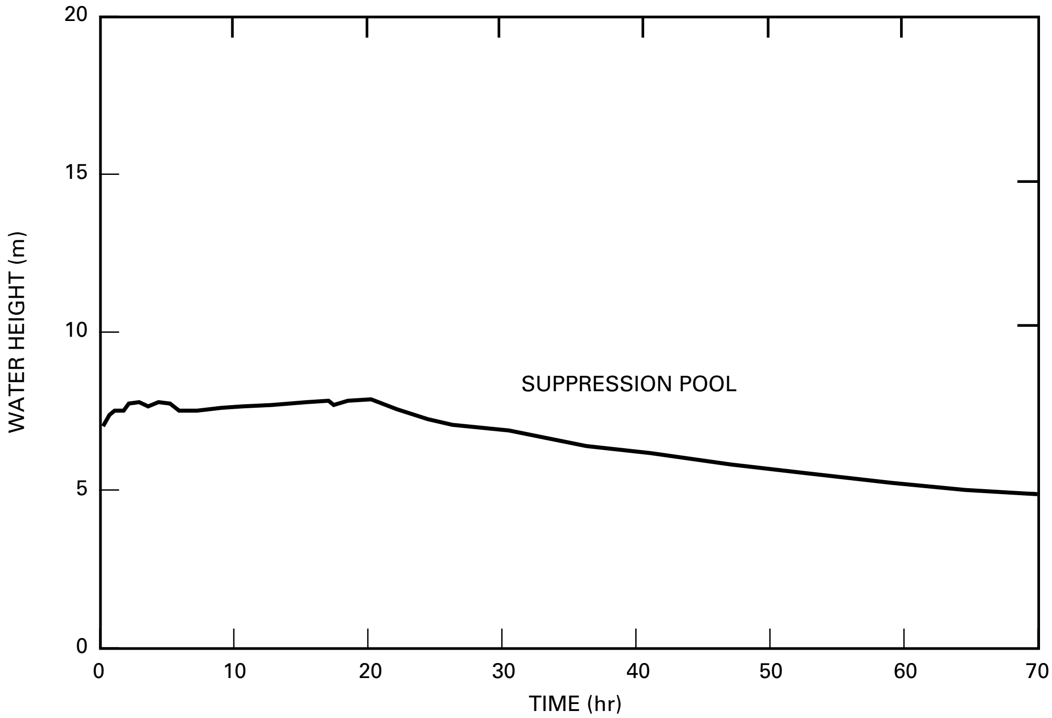


Figure 19E.2-2i LCLP-PF-R-N: Loss of All Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Water Height

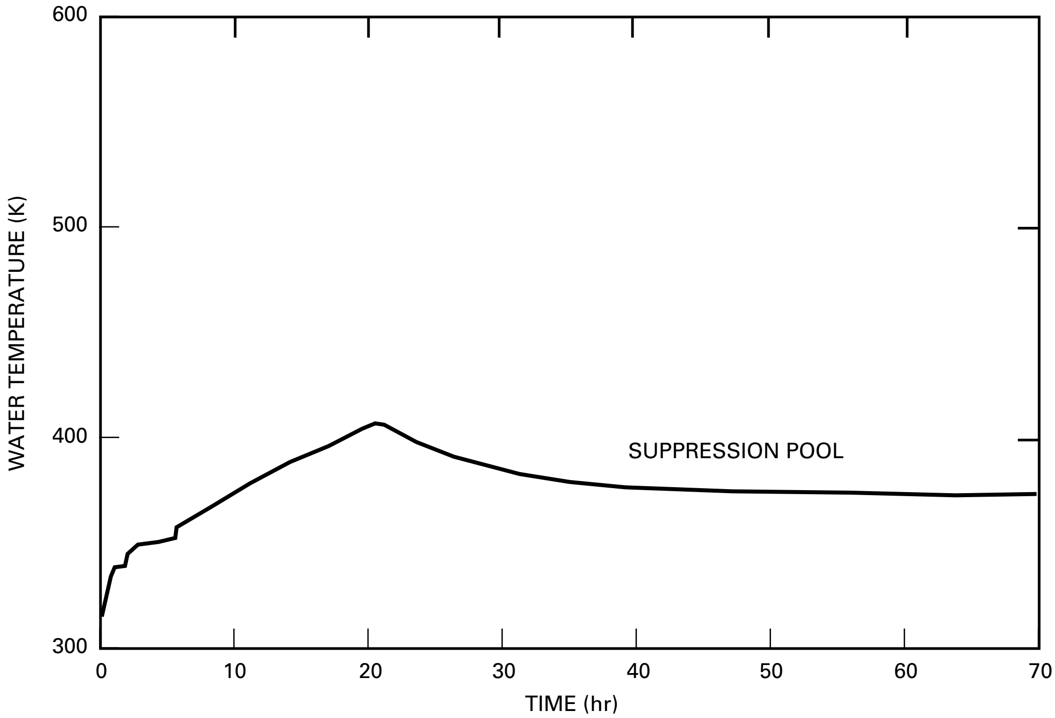


Figure 19E.2-2j LCLP-PF-R-N: Loss of All Core Cooling with Vessel Failure at Low Pressure, Passive Flooder Operates and Rupture Disk Opens: Water Temperature

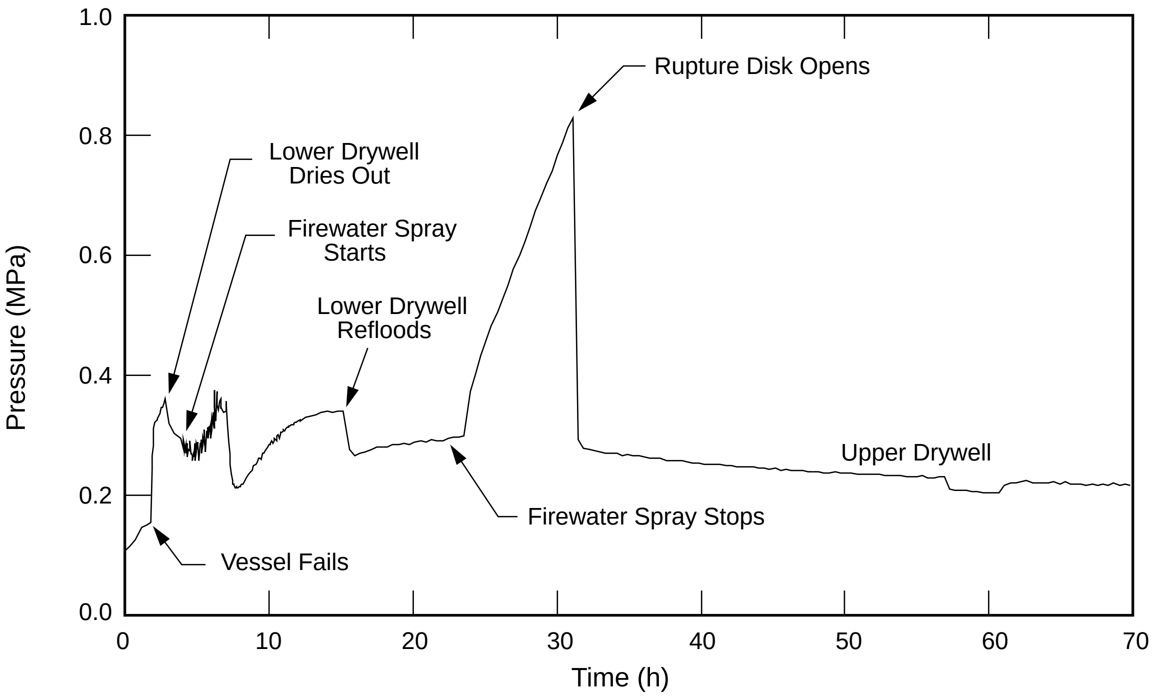


Figure 19E.2-3a LCLP-FS-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Firewater Spray Operates and Rupture Disk Opens: Drywell Pressure

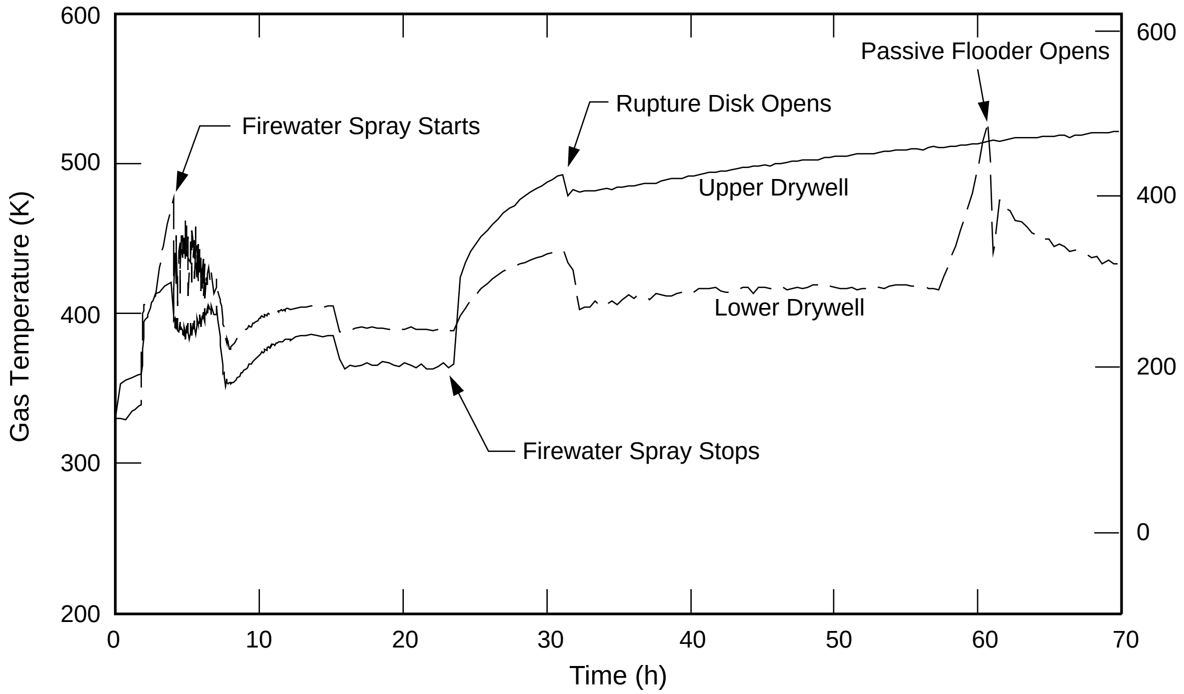


Figure 19E.2-3b LCLP-FS-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Firewater Spray Operates and Rupture Disk Opens: Gas Temperature

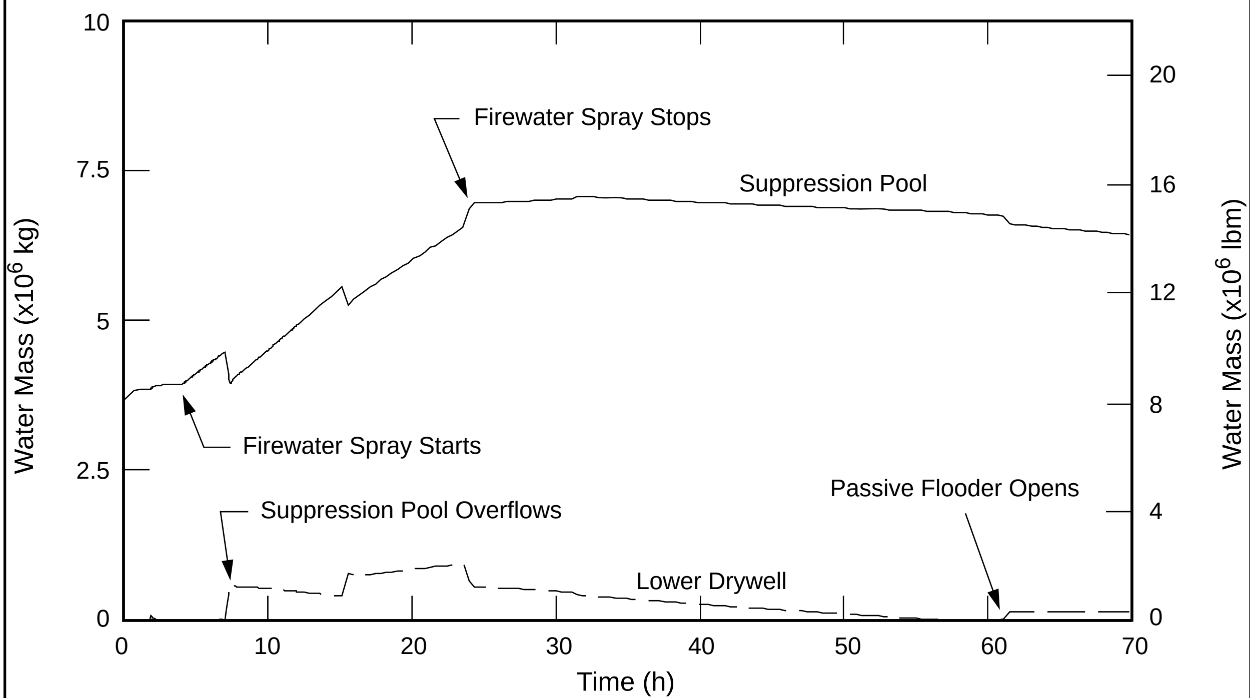


Figure 19E.2-3c LCLP-FS-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Firewater Spray Operates and Rupture Disk Opens: Water Mass

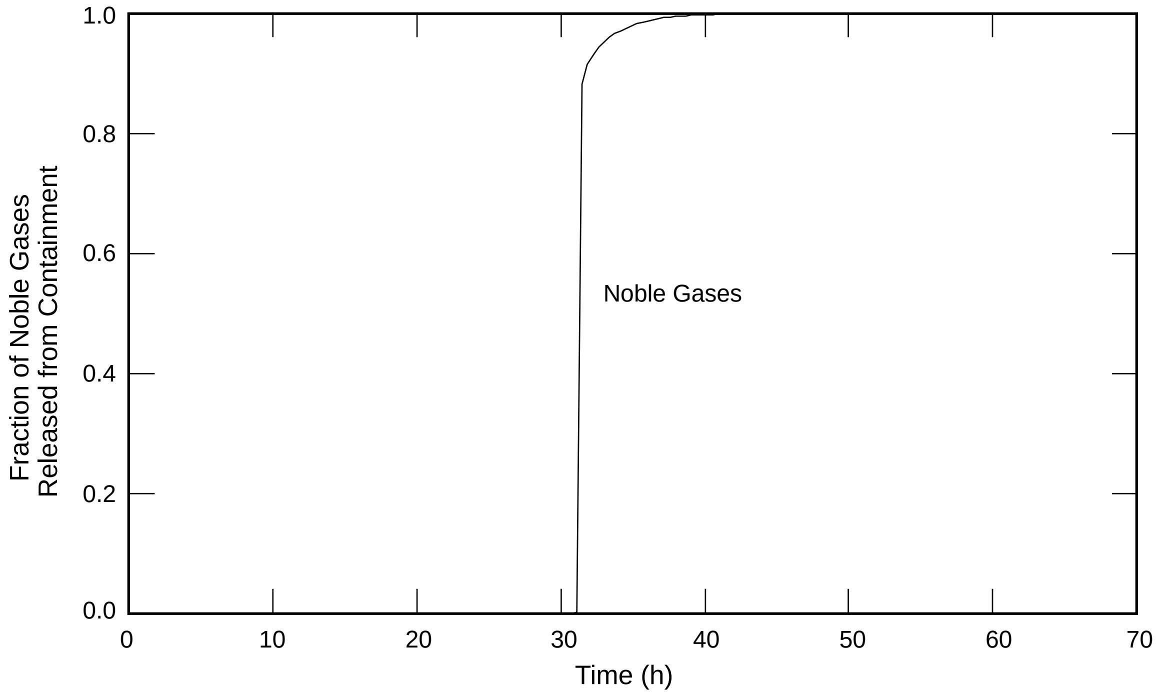


Figure 19E.2-3d LCLP-FS-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Firewater Spray Operates and Rupture Disk Opens: Noble Gas

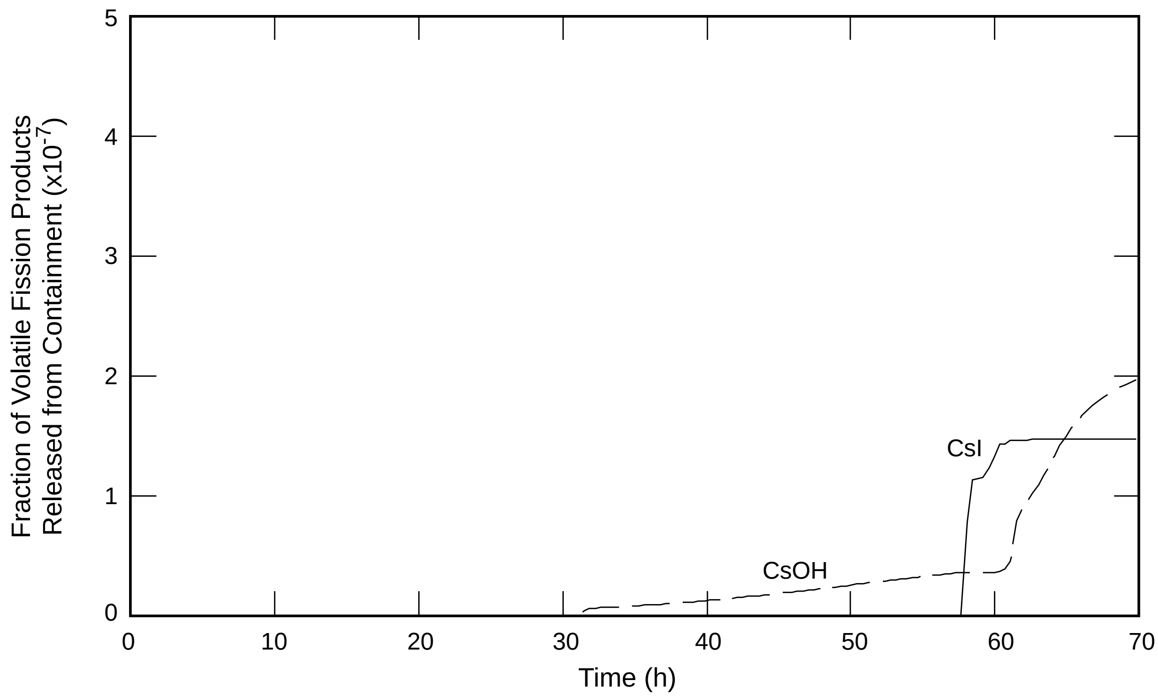


Figure 19E.2-3e LCLP-FS-R-N: Loss of all Core Cooling with Vessel Failure at Low Pressure, Firewater Spray Operates and Rupture Disk Opens: Volatile Fission Products

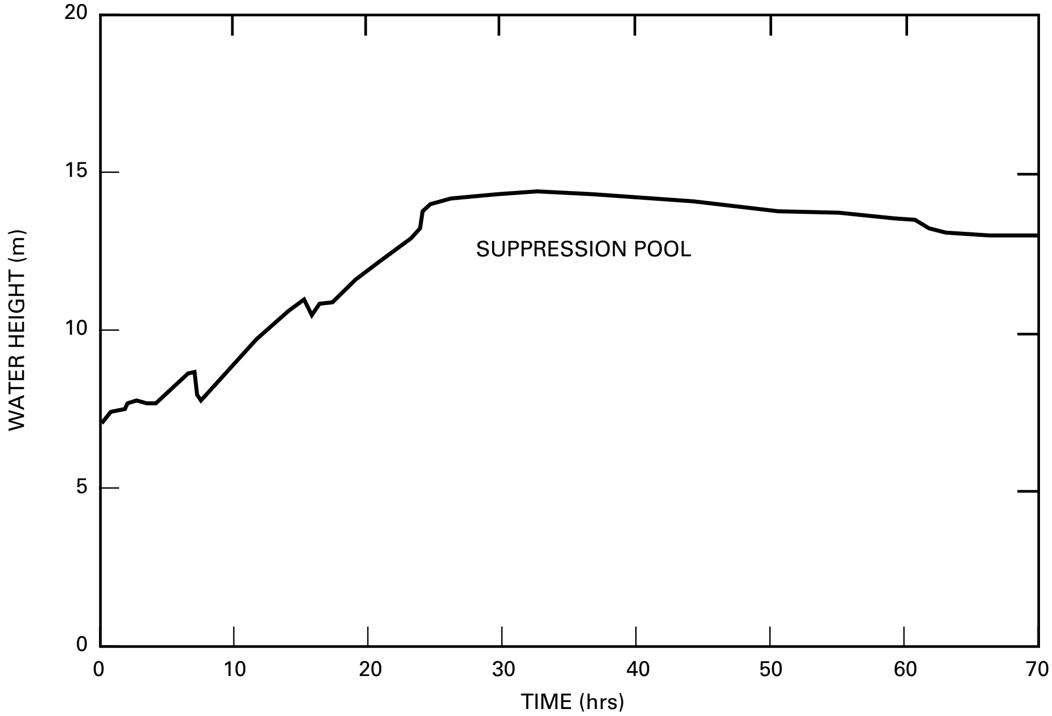


Figure 19E.2-3f LCLP-FS-R-N: Loss of All Core Cooling with Vessel Failure at Low Pressure, Firewater Spray Operates and Rupture Disk Opens: Water Height

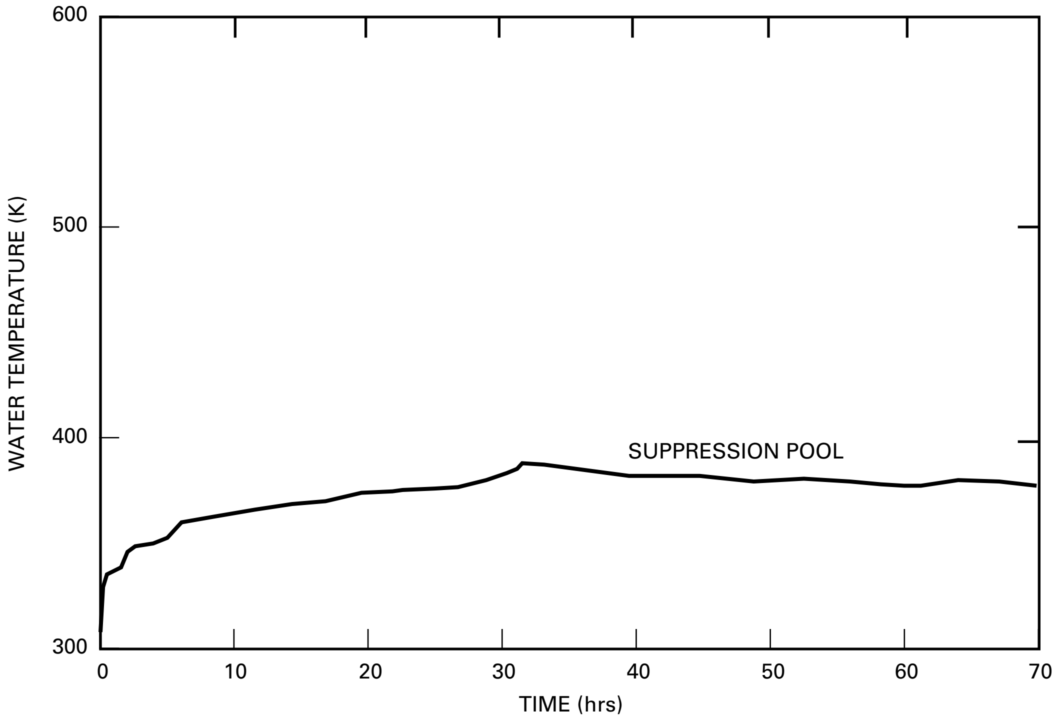


Figure 19E.2-3g LCLP-FS-R-N: Loss of All Core Cooling with Vessel Failure at Low Pressure, Firewater Spray Operates and Rupture Disk Opens: Water Temperature

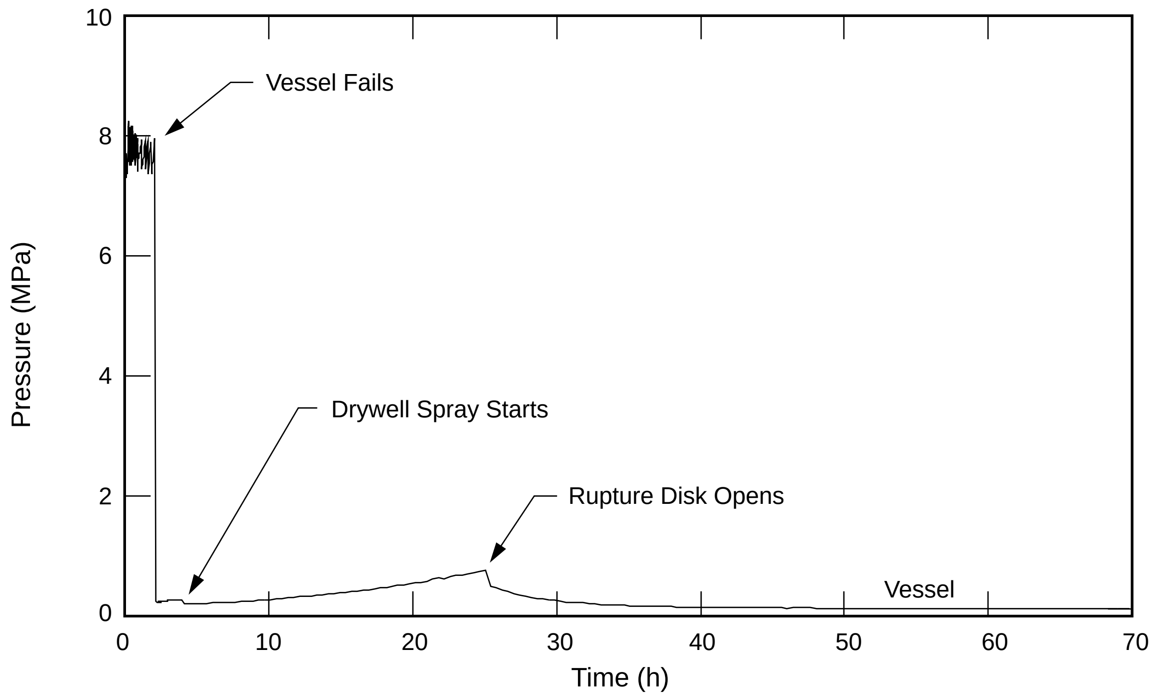


Figure 19E.2-4a LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: Vessel Pressure

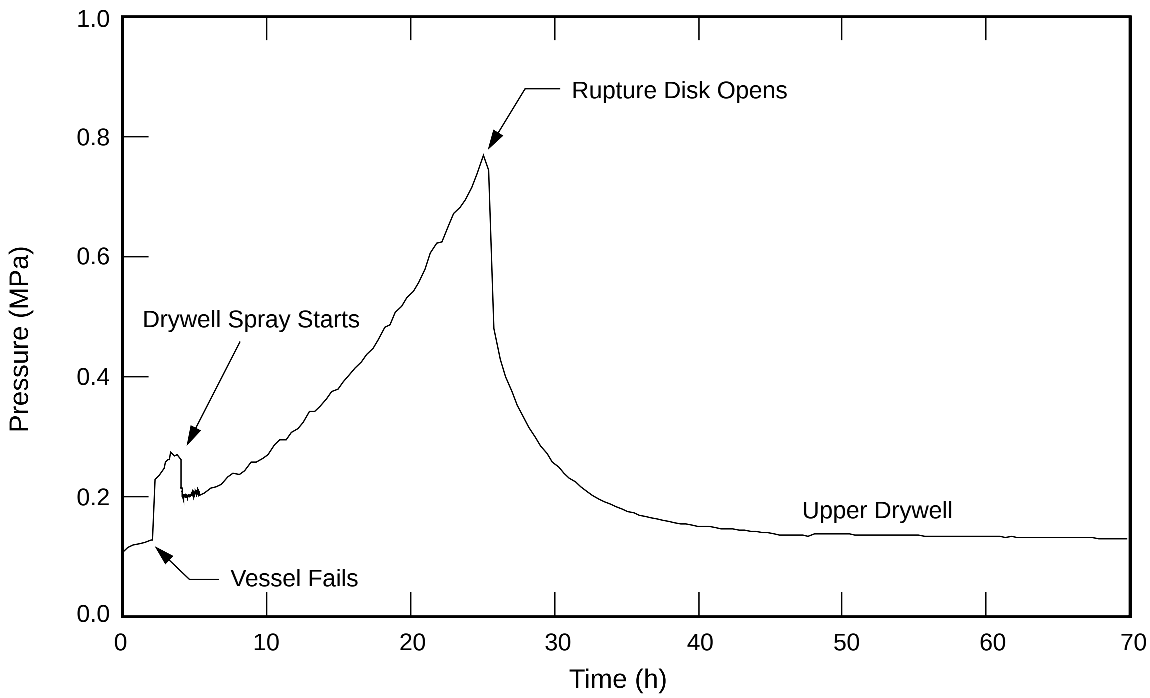


Figure 19E.2-4b LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: Drywell Pressure

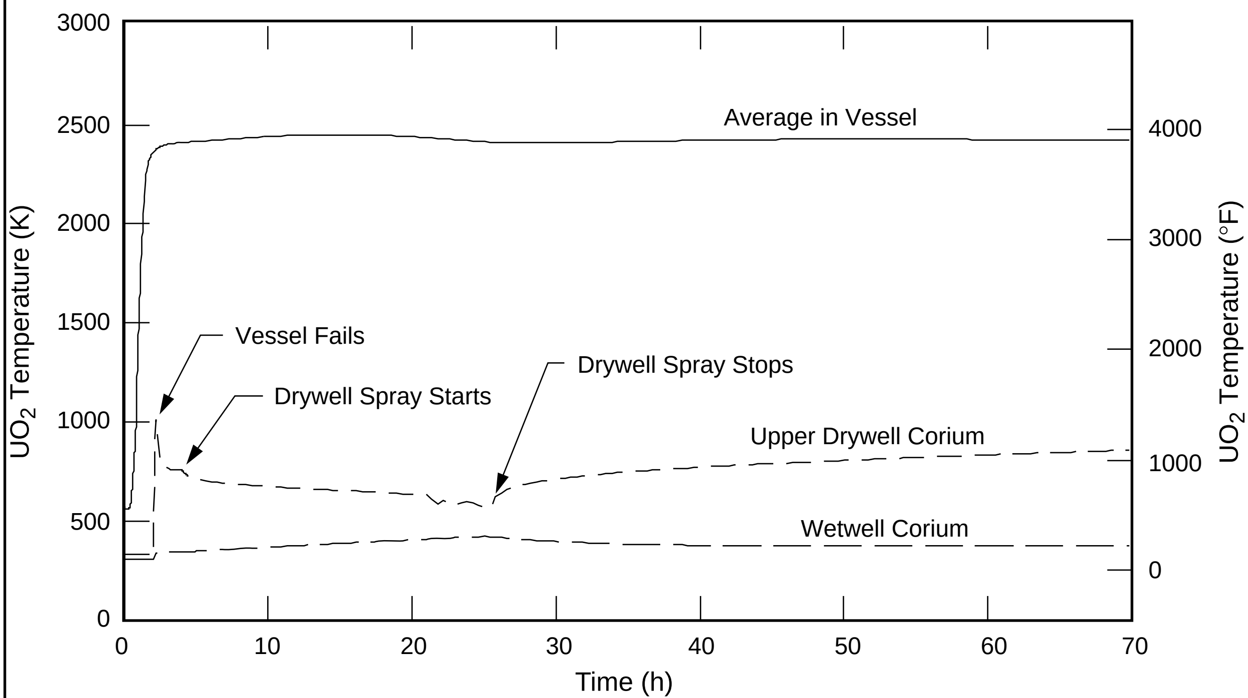


Figure 19E.2-4c LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: UO_2 Temperature

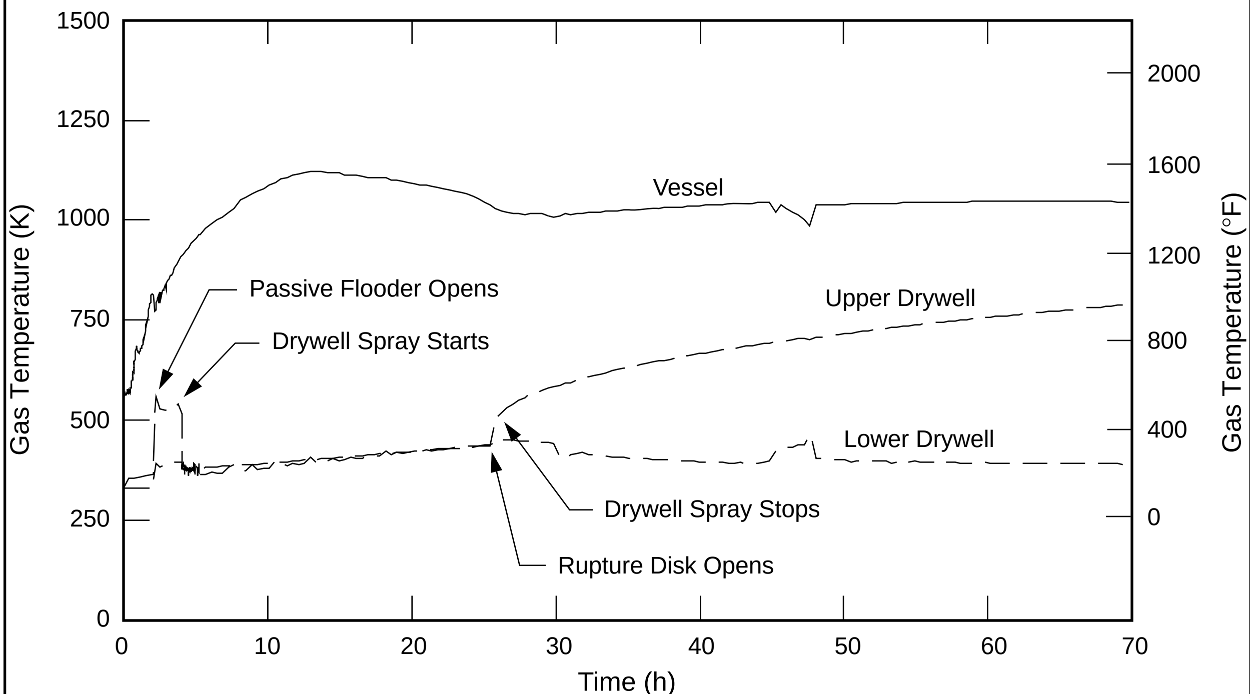


Figure 19E.2-4d LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: Gas Temperature

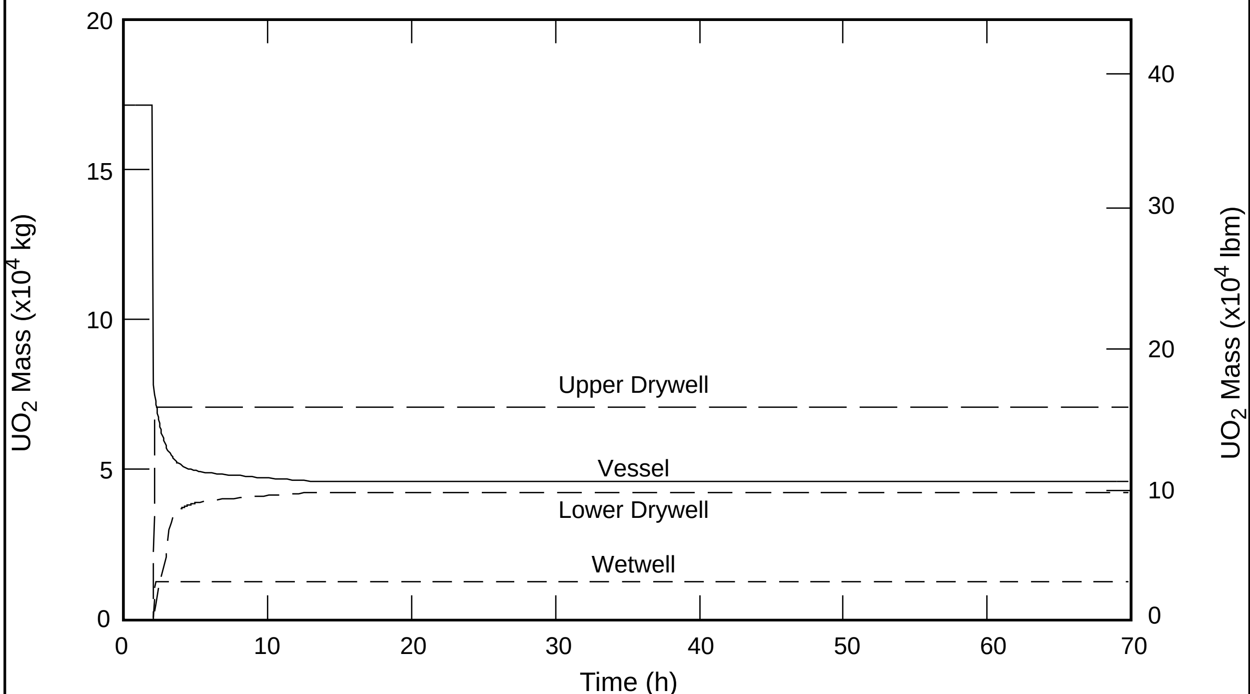


Figure 19E.2-4e LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: UO_2 Mass

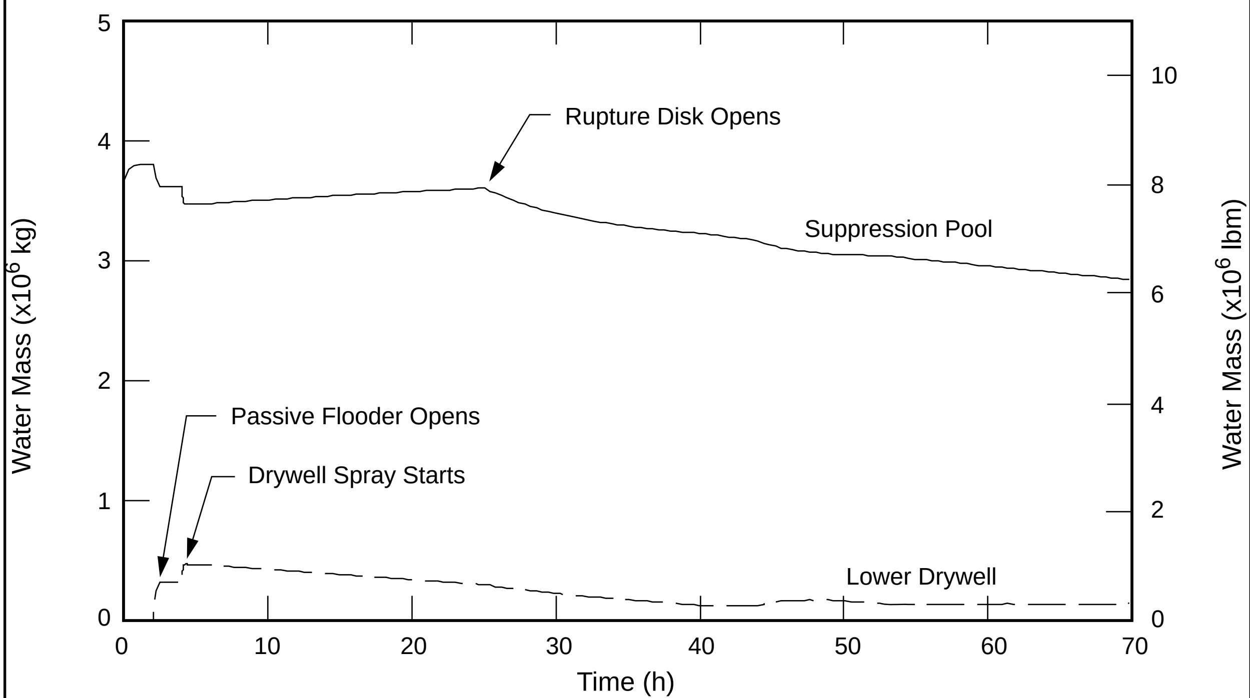


Figure 19E.2-4f LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: Water Mass

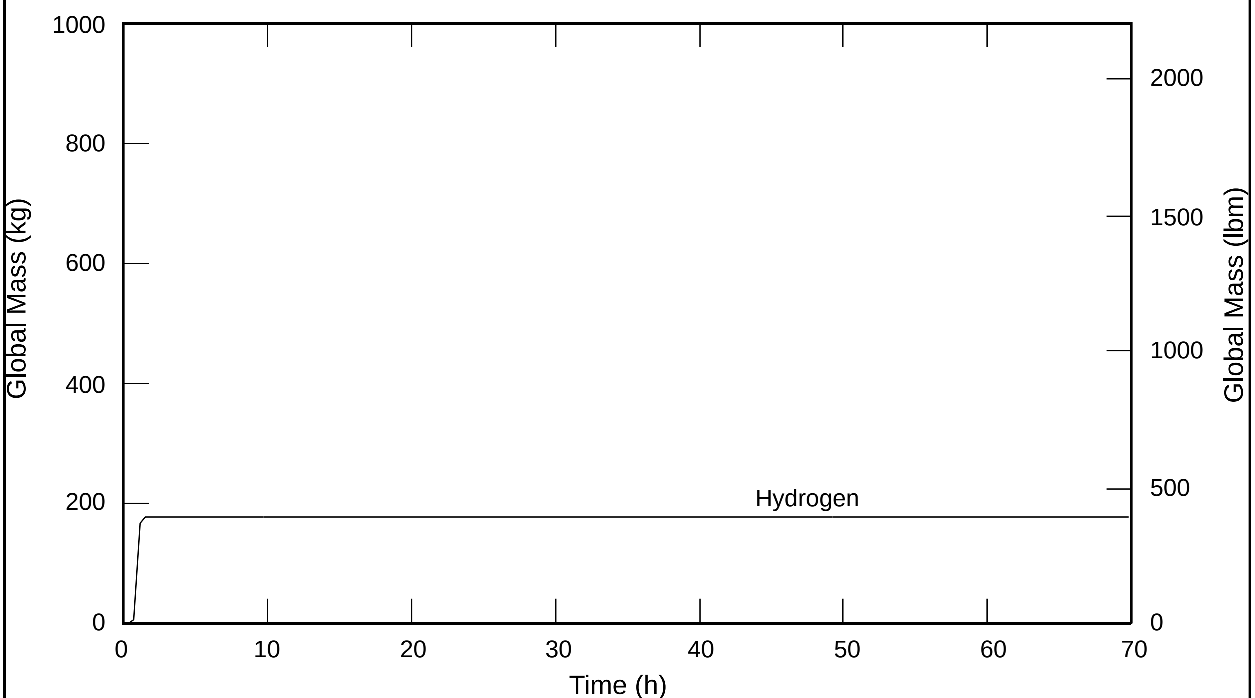


Figure 19E.2-4g LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: Global Mass

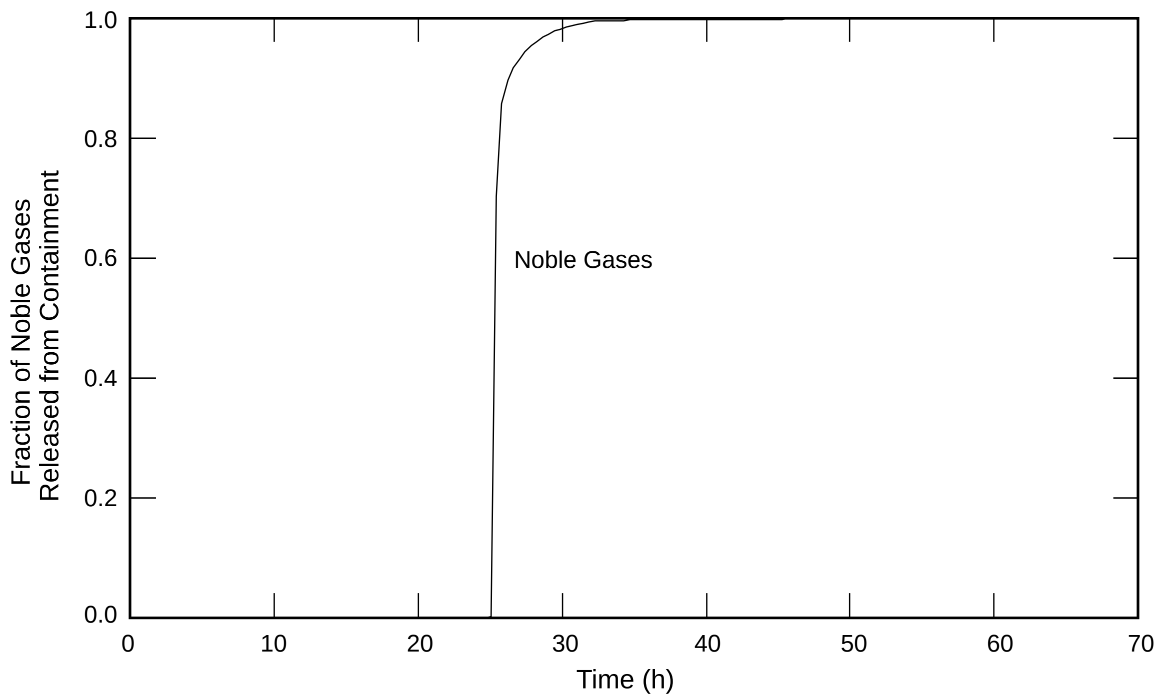


Figure 19E.2-4h LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: Noble Gases

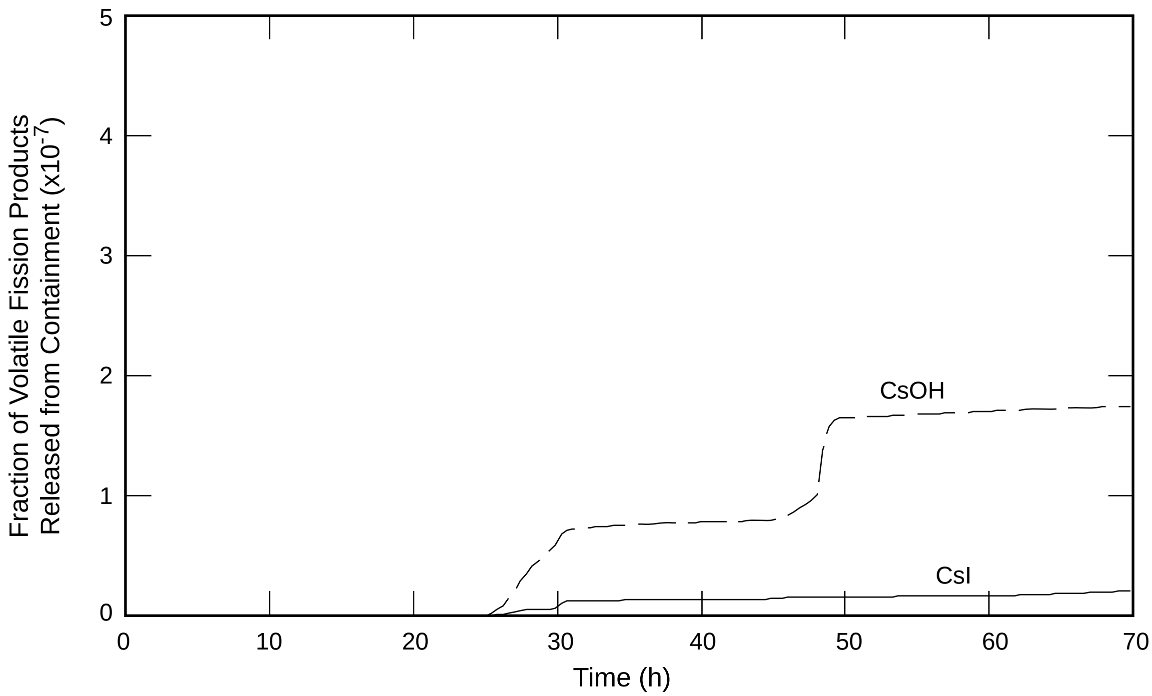


Figure 19E.2-4i LCHP-PS-R-N: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder and Drywell Sprays Operate, Rupture Disk Opens: Volatiles

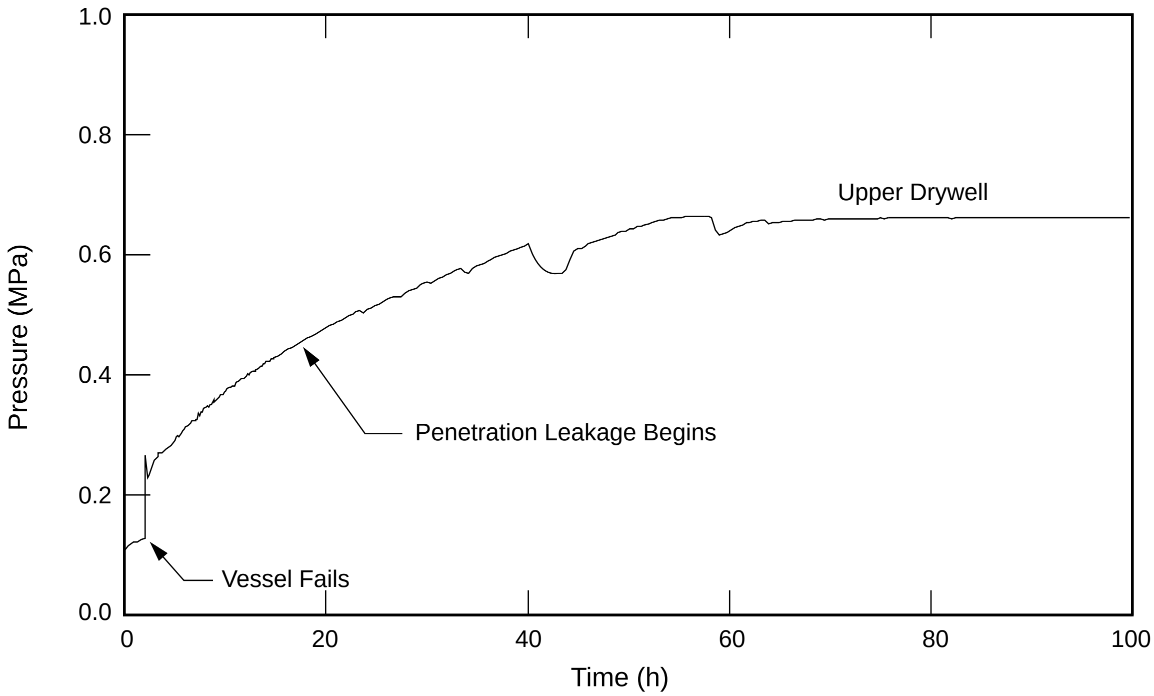


Figure 19E.2-5a LCHP-PF-P-M: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder Operates, Penetration Leakage: Drywell Pressure

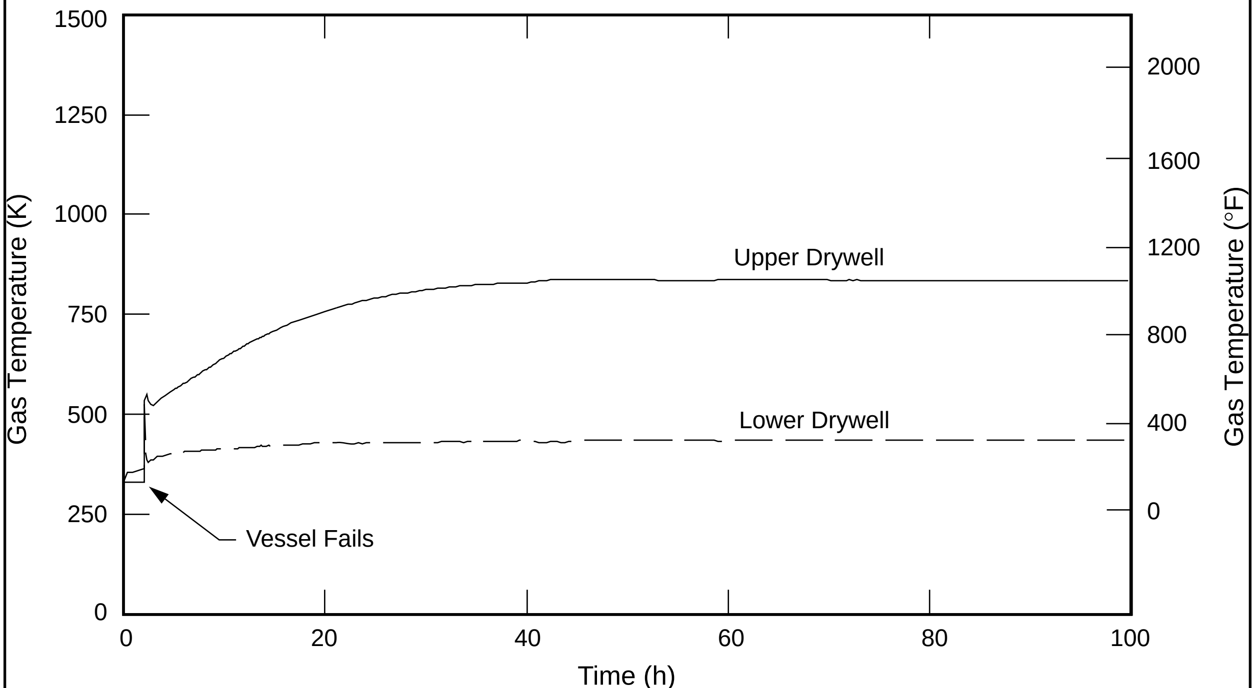


Figure 19E.2-5b LCHP-PF-P-M: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder Operates, Penetration Leakage: Gas Temperature

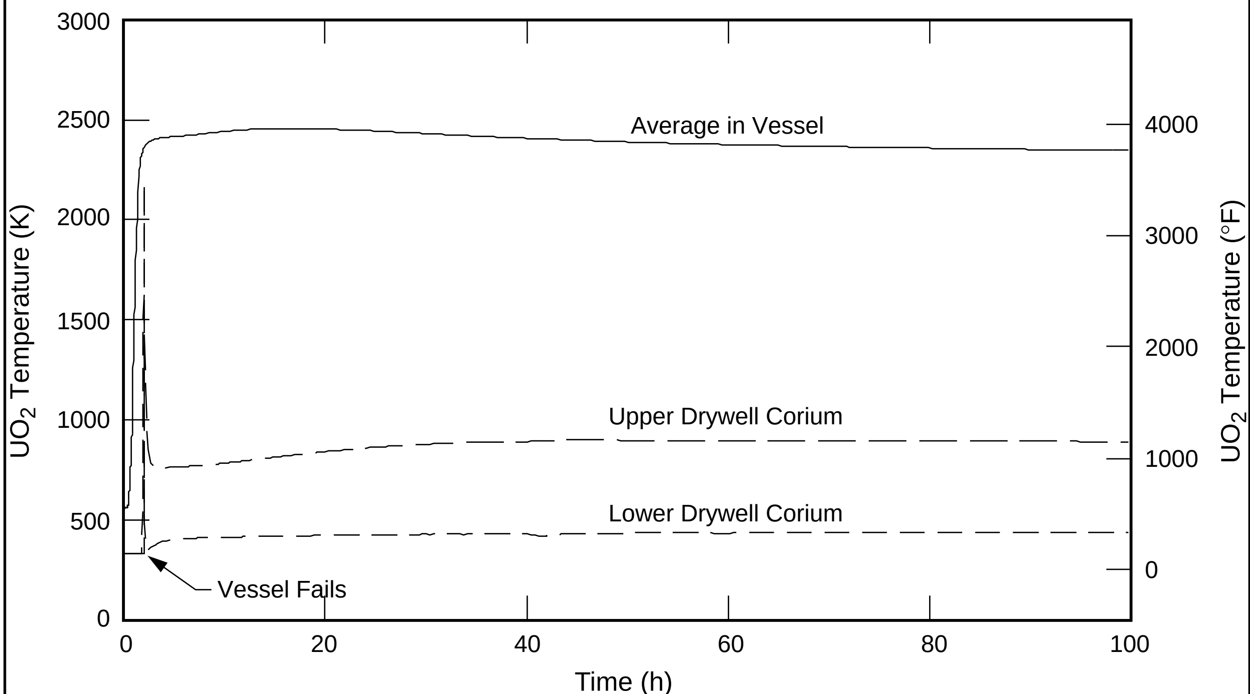


Figure 19E.2-5c LCHP-PF-P-M: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder Operates, Penetration Leakage: UO_2 Temperature

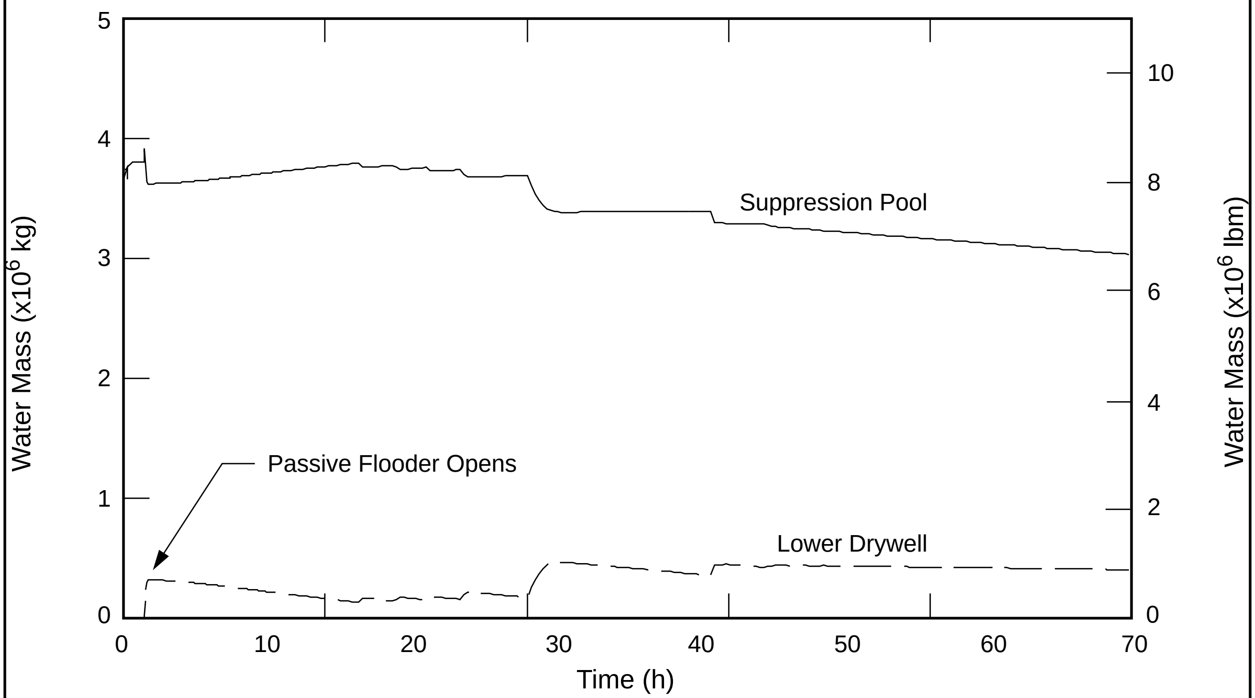


Figure 19E.2-5d LCHP-PF-P-M: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder Operates, Penetration Leakage: Water Mass

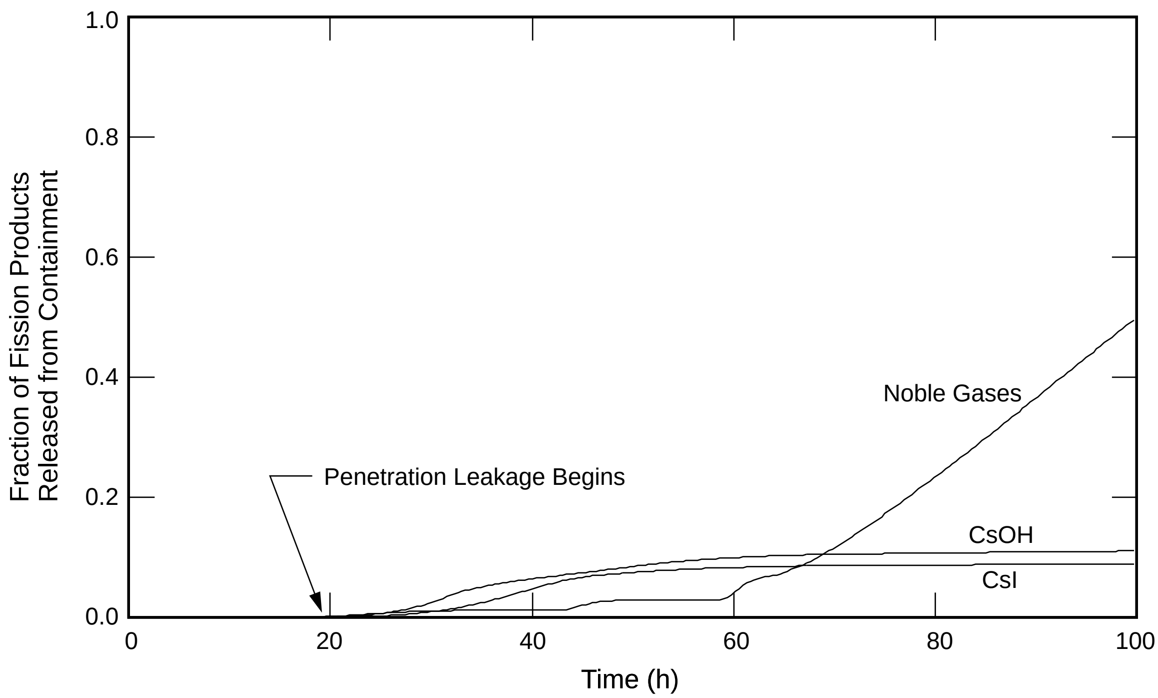


Figure 19E.2-5e LCHP-PF-P-M: Loss of all Core Cooling with Vessel Failure at High Pressure, Passive Flooder Operates, Penetration Leakage: Fission Product Release

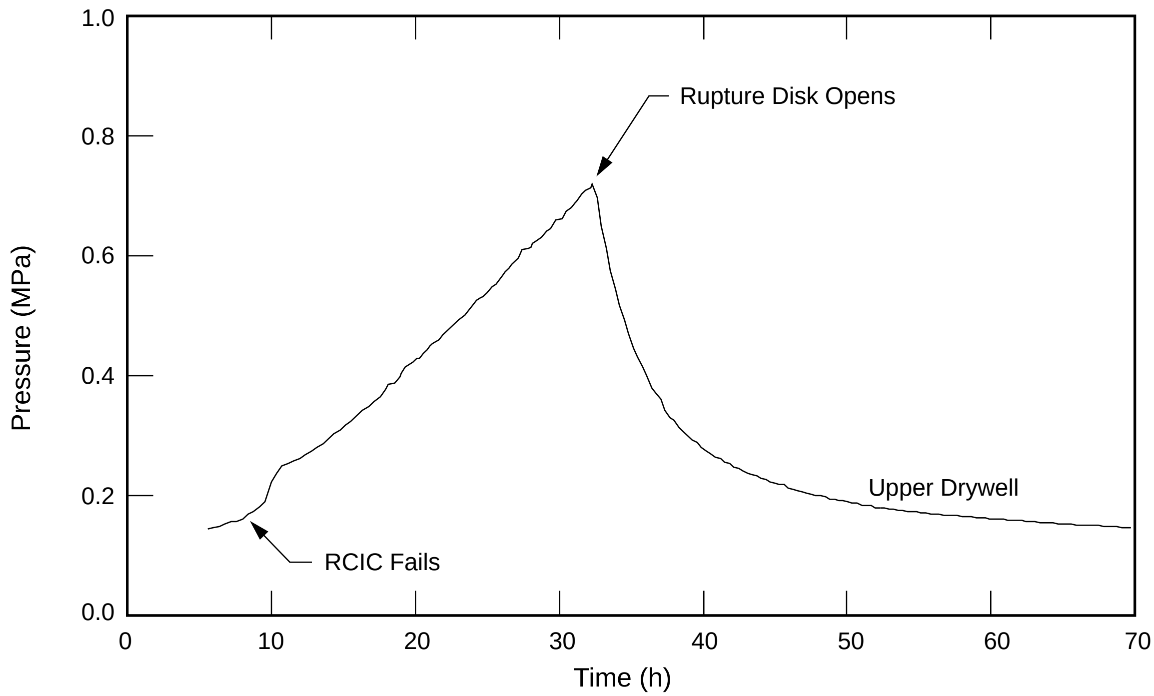


Figure 19E.2-6a SBRC-FA-R-0: Station Blackout, RCIC Runs Eight Hours, Firewater Addition Prevents Core Damage, Rupture Disk Opens: Drywell Pressure

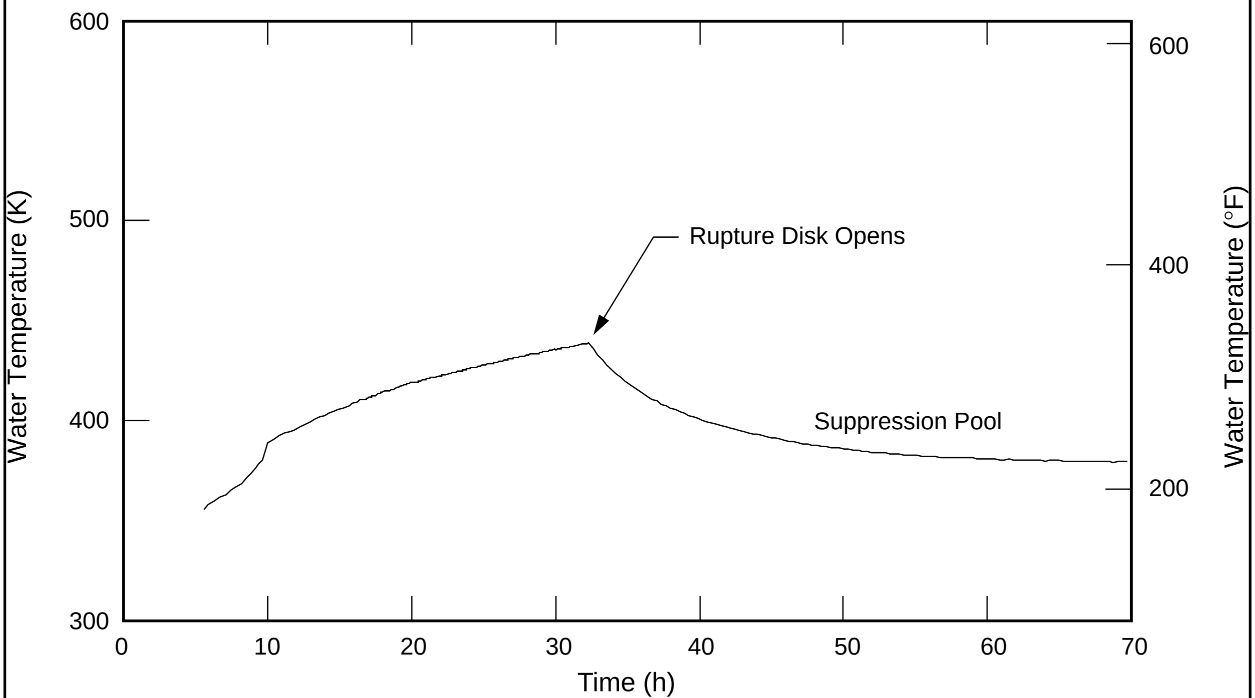


Figure 19E.2-6b SBRC-FA-R-0: Station Blackout, RCIC Runs Eight Hours, Firewater Addition Prevents Core Damage, Rupture Disk Opens: Water Temperature

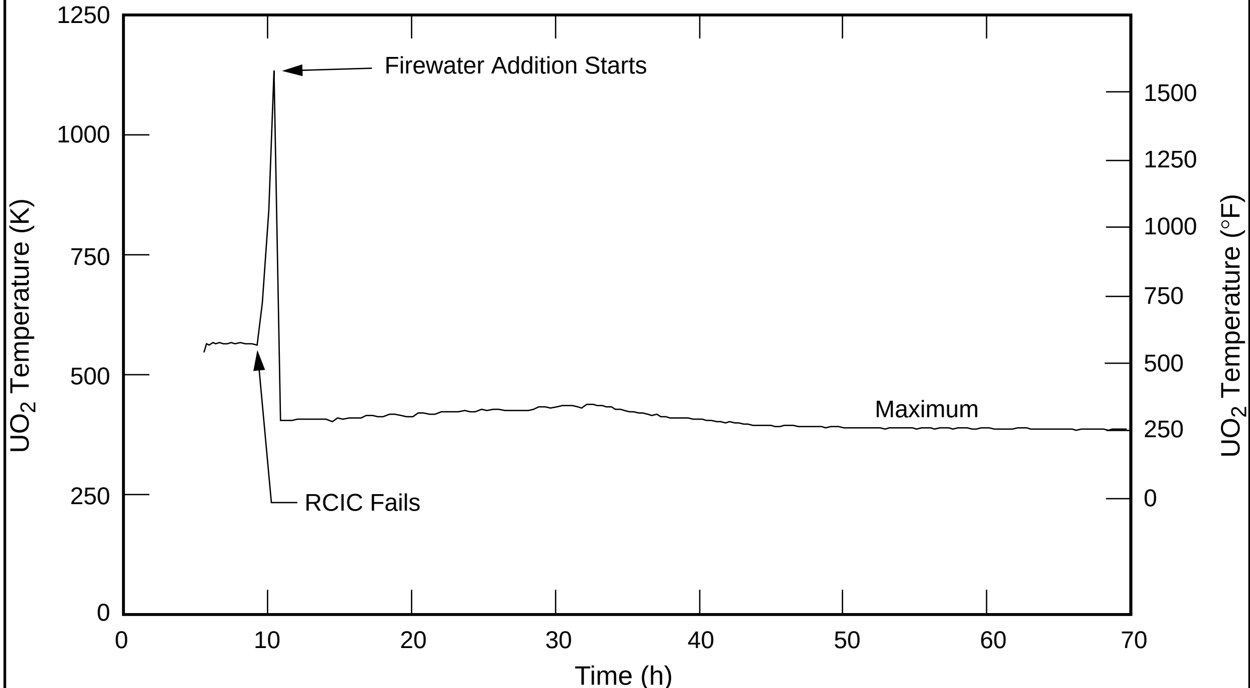


Figure 19E.2-6c SBRC-FA-R-0: Station Blackout, RCIC Runs Eight Hours, Firewater Addition Prevents Core Damage, Rupture Disk Opens: UO_2 Temperature

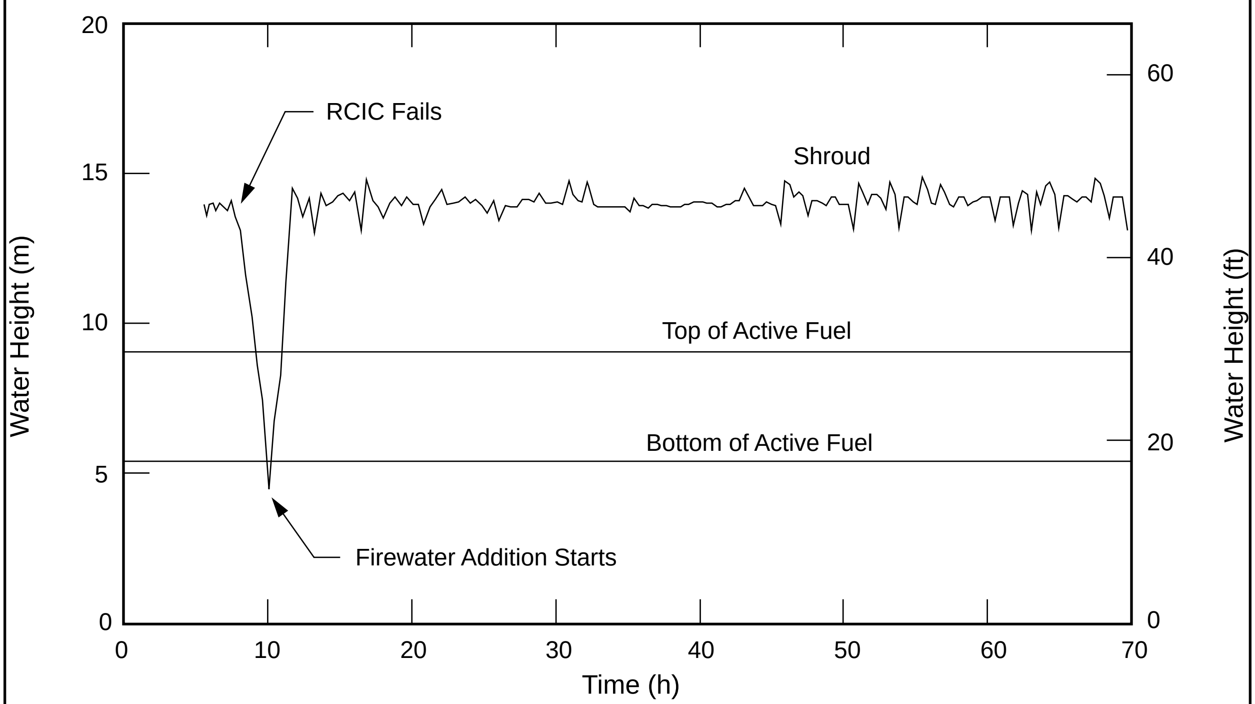


Figure 19E.2-6d SBRC-FA-R-0: Station Blackout, RCIC Runs Eight Hours, Firewater Addition Prevents Core Damage, Rupture Disk Opens: Vessel Water Height

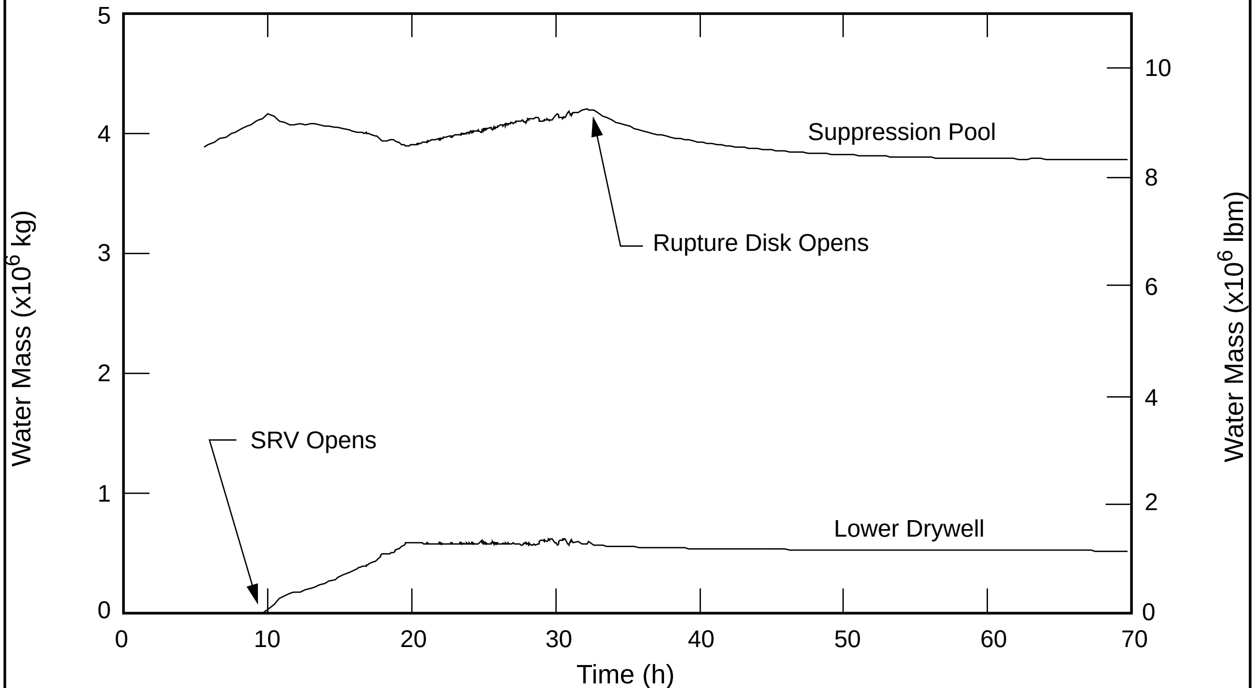


Figure 19E.2-6e SBRC-FA-R-0: Station Blackout, RCIC Runs Eight Hours, Firewater Addition Prevents Core Damage, Rupture Disk Opens: Water Mass

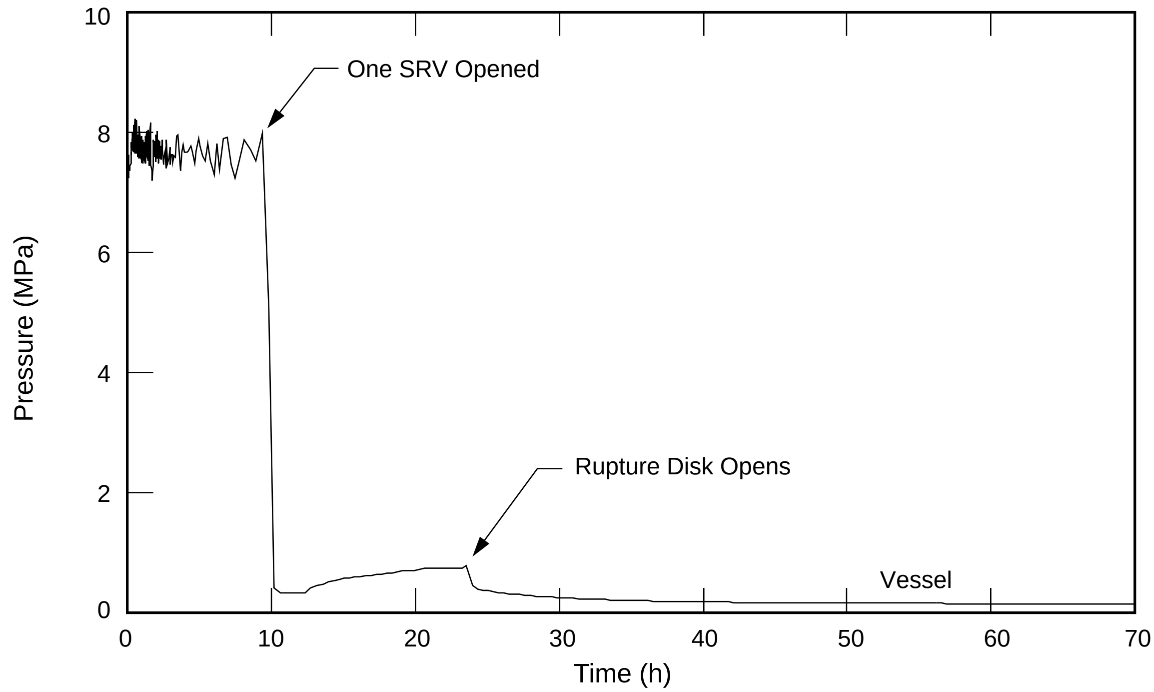


Figure 19E.2-7a SBRC-PF-R-N: Station Blackout with RCIC Operating, Passive Flooder Operates and Rupture Disk Opens: Vessel Pressure

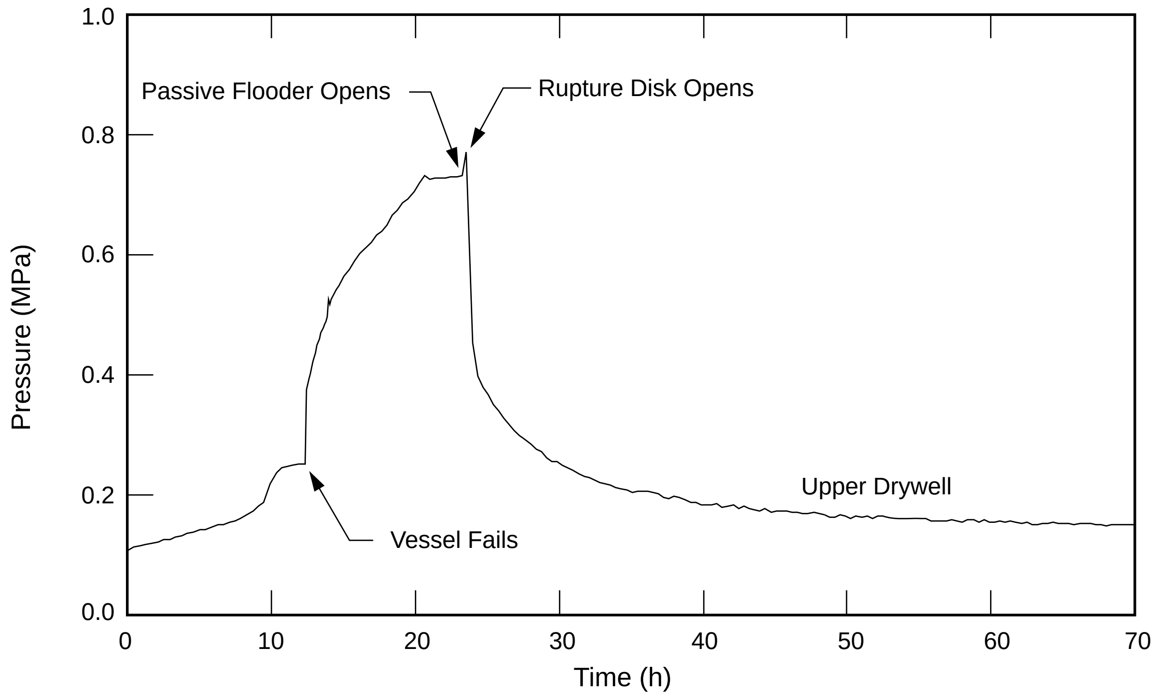


Figure 19E.2-7b SBRC-PF-R-N: Station Blackout with RCIC Operating, Passive Flooder Operates and Rupture Disk Opens: Drywell Pressure

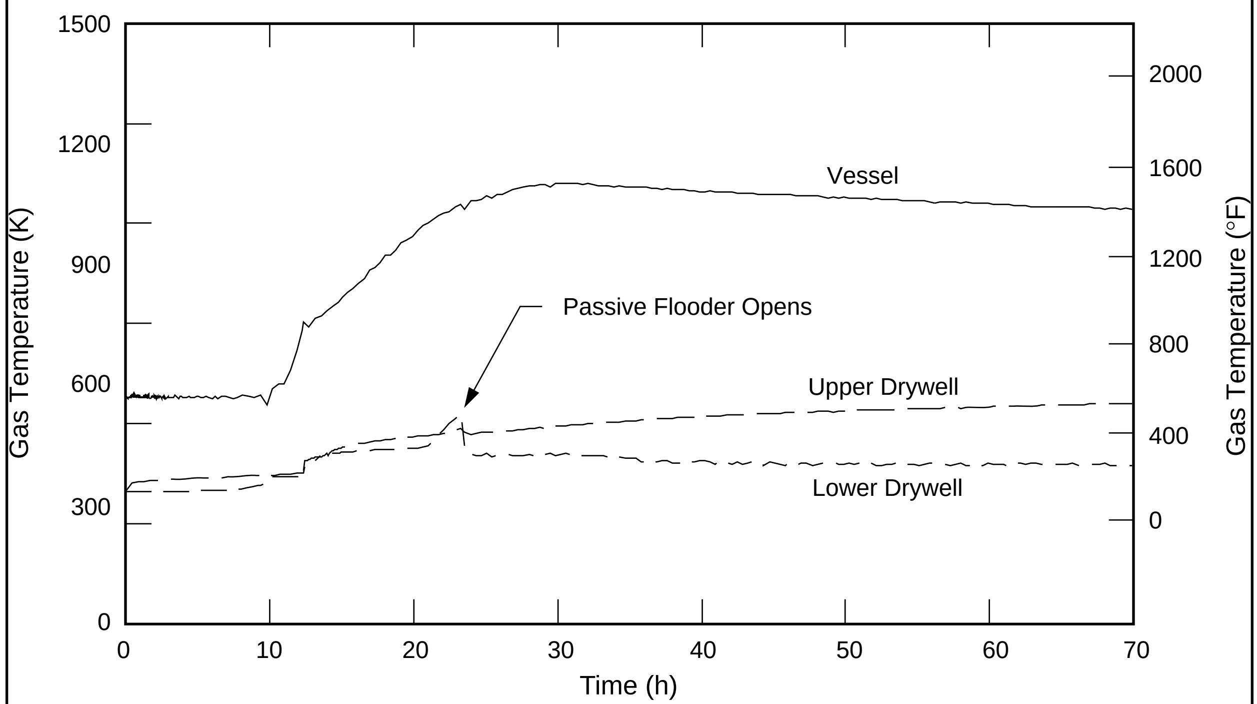


Figure 19E.2-7c SBRC-PF-R-N: Station Blackout with RCIC Operating, Passive Flooder Operates and Rupture Disk Opens: Gas Temperature

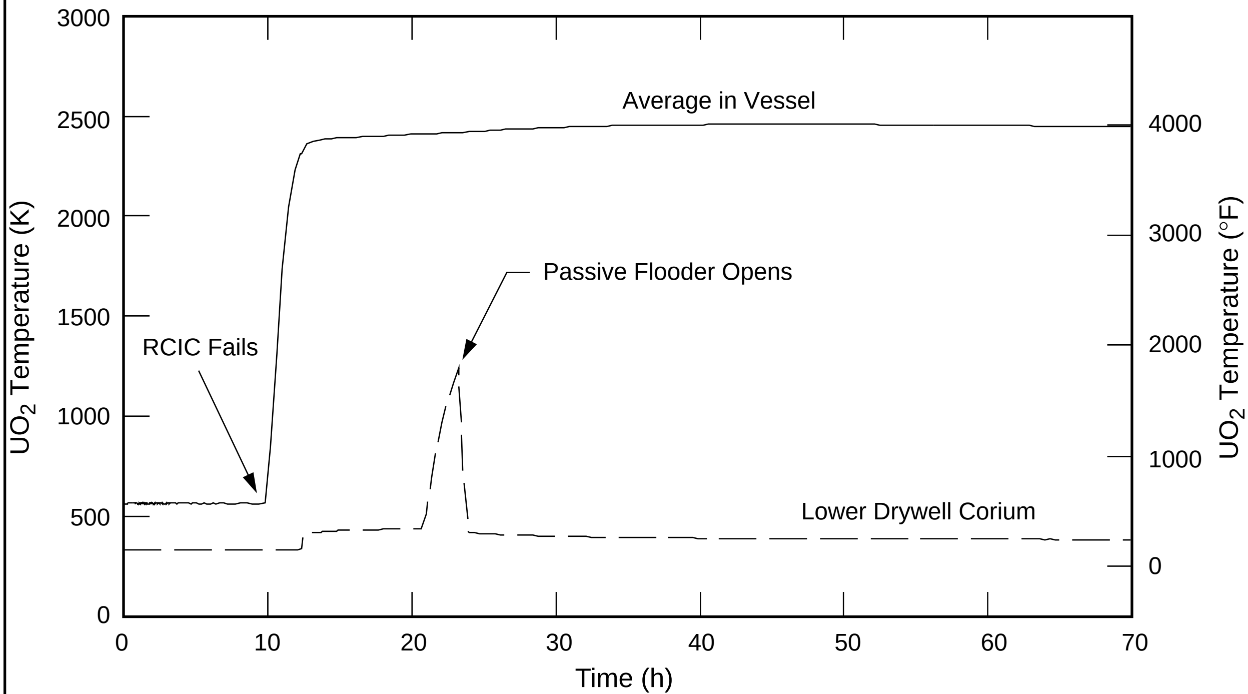


Figure 19E.2-7d SBRC-PF-R-N: Station Blackout with RCIC Operating, Passive Flooder Operates and Rupture Disk Opens: UO_2 Temperature

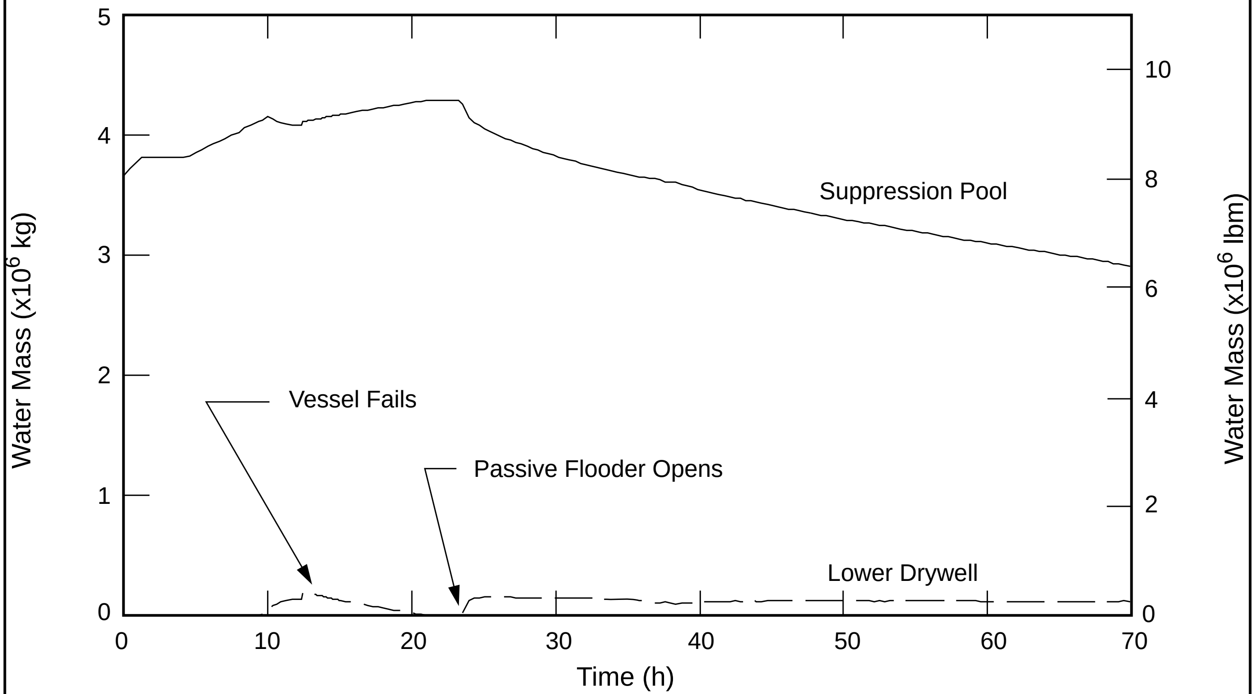


Figure 19E.2-7e SBRC-PF-R-N: Station Blackout with RCIC Operating, Passive Flooder Operates and Rupture Disk Opens: Water Mass

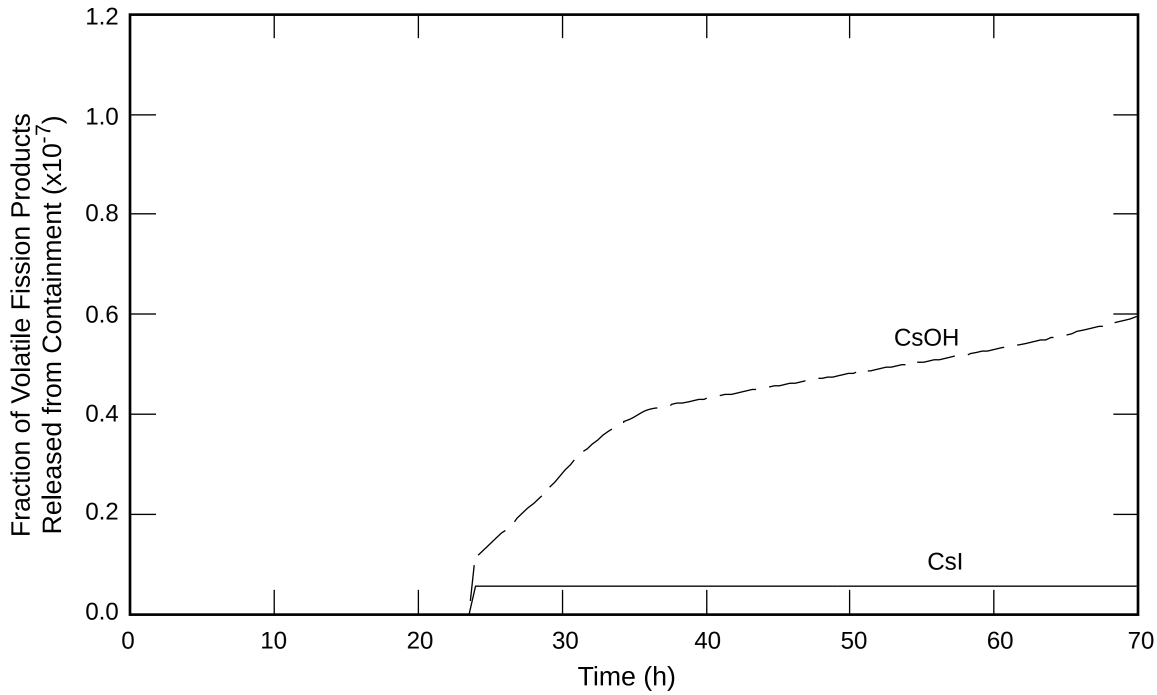


Figure 19E.2-7f SBRC-PF-R-N: Station Blackout with RCIC Operating, Passive Flooder Operates and Rupture Disk Opens: Volatile Fission Product Release

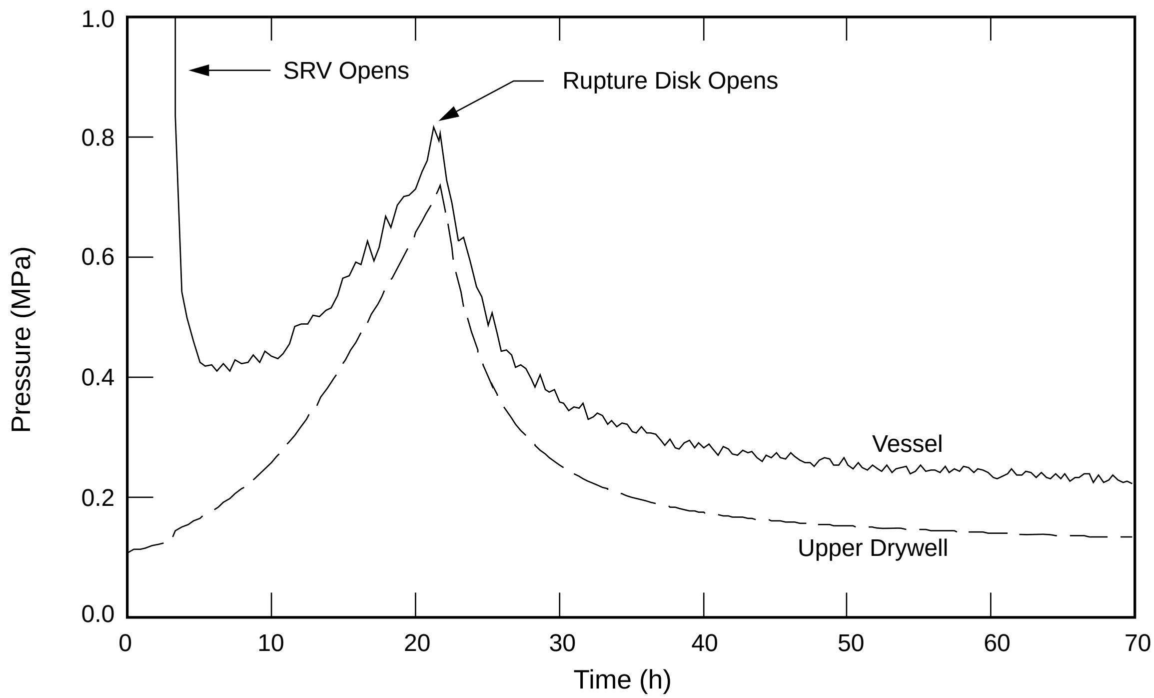


Figure 19E.2-8a LHRC-00-R-0: Isolation with Loss of Containment Heat Removal and Rupture Disk Opens: Pressure

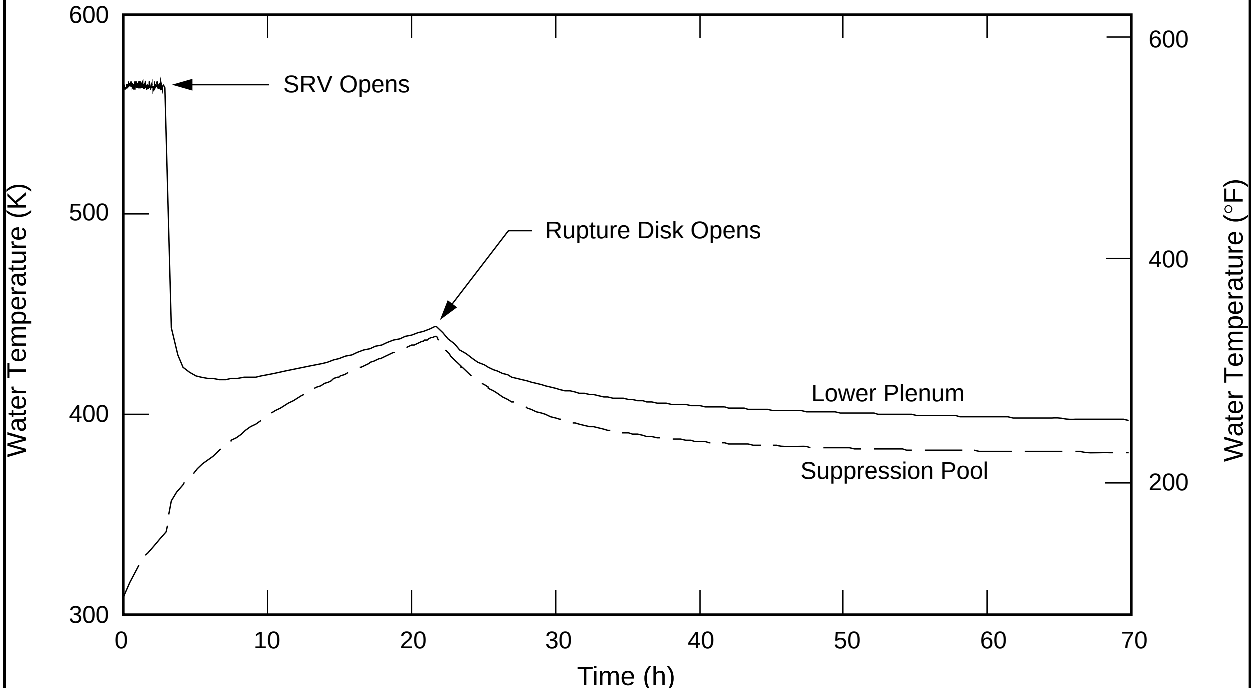


Figure 19E.2-8b LHRC-00-R-0: Isolation with Loss of Containment Heat Removal and Rupture Disk Opens: Water Temperature

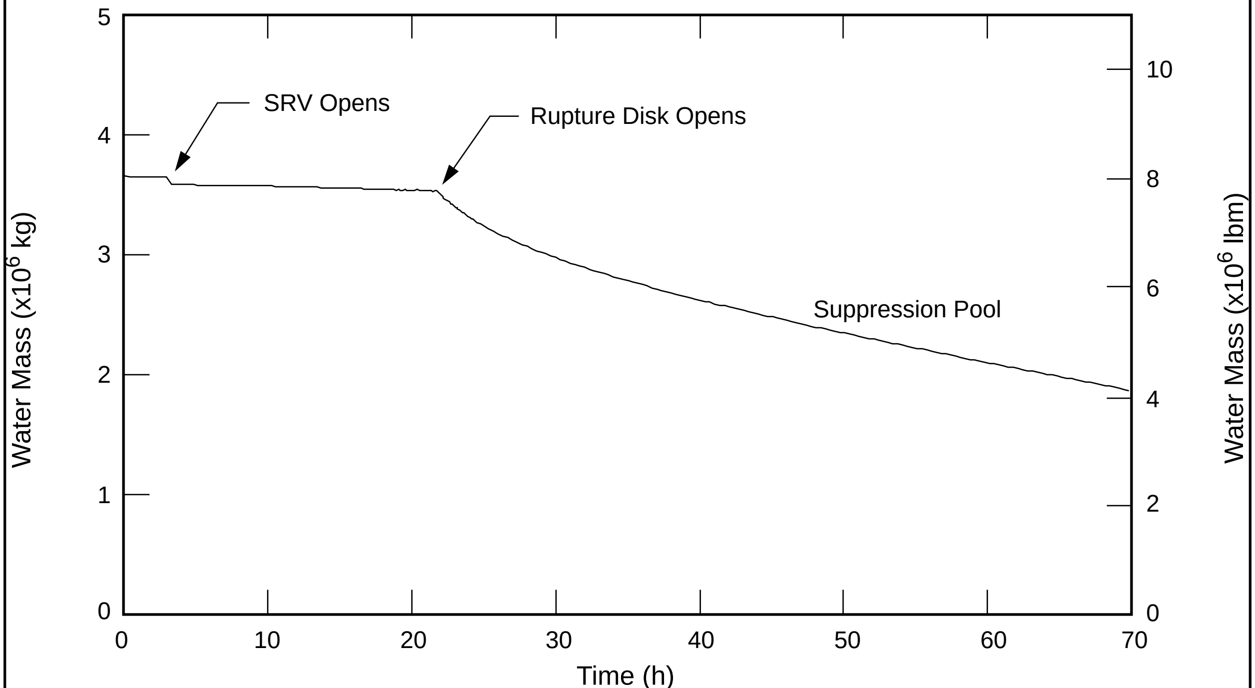


Figure 19E.2-8c LHRC-00-R-0: Isolation with Loss of Containment Heat Removal and Rupture Disk Opens: Water Mass

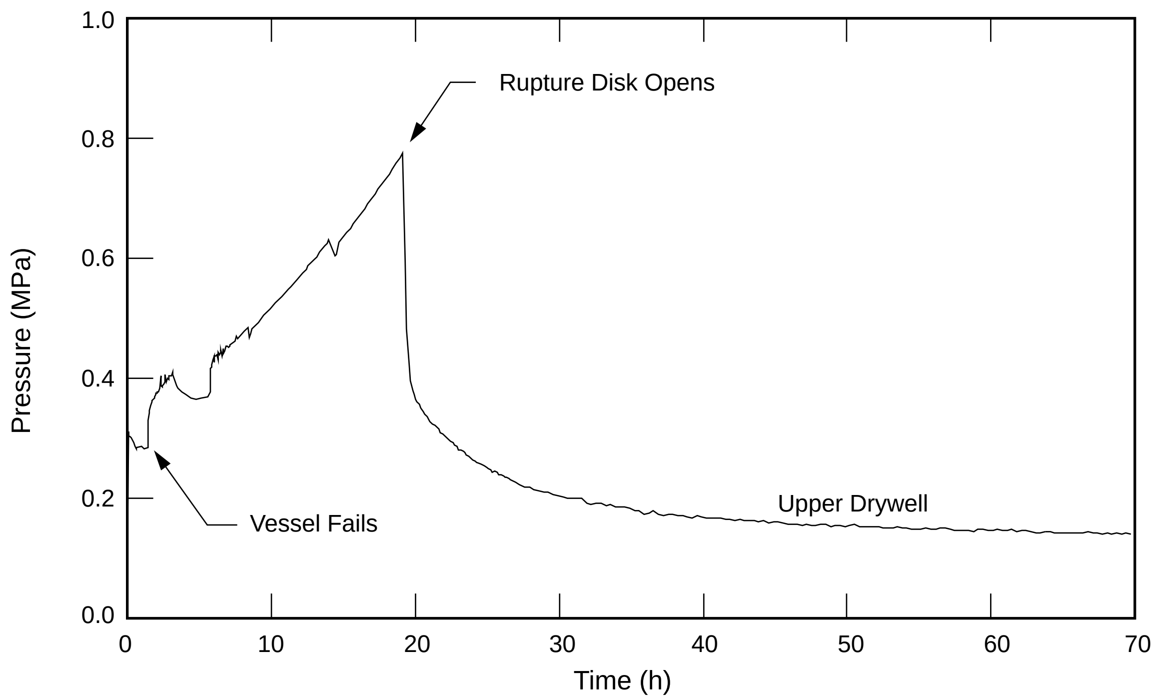


Figure 19E.2-9a LBLC-PF-R-N: Large Break LOCA with Loss of all Core Cooling, Passive Flooder Operates and Rupture Disk Opens: Drywell Pressure

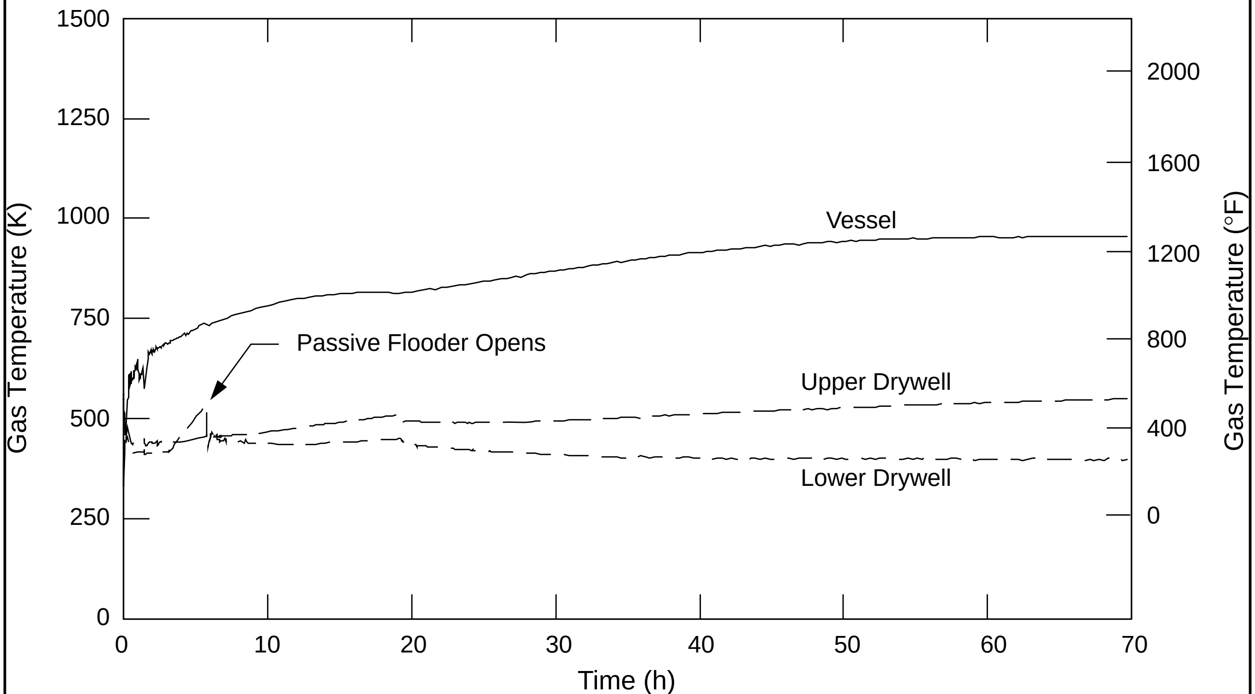


Figure 19E.2-9b LBLC-PF-R-N: Large Break LOCA with Loss of all Core Cooling, Passive Flooder Operates and Rupture Disk Opens: Gas Temperature

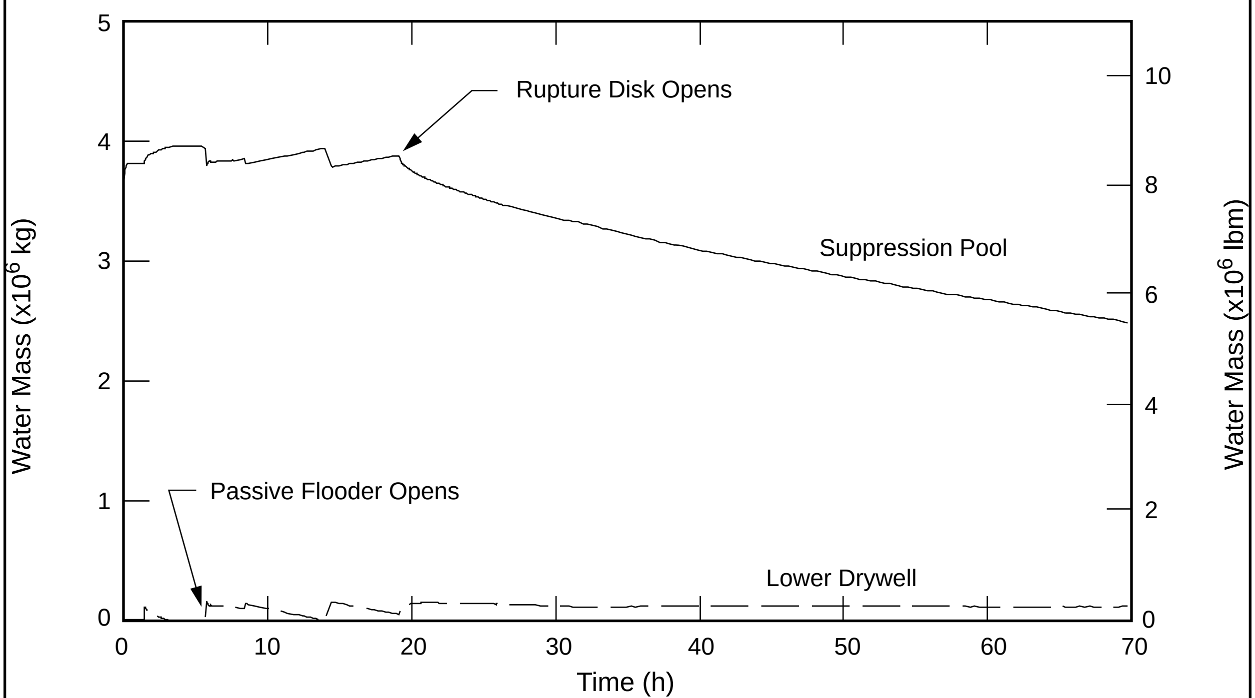


Figure 19E.2-9c LBLC-PF-R-N: Large Break LOCA with Loss of all Core Cooling, Passive Flooder Operates and Rupture Disk Opens: Water Mass

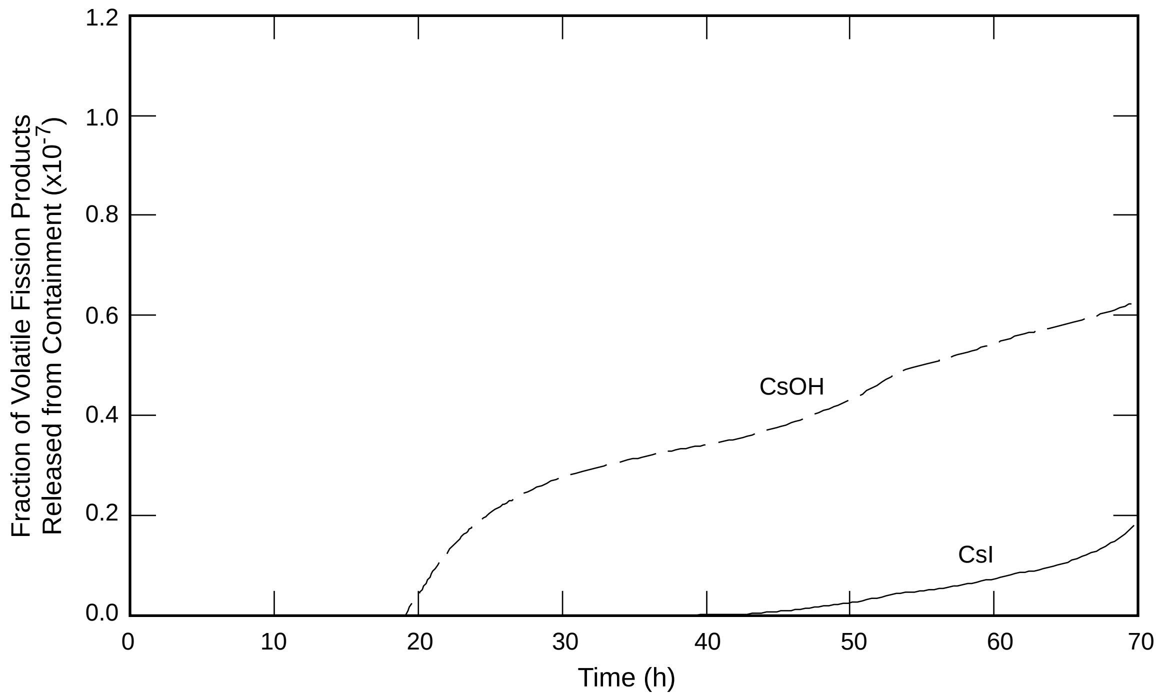


Figure 19E.2-9d LBLC-PF-R-N: Large Break LOCA with Loss of all Core Cooling, Passive Flooder Operates and Rupture Disk Opens: Volatile Fission Product Release

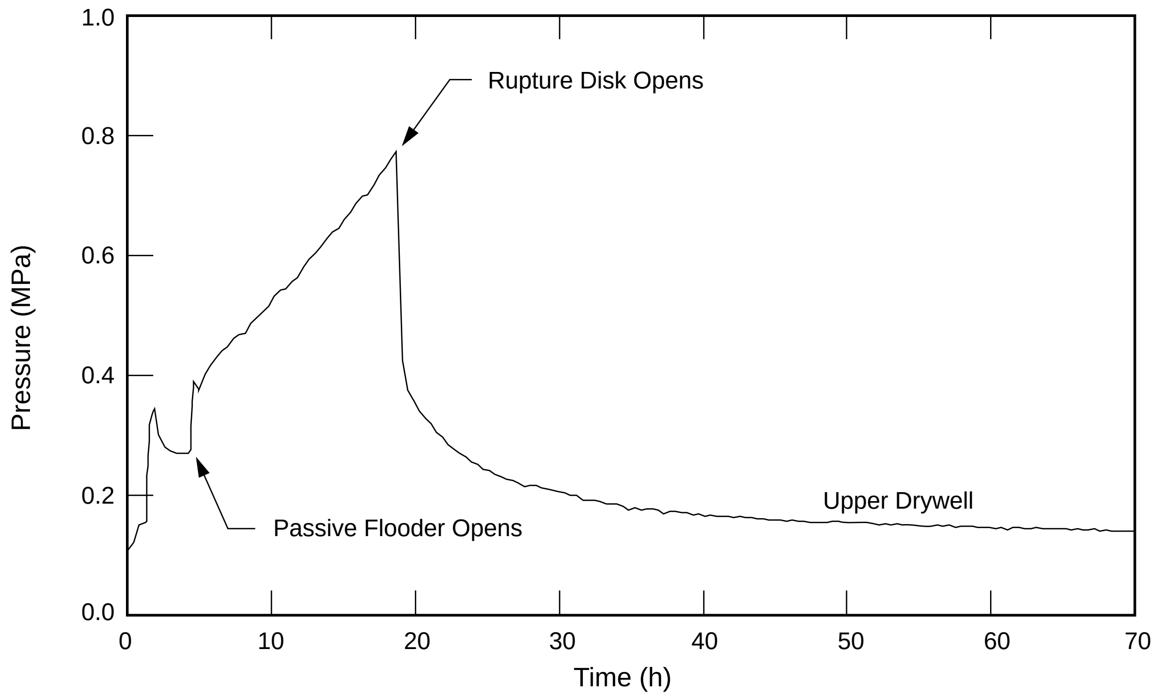


Figure 19E.2-10a NSCL-PF-R-N: Concurrent Loss of all Core Cooling and ATWS with Vessel Failure at Low Pressure, Passive Flooder and Rupture Disk: Drywell Pressure

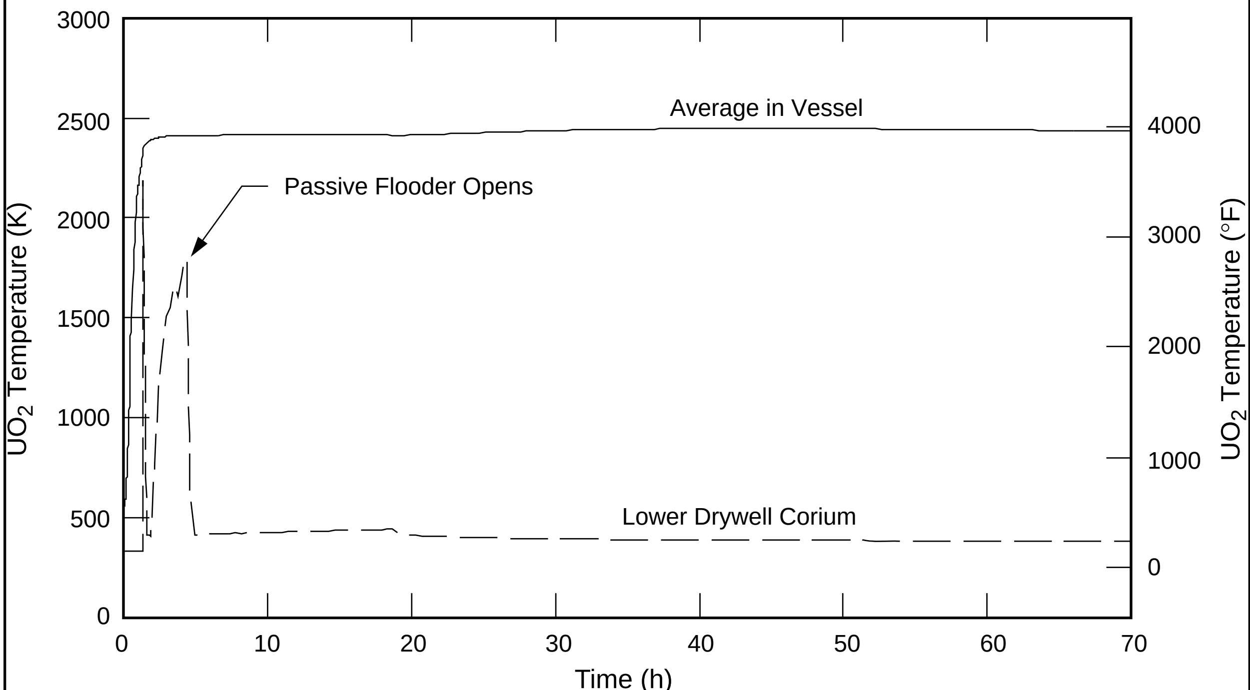


Figure 19E.2-10b NSCL-PF-R-N: Concurrent Loss of all Core Cooling and ATWS with Vessel Failure at Low Pressure, Passive Flooder and Rupture Disk: UO_2 Temperature

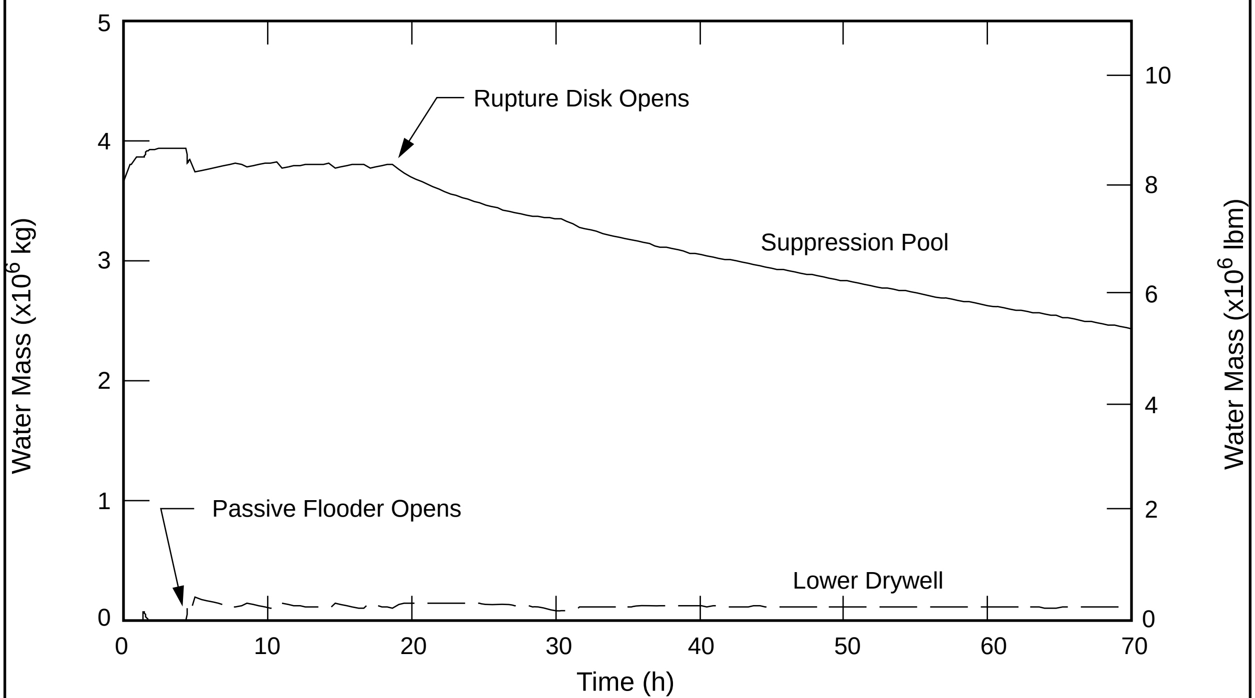


Figure 19E.2-10c NSCL-PF-R-N: Concurrent Loss of all Core Cooling and ATWS with Vessel Failure at Low Pressure, Passive Flooder and Rupture Disk: Water Mass

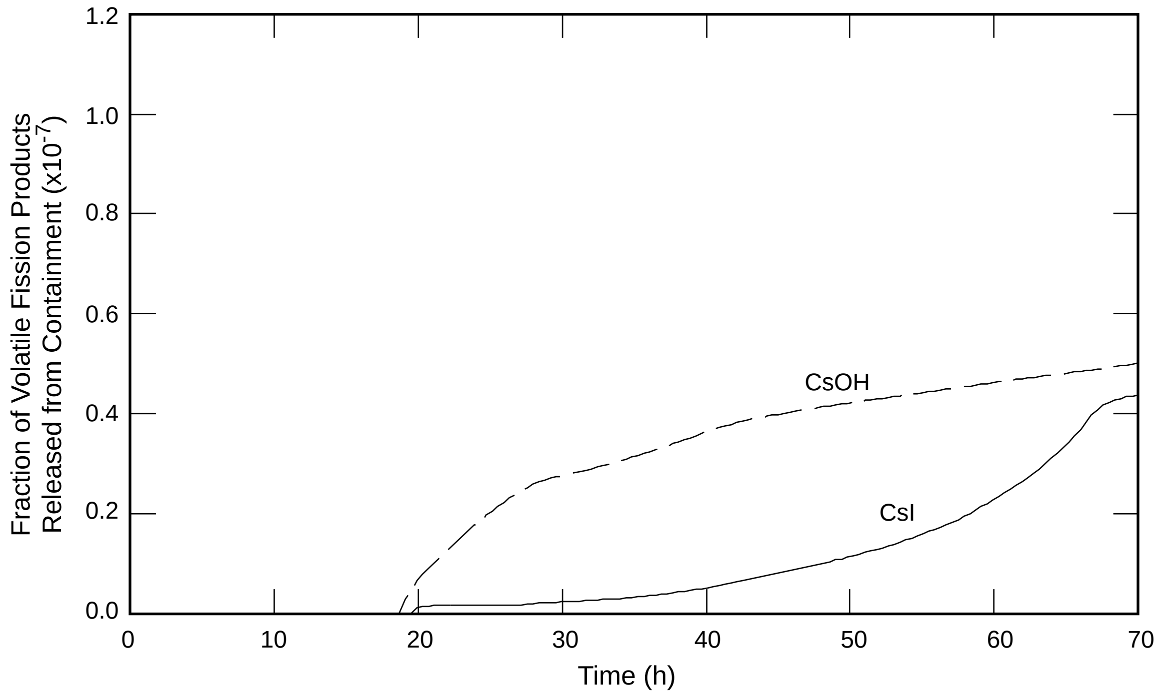


Figure 19E.2-10d NSCL-PF-R-N: Concurrent Loss of all Core Cooling and ATWS with Vessel Failure at Low Pressure, Passive Flooder and Rupture Disk: Volatile Fission Products

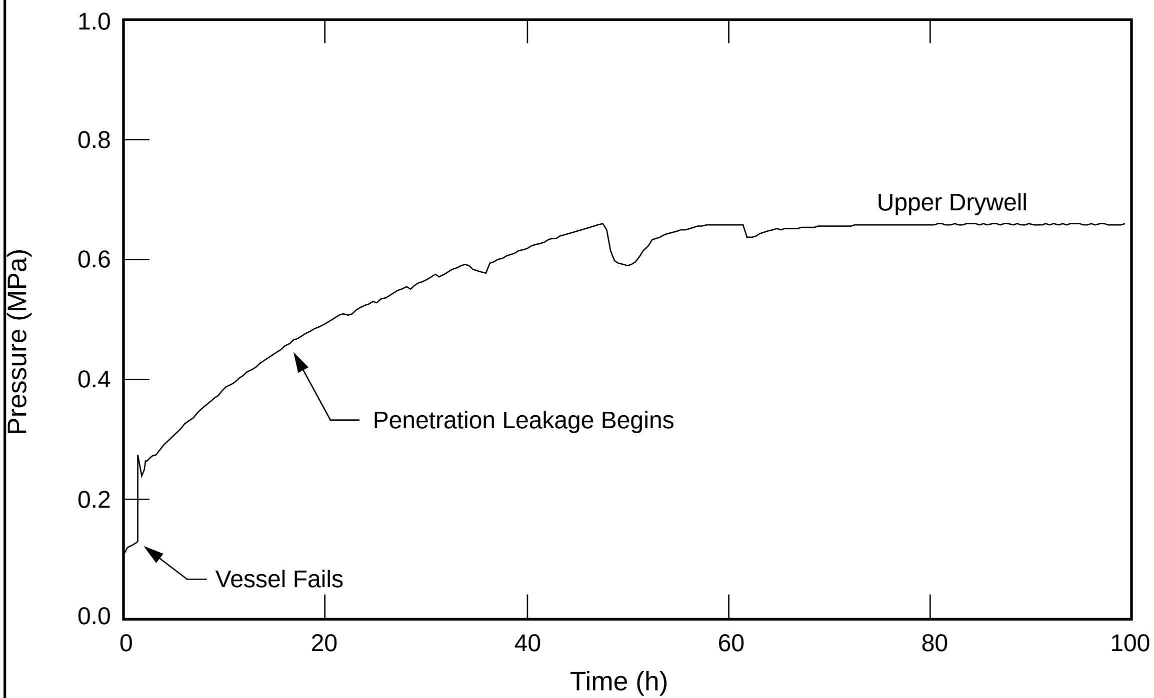


Figure 19E.2-11a NSCH-PF-P-M: Concurrent Loss of all Core Cooling and ATWS with Vessel Failure at High Pressure, Passive Flooder, Penetration Leakage: Drywell Pressure

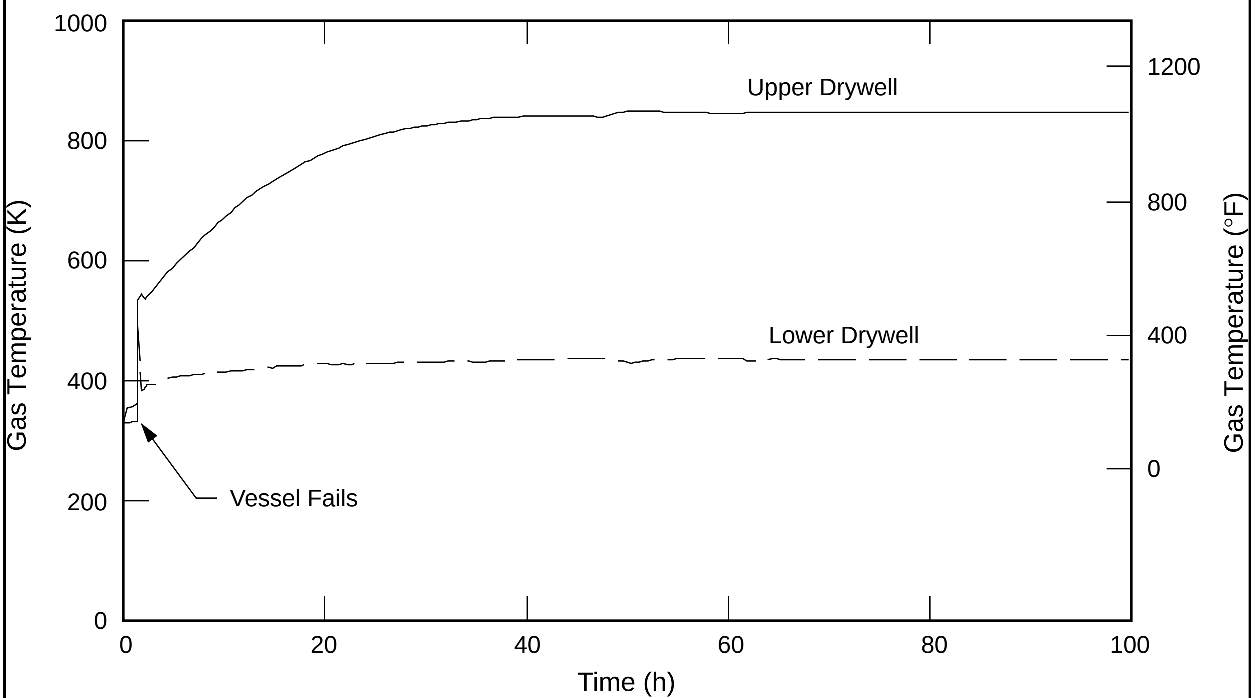


Figure 19E.2-11b NSCH-PF-P-M: Concurrent Loss of all Core Cooling and ATWS with Vessel Failure at High Pressure, Passive Flooder, Penetration Leakage: Gas Temperature

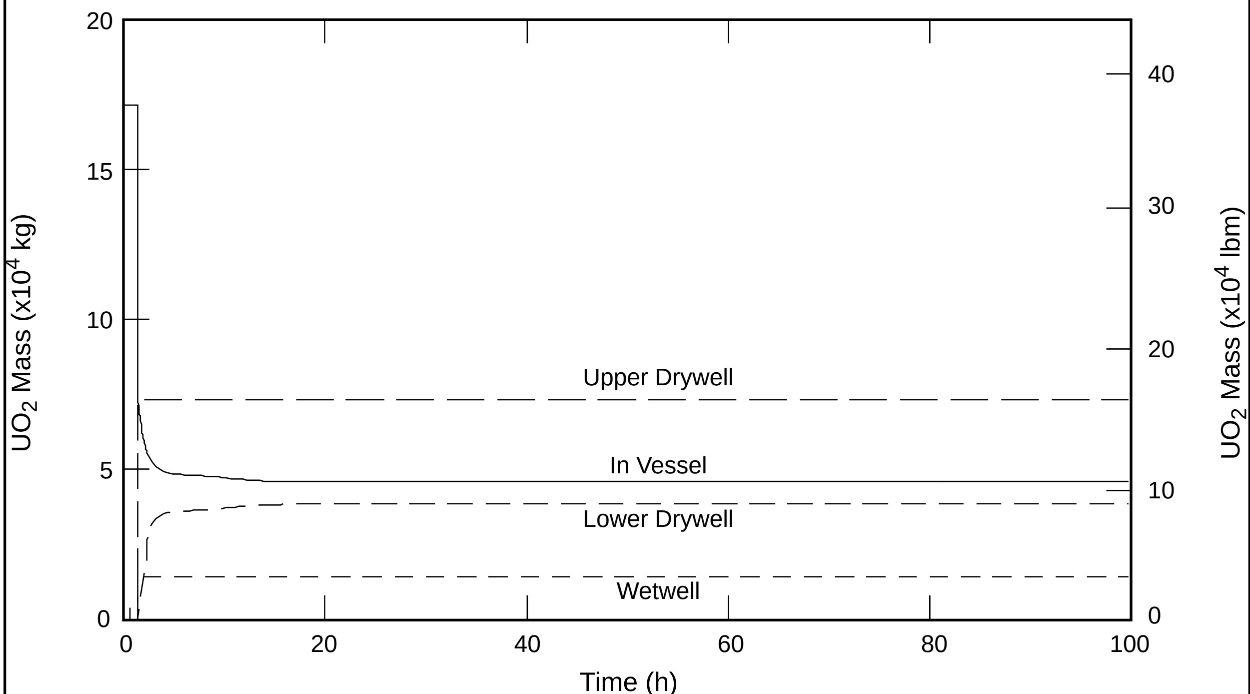


Figure 19E.2-11c NSCH-PF-P-M: Concurrent Loss of all Core Cooling and ATWS with Vessel Failure at High Pressure, Passive Flooder, Penetration Leakage: UO_2 Mass

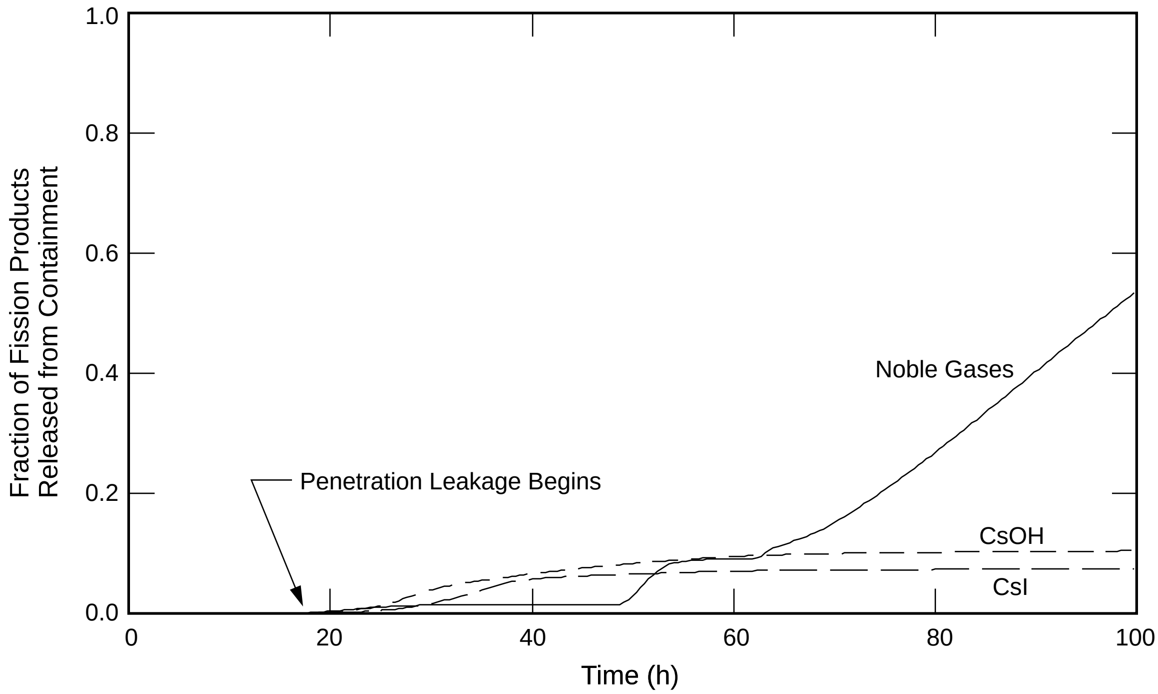


Figure 19E.2-11d NSCH-PF-P-M: Concurrent Loss of all Core Cooling and ATWS with Vessel Failure at High Pressure, Passive Flooder, Penetration Leakage: Fission Products

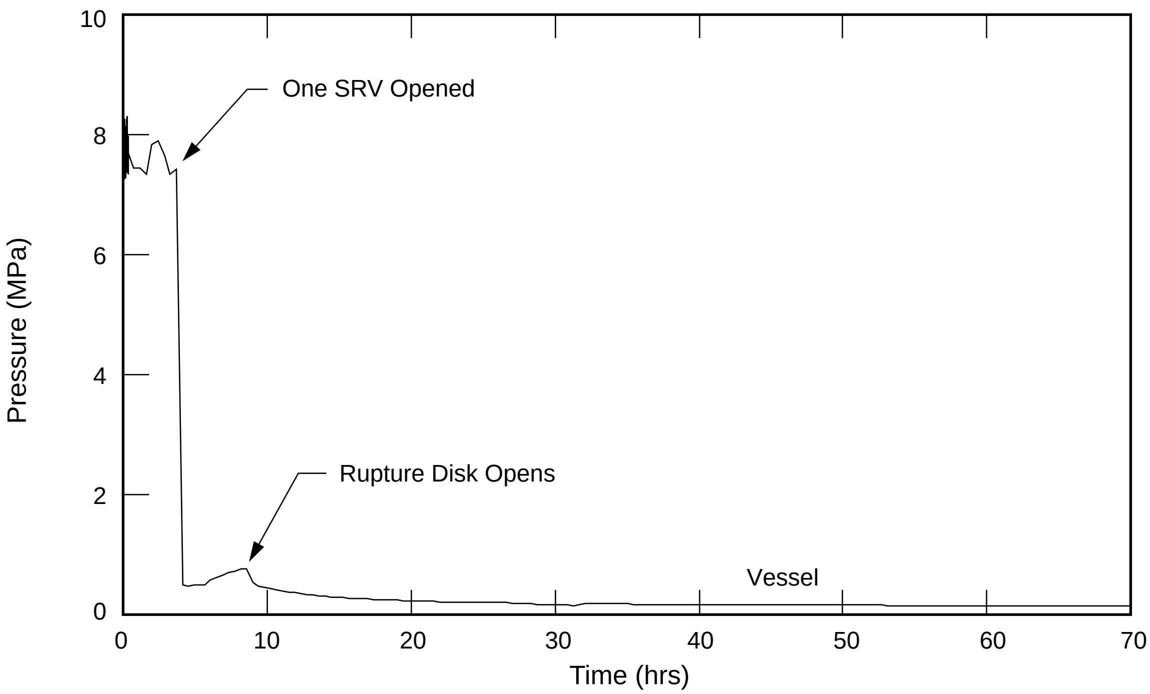


Figure 19E.2-12a NSRC-PF-R-N: Concurrent Station Blackout with ATWS, Passive Flooder Operates and Rupture Disk Opens: Vessel Pressure

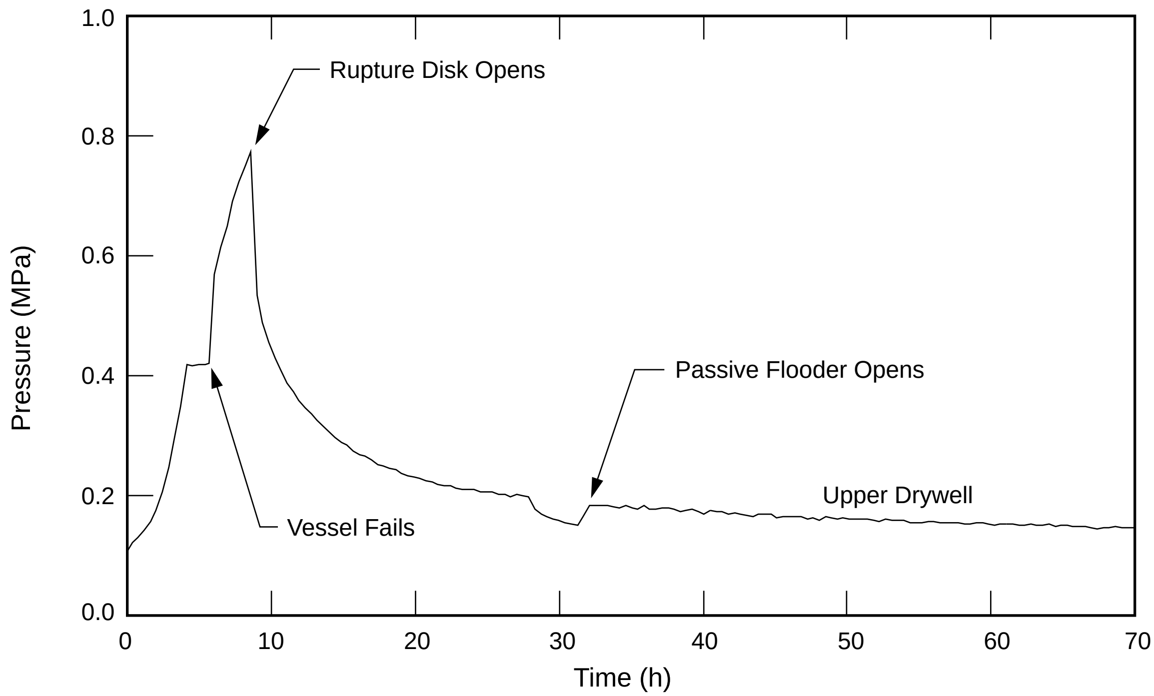


Figure 19E.2-12b NSRC-PF-R-N: Concurrent Station Blackout with ATWS, Passive Flooder Operates and Rupture Disk Opens: Drywell Pressure

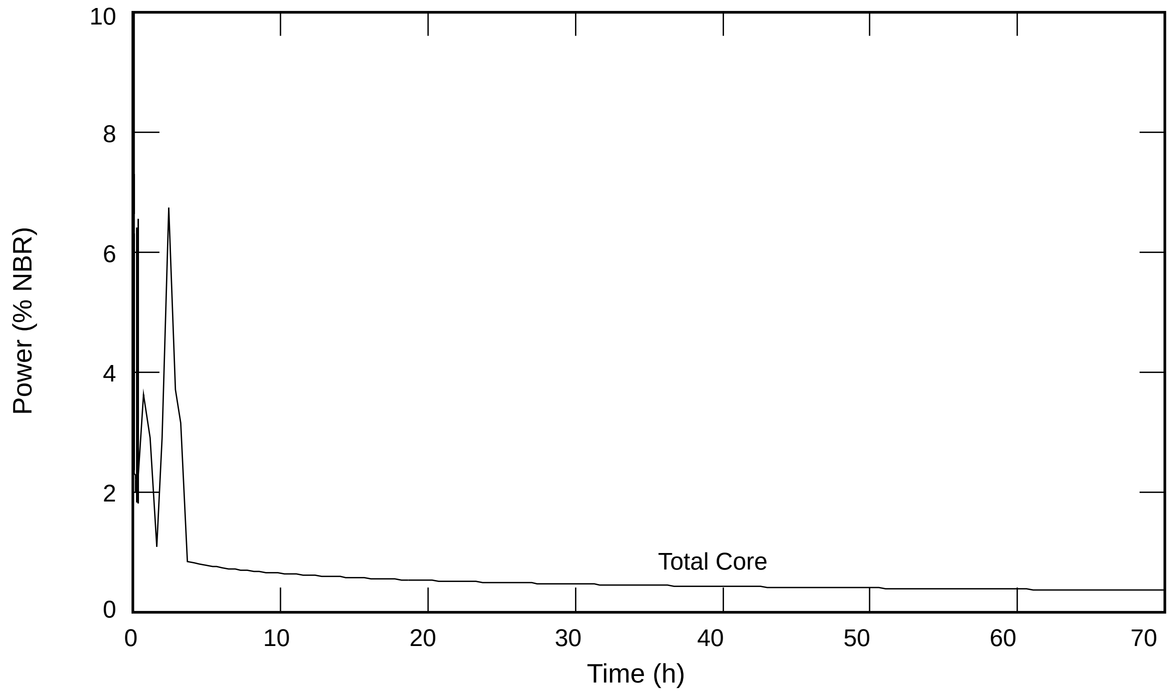


Figure 19E.2-12c NSRC-PF-R-N: Concurrent Station Blackout with ATWS, Passive Flooder Operates and Rupture Disk Opens: Power

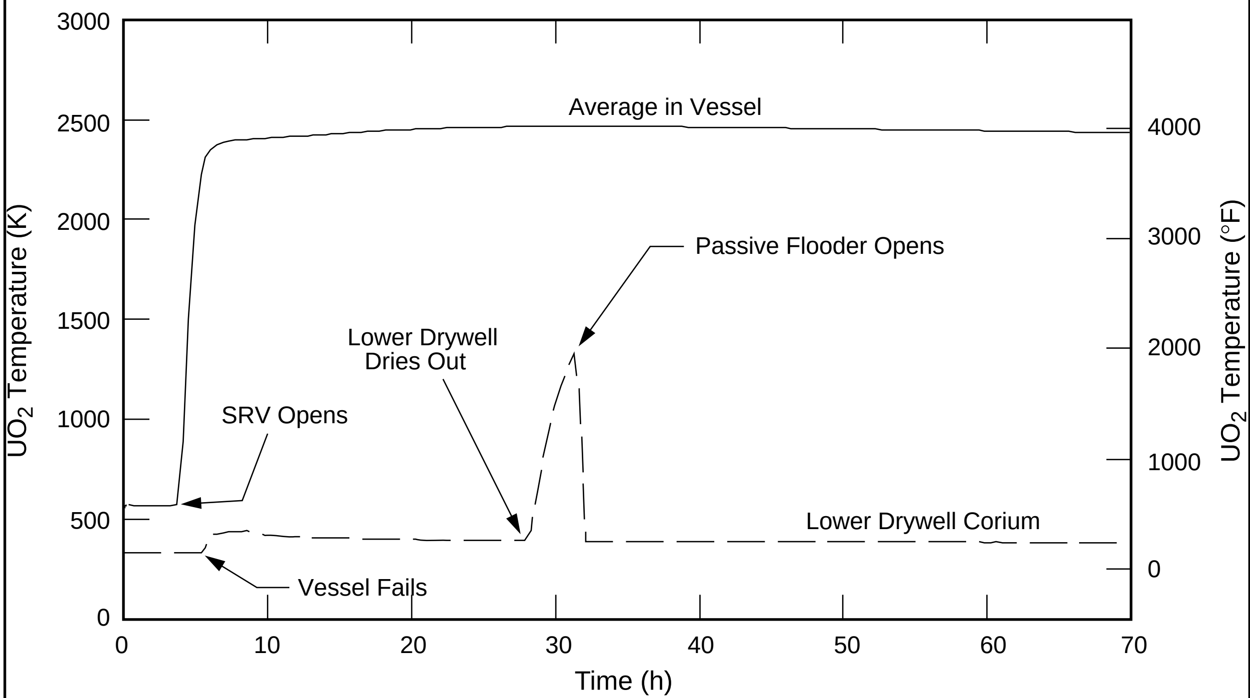


Figure 19E.2-12d NSRC-PF-R-N: Concurrent Station Blackout with ATWS, Passive Flooder Operates and Rupture Disk Opens: UO_2 Temperature

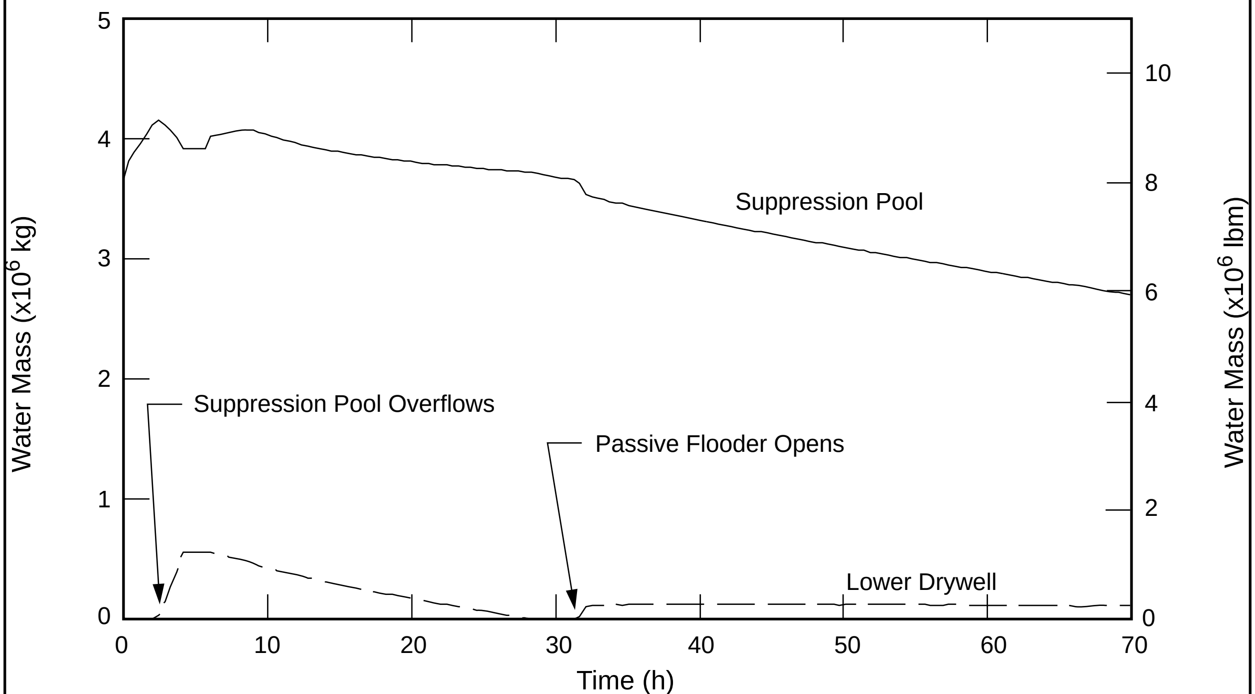


Figure 19E.2-12e NSRC-PF-R-N: Concurrent Station Blackout with ATWS, Passive Flooder Operates and Rupture Disk Opens: Water Mass

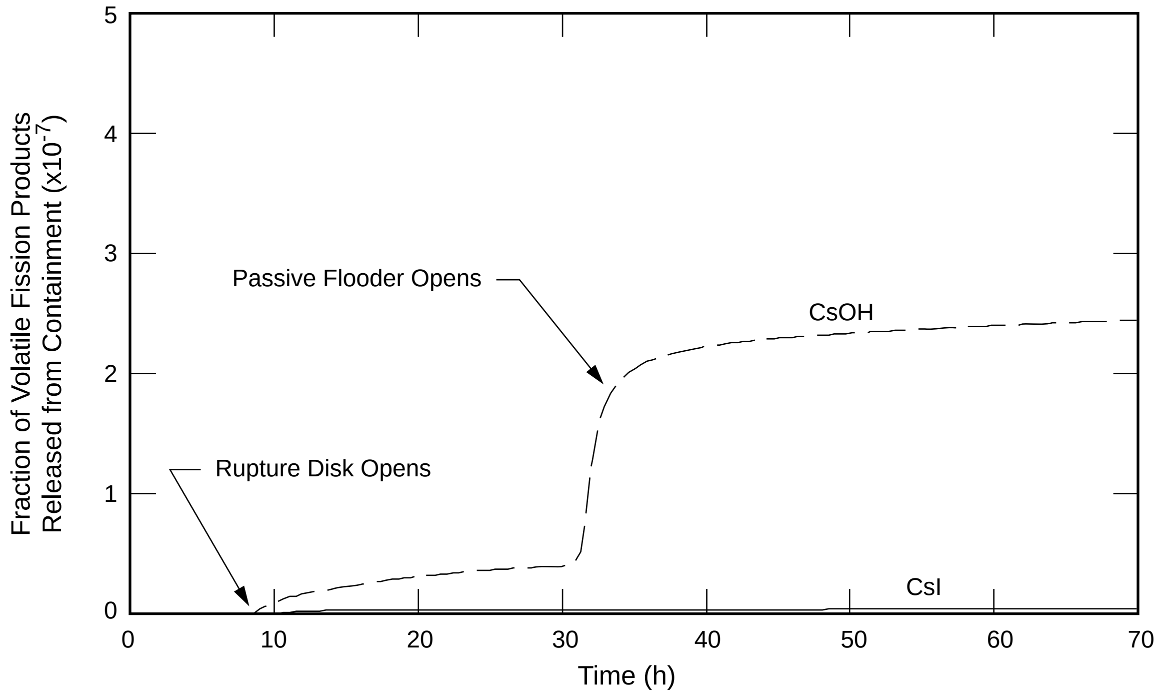


Figure 19E.2-12f NSRC-PF-R-N: Concurrent Station Blackout with ATWS, Passive Flooder Operates and Rupture Disk Opens: Volatile Fission Product Release

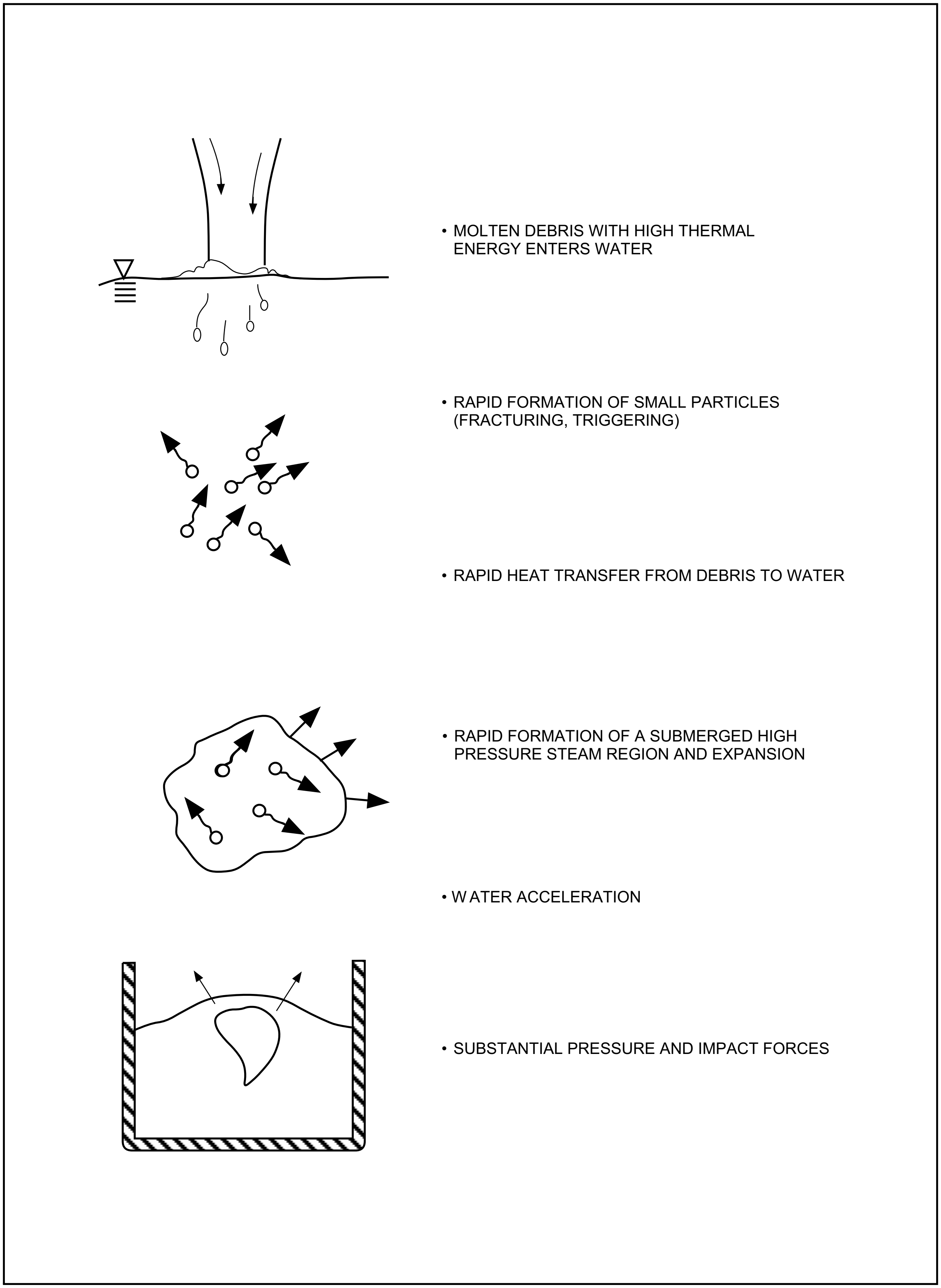


Figure 19E.2-13 Steam Explosion Process

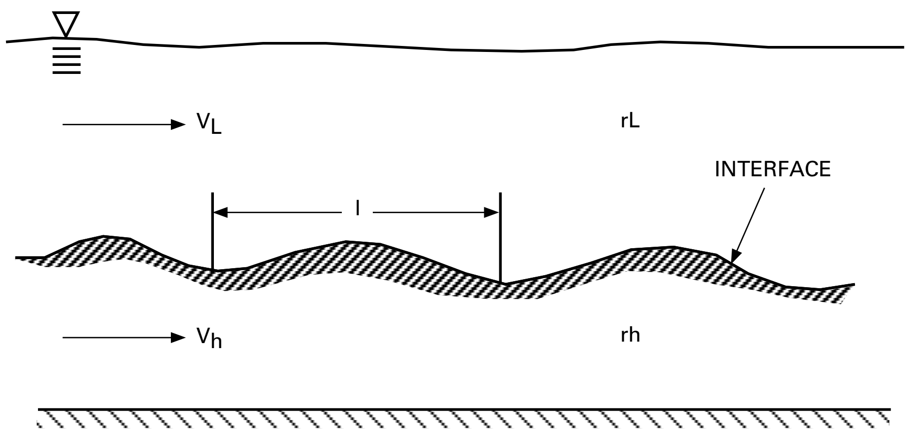


Figure 19E.2-14a Interfacial Instability

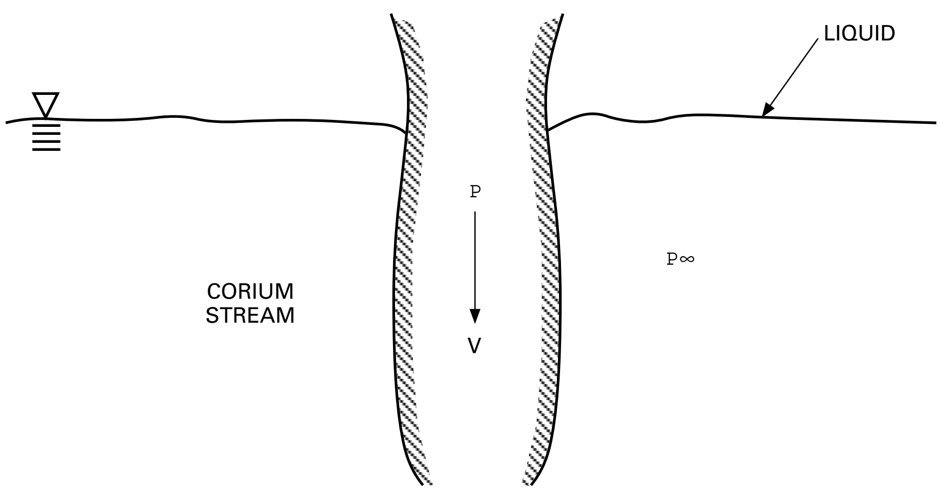


Figure 19E.2-14b Corium Stream in Liquid

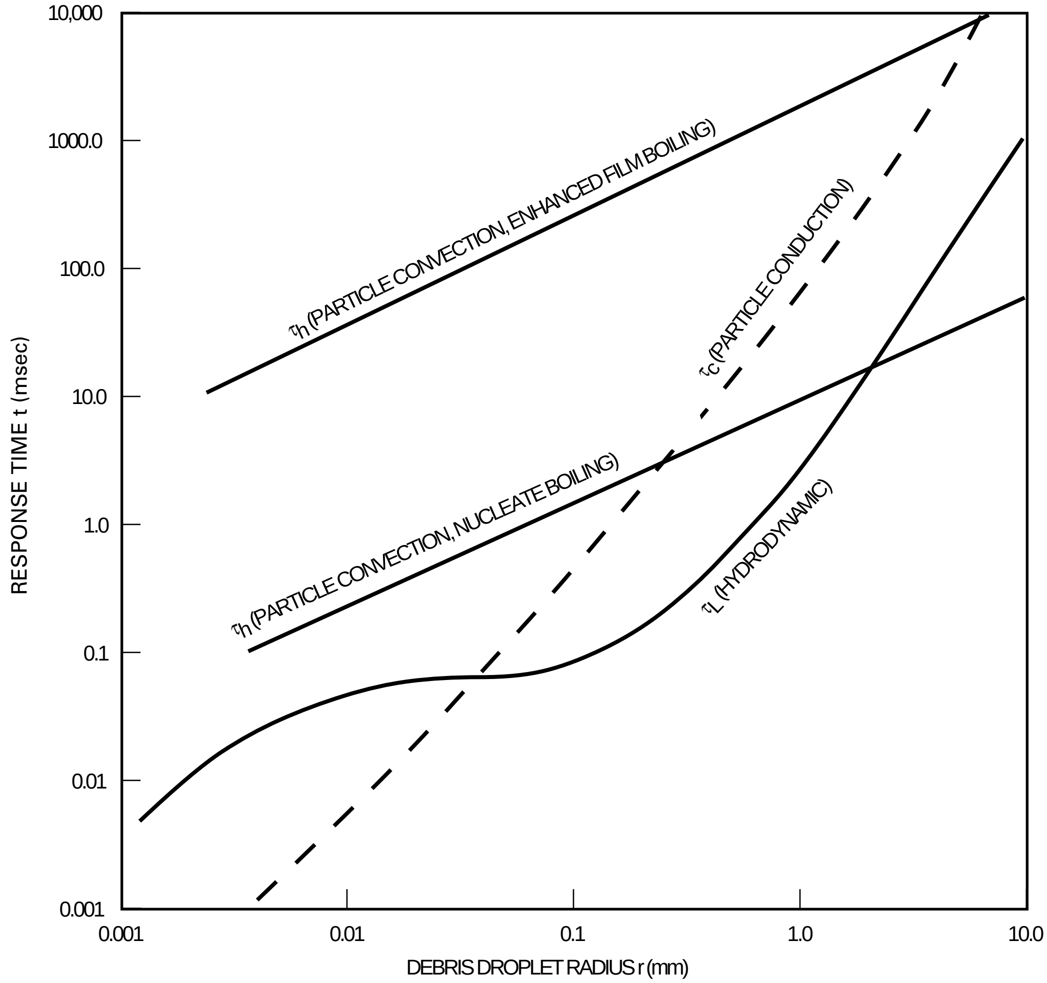


Figure 19E.2-15 Important Response Times

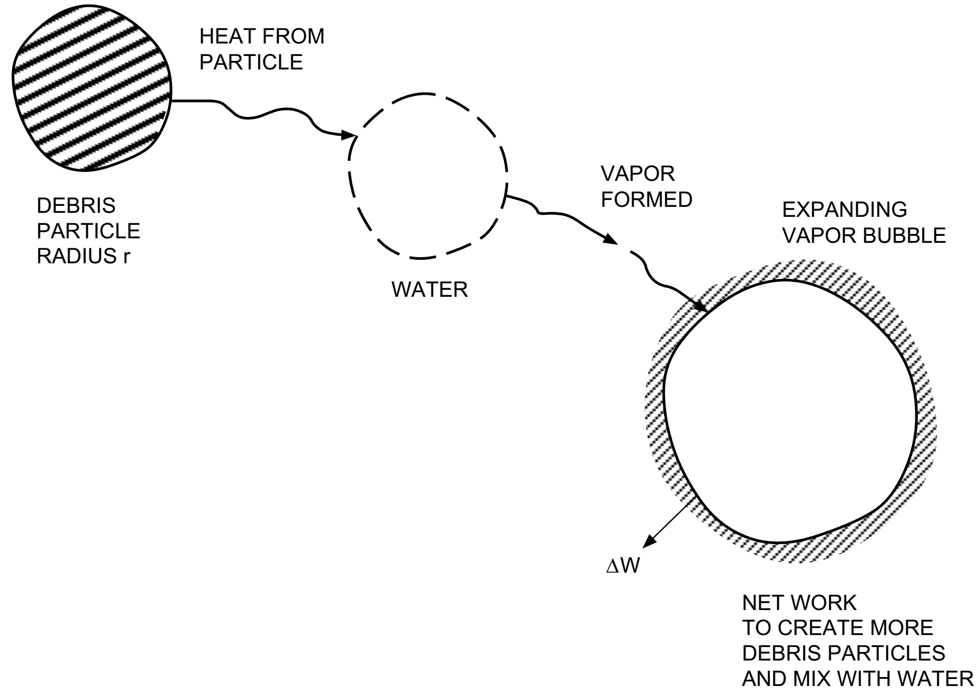


Figure 19E.2-16 Self-Triggering Process

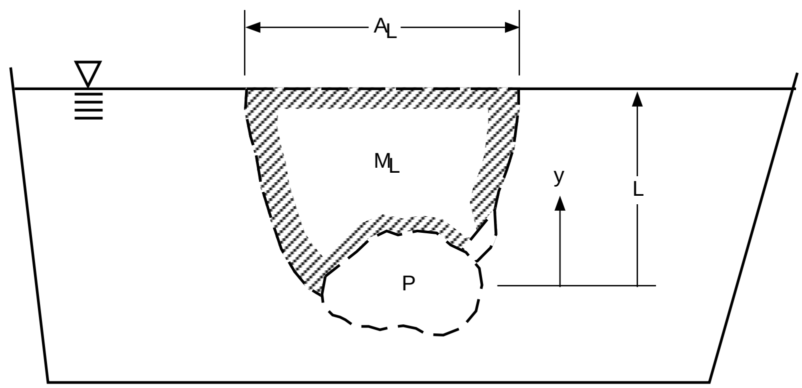


Figure 19E.2-17 Conditions for Steam Explosion

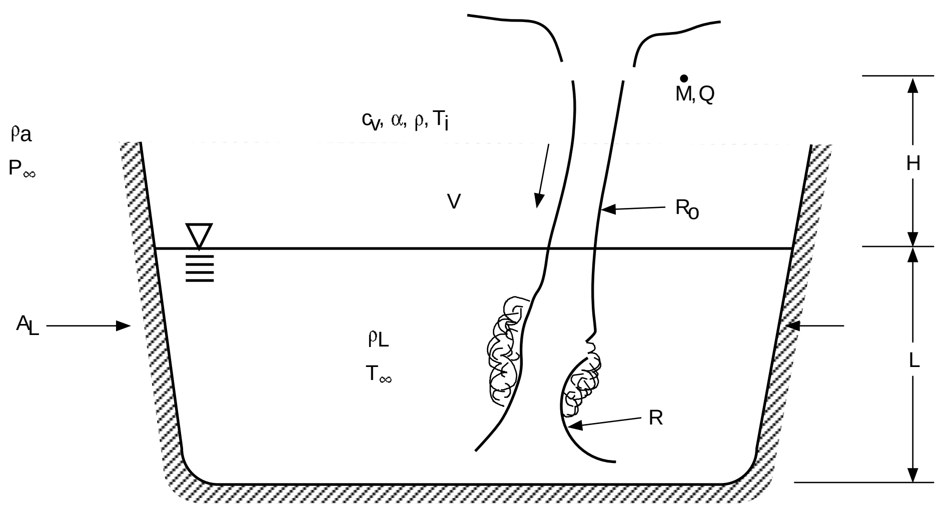


Figure 19E.2-18 Application to ABWR

Figure 19E.2-19a Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19b Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19c Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19d Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19e Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19f Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19g Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19h Suppression Pool Bypass Paths and Configurations
Not Part of DCD (Refer to SSAR)

Figure 19E.2-19i Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19j Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-19k Suppression Pool Bypass Paths and Configurations

Not Part of DCD (Refer to SSAR)

Figure 19E.2-20a Small LOCAs Outside Containment

Not Part of DCD (Refer to SSAR)

Figure 19E.2-20b Intermediate LOCAs Outside Containment

Not Part of DCD (Refer to SSAR)

Figure 19E.2-20c Large LOCAs Outside Containment

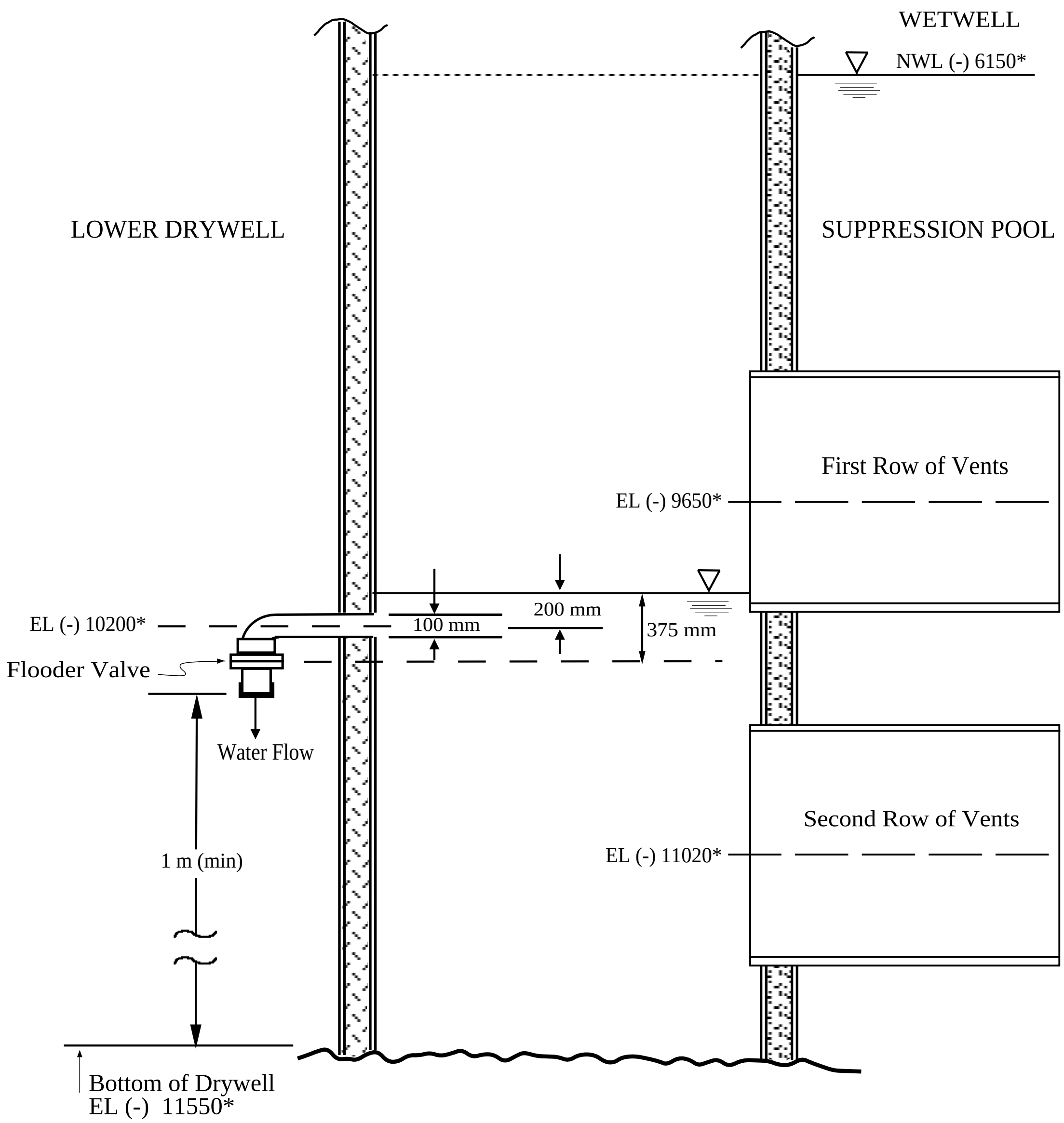
Not Part of DCD (Refer to SSAR)

Figure 19E.2-21 Sensitivity to Suppression Pool Decontamination Factor

Not Part of DCD (Refer to SSAR)

Figure 19E.2-22 Impact of COPS on Risk

Not Part of DCD (Refer to SSAR)



* Elevations based on RPV bottom at EL 0. All dimensions in millimeters.

Figure 19E.2-23 Lower Drywell Flooder System

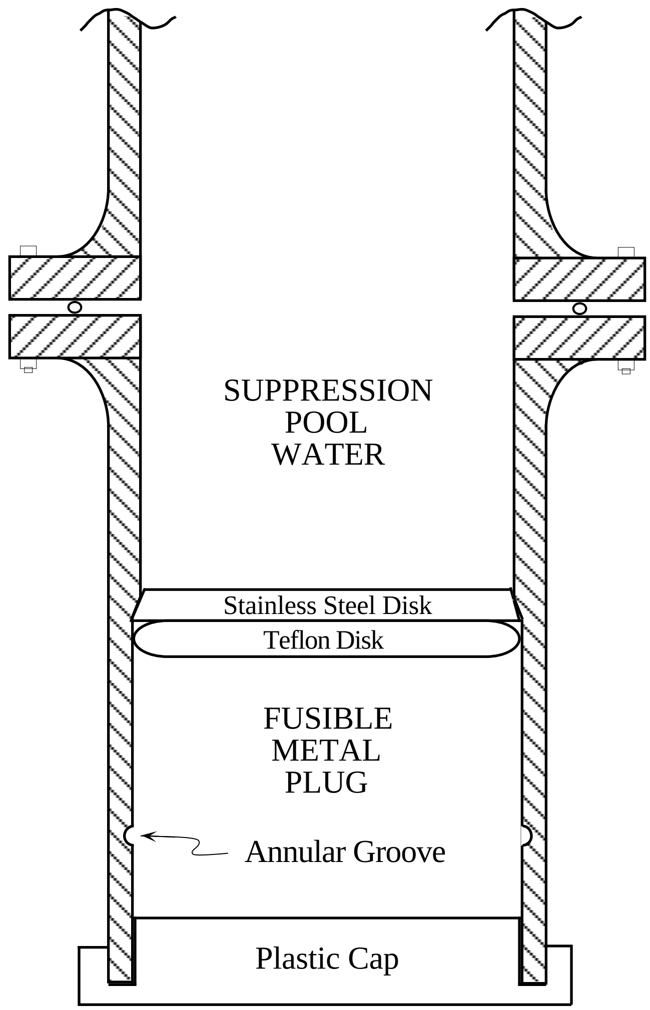


Figure 19E.2-24 Flooder Valve Assembly

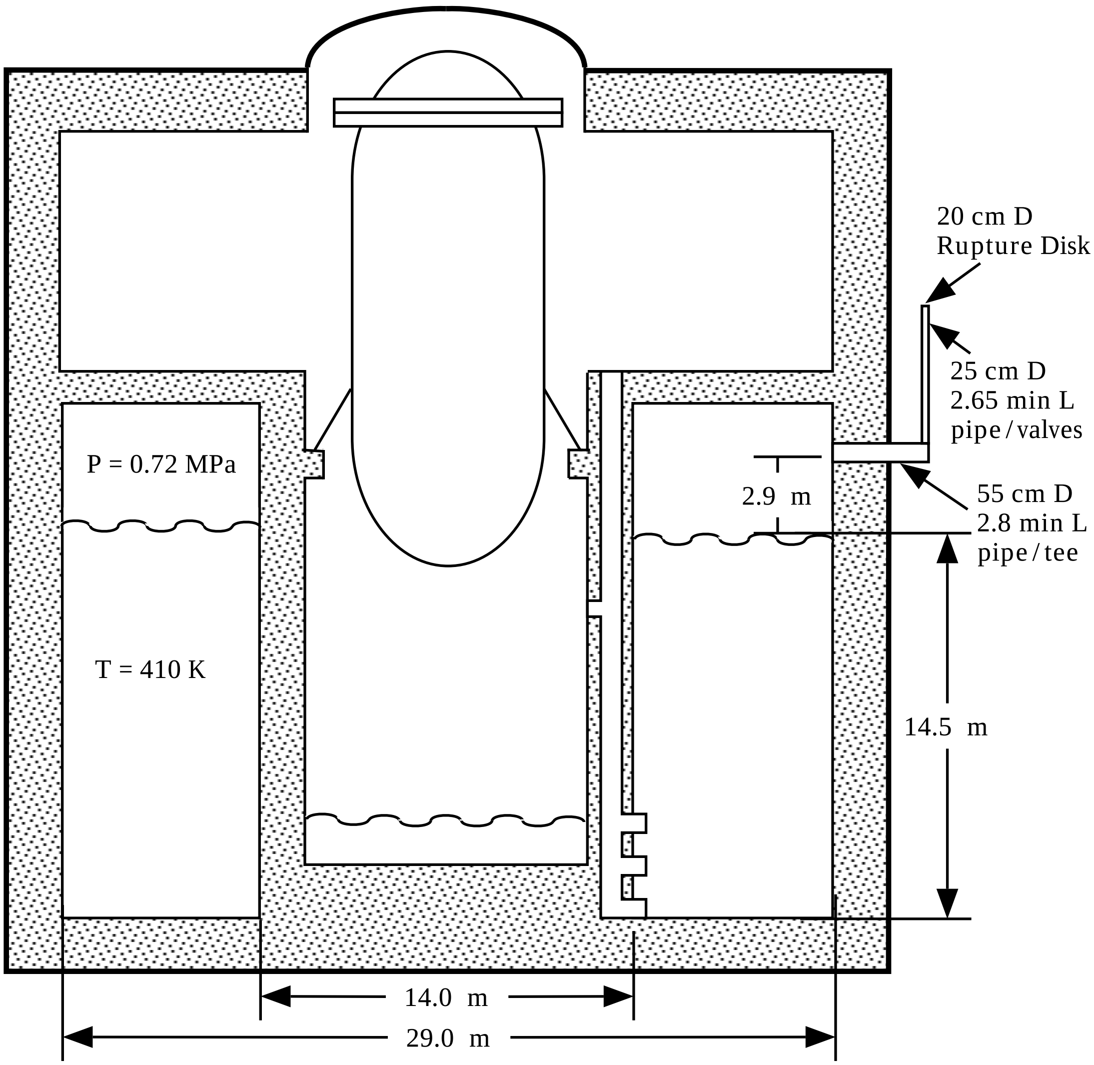


Figure 19E.2-25 Limiting Configuration for COPS Blowdown Study

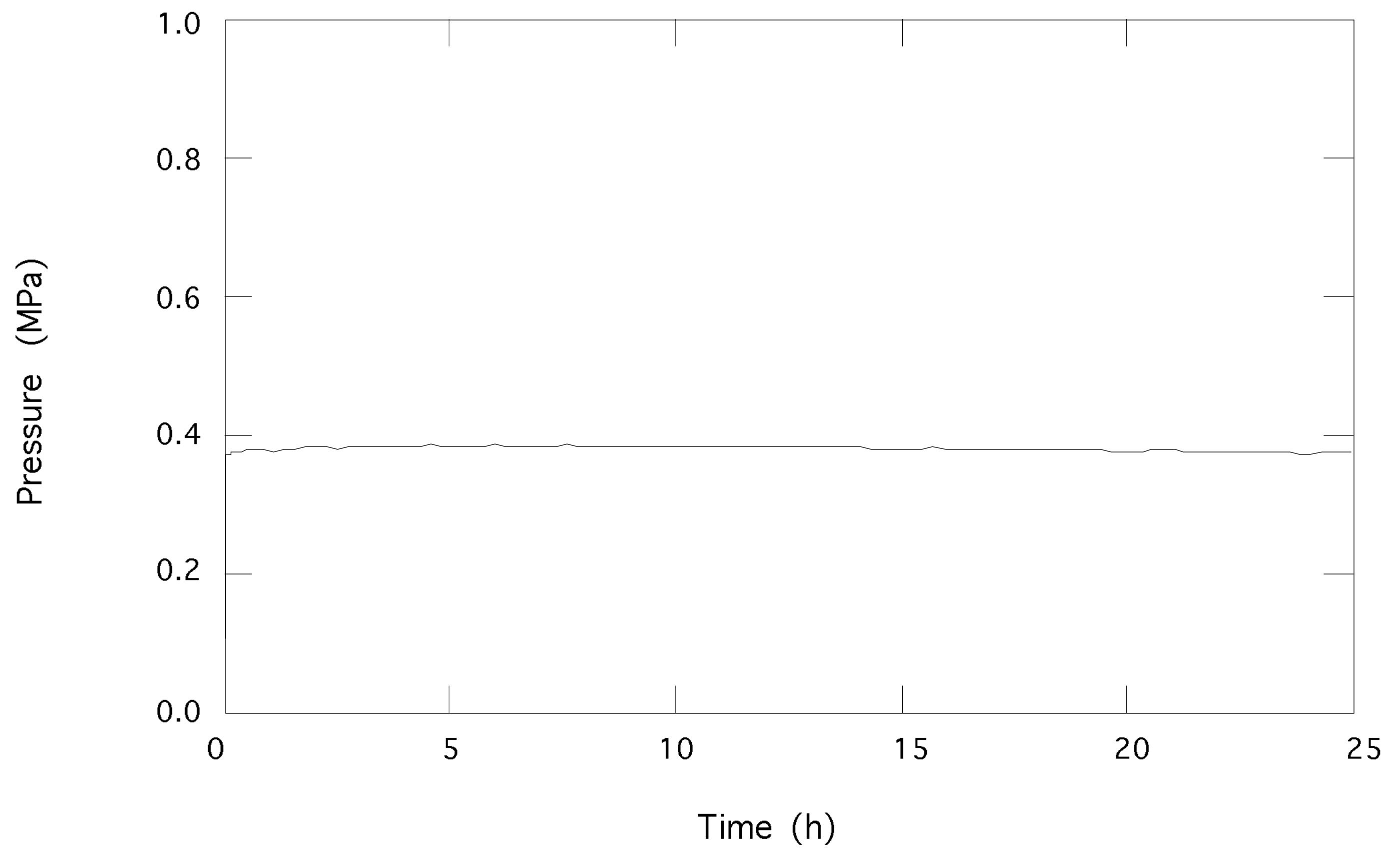


Figure 19E.2-26a Drywell Pressure for 100% Metal-Water Reaction Scenario

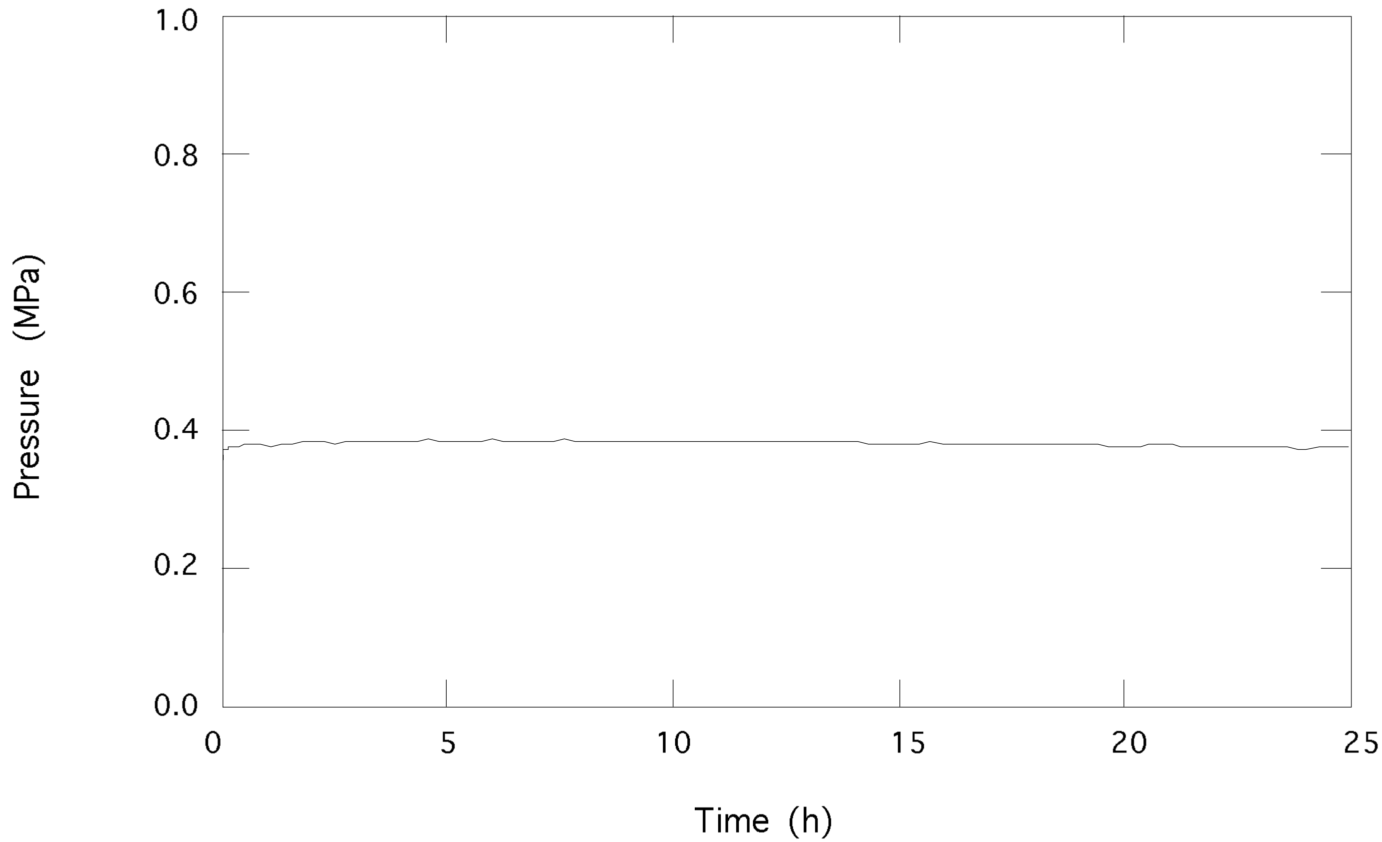


Figure 19E.2-26b Wetwell Pressure for 100% Metal-Water Reaction Scenario

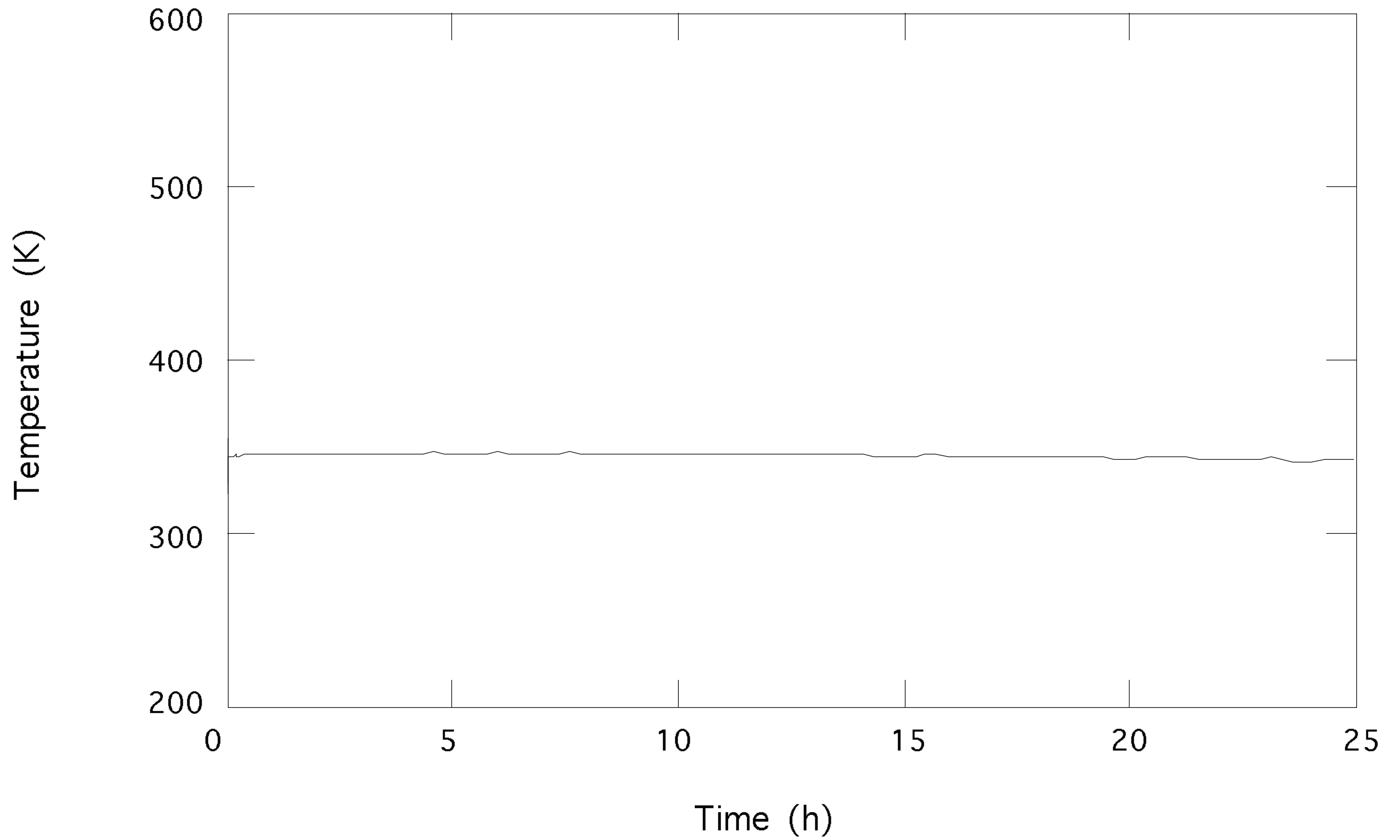


Figure 19E.2-26c Drywell Temperature for 100% Metal-Water Reaction Scenario

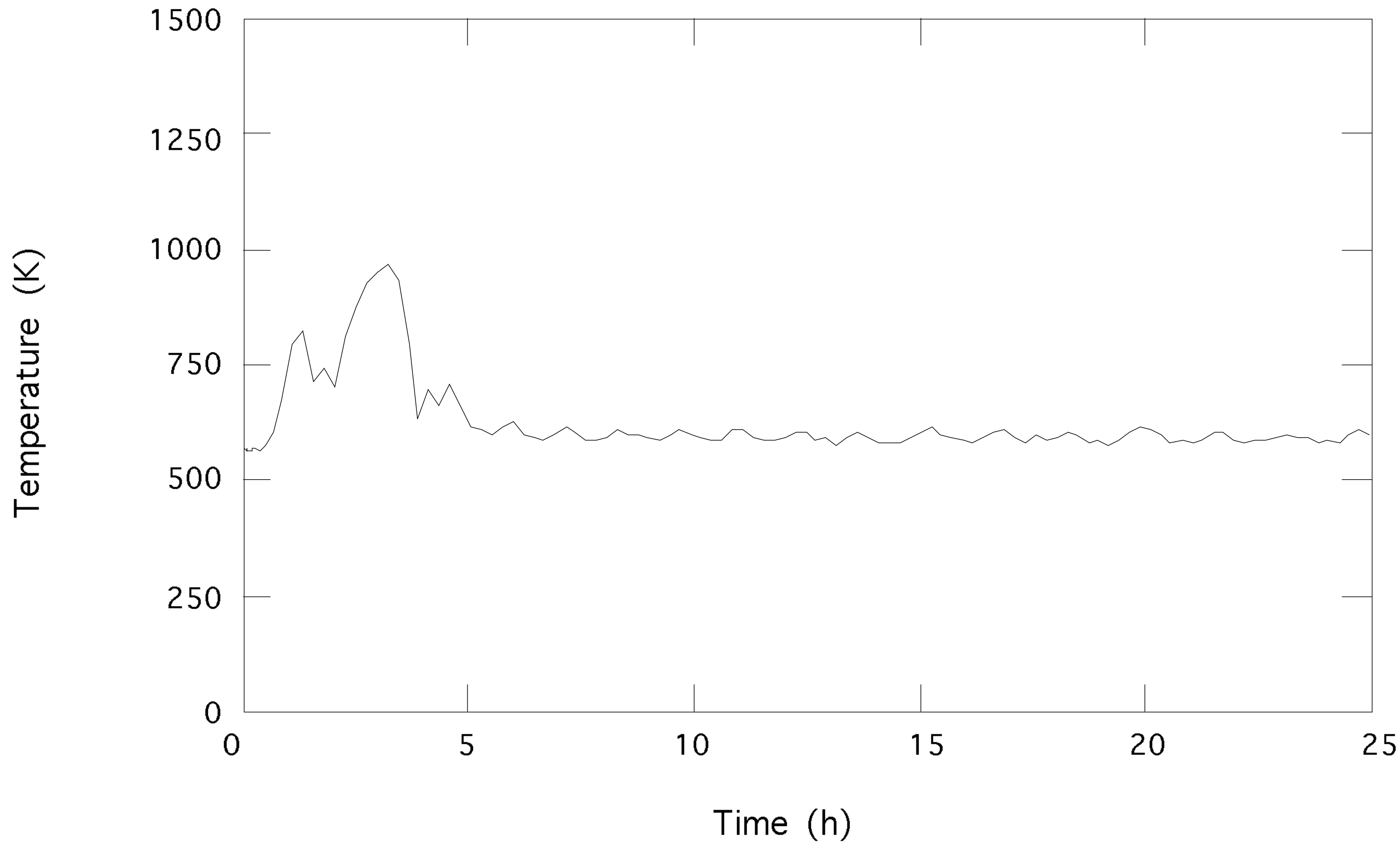


Figure 19E.2-26d Vessel Temperature for 100% Metal-Water Reaction Scenario

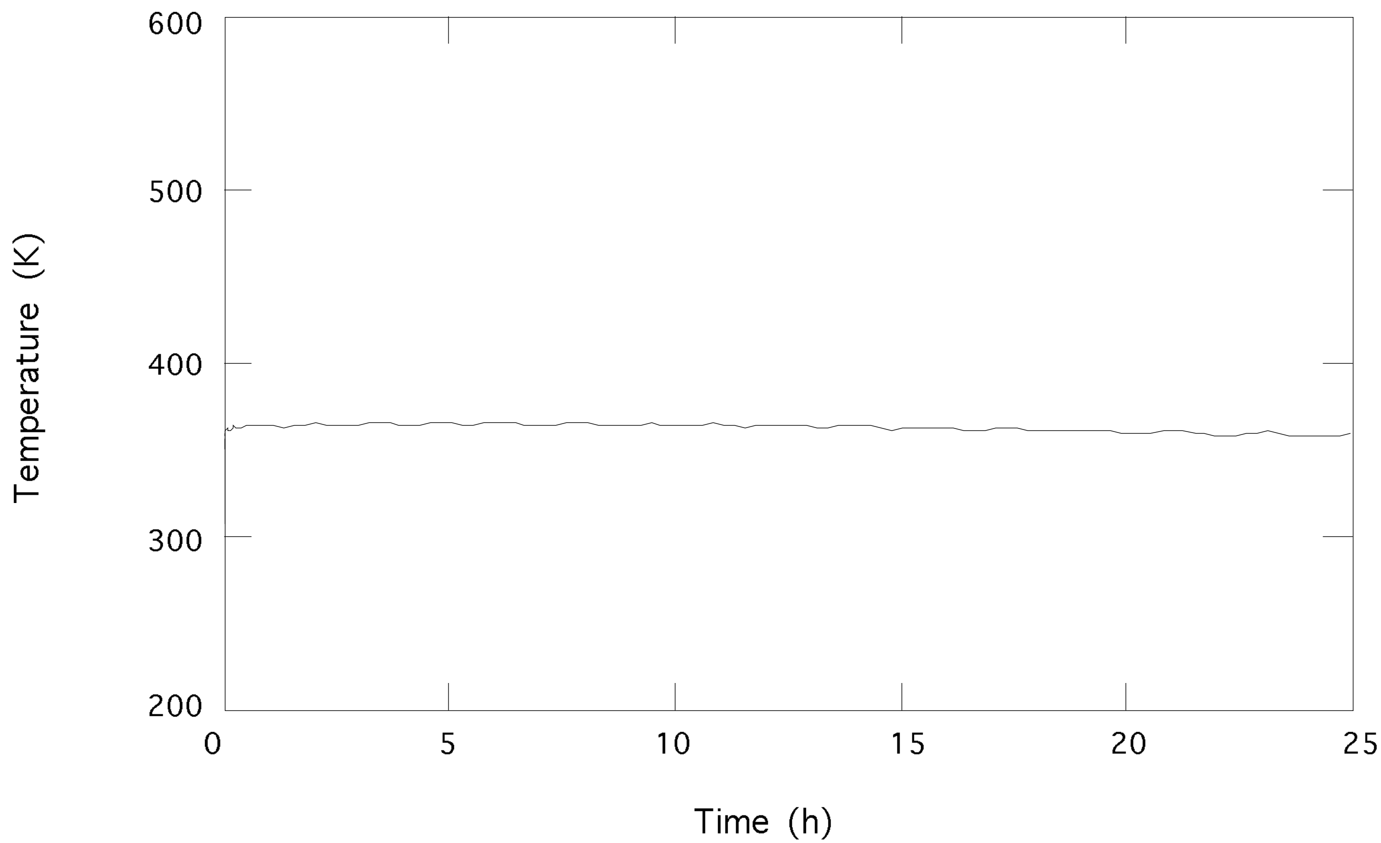


Figure 19E.2-26e Suppression Pool Water Temperature for 100% Metal-Water Scenario

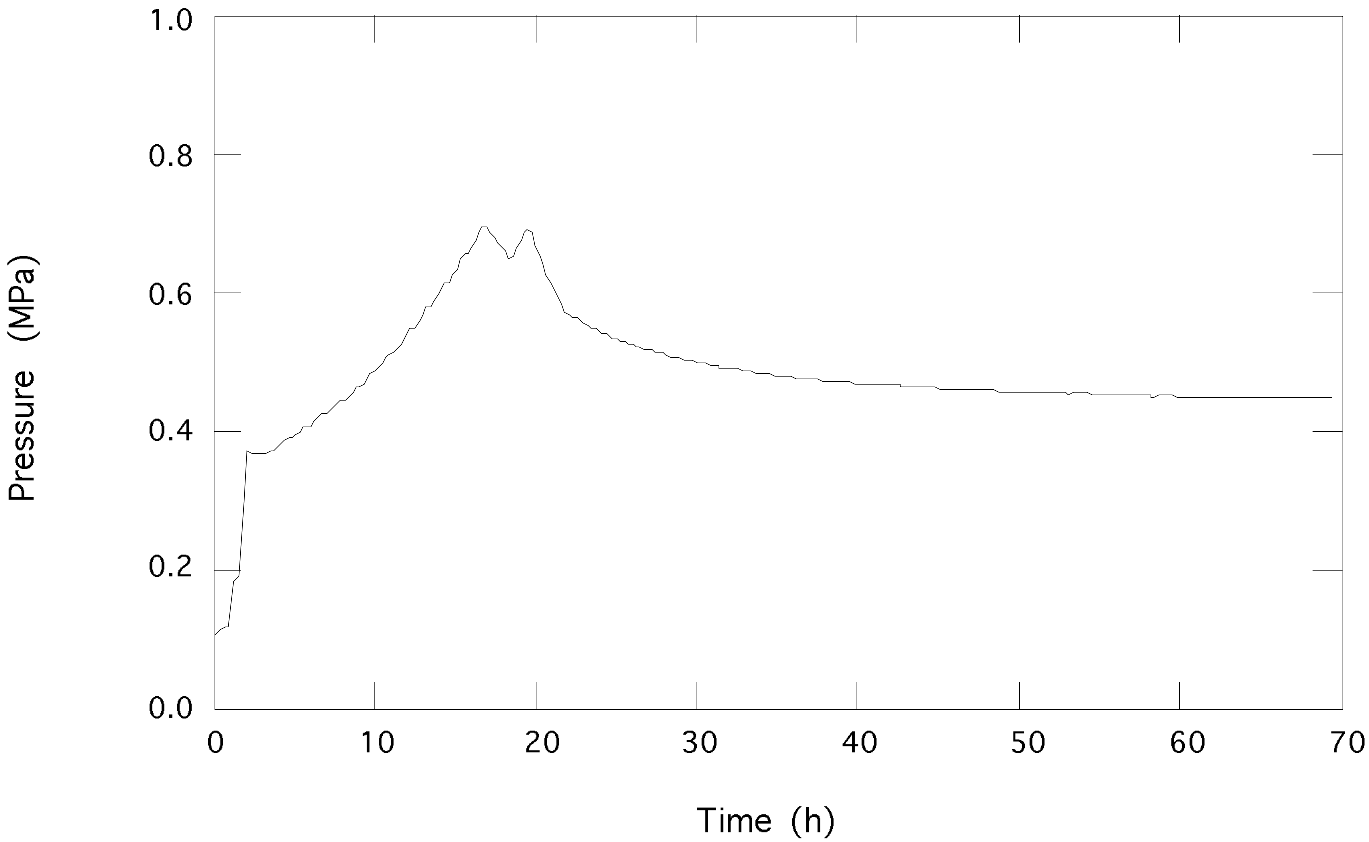


Figure 19E.2-27a Drywell Pressure for In-Vessel Core Melt Scenario

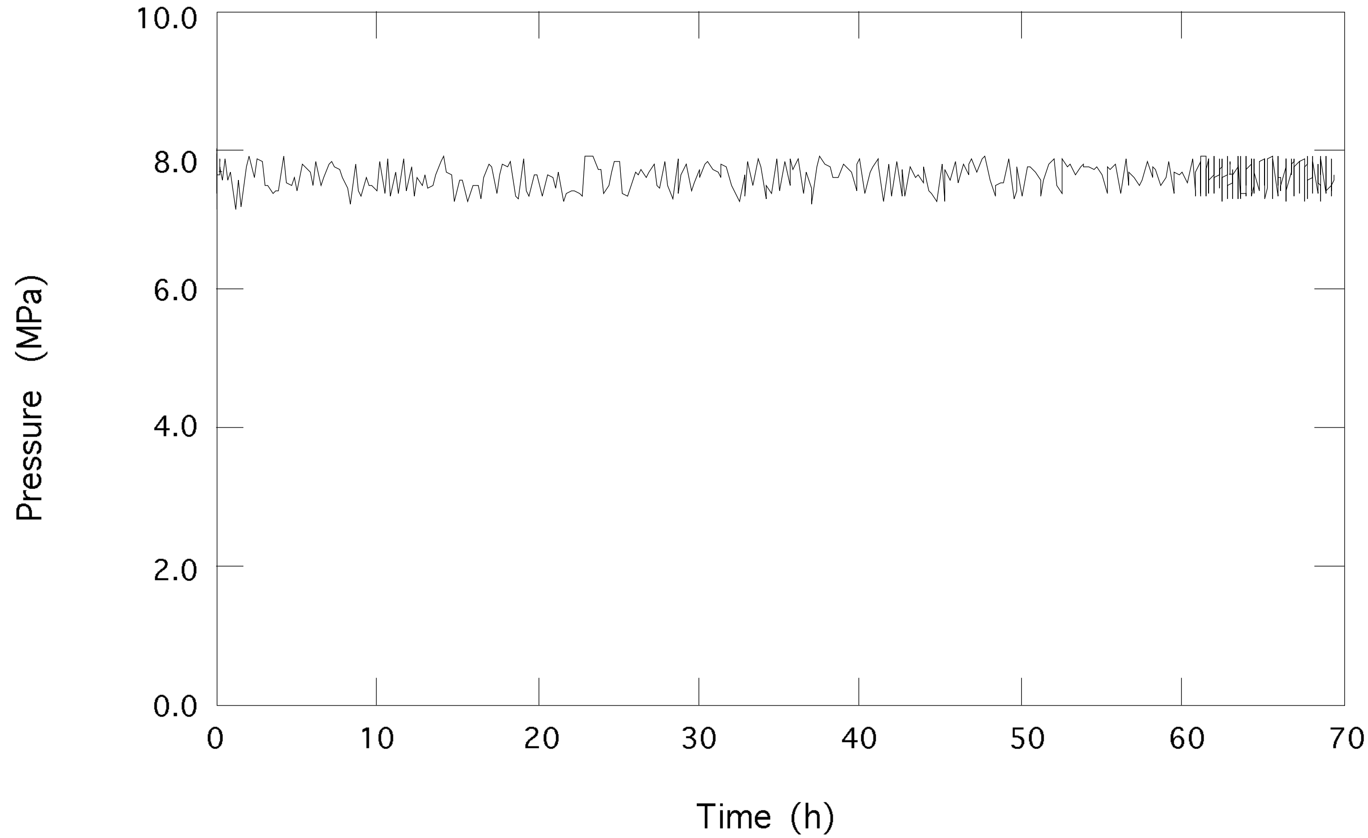


Figure 19E.2-27b Vessel Pressure for In-Vessel Core Melt Scenario

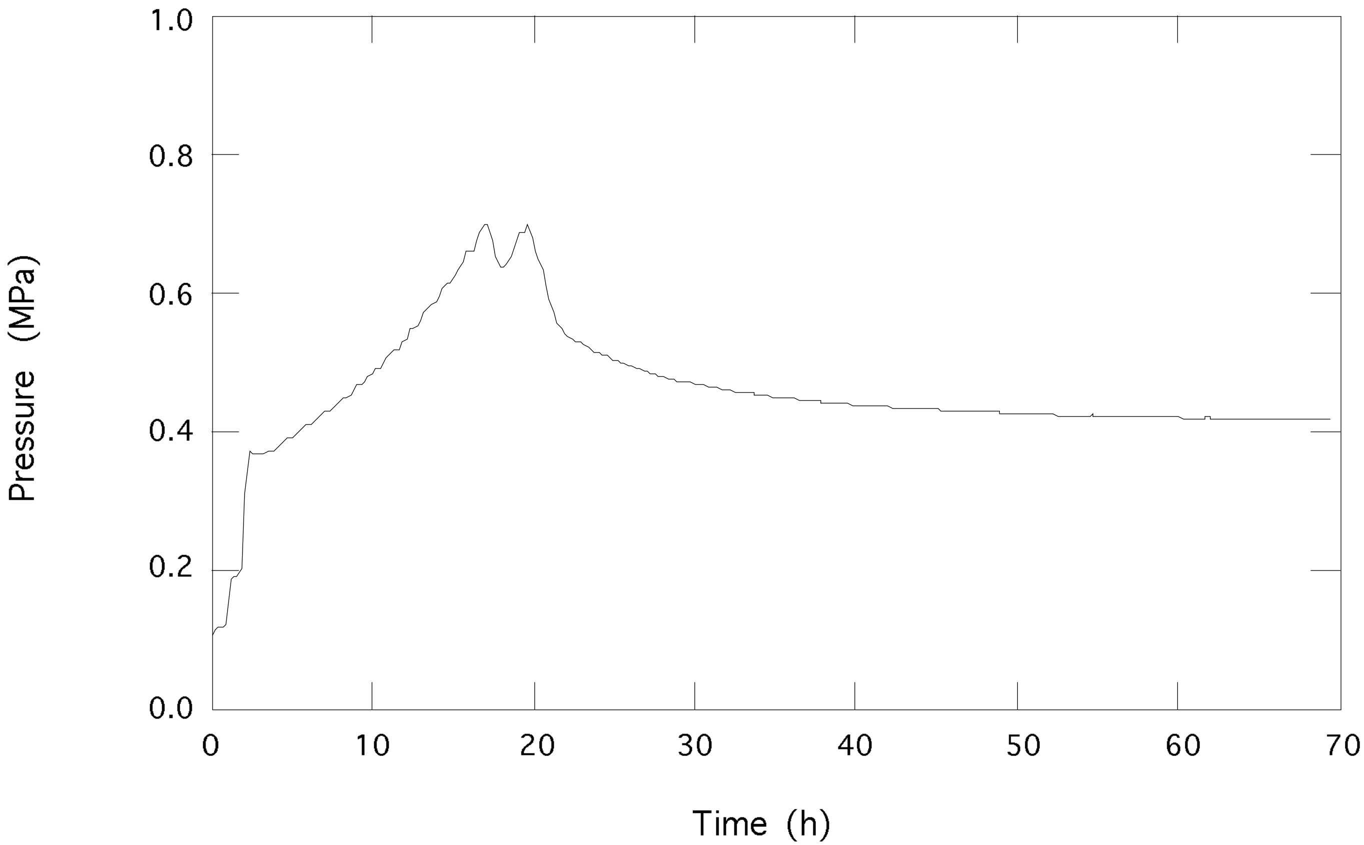


Figure 19E.2-27c Wetwell Pressure for In-Vessel Core Melt Scenario

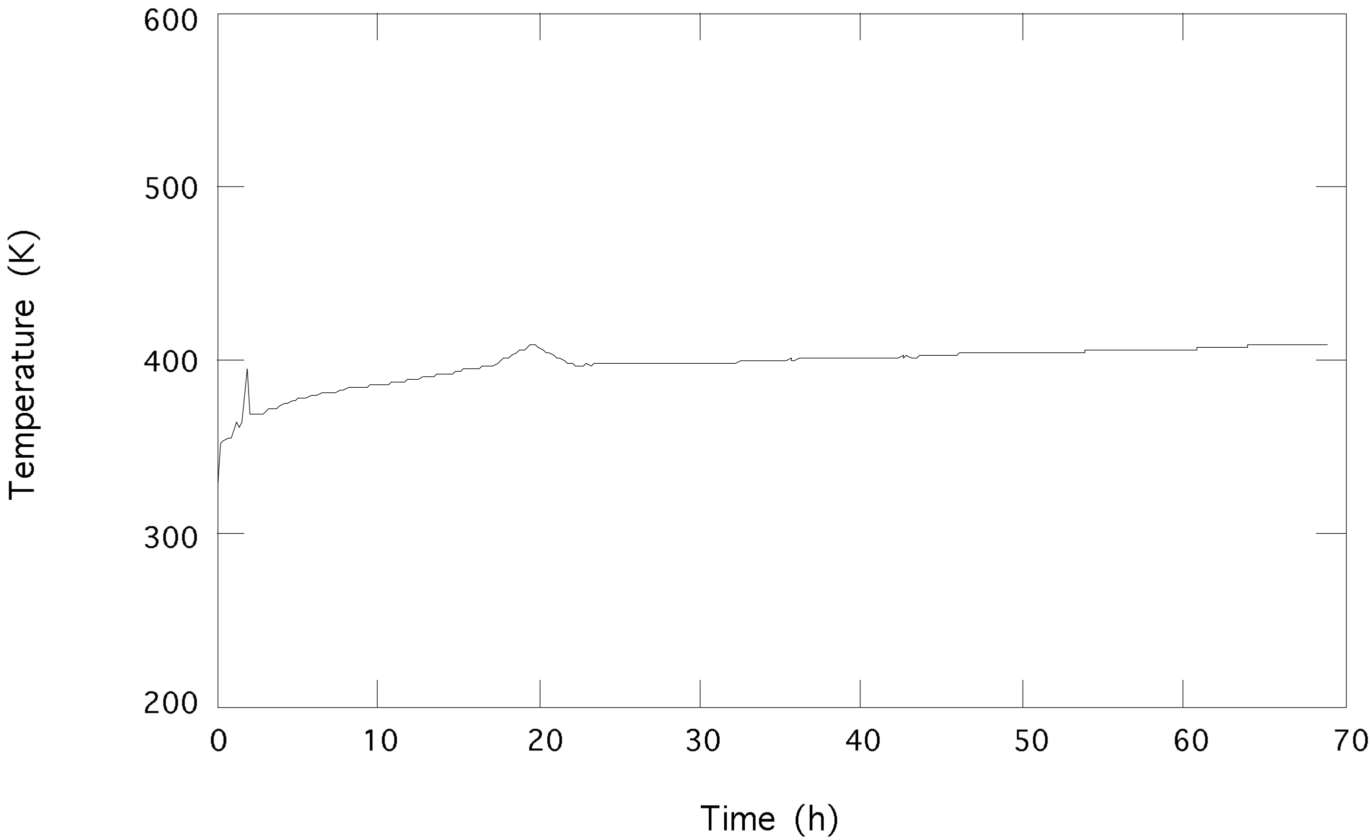


Figure 19E.2-27d Drywell Temperature for In-Vessel Core Melt Scenario

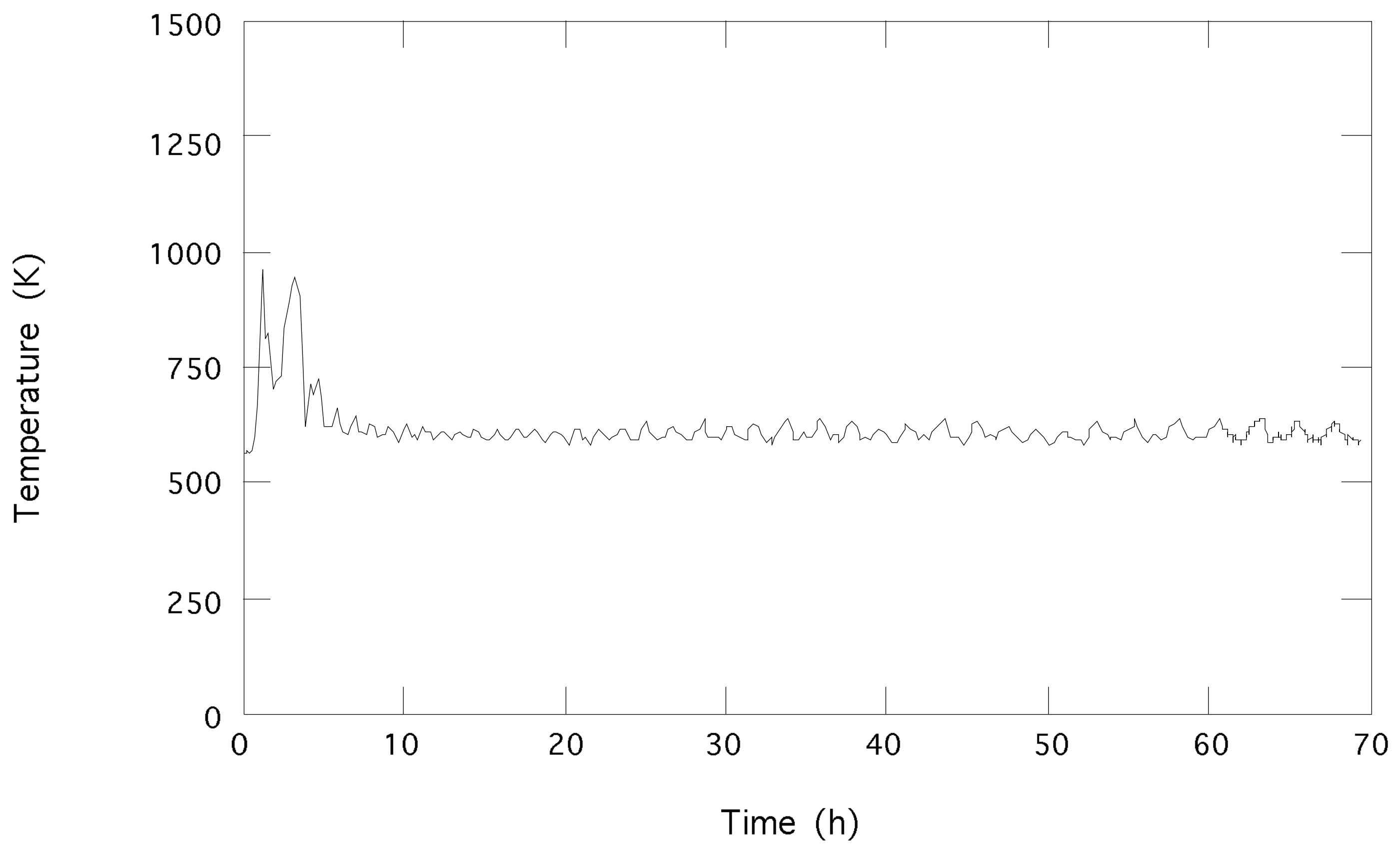


Figure 19E.2-27e Vessel Temperature for In-Vessel Core Melt Scenario

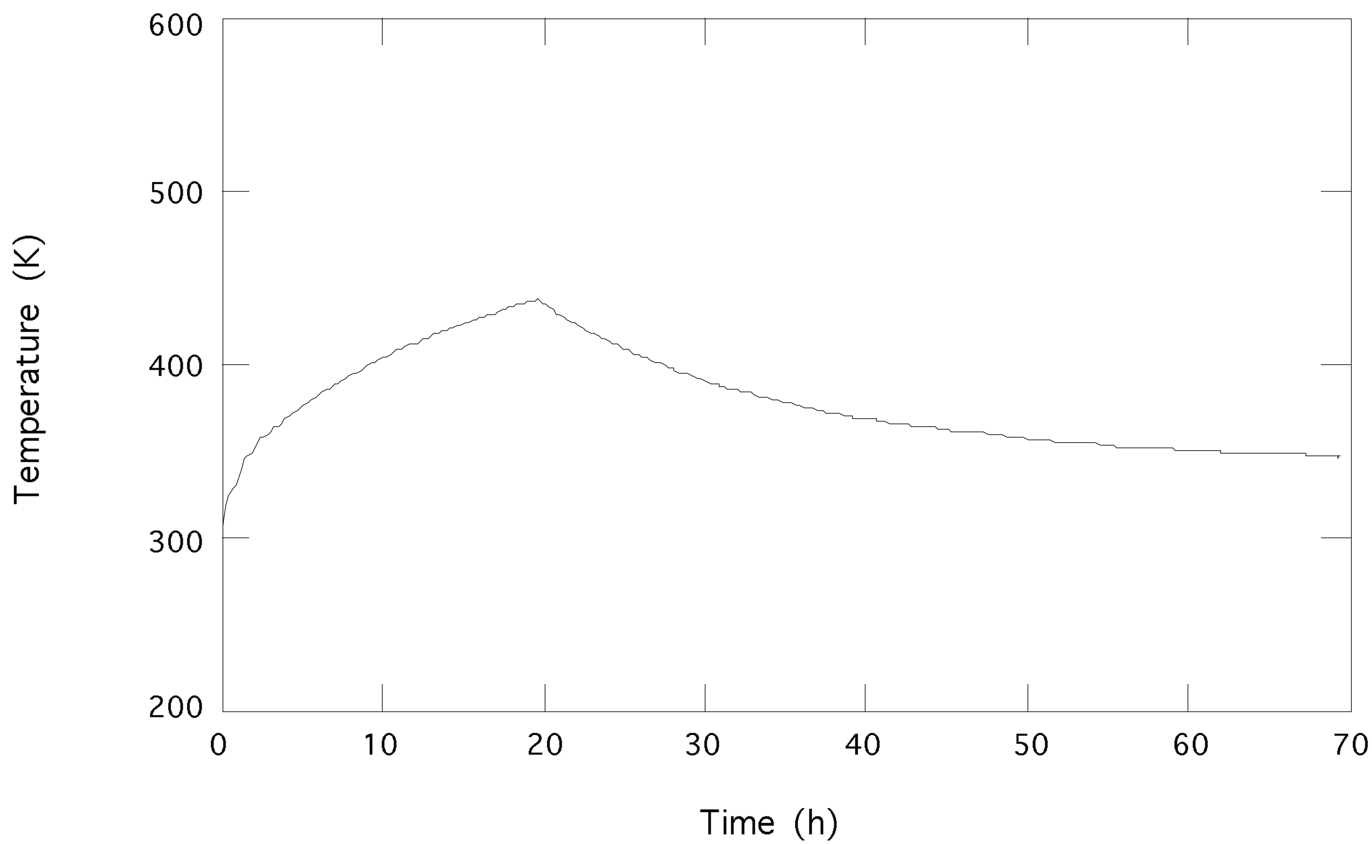


Figure 19E.2-27f Suppression Pool Water Temperature for In-Vessel Core Melt

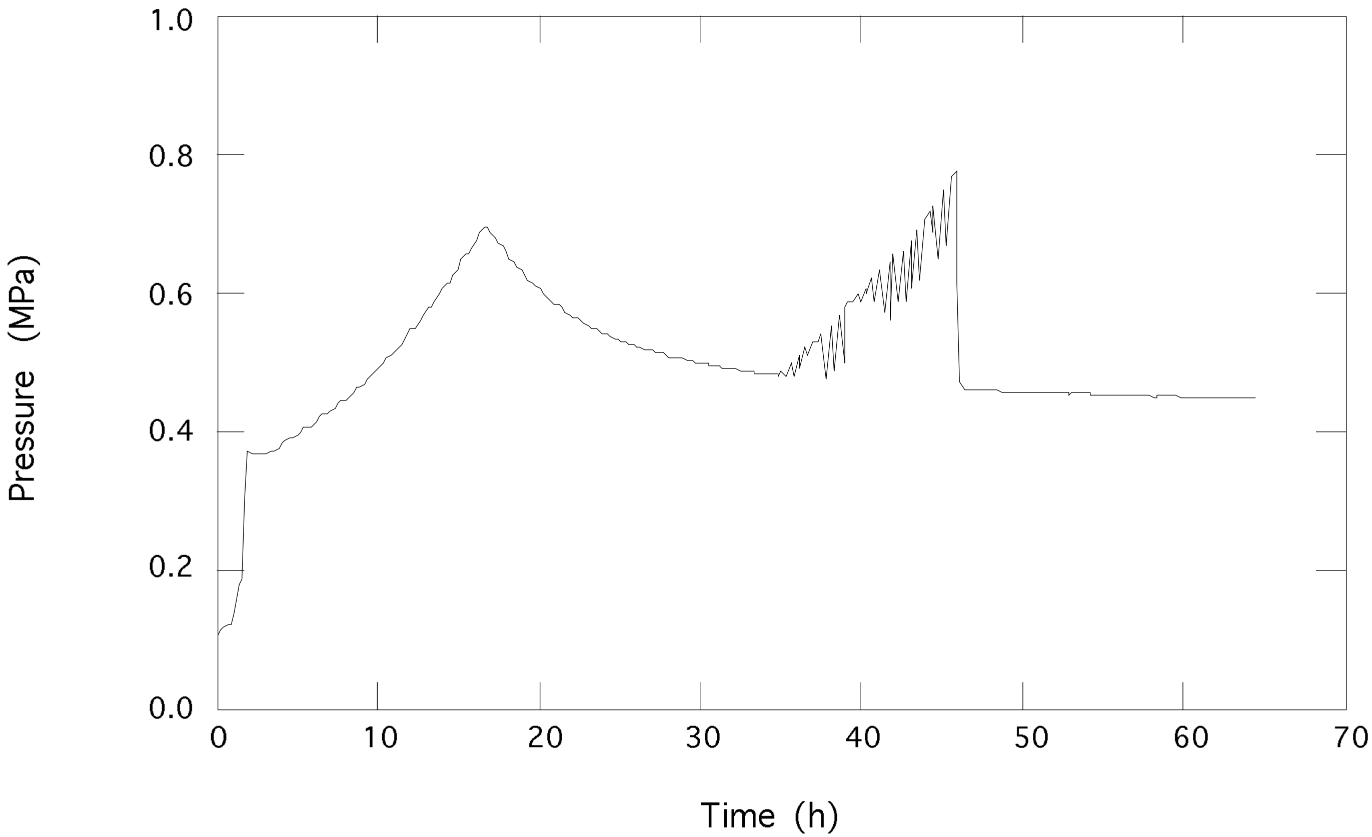


Figure 19E.2-28a Drywell Pressure for High Pressure Ex-Vessel Core Melt Scenario

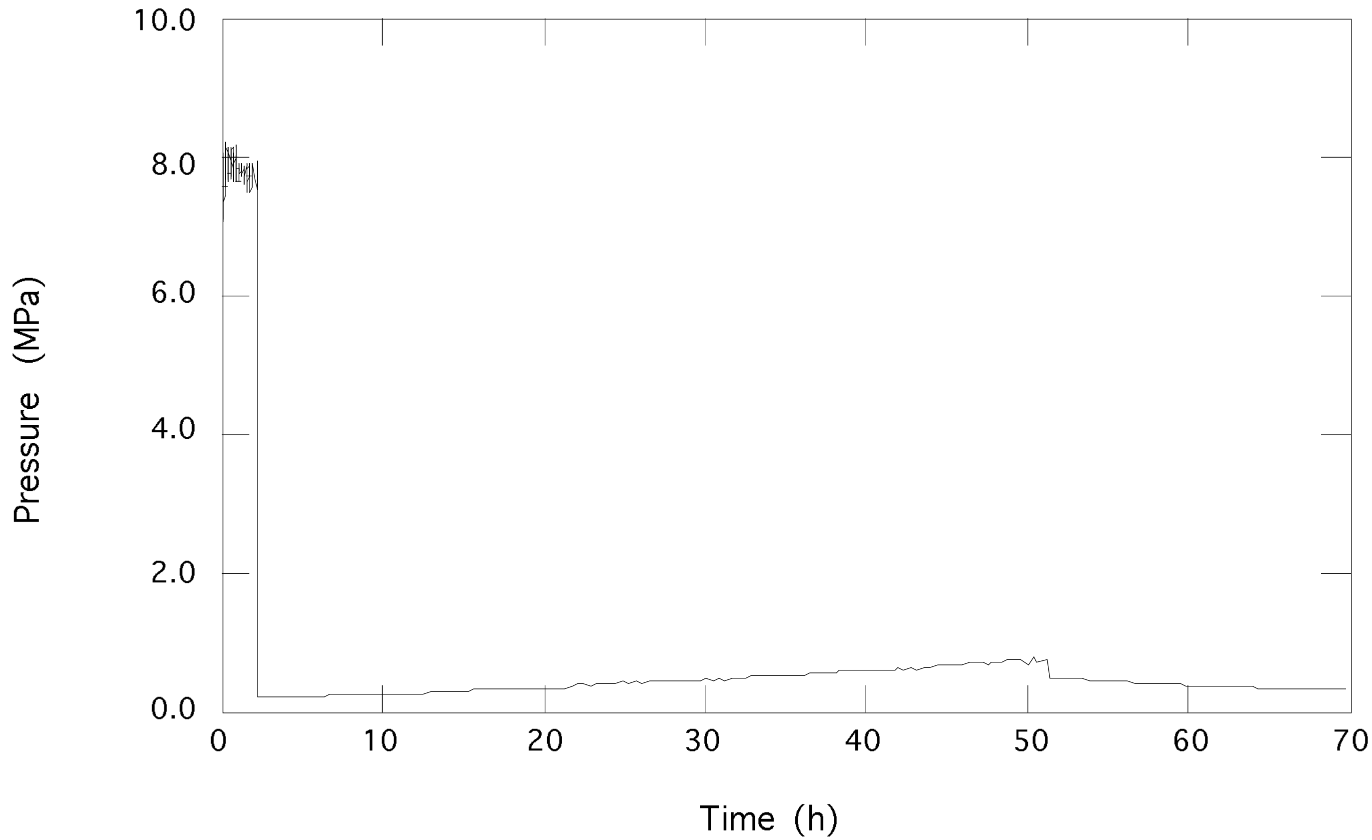


Figure 19E.2-28b Vessel Pressure for Ex-Vessel High Pressure Core Melt Scenario

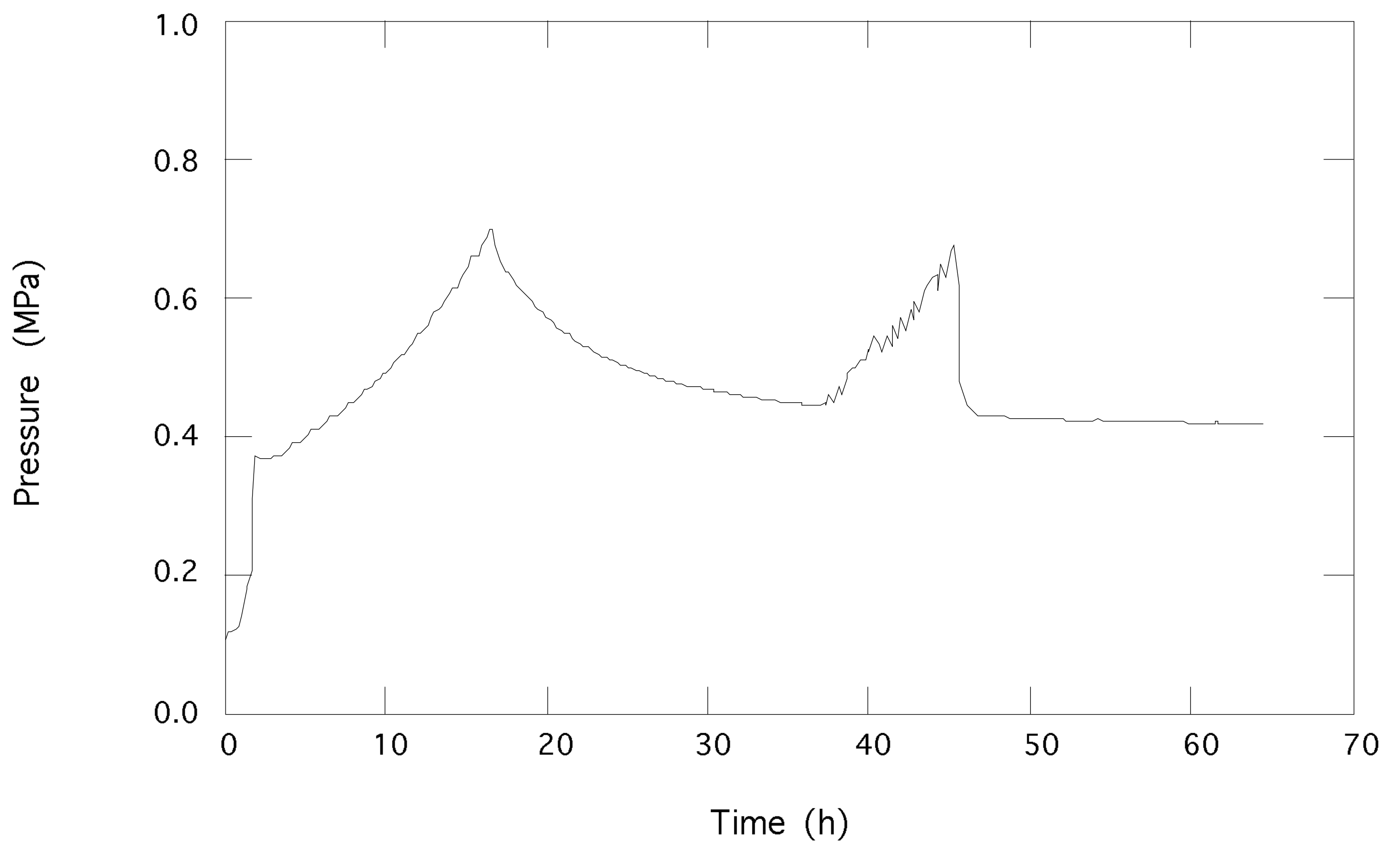


Figure 19E.2-28c Wetwell Pressure for Ex-Vessel High Pressure Core Melt Scenario

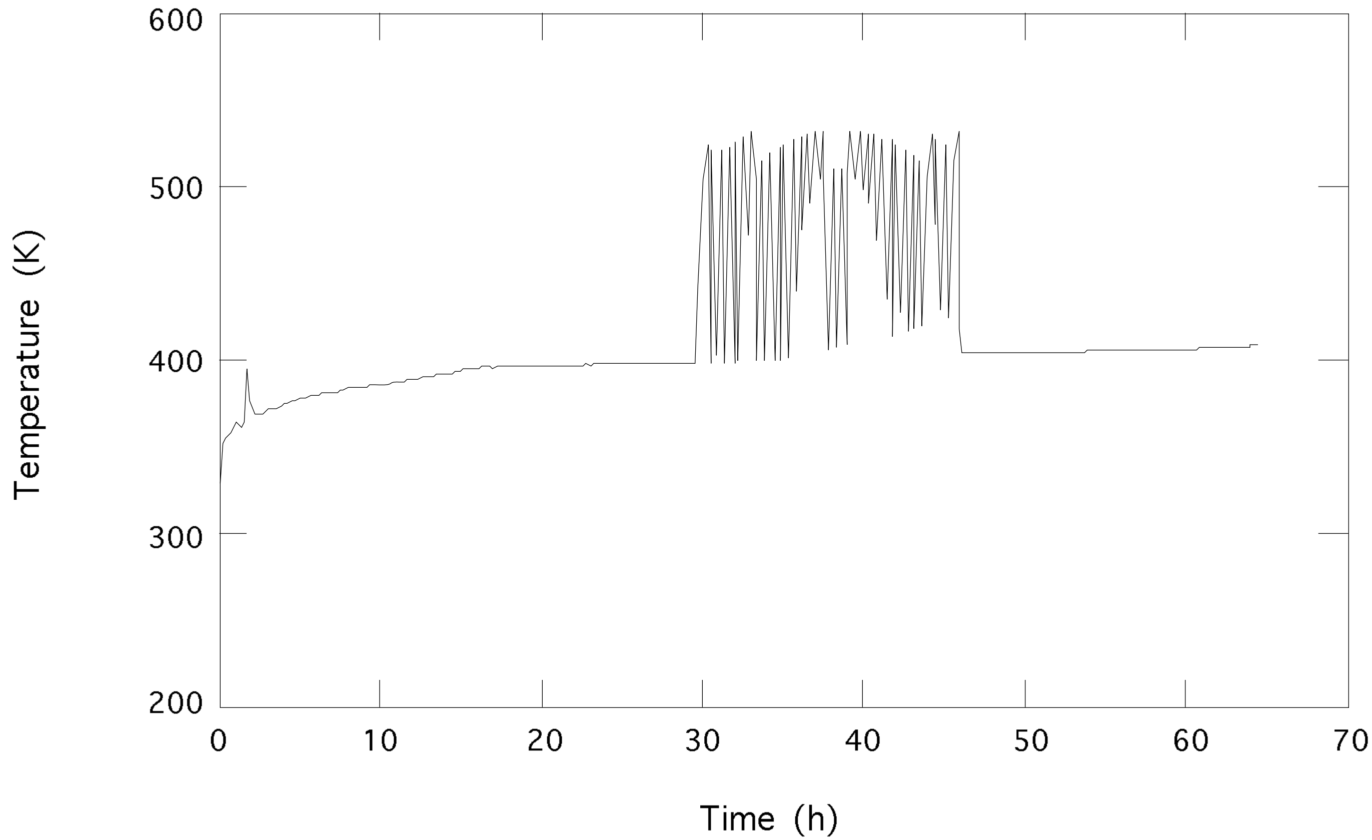


Figure 19E.2-28d Drywell Temperature for Ex-Vessel High Pressure Core Melt Scenario

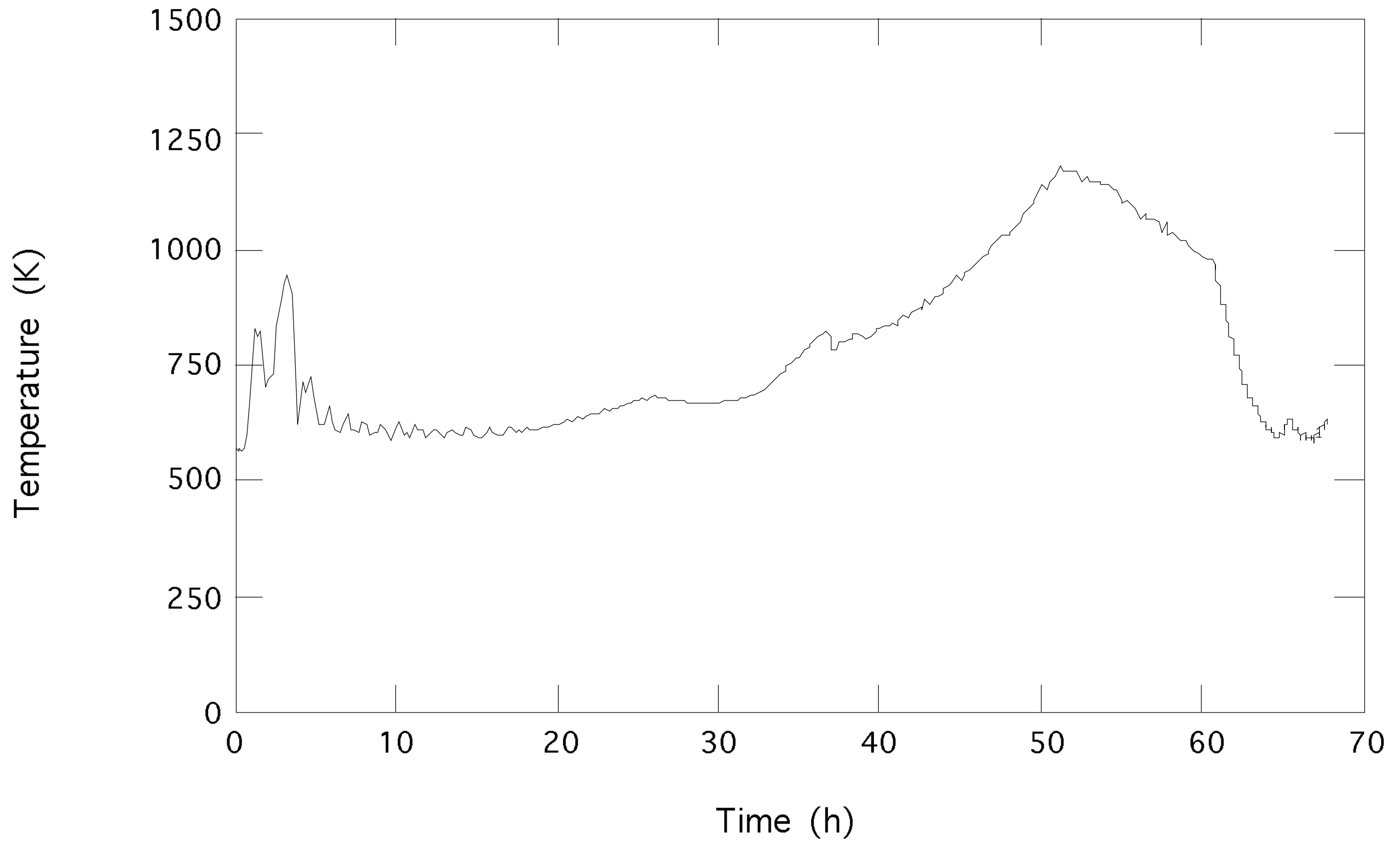


Figure 19E.2-28e Vessel Temperature for Ex-Vessel High Pressure Core Melt

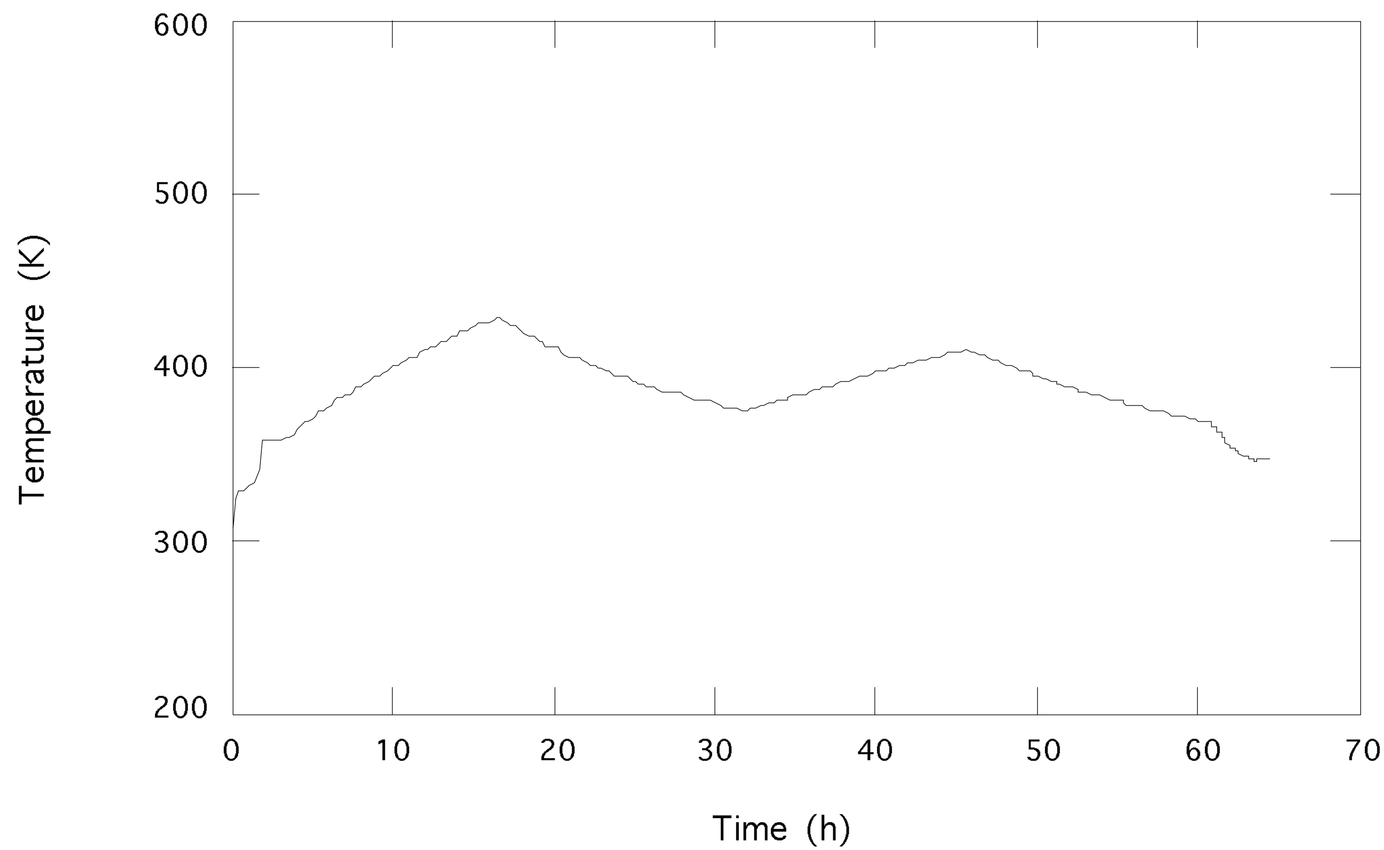


Figure 19E.2-28f Suppression Pool Water Temperature for High Pressure Ex-Vessel Core Melt Scenario

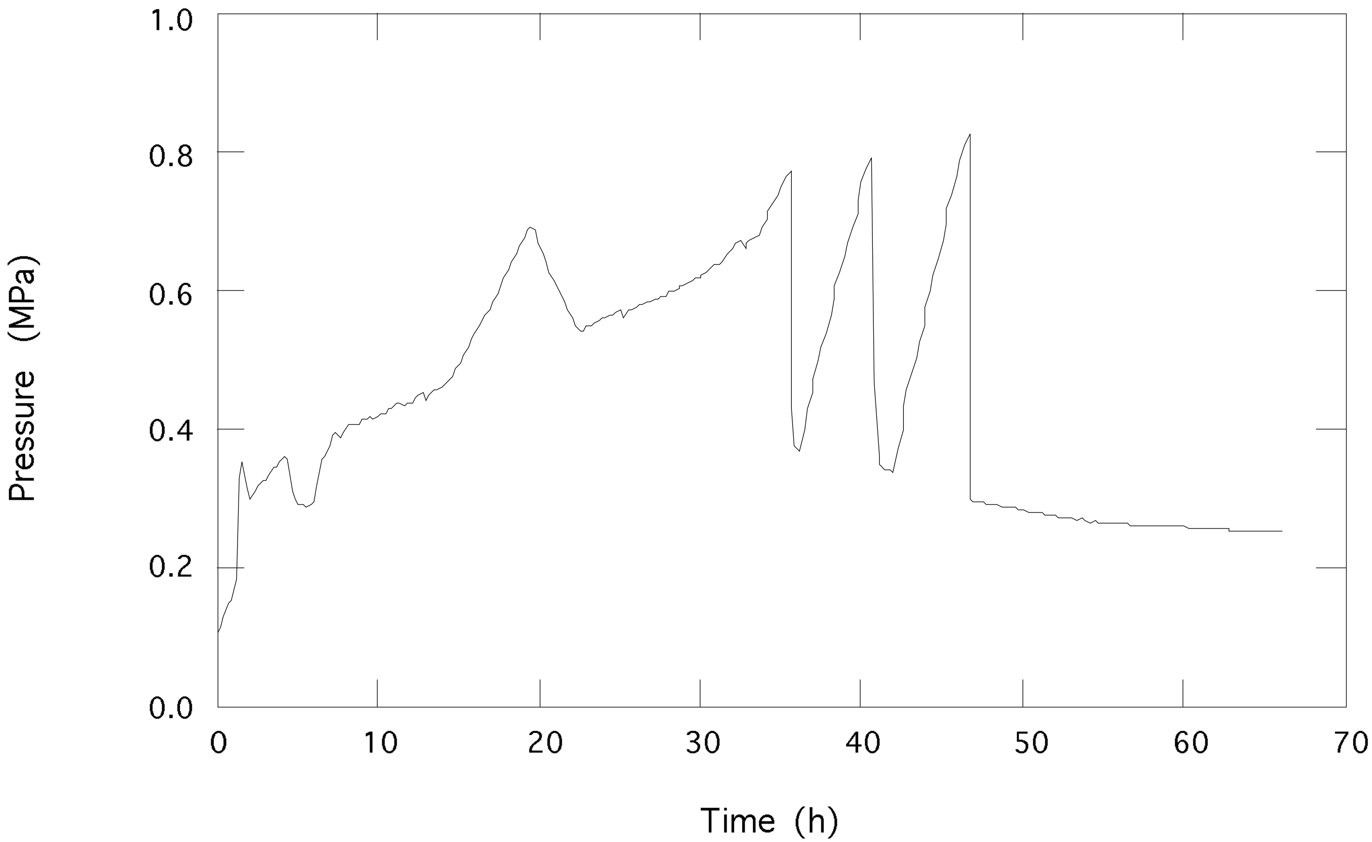


Figure 19E.2-29a Drywell Pressure for Low Pressure Ex-Vessel Core Melt Scenario

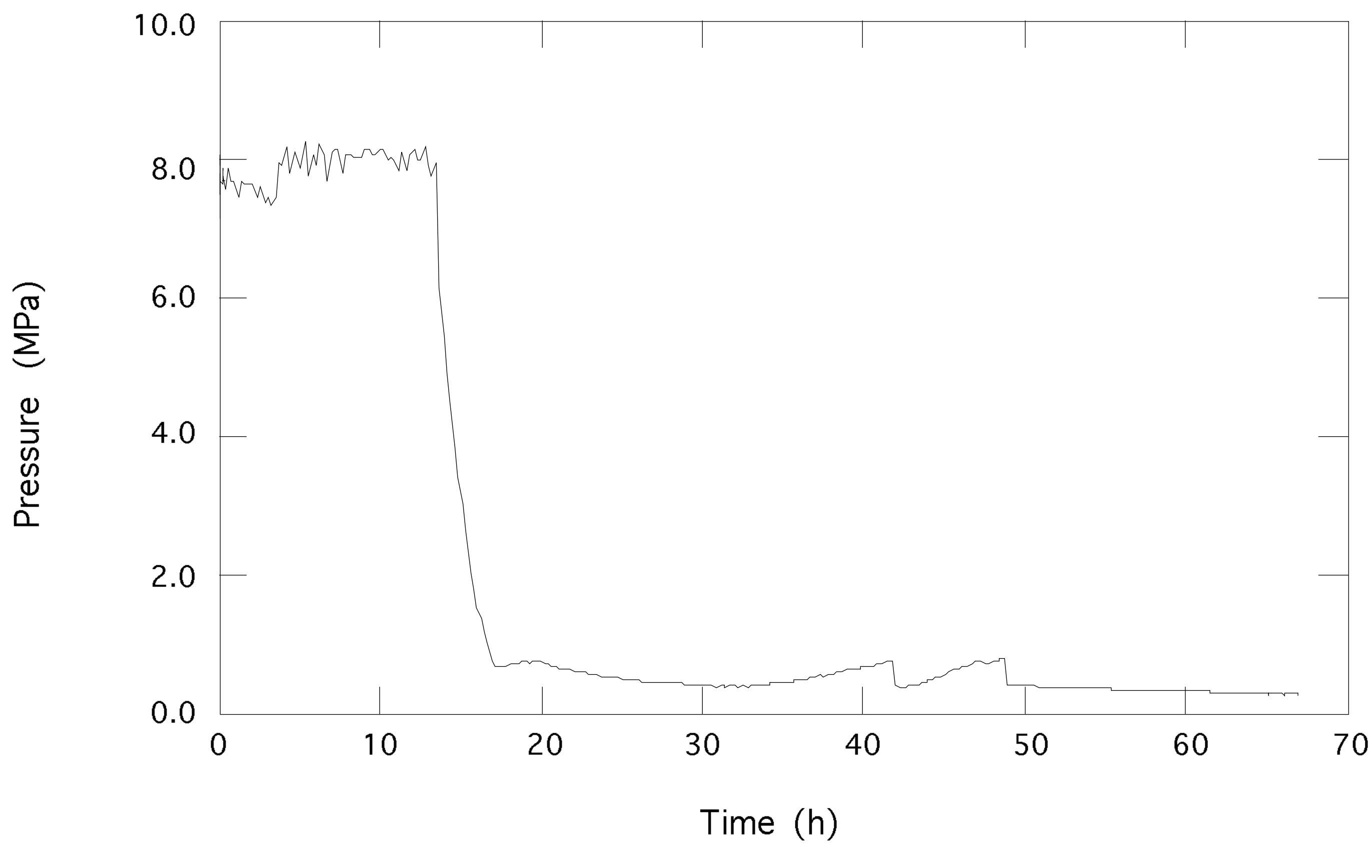


Figure 19E.2-29b Vessel Pressure for Ex-Vessel Low Pressure Core Melt Scenario

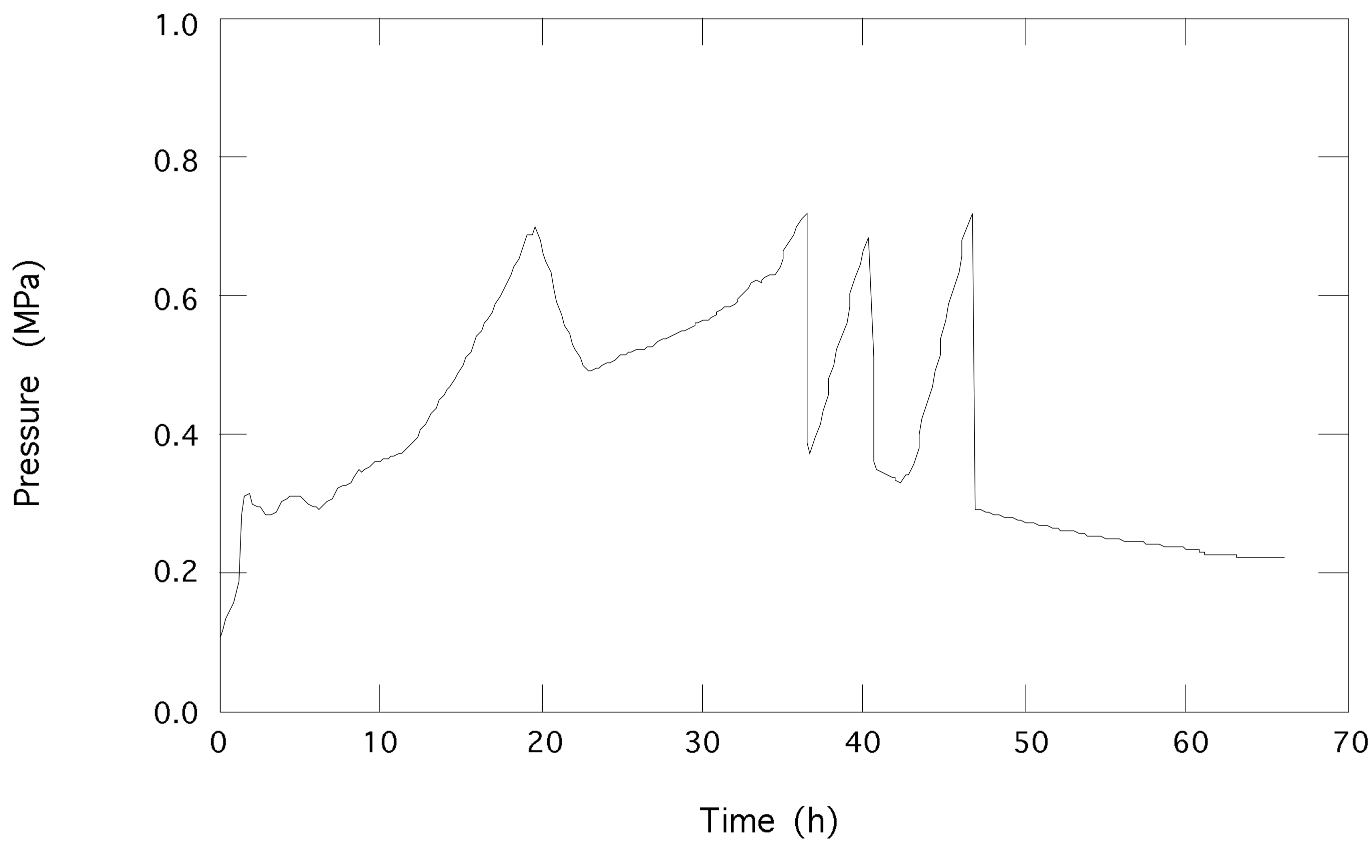


Figure 19E.2-29c Wetwell Pressure for Ex-Vessel Low Pressure Core Melt Scenario

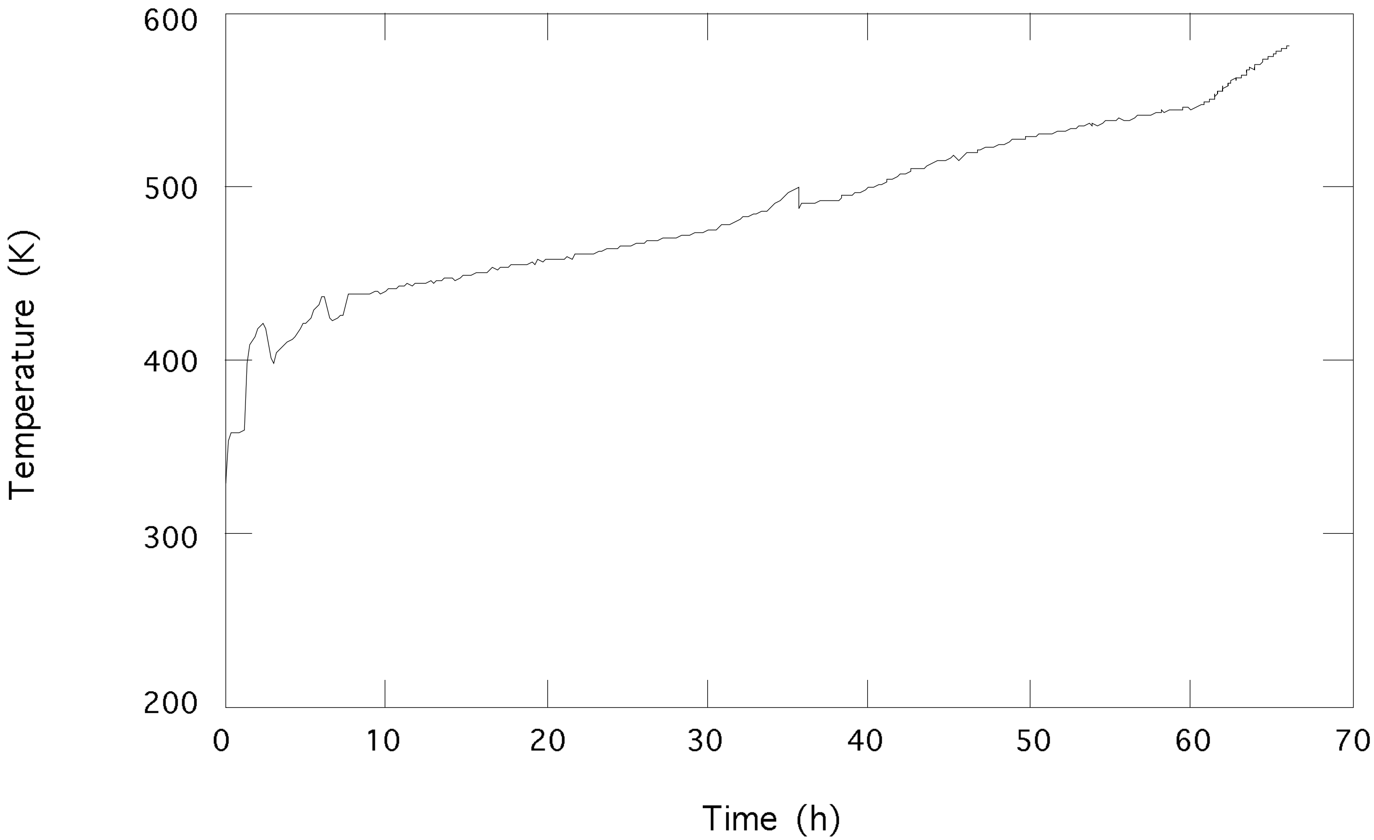


Figure 19E.2-29d Drywell Temperature for Ex-Vessel Low Pressure Core Melt

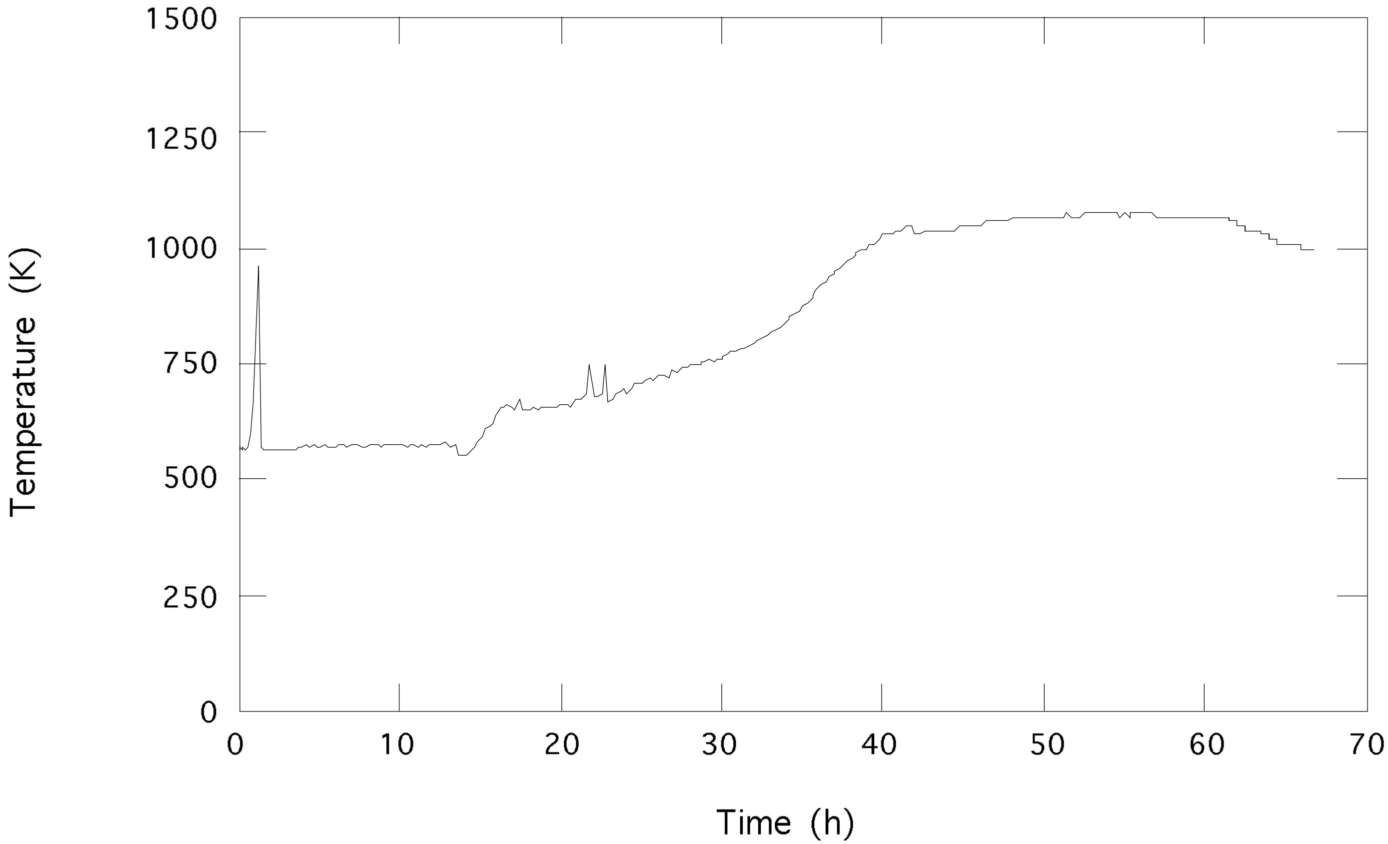


Figure 19E.2-29e Vessel Temperature for Ex-Vessel Low Pressure Core Melt Scenario

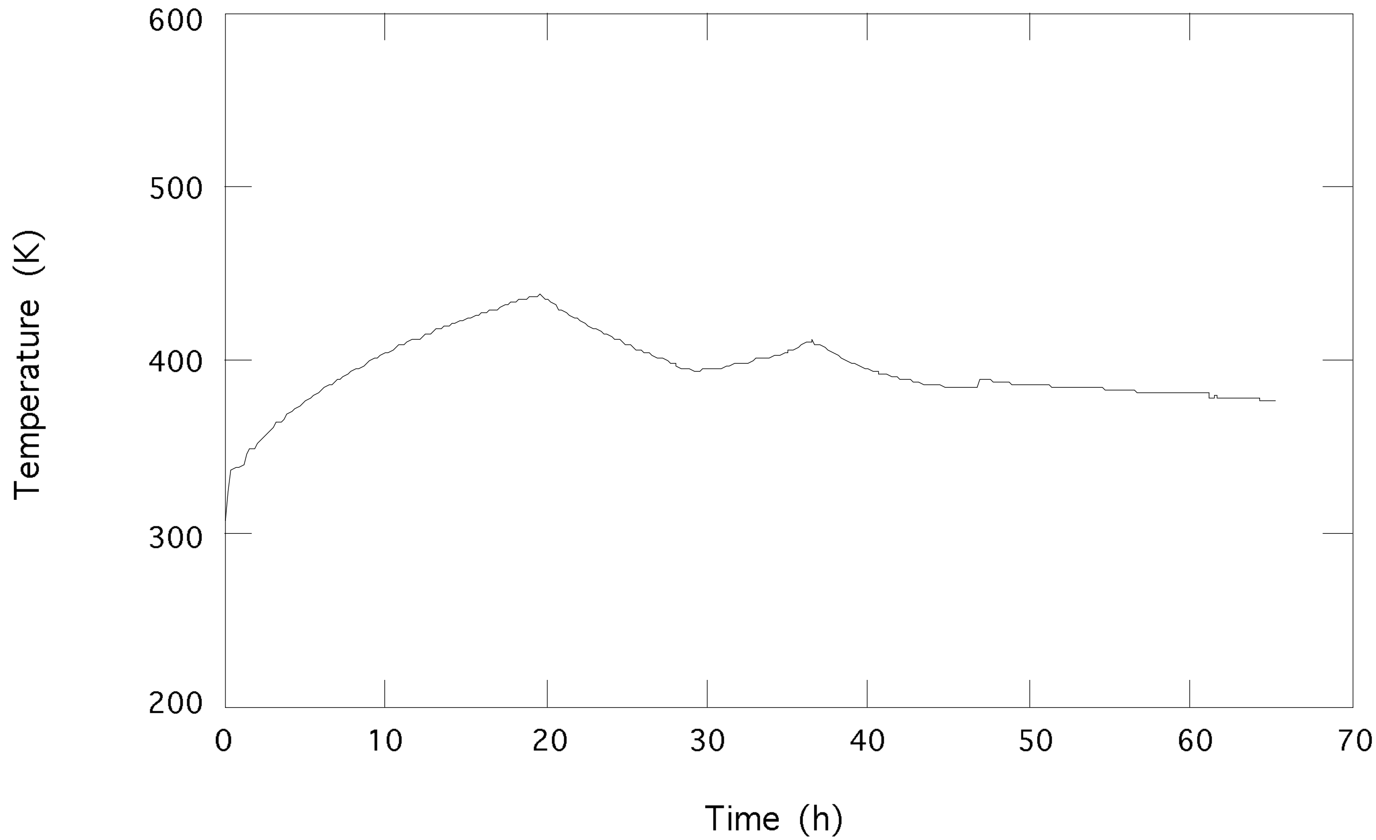


Figure 19E.2-29f Suppression Pool Water Temperature for Low Pressure Ex-Vessel Core Melt Scenario

19E.3 Consequence Analysis

This subsection describes the consequence evaluation. Key inputs and assumptions are described. The calculated results are compared to consequence-related goals to show that the goals are satisfied.

The CRAC-2 computer code (Reference 19E.3-1) was used to determine the consequences of potential reactor accidents. The CRAC code evaluates offsite dose and consequences for each accident category over a range of possible weather conditions and evacuation assumptions. The CRAC code models are described in Reference 19E.3-2. The rationale for site related input selection is presented in Subsection 19E.3.1. This data and data from the plant performance analysis is presented in Subsection 19E.3.2. The calculated results are compared to the goals in Subsection 19E.3.3.

19E.3.1 Site Assumptions

The evaluation of the consequences of a reactor accident are closely tied to the site parameters (e.g., weather, population, and land use). Envelope site parameters for deterministic evaluations are provided in Chapter 2. For probabilistic consequence evaluations, additional site related assumptions were required. They are described below.

19E.3.1.1 Meteorology

In the original WASH-1400 analysis (Reference 19E.3-3), a number of actual site meteorologies were used. However, the original WASH-1400 meteorology data files are not compatible with the CRAC-2 code. A set of meteorological data files suitable for use with the CRAC-2 code was obtained from Sandia National Laboratory. This data was used in the study given in Reference 19E.3-4. These files define hourly weather data for a one year period for twenty-six U.S. Sites. Five sites representing five geographical regions throughout the U.S. were chosen for this ABWR study. These regions were termed NE(northeast), NW (northwest), S (south), W (west), and SW (southwest) as is shown in Figure 3-1 of Reference 19E.3-4.

For each of these geographical regions, one meteorological data file was chosen. The basis for this choice was an evaluation for each meteorology using reactor release parameters for five accidents representing a very large percentage of the risk calculated in the GESSAR II PRA (Reference 19E.3-5). This accident data set is given in Table 19E.3-1. It was chosen since the GESSAR II design is closer to the ABWR design in terms of offsite releases than other designs for which PRA's were available. In determining the variations in consequence due to different meteorological data sets, each data file was input to the CRAC-2 code with all other information being identical. From these results, the site in each geographical region most closely approximating the mean total latent fatality result for that region was chosen to represent the region. The consequence results reported here (Subsection 19E.3.3) represent the average of five runs, one for each meteorological region.

19E.3.1.2 Population

For the ABWR consequence evaluation, the population density tables from Reference 19E.3-4, Tables 3-2 and 3-3, were used to develop regional populations corresponding to each regional meteorology. The mean values used are given in Table 19E.3-2.

19E.3.1.3 Evacuation

Many evacuation related characteristics (local roads, population demographics, emergency services) are quite site specific. No general guidance has been given for generic evacuation evaluations by the NRC. The evacuation parameters used in this study are given in Table 19E.3-3. Five percent of the people are assumed not to evacuate. Ninety-five percent are assumed to wait 1.5 hours after notification and then move radially outward at 4.47 meters per second (10 mph). Values used for shielding were the standard CRAC assumptions. Definitions for the parameters given in Table 19E.3-3 are provided in Table 19E.3-4.

These evacuation assumptions were used for individual and societal risk calculations. For the purposes of evaluating dose levels for comparison to the dose goal (Subsection 19E.3.3.1 item 3), no evacuation or shielding was assumed.

19E.3.2 CRAC Input Data

19E.3.2.1 Input Which Differs From Standard CRAC Assumptions

The following table describes these inputs.

Table	Inputs
19E.3-2	Population Density
19E.3-3	Evacuation Parameters
19E.3-5	Site and Reactor Data for Meteorological Modeling
19E.3-6	Event Release Parameters

19E.3.2.2 Input to CRAC from Performance Analysis

The plant performance analysis results which are input parameters to the CRAC-2 code are described here and are shown in Table 19E.3-6. These inputs describe the data used which are plant specific and are not related to radiological modeling which is discussed in Subsection 19E.3.1. The plant input parameters are described below with the subsection of Tier 2 in which the parameters are developed indicated at the end of each section in parenthesis.

For each accident case, which represents the accident sequence listed below it, the following data are used (Table 19E.3-6):

Release Category Name, LNAME(j) - Abbreviated name given to release which results from the event. (Subsection 19E.2)

Release Probability, P(j) - the probability per year associated with release LNAME(j). (Subsection 19D.4)

TL(j) - time(hr) from reactor shutdown (defined as the end of neutron generation) to release to the atmosphere. The value is used to determine isotopic decay prior to release from the plant. For an ATWS event, containment failure is postulated to occur before core damage. Since neutron production may continue up to the time of core melt, TL may be zero for an ATWS event. (Subsection 19E.2)

DR(j) - duration of initial release (h) of radionuclides from the plant. This value is used to determine the expansion of the cloud. The maximum value of this parameter is 10 hours (CRAC limitation for plume modeling). (Subsection 19E.2)

TLL(j) - warning time (h) between official notification of public and release of radioactivity from the plant. The basis for the warning time is the onset of severe core damage. The emergency action levels specified in Reference 19E.3-6, Appendix I require that a site area emergency be delayed when “delayed core with possible loss of coolable geometry” occurs.

FPR(j) - Sensible heat release rate in calories/s in the release cloud. This value is used to determine the initial buoyancy of the released cloud plume.

RH(j) - Plume release height in meters from the ground. If this value is less than the building height, a ground release with building wake effect is assumed. Otherwise, the plume will be buoyed to a height equal to the release height plus a buoyancy height. (Subsection 19D.5)

FLEAK(j,k) - fraction of core inventory at the beginning of the accident for each isotope group which is eventually released into the atmosphere. The standard isotopes groups are:

(1) Noble gases (Kr, Xe)

(2) Not used, originally used for organic iodide

(3) Iodine, including organic iodide

(4) Cesium, including Rb

(5) Tellurium, including Sb

(6) Barium, including Sr

(7) Cobalt, including Mo, Tc, Ru, Rh

(8) Lanthanum, including Y, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm

19E.3.3 Comparison of Results to Goals

19E.3.3.1 Goals

Three major consequence-related goals were established in the GE ABWR Licensing Review Bases (Reference 19E.3-7) which referenced the Safety Goal Policy Statement. These goals are:

(1) Individual Risk Goal

The risk to an average individual in the “vicinity” of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of “prompt fatality risks” resulting from other accidents to which members of the U.S. Population are generally exposed. As noted in the Safety Goals Policy statement, “vicinity” is defined as the area within 1.61 km (1 mile) of the plant site boundary. “Prompt Fatality Risks” are defined as those risks to which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities. Such risks are the sum of risks which result in fatalities from such activities as driving, household chores, occupational activities, etc. For this evaluation, the sum of prompt fatality risks was taken as the U.S. accidental death risk value of 39.1 deaths per 100,000 people per year based upon Reference 19E.3-8.

(2) Societal Risk Goal

The risk to the population in the area “near” a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of the “cancer fatality risks” resulting from all other causes. As noted in the Safety Goal Policy Statement, “near” is defined as within 16.1 km (10 miles) of the plant. The “cancer fatality risk” was taken as 169 deaths per 100,000 people per year based upon 1983 statistics in Reference 19E.3-9.

(3) Radiation Dose Goal

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

The calculated results are compared to these goals in the following subsections.

19E.3.3.2 Results

The results from the internal events analysis and the seismic event analysis (the average of the individual results over all five meteorological regions evaluated) are shown in Table 19E.3-7. A plot of whole body dose at a distance of 805 m (one-half mile) against cumulative probability is shown in Figure 19E.3-1. Based upon these results, the ABWR meets the established consequence related goals.

19E.3.4 References

19E.3-1 Ritchie, L.T., et al, “Calculation of Reactor Accident Consequences Version 2 CRAC2: Computer Code”, NUREG/CR-2326, February 1983.

19E.3-2 Ritchie, L.T., et al, “CRAC2 Model Description”, NUREG/CR-2552, March 1984.

19E.3-3 “Reactor Safety Study, Appendix 6: Calculation of Reactor Accident Consequences”, WASH-1400 (NUREG 75/014), October 1975.

19E.3-4 Aldrich, D.C., et al, “Technical Guidance for Siting Criteria Development”, NUREG/CR-2239, December 1982.

19E.3-5 “General Electric Company GESSAR II BWR/6 Nuclear Island Design (22A7007)”, March 1982.

19E.3-6 “Criteria for preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants”, NUREG-0654.

19E.3-7 Murley, T.E., “Advanced Boiling Water Reactor Licensing Review Bases”, Project No. 671, August 7, 1987.

19E.3-8 “Accident Facts”, 1988, National Safety Council.

19E.3-9 “1986 Cancer Facts & Figures”, American Cancer Society, 90 Park Ave, New York, NY 10016.

Table 19E.3-1 GESSAR Reactor Release Parameters

Category	$P_{(j)}$*	$TL_{(j)}$	$DR_{(j)}$	$TLL_{(j)}$	$FPR_{(j)}$	$RH_{(j)}$	Isotopic Release Fractions by Group						
Group =							1	3	4	5	6	7	8
C1-TR-E2		1.66	0.1	0.7	4.0E+07	10.0	1.0E+0	1.3E-03	1.0E-03	1.0E-03	1.1E-03	2.6E-04	1.5E-07
C1-TR-E3		1.7	4.3	0.7	1.5E+06	10.0	1.0E+0	1.3E-03	1.0E-03	1.0E-03	2.2E-04	3.1E-04	4.9E-05
C1-TR-L3		11.9	10.0	10.9	5.0E+05	49.0	1.0E+0	4.8E-04	1.8E-04	1.8E-04	3.9E-05	5.7E-05	9.0E-06
C1-TR-12		3.0	0.1	2.0	4.4E+07	10.0	1.0E+0	1.3E-03	9.9E-04	9.9E-04	1.0E-03	2.5E-04	1.5E-07
C1-TR-13		3.0	3.6	2.0	2.1E+06	10.0	1.0E+0	1.3E-03	1.0E-03	1.0E-03	2.2E-04	3.1E-04	4.8E-05

* Probabilites not part of DCD (Refer to SSAR).

Note:

See Subsection 19E.3.2.2 for definition of parameters in this table.

Table 19E.3-2 Population Density for Each Geographical Region

Radial Interval (mi)	Mean Population by Geographic Sector (people per sq. mi.) NE	MW	S	W	SW
0-5	100	60	30	20	10
5-10	130	60	80	30	20
10-20	170	90	70	60	30
20-30	180	120	100	50	40
30-50	400	100	80	40	130

Note:
Data taken from Reference 19E.3-4, Table 3-2.

Table 19E.3-3 Evacuation Parameters

Parameter	Strategy 1	2
Fraction of Population Evacuating	0.95	0.05
Time Delay Before Evacuation — h	1.5	0
Evacuation Speed — m/s (mph)	4.47 (10)	0
Maximum Distance of Evacuation — m (mi)	4827 (3)	0
Distance Moved by Evacuees — m (mi)	1260 (7)	0
Sheltering Radius — m (mi)	24140 (15)	0

Note:
See Subsection 19E.3.1.3 for additional description of parameters in this table.

Table 19E.3-4 Evacuation Parameter Definition

Parameter	Definition
Fraction of Population Evacuating	Fraction of population following the evacuation strategy.
Time Delay Before Evacuating	Time between notice to evacuate and start of evacuation.
Evacuation Speed	Once evacuation begins, it is assumed that the public moves directly outward and away from the plant site at this speed.
Maximum Distance of Evacuation	Once evacuation begins, individuals within this distance are assumed to evacuate as above with their exposure determined by detailed tracking of their position relative to the radioactive cloud plume. People living beyond this distance are assumed to not be evacuated initially. They are assumed to be exposed to ground contamination for 24 hours and then evacuated.
Distance moved by Evacuees before Sheltering	Distance at which evacuees are assumed to take shelter. This parameter is nominally designed to represent the use of prearranged evacuation shelters.
Sheltering Radius	People living within this distance are assumed to take shelter if they do not evacuate. Sheltering is assumed for 24 hours at which time these people are assumed to be relocated out of the contaminated area, without further exposure.

Table 19E.3-5 Site and Reactor Data for Meteorological Modeling

Reactor Building Length	54.0 m	177 ft.
Reactor Building Height	37.7 m	124 ft.
Interval for Special Wake Effects	2.41 km	1.5 mi.

Table 19E.3-6 Event Release Parameters

							Release Fractions†		
Accident	**P(i)***	**TL**	**DR**	**TLL**	**FPR**	**RH**	**NG**	**Iodine**	**Cesium**
NCL		2.7	10	1.7	3.3E+5	37	0.044	2.3E-05	2.3E-05
CASE 1		20	1	19.2	3.3E+5	37	1	1.5E-07	1.3E-05
LCHPFSRN									
LCHPPSRN									
LBLCFSRN									
SBRCPFRN									
LCLPPFRN									
LCPFSRN									
CASE 2		19	1	18.2	3.3E+5	37	1	5.0E-06	5.0E-06
LCLPPFCR									
LCLPFSCR									
CASE 3		50	10	49.2	3.3E+5	37	1	2.8E-04	2.2E-03
LCHPFSD90									
CASE 4		20	1	19.2	3.3E+5	37	1	1.6E-03	1.6E-03
DF100FSR									
DF100PFR									
CASE 5		19	1	19.2	3.3E+5	37	1	6.0E-03	5.3E-04
LBLCPFRN									
CASE 6		19	10	18.2	3.3E+5	37	1	3.1E-02	7.7E-02
LCHPPSD90									
LBLCPFD90									
LBLCFSD90									
CASE 7		20	10	19.2	3.3E+5	37	1	8.9E-02	9.9E-02
LCLPFSD90									
LCHPPFPM									
LCLPPFD90									
CASE 8		2	10	1.2	1.0E+6	37	1	1.9E-01	2.5E-01
LCHPPFEH									
LCHPPFBR									
LCHPPFBD									
CASE 9		23.6	10	12.2	3.3E+5	37	1	3.7E-01	3.6E-01
SBRCPFD90									

* Probabilites not part of DCD (Refer to SSAR).
† Group 5-8 negligible release

Note:

See Subsection 19E.3.2.2 for definition of parameters in this table.

Table 19E.3-7 Consequence Goals and Results

Goal	Numerical Goal	ABWR*
Individual Risk	$<3.9 \times 10^{-7}$ (0.1%)	
Societal Risk	$<1.7 \times 10^{-6}$ (0.1%)	
Radiation Dose Probability at 0.25 Sv	$<10^{-6}$	

* Probabilities not part of DCD (Refer to SSAR).

Figure 19E.3-1 Whole Body Dose at 805 m (0.5 Mile) as Probability of Exceedence

Not Part of DCD (Refer to SSAR)

19EA Direct Containment Heating

19EA.1 Summary Description

Direct Containment Heating (DCH) is the sudden heatup and pressurization of containment resulting from the fragmentation and dispersal of core material in the containment atmosphere. DCH is a concern for sequences in which the vessel fails at high pressure since the steam flow from the vessel provides the motive force for entrainment. In the event of a sufficiently large DCH event, the containment could fail at the time of vessel failure. Since this could lead to very high releases to the environment, a study has been carried out to investigate the uncertainties in the challenge to containment due to a DCH event. In the past, this issue has been primarily addressed for Pressurized Water Reactors (Reference 19EA-1) since BWRs have very reliable vessel depressurization systems. Thus, the frequency of accidents with the vessel remaining at high pressure is extremely low.

Subsection 19D.6.2.5 provides an evaluation of the ADS System reliability including the nitrogen, control and instrumentation systems. Additional information about the SRVs and the ADS System may be found in Subsections 5.2.2, 7.3.1.1.1.2 and 19E.2.1.2.2.2.

Subsection 7.3.1.1.1.2(3)(h) indicates that the signal cables, solenoid valves, safety/relief valve operators and accumulators are located inside the drywell and are designed to operate in the most severe accident resulting from a DBA LOCA, including the radiation effects. The conditions in the containment during the early stages of a severe accident (before vessel failure) which require depressurization using the SRVs are less challenging than those specified by a DBA LOCA. Additional analyses of the ADS System capability were performed in support of station blackout performance analysis. This discussion is included in Subsection 19E.2.1.2.2.2. The conclusions of that analysis are that there is ample DC power for the operation of the SRVs for many days after the 8 hour capability required by the station blackout rule.

Subsection 5.2.2.4.1 indicates that the nitrogen accumulator capacity for each valve is designed to be sufficient to open for one actuation at drywell design pressure even if the air supply to the accumulators is lost. The risk significant severe accidents in the ABWR PRA remain below the design pressure of the containment in the time period before vessel failure. Valve operability at high containment pressure conditions are also discussed in Subsection 19E.2.1.2.2.2(2)(b). Based on the presence of the containment overpressure protection system, the maximum drywell pressure is approximately 0.79 MPa (100 psig). Subsection 19E.2.1.2.2.2(2)(b) indicates the operator actions which could be taken to assure SRV operability under these conditions. The appropriate operator actions are specified in the ABWR EPGs. Since the containment pressurizes very slowly, over a period of about a day, there is ample time for the operators to take the appropriate actions.

Given the above discussions, one may conclude that the ADS System will not be compromised before vessel failure in the unlikely event of a severe accident, and the frequency of severe accident sequences in which the vessel fails at high pressure is extremely low. However, with the many sources of low-pressure injection available to the ABWR to prevent core damage, the frequency of all core damage sequences is very low. Therefore, high-pressure core melts appear as contributors to the total core damage frequency, albeit with a very low probability.

A detailed study utilizing event trees was performed to assess the peak drywell pressure resulting from a DCH event. This attachment outlines the analysis and the results.

19EA.2 Description of Event Tree Analysis

The early containment failure event tree analysis consists of a main tree (Figures 19EA-1, 19EA-2, and 19EA-3) and three supplemental decomposition event trees (DETs) (Figures 19EA-4, 19EA-5, and 19EA-6). The first two events on the main event tree sort the sequence classes by Reactor Pressure Vessel (RPV) pressure and pre-existing containment pressure at the time of vessel failure. These parameters are uniquely determined by the accident class attributes. The last event on the main event tree assesses the probability of drywell head failure following vessel failure. The probabilities for this event are evaluated in supplemental DETs. Three DETs were constructed to assist in the quantification for accident classes with high RPV pressure. (Low RPV pressure sequence classes are not expected to lead to containment pressures which would challenge the integrity of containment.) The three DETs assess the probability of containment failure for different pre-existing containment pressures at the time of RPV failure.

The DETs consider the major phenomena which contribute to early over pressurization of containment from high RPV pressure sequences including debris entrainment from the lower drywell, Direct Containment Heating (DCH), the pre-existing pressure in containment prior to RPV failure, and the pressure rise due to blowdown of the RPV. Each pathway through a DET represents a possible accident progression pathway given the uncertainties in the underlying phenomena. A peak containment pressure is associated with each pathway. These pressures have been estimated from a deterministic DCH model (described in Subsection 19EA.3) with input conditions which reproduce the parameter values and assumptions along each sequence pathway on the tree. These pressures were then compared with the containment fragility curve (developed in Attachment 19FA) to determine the probability of containment failure.

The probabilities for each sequence pathway with similar end states were summed and these results transferred as the branch probabilities for the last event on the main event tree.

The spectrum of pressures and associated probabilities represented by the quantified DETs represents a discrete probability distribution of containment pressurization following vessel failure. This distribution is a representation of the uncertainties associated with the estimation of containment pressurization due to the phenomena occurring at vessel failure.

19EA.2.1 Event Headings

The important parameters and assumptions which are considered as headings on the main event tree and the DETs and the reason for their use are discussed below.

19EA.2.1.1 Containment Pressure Prior to RPV Failure (CONTPRES)

The pre-existing pressure of the containment is obviously important in the assessment of containment pressurization following vessel failure. Three pressure regimes have been selected to represent the range of possible pre-RPV failure containment pressures. MAAP-ABWR calculations (described in Subsection 19E.2.2 and summarized on Table 19EA-1) indicate that ABWR accident sequences can be grouped into three classes. These pressure regimes are similar to those selected to represent pre-RPV failure containment pressures in NUREG/CR-4551 (Reference 19EA-2).

Class	Pressure Range	Examples
Low	0.134-0.264 MPa (Nominal = 1.5 atmosphere.)	Non ATWS sequences with operable DHR or with rapid core damage (i.e., all invessel injection failed).
Inter	0.264-0.310 MPa (Nominal = 2.5 atmosphere)	Large LOCAs with early failure of DHR. SBO with RCIC and failure of DHR.
High	>0.310 MPa (Nominal = 4 atmosphere)	ATWS with RCIC.

This event is quantified based on the sequence accident class. This is a sorting type event. The probability of each branch is either 0 or 1 depending upon the attributes of the accident class.

19EA.2.1.2 RPV Pressure at RPV Failure (RVPRES)

The RPV pressure at the time of vessel failure is a major parameter impacting a number of processes which contribute to containment pressurization at RPV failure. Blowdown of the reactor vessel following failure at elevated pressure contributes directly to containment pressurization. High RPV pressures promote entrainment of the debris from the lower drywell and debris fragmentation.

Subsection 19EA.3.1 describes the mechanism for entrainment and the potential for debris dispersal in the ABWR.

Two pressure regimes are considered:

High > 1.37 MPa

Low ≤ 1.37 MPa

For sequences with low RPV pressure at the time of vessel failure, the mechanisms which may lead to rapid containment pressurization are generally not operative. As discussed in Subsection 19EA.3.1, entrainment of the debris is an essential prerequisite for DCH. The entrainment of debris from the lower drywell occurs due to levitation by the steam expelled from the vessel after vessel failure. For sequences with the RPV at low pressure at the time of vessel failure, there is no driving force for the steam. Consequently, in the event tree for early containment failure due to DCH, the probability for early containment failure for low-pressure sequences is set to zero.

For high-pressure sequences, on the other hand, mechanisms such as DCH and RPV blowdown may challenge the integrity of the containment. The remaining events on the event tree assess those mechanisms which impact containment loading for high-pressure sequences.

This event is quantified based on the sequence accident class. This is a sorting type event. The probability of each branch is either 0 or 1 depending upon the attributes of the accident class.

19EA.2.1.3 Mode of RPV Failure (MODRVFAIL)

Following slumping of the molten core debris into the lower RPV head, thermal attack on the lower head and lower head penetrations will eventually result in bottom head failure (unless the debris is cooled in-vessel). Several modes of vessel failure have been considered to be possible ranging from a limited area failure of one or more instrument tubes, drains or control rod drives to creep-rupture failure of the lower head resulting in a large diameter failure.

This event is a split fraction, representing uncertainties in the phenomenology. Two size classes were defined for this study:

Small: Initial area equal to the area of one control rod drive penetration ($< 0.1\ m^2$)

Large: Nominal failure area of $2.0\ m^2$

For quantifying this event, the results from NUREG/CR-4551 were used as guidance. In this analysis, as in NUREG/CR-4551, it was assumed that all breach sizes greater than $2.0\ m^2$ could be treated identically. For all core damage sequences where core damage progression is not terminated in-vessel (and vessel failure is predicted) NUREG/CR-4551 indicated that the mean probability of small penetration failures was 0.75 and the probability of large lower-head failure modes was 0.25.

Analyses performed subsequent to NUREG/CR-4551 indicate that the probability of large creep-rupture lower-head failure modes may have been overestimated (References 19EA-3 and 19EA-4). Even though the best-estimate studies indicate a small penetration failure is expected, this analysis addresses hole sizes up to $2.0\ m^2$. The probability of a larger failure is judged to be quite low based on References 19EA-3 and 19EA-4. Therefore, the probability of large lower-head failure modes has been decreased in this analysis.

The probabilities are considered to be appropriate for both early and late core damage sequences. Vessel ablation is primarily controlled by the superheat in the core debris, and is less influenced by the time of core damage. Thus, the time of core damage will have little effect on the mode of vessel failure.

19EA.2.1.4 Fraction of Core Inventory Molten in Lower RPV Head (RVCORMASS)

This parameter largely defines the potential for large scale DCH events. It is generally considered that only the debris that is molten in the lower head at the time of vessel breach will have the potential for dispersal and fragmentation. Thus, only this material can significantly contribute to DCH.

Two regimes are considered:

Small 0–20% Core Debris Inventory (Nominal 10%)

Large 20–60% Core Debris Inventory (Nominal 40%)

These mass regimes are similar to those chosen in NUREG/CR-4551 to represent the Grand Gulf plant.

NUREG/CR-4551 provides probabilities for three cases which also appear to be applicable to the ABWR. The mean NUREG/CR-4551 probabilities are presented below:

(1) Case 1

For sequences with water injection into the reactor vessel prior to vessel breach by low-pressure or high-pressure injection systems:

P (Small) = 0.975

P (Large) = 0.025

(2) Cases 2 and 3

For high- and low-pressure sequences without in-vessel injection:

P (Small) = 0.9

P (Large) = 0.1

Since the majority of ABWR core damage sequences do not involve late water addition to the core, it is conservatively assumed that the Case 2/3 results apply to all ABWR core damage sequences.

It can be shown that the core debris discharge rates used in the ABWR DCH analysis bound results typical of a BWRSAR calculation (Reference 19EA-5). Table 19EA-2 compares the approximate debris masses participating in DCH at selected intervals after the vessel has failed. The ABWR DCH analysis column shows the values used for a *small* mass of molten debris in the lower head at RPV failure. It should be noted that debris entrainment will occur only until the vessel has depressurized to about 1.379 MPa (200 psia). The BWRSAR results indicate that the RPV depressurizes in about four minutes. The analysis of this study has a much larger vessel failure area due to ablation, thus, the depressurization is more rapid. The pressurization of the containment is most rapid before the wetwell connecting vents clear. Vent clearing will occur within the first second of the blowdown. Therefore, the very early stages of the debris pour and entrainment are the most significant.

The total mass of debris and the zirconium mass used in this analysis are much larger than the masses calculated by BWRSAR. Indeed, the mass of the zirconium bounds the entire zirconium and metal mass calculated by BWRSAR for the critical, early stages of the blowdown. Since the heat of reaction for the oxidation of zirconium is much higher than that of other metals, the use of a high zirconium mass bounds the effects of the other metals. Even at the four minute mark, the distribution of the masses is conservative due to the relative heats of reaction for zirconium and other metals. Thus, the table clearly shows that the assumed masses bound the BWRSAR results.

19EA.2.1.5 High-Pressure Melt Ejection (HPME)

For sequences with high RPV pressure at vessel failure, the core debris is likely to be expelled from the vessel at high velocity. Furthermore, the velocity of the residual gases blowing down from the reactor vessel are likely to be sufficiently high to result in significant entrainment of the debris from the lower drywell and to result in dispersal and fragmentation of the debris. This event is a split fraction indicating the uncertainty in phenomena. The question evaluates whether a substantial fraction of the core debris is expelled from the vessel at high velocity and followed by the blowdown of the vessel. Given these precursors, it is believed that material will be lifted from the lower drywell floor. A subsequent event heading (FRAG) will assess the extent of debris fragmentation, and dispersal into the upper drywell.

For all cases where reactor vessel failure occurred under high-pressure conditions, the probability of an HPME was assessed to be 0.8 for the Grand Gulf plant in the NUREG/CR-4550 study. Based on similarities in the design of the ABWR and BWR-6 vessel bottom heads, it is assumed that these results can also be applied to the ABWR. Additional discussion is provided in Subsection 19EA.3.1. The “No HPME” value of 0.2 represents the potential that the gas from the RPV will break through the core debris and the vessel will be depressurized prior to the release of the core debris and the potential that the initial vessel breach will be near the melt surface. BWRSAR results (Reference 19EA-5) indicate that the RPV depressurizes before any substantial amount of core material is expelled from the vessel.

For reasons similar to those discussed above for low-pressure sequences, if an HPME event does not occur, then the loads imposed on the containment structure at vessel failure will not result in a serious threat to containment integrity and the probability of early containment failure is assumed to be zero.

19EA.2.1.6 Fraction of Entrained Debris Fragmented and Transported to the Upper Drywell (FRAG)

This branch in the DETs is a split fraction event. For high-pressure sequences where an HPME has occurred, this event assesses the extent of dispersal and fragmentation of the entrained debris. In order for a serious overpressure challenge of the containment by direct containment heating (DCH) to occur, a significant fraction of the debris that was molten in the lower RPV head at vessel failure must be transported from the lower drywell into the upper drywell and fragmented. The mechanisms which may limit the transport of the molten debris from the lower cavity to the upper cavity are discussed below. These include:

(1) Trapping of the debris in the lower drywell.

(2) Impaction and removal of the debris in the gas transport pathway connecting the lower and upper drywell compartments.

(3) Partitioning of the gas (and entrained debris) flow exiting the lower drywell between the upper drywell and the wetwell.

(4) Debris dispersal by wave formation rather than by small particles.

The above mechanisms can impact the extent of debris dispersal to the upper drywell as small debris particles which is the critical parameter for determining the potential threat from DCH.

The basic configuration of the ABWR lower drywell is shown in Figure 19EA-7. Additional details can be seen in the arrangement elevation drawing (Figures 1.2-2 and 1.2-2a), the lower drywell elevation (Figures 1.2-3b and 1.2-3c) and the arrangement plan (Figures 1.2-13e through 1.2-13h). The vessel skirt of the ABWR is solid, there are no openings in it which could connect the upper drywell to the lower drywell. This precludes water transport from the upper drywell into the lower drywell following a LOCA. Hence, the flow path for gases and debris expelled from the lower drywell will be through the downcomers. The upper drywell to wetwell downcomers are imbedded in the lower drywell wall. The downcomers are also connected to the lower drywell gas space via horizontal pipes which penetrate the lower drywell wall at an elevation approximately two-thirds of the height between the lower drywell floor and the top of the lower drywell.

Because of the lower drywell configuration, it is expected that some fraction of the molten debris which is released at vessel failure will be trapped in the lower drywell. The region of the lower drywell above the downcomers does not have any open flow paths. Furthermore, the control rod drive mechanisms are located in this region. Therefore, the velocities in this region will be lower than that in the region below the downcomers. Material which has been lifted off the floor could become trapped in these more stagnant regions of the lower drywell above the downcomer. Thus, one would not expect that all of the debris entrained in the gas flow would exit the lower drywell.

Once debris is assumed to leave the lower drywell and enter the downcomer, two mechanisms govern the final distribution of core material. These mechanisms are the impaction of core debris on structures and the transport to the suppression pool due to flow toward the wetwell in the downcomer. The gas transport pathway to the upper drywell is relatively convoluted. For the debris to enter the upper drywell, it must be entrained off the drywell floor, flowing vertically along the drywell wall. It then turns 90 degrees to enter the horizontal piping. After flowing a short distance through the horizontal piping, the flow will encounter a Tee type junction with the vertical downcomer. At this point the entrained debris must again turn 90 degrees. There is potential for impaction on the downcomer wall at each turn. This impacted debris is likely then to flow downward along the downcomer wall toward the wetwell vents. This effectively removes the material from participating in the DCH event.

In addition, if the horizontal wetwell vents have cleared then the entrained flow will split between that going upward toward the upper drywell and that going downward toward the wetwell vents. If the vents have not (yet) cleared, then all the flow will go upward toward the upper drywell. The debris that partitions with the gas flow going downward toward the wetwell vents will not participate in DCH.

Finally, since DCH relies on the rapid heat transfer from the corium to the surrounding gas, any debris which is transported via a wave type motion will not participate in the DCH event. As discussed above and described in detail in Subsection 19EA.3, wave formation is not expected to be the dominant transport mechanism. Most of the debris transported to the upper drywell is expected to be in the form of particulate.

The fraction of the molten debris in the lower head that is dispersed into the upper drywell and fragmented can be represented by the following relationship:

$$f_{frag} = f_{downcomer} \times f_{impact} \times f_{split\text{-}updw} \times f_{part} \qquad (19EA\text{-}1)$$

where:

f_{frag} = The fraction of debris transported to the upper drywell and fragmented,

$f_{downcomer}$ = The fraction of debris which gets entrained out of the lower drywell into the downcomers,

f_{impact} = The fraction of the debris entering the downcomers which does not remain permanently impacted on the downcomer walls (i.e. either is not impacted or is impacted and re-entrained),

$f_{split\text{-}updw}$ = The fraction of the gas flow which goes upward toward the upper drywell (as opposed to the wetwell),

f_{part} = The fraction of the debris entrained into the upper drywell which enters in the form of small particles.

The uncertainty ranges for these four parameters were chosen based on the physical layout of the lower drywell along with engineering judgment. The value chosen to represent the amount of material expelled from the vessel which exits the lower drywell represents the potential for material to be trapped in the stagnant region above the horizontal vent pipes. The ANL experiment described in Subsection 19EA.3.1.2.2 did not include any below vessel structures. It may be possible to freeze and hold material on these massive structures. Furthermore, the openings from the ANL cavity are much wider than the openings in the ABWR. Since the debris will have to make a turn to enter the ABWR openings, the smaller area makes the debris less likely to entrain. Based on the above discussion, about half of the debris is assumed to reach the connecting vents.

The debris leaving the lower drywell will then travel a short distance before entering a tee junction. It is expected that some of the debris will impact on the wall of the pipe at the junction and flow down into the wetwell. Based on the physical characteristics of the debris flow path, it was judged that most of the debris does not get removed due to impaction.

Prior to clearing of the horizontal vents, gas flow into the vent pipes will be directed up into the upper drywell. As the upper and lower drywells pressurize, the water level within the vent pipes will be depressed, and venting into the wetwell will begin. The average vent clearing time is 0.5 seconds. After the vents have cleared, the gas would preferentially flow into the wetwell since the upper drywell would be increasing in pressure. If the wetwell and upper drywell pressures were conservatively assumed to be equal, then a 50/50 split would occur based on equal flow areas in both directions. As described in Subsection 19EA.3, the entrainment dispersal time is estimated to be two seconds. If the debris is dispersed linearly, then the vents would clear after only 25% of the debris was released from the lower drywell. The debris leaving the lower drywell after vent clearing would then conservatively be split 50/50 between the wetwell and the upper drywell. Based on this discussion, it was conservatively assumed that most of the debris flows to the upper drywell.

The final phenomenon that could influence the amount of the debris that would participate in DCH is the wave formation. If the debris enters the upper drywell in the form of a coherent wave, it would not be expected to participate in mixing with the gas. Experiments performed at ANL for PWR cavity configurations have resulted in this wave type of sweepout. As discussed in Subsection 19EA.3, wave formation is not expected to be the dominant removal mechanism for the ABWR configuration. However, as debris flows through the wetwell/drywell connecting vents to the upper drywell, it is possible that some of the debris forms wave-like sheets. Therefore, engineering judgment has been used to estimate that the median value for the fraction of debris that is dispersed as particulate debris is 0.875.

Assuming a uniform distribution for each of these parameters between their assessed upper and lower bounds results in the distribution for f_{frag} shown in Figure 19EA-8.

Based on the above discussion, three regimes were selected to represent this parameter in the event tree:

- Low ($f_{frag} \leq 0.35$),

- Intermediate ($0.35 < f_{frag} \le 0.60$), and
- High ($f_{frag} > 0.60$).

In the deterministic DCH pressure calculations described in Reference 19EA-3, the nominal values for f_{frag} used to represent these three regimes were 0.25, 0.5, and 0.75.

Figure 19EA-9 shows the comparison of the calculated cumulative distribution, as determined from the above parameters, and the distribution assumed for the deterministic DCH analysis. As the figure shows quite clearly, the assumed distribution is conservative. The entire range assumed for high fragmentation lies above the calculated range. For the low and intermediate fragmentation ranges, only a very small portion of the assumed distribution lies below the calculated distribution. Therefore, the discretization of f_{frag} used for this analysis is conservative.

19EA.2.1.7 Peak Containment Pressure Following RPV Failure

This event assesses the peak drywell pressure following RPV failure. There is only one branch for this event. This event summarizes the deterministically calculated drywell pressure for the set of conditions and assumptions specified in the event sequence pathway leading to this event. A description of the calculational methodology and calculated results are presented in Subsection 19EA.3.

19EA.2.1.8 Drywell Head Fails Following Vessel Failure

This event assess the probability of drywell head failure given the pressure determined in the previous event. The probabilities for failure are determined from the drywell head fragility curve described in Attachment 19FA.

19EA.3 Deterministic Model for DCH

A computer program has been developed to provide scoping calculations for DCH events in the ABWR. Several simplifications and assumptions exist in this model. This model, and its application to the ABWR design are described below.

19EA.3.1 Debris Dispersal in the ABWR

The purpose of this subsection is to briefly summarize the available information on debris dispersal from a configuration like that of the ABWR lower drywell.

19EA.3.1.1 Velocity Required to Transport Debris Particles

The velocity required to transport debris particles out of a compartment by entrainment can be easily estimated (Reference 19EA-6). To lift a particle of radius r against the force of gravity requires a velocity given by:

$$(\rho_f - \rho_g)\frac{4}{3}\pi r^3 g = c_d \pi r^2 \frac{\rho_g u_g^2}{2} \qquad (19EA\text{-}2)$$

where:

ρ_f	=	Fluid density
ρ_g	=	Gas density
r	=	Particle radius
g	=	Acceleration of gravity
c_d	=	Drag coefficient
u_g	=	Gas velocity

If we assume complete hydrodynamic breakup, the maximum particle radius is given by equating the force imparted by the gas stream to the surface tension force holding the droplet together. This is usually cast in terms of a Weber number:

$$We = \frac{2c_d \rho_g u_g^2 r}{\sigma} \qquad (19EA\text{-}3)$$

where:

σ	=	Surface tension

There is some ambiguity on the form of this equation and the choice of the Weber number, We. Most authors fold the drag coefficient into We. For reasons that will soon become clear, we follow Henry (Reference 19EA-7) and leave the coefficient explicitly in the expression. Typical values of Weber number are 6 to 12 when the drag coefficient is left out. Strictly speaking, these would specify the maximum particle size. The mass median diameter, for example, would be about half this value (Reference 19EA-8).

If we substitute Equation 19EA-3 into 19EA-2 and neglect the gas density compared to the liquid density, we obtain:

$$u_g^4 = \left(\frac{4We}{3c_d^2}\right)\frac{\rho_f \sigma g}{\rho_g^2} \qquad (19EA\text{-}4)$$

Define the Kutateladze number by:

$$Ku = \frac{u_g}{\left(\frac{\rho_f \sigma g}{\rho_g^2}\right)^{1/4}} \qquad (19EA\text{-}5)$$

so that:

$$Ku = \left(\frac{4We}{3c_d^2}\right)^{1/4} \qquad (19EA\text{-}6)$$

A free particle in the high Reynolds number limit has a drag coefficient of about 0.44. If we assume a Weber number of 8 (the results obviously depend weakly on this choice), we obtain:

$$Ku = 2.7 \qquad (19EA\text{-}7)$$

When substituted into Equation 19EA-5, this gives a velocity close to the experimentally measured value required to entrain particles off a free surface (Reference 19EA-9).

Consider a different situation in which gas is sparging a pool from below. If we consult Figure 19EA-10 (Reference 19EA-6) which shows the drag coefficient for various porosities (ε), we note that the drag coefficient for a particle bed is in the range 20–80. If we use 80 and leave the We number at 8, we obtain:

$$Ku = 0.2 \qquad (19EA\text{-}8)$$

This does yield just the experimentally measured velocity required to fluidize a pool (Reference 19EA-10).

Thus, we see that the velocity required to lift liquid droplets is a strong function of the configuration. A gas stream passing horizontally over a liquid pool requires on the order of 10 times the velocity required to fluidize a bed (i.e., a situation in which the gas stream proceeds vertically from below). This effect results from the different drag coefficients which apply to the two situations. Thus, one would expect the required velocity for entrainment in the ABWR to be 10% of the value for the Zion cavity.

19EA.3.1.2 Argonne Experiments on Debris Dispersal

Spencer, et. al., at Argonne National Laboratory have conducted a number of debris dispersal experiments in several geometries. Extensive information on these is available (References 19EA-10, 19EA-11, 19EA-12, and 19EA-13). Here we shall briefly summarize the results of two sets of quasi-steady experiments designed to determine the threshold velocities required to transport debris from simulated reactor cavity/pedestal regions.

19EA.3.1.2.1 Experiment on Zion Configuration

Figure 19EA-11 shows a schematic view of the Zion reactor cavity. Experiments conducted with this geometry (Reference 19EA-11) indicate that the threshold velocity required to disperse liquid droplets into the gas stream and move them from the cavity is approximately given by a Kutateladze number of 2.5. This is not unexpected, since the geometry of the cavity is such as to cause the gas jet leaving the reactor to stagnate at the floor of the cavity, to turn and proceed horizontally down the cavity keyway over the liquid pool. Thus, the considerations which lead to Equation 19EA-7 would seem to apply.

It should be noted that sweepout, in which a continuous liquid film was observed to be flooded from the reactor cavity, occurred at about the same velocity as entrainment of droplets. Thus, as noted in Reference 19EA-14, the amount of material transported from a Zion-like cavity as droplets relative to the amount transported as a film is determined by the relative rates of the two processes.

An additional experiment was run in which steel shot of diameter 780×10^{-6} m was used instead of a liquid pool. By substituting into Equation 19EA-2, the drag coefficient necessary to explain the observed velocity threshold for debris dispersal of ~16 m/s (Reference 19EA-13) is found to be about 0.3. It is not known why this is less than the expected value of 0.44, but the discrepancy is not considered large.

19EA.3.1.2.2 Experiments on Grand Gulf Configuration

Figure 19EA-12 shows a schematic view of the Grand Gulf pedestal region. Experiments were also conducted at Argonne on a scale model of this configuration. Quite different behavior was observed in these tests. Due to the more or less symmetric orientation of the scale model CRD ports around the circumference of the pedestal, the gas jet leaving the simulated reactor vessel was observed to stagnate, proceed horizontally so as to undercut the entire liquid pool, turn and proceed vertically using virtually the entire cross-sectional area of the pedestal. (Only a small fraction of the area was used by the jet moving downward from the simulated vessel.)

As noted in Reference 19EA-10, this configuration is reminiscent of a pool sparged from below. Using this reasoning, Equation 19EA-8 would be expected to apply. Indeed, a Kutateladze number of 0.2 was found to accurately predict the velocity required to initiate removal of debris. Visual observations of the experiment support this conclusion, as it was seen that the entire pool became fluidized, the liquid rose up to the level of the ports, and was then swept out by the gas stream.

An experiment was also conducted in this geometry with steel shot. A drag coefficient of about 7 is required to explain the measured velocity threshold for sweepout of ~3.5 m/s. By consulting Figure 19EA-10, we see that such a drag coefficient is appropriate for a bed of porosity about 0.8. This is, in fact, the porosity that would be obtained if the entire bed of shot was uniformly fluidized up to the elevation of the CRD ports. This suggests that the threshold velocity for sweepout in the Grand Gulf configuration might be a function of the ratio of the initial pool volume to the total volume which exists in the pedestal under the elevation of the gas flow paths. In other words, fluidization could begin at a low velocity (when the porosity is low and the drag coefficient is high), but as the pool tries to grow toward the exit flowpaths, the velocity required to continue to levitate droplets would increase. This conjecture cannot be confirmed by the few tests run with liquid pools, however.

19EA.3.1.2.3 Application to GE ABWR Configuration

The ABWR configuration, Figure 19EA-7, is similar to that for Grand Gulf. Thus, we expect that a Kutateladze number on the order of 0.2 should be applied to calculate the dispersal threshold. With the exception of the ANL Grand Gulf work, the documented experiments performed to date have focused on PWR type cavities such as Zion. As discussed above, these are not directly applicable to the ABWR configuration.

19EA.3.2 Pressurization Due to DCH

The pressurization of the drywell is affected by the blowdown of gases from the vessel and by the heat transfer from fragmented corium into the drywell. An explicit method is used to calculate the response of the system. The gas is assumed to be an ideal gas with the rate of change of pressure, $\dot{P}$, calculated from:

$$\dot{P} = \frac{M_g R \dot{T}_g}{MW_g V} + \frac{\dot{M}_g R T_g}{MW_g V} \qquad (19EA\text{-}9)$$

where:

M_g	=	Total mass of gas in containment (steam and non-condensable gas)
R	=	Gas constant (8314 N•m/kg•mole•K)
T_g	=	Gas temperature
MW_g	=	Average molecular weight of the gas mixture
V	=	Drywell volume

and a dot over a variable indicates its rate of change with time.

The temperature change of the gas, $\dot{T}_g$, is calculated by assuming that the gas and the fragmented debris are in equilibrium at each time step. Since the DCH event is very rapid, no credit is taken for heat transfer to containment heat sinks. The specific heat capacity for steam is evaluated using a curve fit to saturated steam properties (Reference 19EA-15). Constant specific heat is used for the non-condensable gas.

The rate of change of mass in the containment, $\dot{M}_g$, considers the gas blowdown from the vessel, any flow to the suppression pool through the connecting vents, and hydrogen generation which occurs as a result of the reaction between the steam and the zirconium. The mass flow rates through the downcomers and from the vessel are evaluated using a compressible flow model (Reference 19EA-15). The pressurization of the wetwell due to any addition of non-condensable gases is considered. Steam which passes through the connecting vents is assumed to be quenched.

The debris conditions are calculated by conservation of energy in the system. The mass of debris participating in the DCH event is assumed to increase linearly over the time constant for the event (discussed in Subsection 19EA.3.4). The fraction of the debris allowed to oxidize is a user input (discussed in Subsection 19EA.3.5). The energy of reaction is taken to be that for the zirconium steam reaction. Oxidation of the zirconium participating in the DCH event is assumed to be instantaneous.

The temperature of the debris is calculated based on the amount of energy remaining with the phase change energy accounted for and assumed to take place at a uniform temperature of 2500 K. Constant specific heat and latent heat of fusion are assumed.

19EA.3.3 Calculation of Vent Clearing Time

The DCH program previously described includes a model to predict the time required to clear the horizontal vents and begin gas flow to the wetwell. The model, based on analysis by Moody (Reference 19EA-16), requires as input the pressurization rate for the upper drywell. The DCH model computes the pressurization rate for each time step. Given this, Moody has derived a simple formula for the water velocity resulting from this ramp pressure. The DCH model then computes the water movement as a function of time; and, based on a table look up of vent area vs. water level, calculates the appropriate drywell vent area at any point in time.

19EA.3.4 Calculation of Dispersal Time Constant

For the parametric modeling of DCH in this analysis, a timescale for dispersal must be input. This influences the rate of containment pressurization by defining the entrainment rate of the debris.

There is a dearth of good models for DCH. The only models which were identified are:

(1) The CONTAIN Model

This is a lumped parameter model in which the rate of entrainment is input by the user. It provides no insights for this study.

(2) Henry has developed a model for ARSAP (Reference 19EA-17) that explicitly compares the time-scale for dispersal due to the acceleration of the liquid film as a whole to the time required to entrain the debris as droplets.

This model is very attractive from the standpoint that it produces closed-form answers and illustrates that the competition between the two modes of debris removal from the cavity may be an important consideration for designing and interpreting experiments. However, there appear to be several problems with this model. First, it assumes a very schematic debris configuration, i.e. an initially static debris pool lying on the floor of the cavity. It seems more reasonable to assume that there is debris splashing throughout the cavity, as point out in Levy's WRSIM papers (Reference 19EA-18) and in Spencer's work at ANL. Next, it is questionable that the entrainment rate formula that is used, the one developed by Ricou and Spalding for gas-gas entrainment, applies to this situation. There is evidence (cited by Levy) that non-uniform gas velocities in cavities may play an important role in enhancing entrainment rate. Finally, only very limited comparisons to data have been offered.

Taken at face value, Henry's model tends to predict very rapid removal of the debris from the cavity, mainly as a liquid film. Oddly enough, the time-scale for removal of the film depends only in a very weak way on the hole size in the vessel (i.e. through the gas density in the cavity and even this matters only as the 0.25 power).

(3) BNL has written a one-dimensional model called DCHVIM. In a summary paper presented at the Pittsburgh Heat Transfer conference in 1987, the model is applied to the SNL DCH-1 experiment. For this calculation, however, the entrainment rate was taken directly from experimental data. In addition, the model was not applied to a full, reactor-scale scenario, only to DCH-1.

(4) Sienicki and Spencer at ANL have written a relatively sophisticated one-dimensional hydrodynamics model called HARDCORE (Sienicki and Spencer, undated). Separate mass, momentum and energy equations are written for the liquid film, the droplets and the gas. The entrainment correlation is based on liquid jet breakup formulas developed by De Jarlais,

Ishii, and Linehan. Being one-dimensional, the model does not, of course, taken into account non-uniformities in velocity, though there is consideration given to entrainment from annular films on the cavity walls.

The model was applied to the ANL CWTI-13 experiment and to DCH-1. In both cases, it is stated that the debris entrainment time was predicted fairly accurately by the code (time-scales on the order of 0.1 seconds). When the code was then applied to a full-scale Zion TMLB accident, the predicted time-scale for sweep-out of the debris from the cavity was of order 2.5 seconds, i.e. the numerical results are fit rather well by:

$$m_e = m_o\,(1 - e^{-t/2.5}) \qquad (19EA\text{-}10)$$

where:

t	=	Time in seconds since the blowdown begins
m_e	=	mass entrained
m_o	=	initial mass available for entrainment

(5) The recent papers by Levy, mentioned above, contain an explicit closed-form expression for the time-dependent entrainment of debris from the reactor cavity. This formula has been compared to a wide variety of small-scale test data with remarkably good results. The formula was applied to calculate the entrainment rate for a full-scale Zion-like cavity in a TMLB-type sequence. If one assumes that steam exists in the cavity (the results are apparently quite sensitive to the gas density there due to the strong dependence on Euler number), one obtains the seemingly nonsensical result of 100 seconds. However, it does not appear at this juncture that a constant in his expression can be derived from small-scale experiments and applied to full-scale cavities as was done in the calculation just mentioned.

(6) A code called CORDE is under development in the UK. We have very little information on its models, state of development, or predictions.

Thus, based solely on the ANL paper, the assumption used in this analysis is linear debris removal assumed over a 2-second period. This value appears to be conservative, but not remarkably so. Figure 19EA-13 compares the fraction of debris discharged for the 2-second linear rate used in this analysis with the 2.5-second e-folding time from the ANL study. The sensitivity to this assumption is investigated in Subsection 19EA.3.6.

19EA.3.5 Application of DCH Model to ABWR

The model requires a variety of inputs which describe the geometry of the vessel and containment, the initial and boundary conditions for the event and a few model parameters.

The geometric information required by the model is:

(1) The drywell vessel and wetwell gas free volumes which are used to calculate pressure.

The drywell volume used for this analysis is the total for the ABWR upper and lower drywells. This effectively assumes that there is a large flow area between upper and lower drywell regions. The possible impact on the results from this assumption is considered in Subsection 19EA.3.6.4.

(2) A table of horizontal vent area as a function of distance from the initial water level and the total vent clearing depth when all vents are available.

These are used to calculate the vent clearing time. For this analysis, it is assumed that there is no initial pressure difference between the wetwell and the drywell. Thus, water level in the connecting vents is high, which conservatively delays the time until the vents begin to uncover and gas can flow to the wetwell.

(3) The vessel failure area which is used to calculate the blowdown from the vessel.

This value is specified for each branch point on the DETs.

Initial and boundary conditions are:

(1) Debris Mass Involved in DCH Event.

The value of this variable is specified for each case on the DETs.

(2) Initial Debris Temperature.

If this temperature is specified above 2500 K, then the latent heat of fusion is used in calculating the initial debris energy. If the temperature is at 2500 K or below, then the latent heat of fusion is not included in the initial debris energy. This value was nominally set at 2501 K. The sensitivity to this assumptions is investigated in Subsection 19EA.3.6.3.

(3) The initial containment temperature and pressure are assumed equal in the wetwell and drywell.

The steam mass fraction in the drywell is assumed to be 1.0. The sensitivity to this assumption is investigated in Subsection 19EA.3.6.6.

(4) The initial vessel pressure is used to calculate the source of steam from the vessel to the containment volume.

The pressure is assumed to be the nominal vessel pressure for normal operating conditions. Slight variations in this value (such as might result from a consideration of the SRV setpoints) do not have a significant impact on the results. No attempt is taken in this analysis to take credit for partially depressurized vessel conditions.

(5) Vessel gas temperature and vessel steam enthalpy.

Both values are conservatively taken to be constant. The values used are typical for MAAP analyses of high-pressure core melt scenarios.

Model parameters are:

(1) Fraction of Zr to be Oxidized in The DCH Event.

Of the debris mass that is being entrained at any instant, 20% is assumed to be Zr. This debris is assumed to oxidize immediately as it is entrained. Therefore, if one specifies 0.5 as the oxidation fraction, then half of that 20% mass will oxidize. For every mole of Zr, 2 moles of H_2O will be replaced by 2 moles of H2 in the drywell volume, and the chemical reaction energy will be added to the debris. The sensitivity to this parameter is discussed in Subsection 19EA.3.6.5.

(2) The time for debris entrainment determines the interval during which the specified mass of debris will be entrained.

Refer to Subsection 19EA.3.4 for a discussion of this parameter. The sensitivity to this parameter is investigated in Subsection 19EA.3.6.2.

(3) Time Constant for DCH.

If set to zero, the debris will be entrained linearly. If set to non-zero value, then the debris will entrain at a rate with an e-fold value equal to the time constant. This analysis assumes the debris is entrained linearly. Any sensitivity to this parameter is bounded by the time for debris entrainment sensitivity discussed in Subsection 19EA.3.6.2.

(4) The time step for the computer code calculations was selected to be one millisecond.

Since the time constant for the DCH event is on the order of a few seconds, there should be no sensitivity to reasonable variations in this parameter.

The DET methodology addresses the variation in debris mass, initial containment pressure, and vessel failure area. Subsection 19EA.3.6 provides a discussion of the importance of the debris temperature, Zr fraction, dispersal rate, nodalization and initial drywell steam fraction.

The code calculates the containment response to DCH events. The most important output of the calculation is the peak containment pressure. The results of the model analysis for each branch of the DETs are summarized in the penultimate column of Figures 19EA-4, 19EA-5, and 19EA-6.

19EA.3.6 Sensitivity to Various DCH Parameters

As indicated in Subsection 19EA.2, the DET methodology addresses the variation of several key DCH parameters. This subsection looks at the importance of the debris temperature, amount of Zr oxidized ex-vessel, and the dispersal time constant to the overall pressurization. These parameters were assumed to be constant in the scoping calculations and were judged not to have a significant impact on the results. The results of the sensitivity studies confirm that these parameters have a second order effect on the peak containment pressure.

19EA.3.6.1 Base Case

For the purpose of comparison, the following case was analyzed using the DCH model:

(1) Fraction of core molten at vessel failure—40%.

(2) Fraction of material dispersed into the upper drywell—50%.

(3) Dispersal time—2 seconds.

(4) Initial containment pressure—1.5 atmosphere.

The result of the analysis indicates a peak drywell pressure of 0.903 MPa. Referring to the containment failure curve, this has a failure probability of about 0.17.

19EA.3.6.2 Dispersal Time

The base case was re-run assuming dispersal times of 1 and 3 seconds. The results are:

	Peak Drywell Pressure MPa
Base Case	0.903
Dispersal Time = 1	1.124
Dispersal Time = 3	0.758

Subsection 19EA.3.4 provides the justification for the 2-second dispersal time and indicates that it may be somewhat conservative. However, a 50% increase in the dispersal time resulted in only a 20% change in the peak containment pressure. This does not represent a very significant change.

Subsequent to the completion of the DCH analysis, Ishii (Reference 19EA-19, Section 3.4.1) developed a model for the calculation of a removal time constant. Ishii's model assumes that droplets are stripped off a film adhering to the walls as compared to the ABWR model of a “sparged pool” to calculate whether gas velocities are high enough to cause debris removal from the lower drywell

To apply the analysis to the ABWR, Ishii's calculations were first duplicated. This proved to be somewhat difficult, because a few of the key physical and geometrical parameters are not supplied in the reference. By using debris property values from the literature and back-calculating some of the geometrical parameters from intermediate results, we obtained a consistent set of parameters that resulted in agreement within about 10 percent on the various results of interest.

If one first assumes the same vessel breach diameter as in the Ishii's calculation (0.2 meter diameter, which is a typical value for an ablated localized failure given the debris masses and temperatures likely to exist), one finds that the flow out of the lower drywell in ABWR never becomes choked. The area-averaged velocity in the lower drywell is an order of magnitude too small to meet the Ishii-Grolmes entrainment criterion.

This is the same result we obtained in the PRA for the low breach area limit. In the PRA, we concluded that it would be appropriate in the interests of conservatism to use Spencer's Mark III results to predict the potential for debris removal from the lower drywell. The original analysis we made of Spencer's experiments recognized that the volume which exists below the pedestal vents could be large compared to the debris volume. Therefore it is not clear that the sparged pool mechanism can always function in a Mark III-like configuration to remove the debris at velocities much lower than the droplet stripping mechanism. As it turns out, the volume beneath the vents in ABWR is very large (~100 m^3) compared to the debris volume (5-10 m^3).

To illustrate, consider the situation as the two phase debris pool level tries to approach the height necessary for debris entry into the connecting vents. This requires such a large pool void fraction in ABWR (e.g. > 90% if the pool is uniformly swelled) that the drag coefficient based on the superficial velocity will drop to values approaching that for isolated drops. In the limit, one merely regains the Ku ~3 result that is used in typical PWR geometries. This suggests that very little debris could be removed from the ABWR lower drywell, which is, of course, a far more favorable result.

A possible alternate mechanism for getting the debris to the connecting vents is to imagine that the inertia of the debris film (or perhaps the inertia of the droplets which are created during the sparging process) carries debris to the vicinity of the connecting vents. The higher gas velocities in the vicinity of the connecting vents would then tend to draw debris into the vents. Considering that the vents are located 10 meters above the floor and that they occupy only about a third of the perimeter at that elevation, this does not seem to be a very effective mechanism either. One could also cite the observations in the Sizewell experiments (Reference 19EA-19, Appendix O) of very high gas velocities near the floor of the cavity and speculate that this might apply near the drywell wall surface. All in all, however, it appears that there is very little likelihood of entrainment in the ABWR PRA for cases with localized vessel failures.

Nevertheless, if one postulates that the debris does make it into the connecting vents by some mechanism, one can then apply Professor Ishii's analysis. At one atmosphere (the pressure used in the Ishii's calculation), the velocities in the pipe are slightly less than the entrainment threshold (~80 m/s) calculated by the Ishii-Grolmes model. The analysis proceeds in a straightforward manner except that it is difficult to estimate the debris film velocity and typical thickness without postulating the mechanism that got the debris to the vents in the first place (this actually presents problems in Ishii's PWR analysis as well). If one uses the Ishii's technique, these quantities are estimated using

$$u_f \delta_{typ} = \frac{A_j u_j}{\pi D_h} \qquad (19EA\text{-}11)$$

where j denotes conditions in the jet leaving the vessel and D_h is the hydraulic diameter of the lower drywell. One then obtains a total entrainment rate from all the connecting vents of about 1 tonne/s (recall, however, that the entrainment criterion is not quite satisfied). This results in a time constant for debris entrainment on the order of 20-70 seconds, depending on the assumed debris mass, which is much longer than assumed in the PRA.

We now turn our attention to the large end of the vessel breach area spectrum. This was taken to be 2 m^2 in the PRA and was assigned a conditional probability. In this case, flow from the cavity will be choked. In the Ishii's calculation, the cavity flow area is the same as the area connecting the cavity to the containment. To calculate the gas velocity, the initial cavity pressure and the vessel gas temperature (isothermal blowdown) were used which presumably maximizes the velocity. If one follows the same procedure here a velocity of 230 m/s is obtained at an initial pressure of 2.5 bar. This is much greater than the entrainment velocity of about 70 m/s.

Gas velocities will be higher still in the connecting vents, which have about one tenth the flow area of the drywell. In the light of the previous discussion, it is assumed that the overall process of debris removal from the lower drywell is limited by the entrainment rate in the lower drywell, not the processes of de-entrainment and subsequent re-entrainment in the connecting vents.

The entrainment rate calculated in the lower drywell at 230 m/s is very large, on the order of 1500 tonne/s. It is doubtful that the correlations would be valid under such extreme conditions. Even if the equations do apply, the drywell pressure and gas density would rapidly increase due to the effect of the rapid transfer of momentum to the debris. This would invalidate the assumptions used in the calculation which maximized gas velocity.

These results supply useful insights. However, the overall impact of these considerations on the results of the PRA are relatively small. The small vessel breach area cases supplied half the conditional failure probability from DCH in the PRA. It appears that the likelihood of DCH-induced failures was overestimated for these cases since it is not at all clear that debris entrainment into the connecting vents can occur. In the large breach area cases, a more refined analysis which took into account the effect of the blowdown and debris entrainment on the thermal-hydraulic conditions in the lower drywell over the period when entrainment was occurring would be necessary to better quantify the likelihood of overpressurizing the lower drywell. For perspective, if all large vessel breach cases were assumed to result in containment failure, the conditional containment failure probability would be small. In any event, it is clear that the chance of significant DCH is determined largely by the assumptions used to characterize the upper extreme of the vessel breach spectrum.

In closing, it is noted that the implications of having a reduced chance of debris transport to the upper drywell in the majority of cases which have a small breach area may be more important for the upper drywell heat-up issue than for DCH. In particular the long term need for the initiation of the ACIWA system in spray mode may have been overestimated.

19EA.3.6.3 Debris Temperature

The core debris will interact with a variety of structures as it exits the reactor vessel. Thus, it is expected to experience substantial cooling by those structures on its way to the upper drywell. The ABWR DCH analysis conservatively assumed that the core debris entering the upper drywell was completely molten at a temperature of 2501 K. A sensitivity case was run assuming 2601 K for the debris temperature entering the upper drywell. The results indicate a peak containment pressure of 0.917 MPa vs. the base case value of 0.903 MPa.

19EA.3.6.4 Nodalization

The DCH analysis combines the lower and upper drywell compartments into a single control volume represented by node 1. The second node is set up to represent the wetwell with the suppression pool in the path connecting node 1 to node 2. A sensitivity case was run to investigate local pressurization in the lower drywell compartment. To do this, the volume included in Node 1 was reduced to that of the lower drywell with the vent area equal to the vent flow area from the lower to the upper drywell (11.3 m^2). Node 2 was used to represent the upper drywell. Since the junction between node 1 and node 2 does not require vent clearing as in the base case analysis, the vent path was assumed already cleared. All of the gas heating was done within the lower drywell compartment. In order to calculate the correct down-stream pressure, zero initial steam was assumed in the drywell compartment. This is necessary because the model assumes that all steam passing from the first node to the second node is condensed, setting the initial steam fraction to zero will correctly account for the increasing upper drywell pressure. A detailed examination of the results of this sequence indicates the zirconium oxidation reaction is essentially steam limited. Since the zirconium oxidation process consumes most of the steam exiting the vessel, only a small amount of steam will actually enter the second node.

The peak pressure computed for the lower drywell was 0.765 MPa. This indicates that the lower-to-upper drywell vent area is sufficient to preclude any substantial pressure buildup in the lower drywell region.

19EA.3.6.5 Zr Oxidation Ex-Vessel

The base case assumed that 20% of the core material that was discharged from the vessel was Zr metal. This is based on a uniform distribution of UO2 and Zr within the lower plenum of the reactor vessel. Of this material 50% of the Zr that was involved in DCH was allowed to oxidize and contribute to the drywell heatup. This is a conservative value since, in the time frame of interest, only the Zr on the surface of the particles would oxidize.

As a sensitivity calculation, the amount of ex-vessel Zr oxidation was doubled. This change in the amount of ex-vessel Zr oxidation is equivalent to assuming that all of the

Zr metal in the debris is oxidized. Alternately, this assumption is equivalent to assuming the fraction of core debris exiting the vessel was 40% Zr instead of 20% assumed in the base case. Thus, this sensitivity addresses both any possible non-uniform core material distribution within the lower head and the potential for increased oxidation.

The peak drywell pressure was computed to be 1.034 MPa vs. the base case value of 0.903 MPa. This is a very small effect given a factor of two variation in the amount of Zr oxidized during the DCH event.

19EA.3.6.6 Initial Drywell Steam Fraction

Since steam passing through the connecting vents will condense, the amount of wetwell pressurization during the DCH event is limited. The base analysis assumed a 100% steam environment in the drywell at the start of the event. To investigate the impact of this parameter on the peak pressure, a case was run assuming the drywell environment is initially 100% nitrogen. The wetwell pressure will be expected to increase faster for this case resulting in a higher drywell pressure. This result indicates that this is true, although the rise in the peak pressure is small. The peak pressure for this scenario is 0.979 MPa, as compared to a peak of 0.903 MPa for the base case. This variation due to initial drywell gas composition does not have a significant impact on the results of this study.

19EA.3.6.7 Hydrogen Combustion

Technical specifications allow the ABWR to be operated with 4% oxygen in the containment. During a DCH event, the drywell gas temperature may exceed the auto-ignition limit of approximately 811 K (1000°F). Burning of the hydrogen in the containment with this residual oxygen could result in an increase in energy of the gas. The appropriate reaction energy was added to the existing corium / gas mixture in order to predict an increase in the peak pressure due to hydrogen combustion. No credit was taken for the reduction in moles which would occur as a result of the burn.

The peak containment pressure increased by 0.103 MPa relative to the base case pressure of 0.903 MPa. The same analysis was performed assuming that the initial steam fraction was 0.0 (as compared to the base case assumption of 1.0). For this second case, the peak pressure increased by 0.228 MPa. Since a 50% steam fraction in the containment more accurately represents the actual conditions, the expected increase in peak pressure resulting from the recombination is 0.172 MPa. This is only a 20% change in the peak containment pressure, and does not represent a very significant effect.

19EA.3.6.8 Vent Clearing

After core debris discharge and before RPV blowdown, it is expected that the containment will begin to pressurize even before debris is dispersed into the upper drywell. A sensitivity study was run assuming that the vents had already cleared prior to debris dispersal. The results show that the peak containment pressure is reduced by 0.221 MPa compared to 0.903 MPa for the base case. This represents a decrease in the peak pressure of about 20%. While this is not very significant, it does provide a measure of the conservatism in the analysis.

19EA.4 Summary of Results

19EA.4.1 Quantification of Decomposition Event Trees

The quantified decomposition event trees are shown in Figures 19EA-1 through 19EA-6. The relationship between the peak drywell pressure and the cumulative probability distribution are shown in Figure 19EA-14. Note that the probability distribution functions (PDFs) are discrete since the discrete probabilities were assigned in developing the trees. The PDFs provide a measure of certainty that the pressure will not exceed a given value. They are not, however, uncertainty distributions in a statistical sense. Rather, they are based on knowledge of DCH and engineering judgment which characterize the ability to accurately characterize the boundary conditions for the problem.

From a deterministic viewpoint, the best estimate for the peak containment pressure is given by the median value of the PDF. As can be seen by comparing Figures 19FA-1 and 19EA-14, this indicates that the containment would not be expected to fail for any of the initial containment pressures studied. A measure of the uncertainty in this study is found by using the weighted sum (mean) of the probability of drywell failure for each of the branches on the DETs. These weighted values are transferred to the containment event trees for use as the conditional probability of drywell failure for sequences in which the vessel fails at high pressure.

19EA.4.2 Impact on Containment Failure Probability

An inspection of the ABWR accident classes shows that the conditional probability of having high RPV pressure at vessel failure is moderate.

A calculated probability of early containment failure (from direct containment heating), conditional on core damage, is very small.

19EA.4.2.1 Sensitivity of Containment Failure Probability to Assumptions

In order to demonstrate the robustness of the containment failure probability to the peak pressures calculated in the deterministic DCH analysis, three additional sensitivity calculations were performed. First, the DET was requantified assuming that the peak containment pressure for each of the low initial containment pressure cases was increased by 0.207 MPa. This could represent the possibility of an initial steam fraction of 50% in combination with a hydrogen burn and no credit for partial clearing of the wetwell connecting vents before the DCH event occurs. The resulting conditional containment failure probability for DCH increased slightly.

The second sensitivity case assumed that the containment would be at intermediate pressure for all cases. This represents potential uncertainties in the hydrogen production during the in-vessel portion of the accident. For this case, the containment failure probability due to DCH increases slightly. Although the conditional probability of failure by DCH is higher in this case than in the base analysis, DCH does not pose a significant threat to the CCFP goal of 0.10.

The third sensitivity calculation assumed that the containment would be at the intermediate pressure for all cases and, in addition, that all peak pressures would be increased by 0.207 MPa. The results show that for this conservative case, the conditional containment failure probability for DCH would be increased. Even with these very conservative assumptions, the DCH containment failure probability is far less than the 0.10 goal for CCFP. This demonstrates a large margin for the ABWR containment design to withstand containment challenges.

19EA.4.3 Impact on Offsite Dose

The final measure of the impact of uncertainties in severe accident phenomena is the effect on offsite dose. The CETs are quantified using the weighted sum of the containment failure probability as discussed above. The results of the CETs are then combined with deterministic accident sequence analysis and consequence analysis to determine the dose associated with the spectrum of severe accidents. In order to indicate the possible variation in dose due to uncertainty in DCH phenomena, other values must be selected for the probability of containment failure due to DCH.

Since the probabilities used in developing the DETs are themselves the uncertainties in the phenomena, one cannot determine the classical 5-50-95 confidence limits. However, one can select pressures corresponding to various cumulative certainty of non-exceedance (shown in Figure 19EA-14) and compare these values to the containment fragility curve (developed in Attachment 19FA) to estimate the probability of drywell failure with varying degrees of certainty. Selecting the 95% value from Figure 19EA-14 and noting that containment failure is not expected at the 50% certainty for peak pressure, one may draw the dose curves shown in Figure 19EA-15. This figure shows, that for the accident frequencies and certainty levels of interest, DCH has no detectable impact on the offsite dose.

19EA.5 Conclusions

The ABWR has a highly reliable depressurization system which results in a very low probability of a core damage event which leads to vessel failure at high pressure. Nonetheless, an evaluation of the potential risk of direct containment heating leading to containment failure in the ABWR has been performed. This study indicates that the design of the ABWR is highly resistant to damage as a result of a DCH event. This is due primarily to the general configuration of the ABWR lower drywell and connecting vent configuration and area. No modifications to the containment design are suggested as a result of this analysis.

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Table 19EA-1 Containment Pressure at RPV Failure

Accident Sequence	Pressure MPa
LCLP-PF-D-M	0.13
LCLP-FS-D-L	0.14
LCHP-PS-D-N	0.12
LCHP-PF-P-H	0.12
SBRC-PF-D-H	0.24
LBLC-PF-D-M	0.29
NSCL-PF-D-H	0.13
NSCH-PF-P-H	0.13
NSRC-PF-D-H	0.43

Table 19EA-2 Comparison of Assumed Debris Mass Participating in DCH with BWRSAR Debris Discharge Results

	Integrated Debris Mass (kg)					
	ABWR DCH Analysis (Minimum Values)			BWRSAR		
Time	Zr	Metals	Oxides	Zr	Metals	Oxides
Vessel Failure	0	0	0	0	0	0
+2 minutes	6500	*	17,000	926	3880	0
+4 minutes	6500	*	17,000	2316	8172	0

* Only a small portion of the core plate was assumed to be added to the debris.

Figure 19EA-1 DCH Event Tree for Sequences with Low Containment Pressure

Not Part of DCD (Refer to SSAR)

Figure 19EA-2 DCH Event Tree for Sequences with Intermediate Containment Pressure,
Not Part of DCD (Refer to SSAR)

Figure 19EA-3 DCH Event Tree for Sequences with High Containment Pressure

Not Part of DCD (Refer to SSAR)

Figure 19EA-4 DET for Probability of Early Containment Failure—High RV Press and Low Cont Press Sequences

Not Part of DCD (Refer to SSAR)

Figure 19EA-5 DET for Probability of Early Containment Failure—High RV Press and Inter Cont Press Sequences

Not Part of DCD (Refer to SSAR)

Figure 19EA-6 DET for Probability of Early Containment Failure—High RV Press and High Cont Press Sequences

Not Part of DCD (Refer to SSAR)

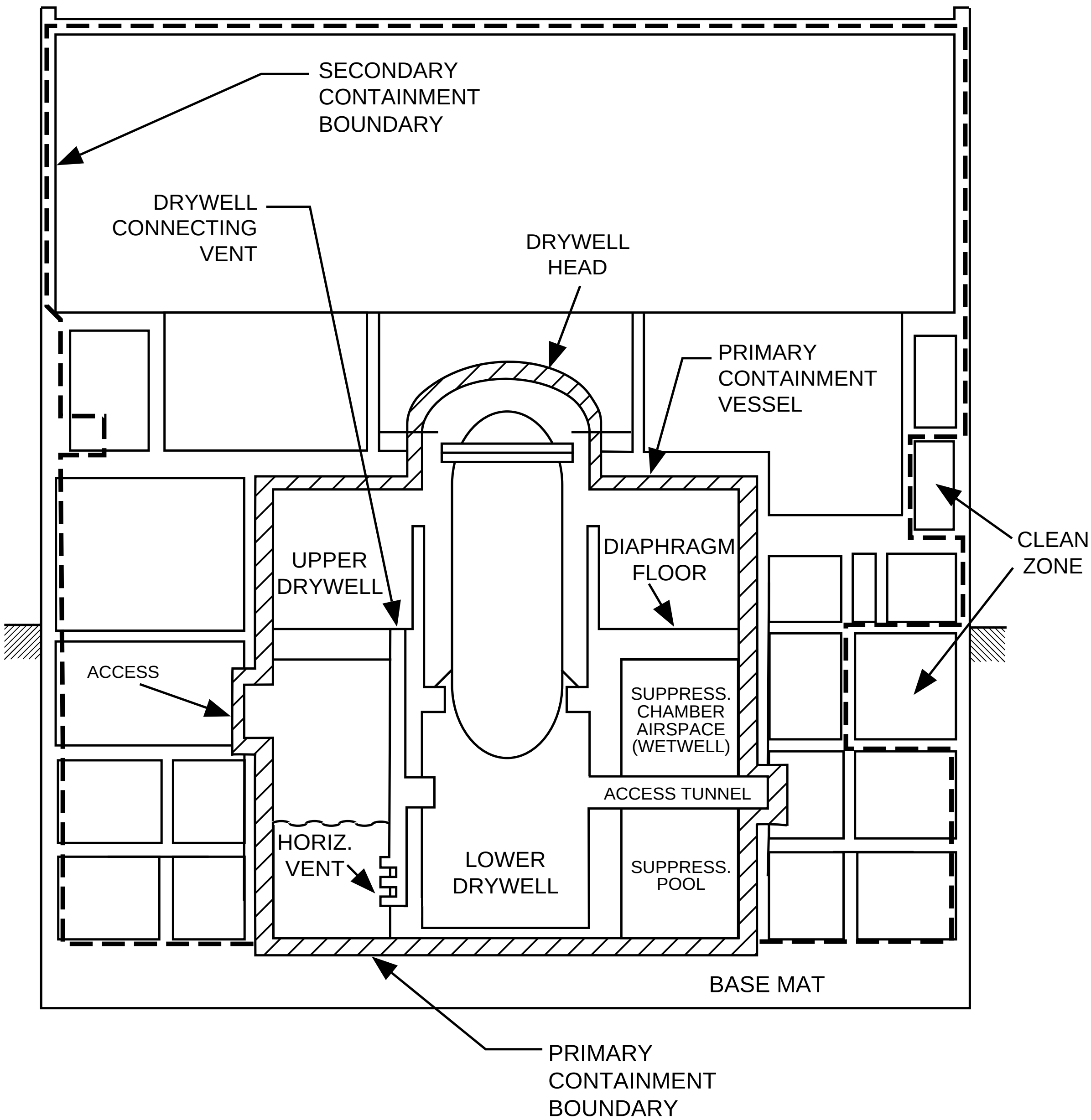

Figure 19EA-7 ABWR Containment Boundary Nomenclature

Figure 19EA-8 Calculated Probability Distribution Function for DCH Parameter F_{frag}, Not Part of DCD (Refer to SSAR)

Figure 19EA-9 Comparison of Calculated and Assumed F_{frag} Distributions Not Part of DCD (Refer to SSAR)

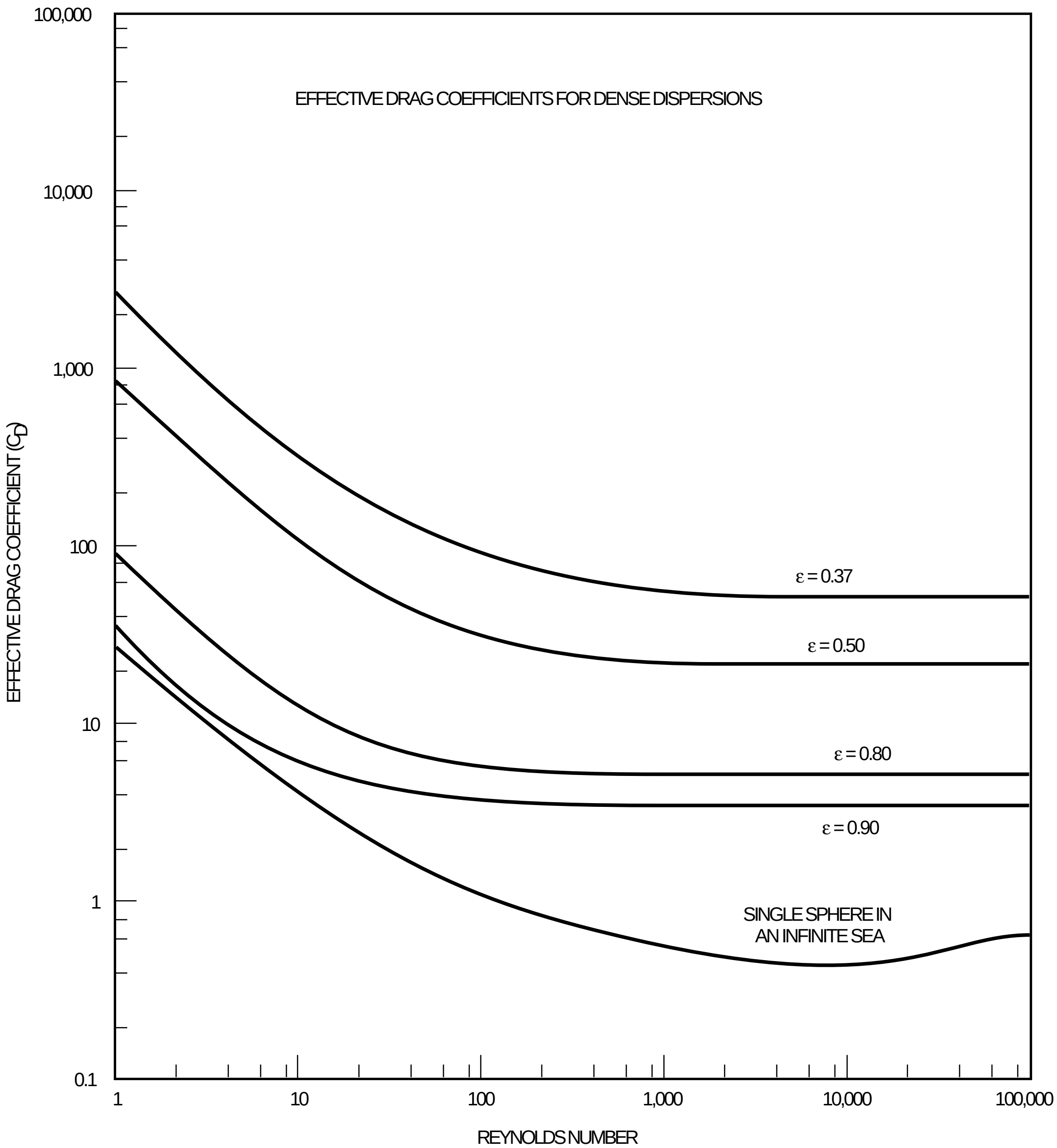

Figure 19EA-10 Effective Drag Coefficient for Dense Dispersions

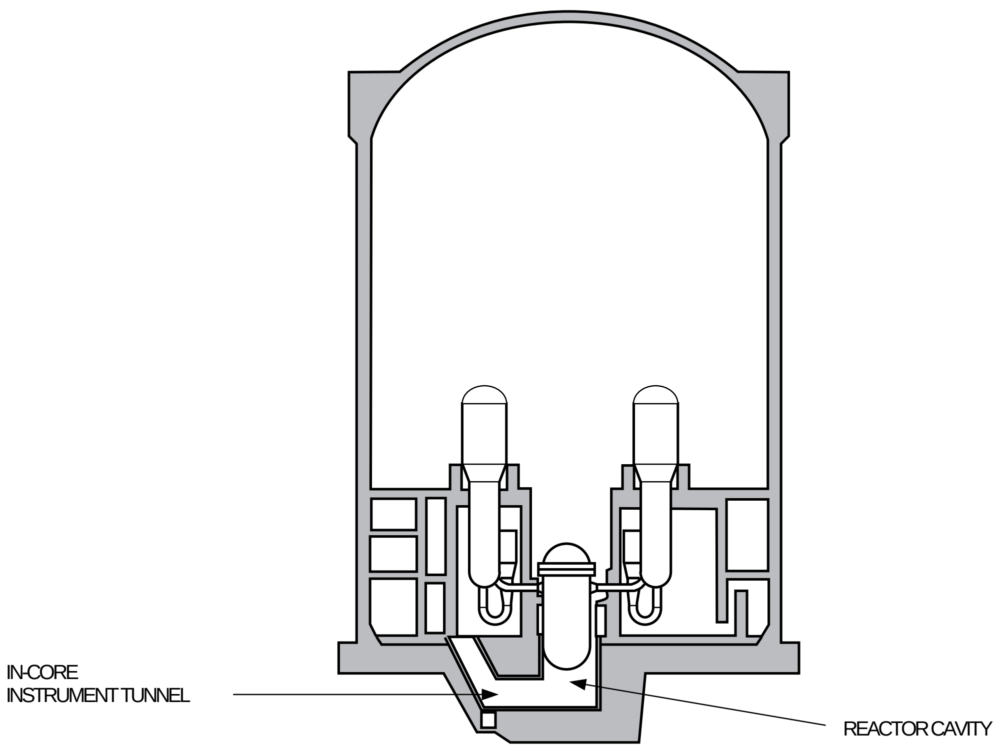

Figure 19EA-11 Zion Reactor Building

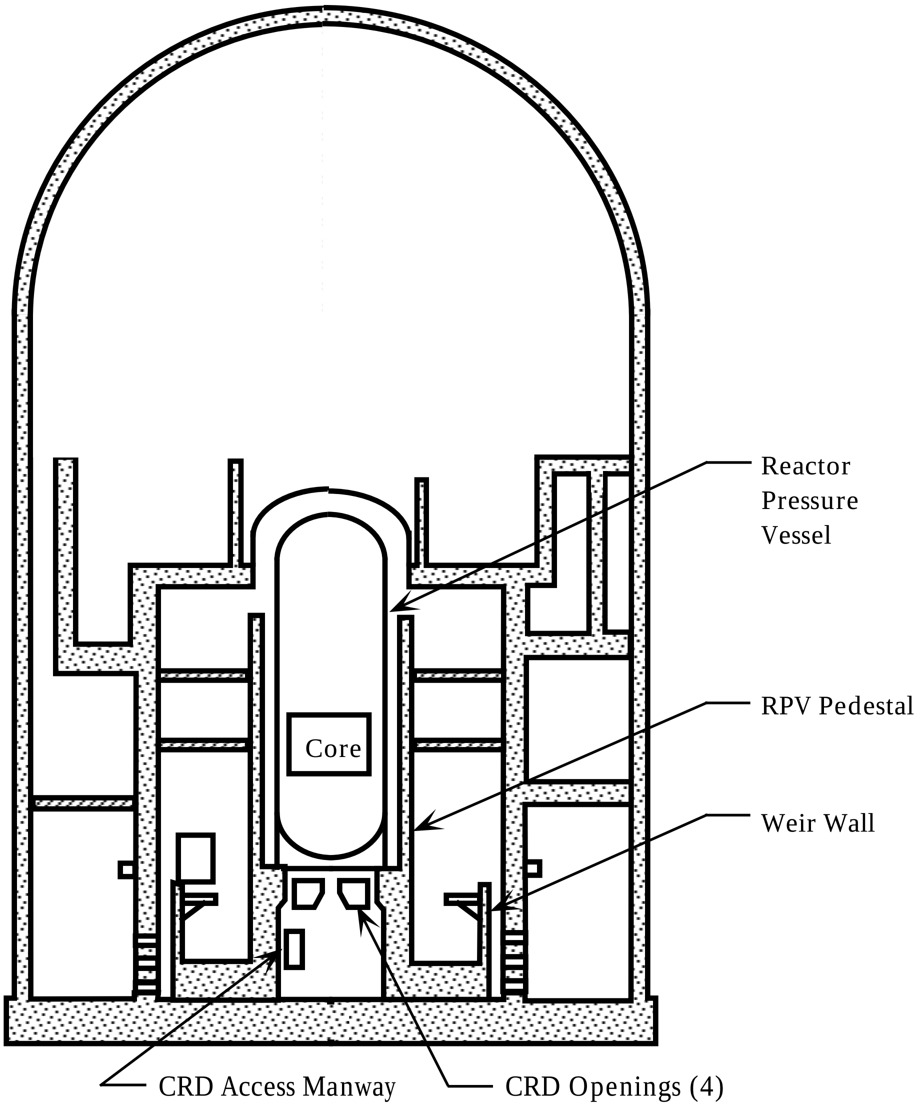

Figure 19EA-12 Schematic of Grand Gulf Containment

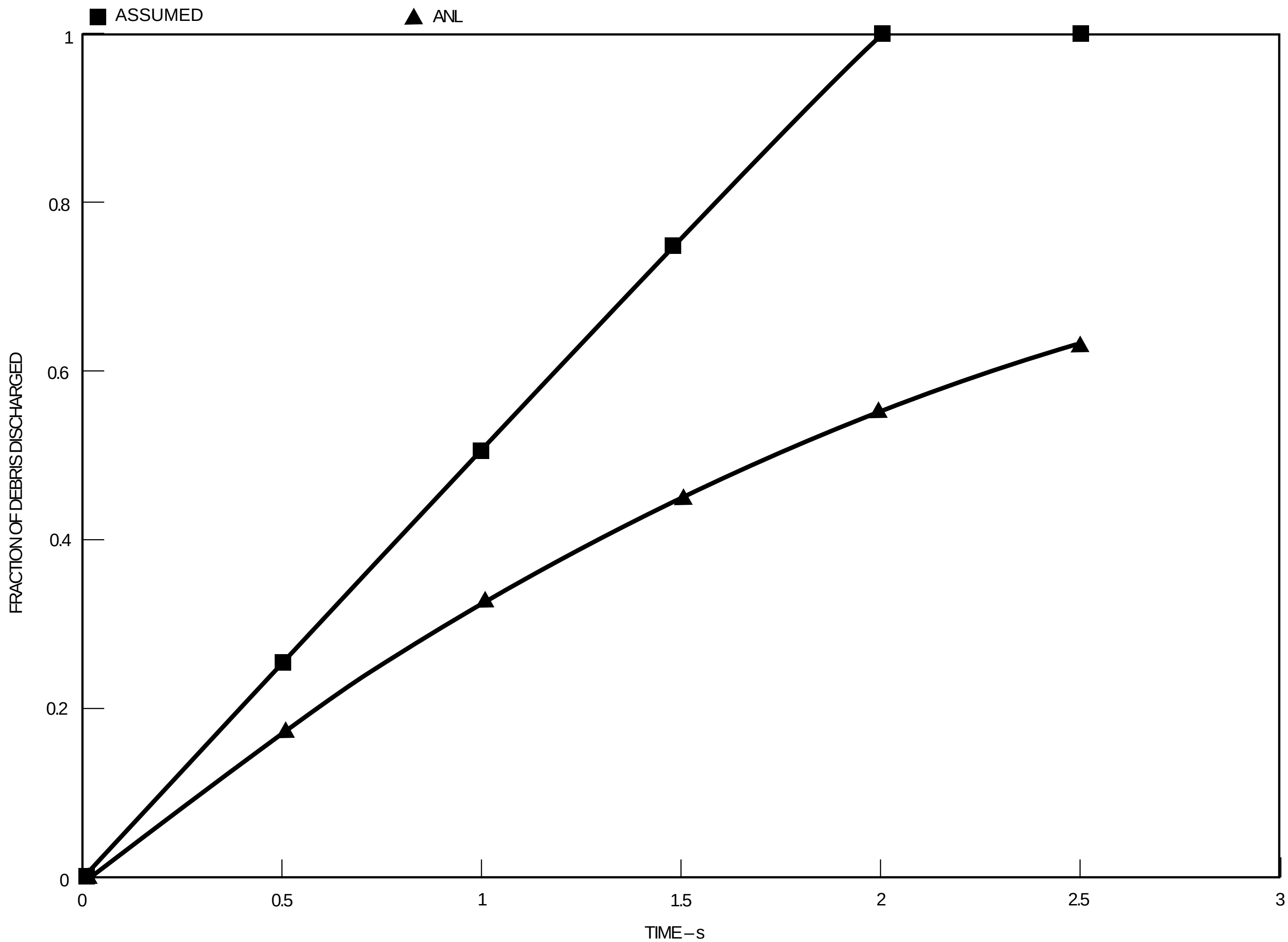

Figure 19EA-13 Comparison of Assumed Debris Discharge to ANL Data Fit

Figure 19EA-14 Cumulative Distribution for Peak Pressure Due to DCH
Not Part of DCD (Refer to SSAR)

Figure 19EA-15 Uncertainty in Whole Body Dose at 805 m (0.5 Mile) Due to DCH
Not Part of DCD (Refer to SSAR)

19EB Fuel Coolant Interactions

19EB.1 Introduction

Fuel coolant interactions were addressed in the early assessment for the ABWR response to a severe accident. Subsection 19E.2.3.1 examined the hydrodynamic limitations for steam explosions and concluded that there was no potential for a large scale steam explosion. The pressurization of the containment from non-explosive steam generation was calculated in the analyses for the accident scenarios. The following subsections examine the available experimental database for its relevance to the ABWR configuration, and provide a simple, scoping calculation to estimate the ability of the ABWR containment to withstand a large, energetic fuel coolant interaction.

Challenges of the containment during a severe accident may result from fuel coolant interactions. Both the impulse and static loads are considered here. Fuel Coolant Interactions (FCI) may occur either at the time of vessel failure when corium and water fall from the lower plenum of the vessel, or when the lower drywell flooder opens after vessel failure has occurred.

The critical time constants for a steam explosion are considered in 19E.2.3.1. This analysis concludes that the critical rates for heat transfer and energy dispersal preclude a large scale steam explosion which could damage the containment. Nonetheless, this study was performed to examine the potential impact of a large steam explosion on the ABWR.

Several experiments which have provided insights to steam explosions are examined, and features of the ABWR are compared to previous plants to indicate the relative resistance of the ABWR to steam explosions. A scoping calculation is also performed to estimate the size of steam explosion the ABWR could withstand.

Four potential failure modes are considered. The transmission of a shock wave through water to the structure may damage the pedestal. Similarly, a shock wave through the airspace can cause an impulse load. However, since the gas is compressible, the shock wave transmitted through the gas will be much smaller than that which can be transmitted through the water. Therefore, this mechanism is not considered here. Third, loading is caused by slugs of water propelled into containment structures as a result of explosive steam generation. Finally, the rapid steam generation may lead to overpressurization of the drywell.

19EB.1.1 Probability of a Pre-flooded Lower Drywell

The configuration of the ABWR containment, shown in Figure 19EB-6, limits the potential for water to be in the lower drywell at the time of vessel failure. The vessel skirt is solid and there are no active injection systems in the lower drywell. Therefore, the only possible sources of water to the containment are the wetwell / drywell connecting vents, the passive flooder and the vessel itself.

The wetwell / drywell connecting vents connect the upper and lower drywell regions to the suppression pool. The connecting vent is a vertical channel which has a horizontal branch leading to the lower drywell. Therefore, in order for water flow from the upper drywell to enter the lower drywell, it would have to fall almost nine meters down the connecting vents, then turn to enter the lower drywell. This is not viewed to be a credible scenario.

For the water level in the wetwell to rise sufficiently to overflow into the connecting vents, approximately 7.2E5 kg (1.6E6 lbm) would have to be added to the containment. If the EPGs are followed, this would occur only if injection was being provided from an external source in the event that flow from the suppression pool was not available. This implies that the only available injection sources are the Firewater and RCIC Systems. The RCIC System may be the only system available in events initiated by station blackout. Examination of the cases in 19E.2.2.3 (SBRC sequences) and 19E.2.2.8 (NSRC sequences) indicates that enough water can be added by the RCIC System to lead to overflow from the suppression pool to the lower drywell. If the station blackout continues and the Firewater Addition System is not used to prevent core damage, vessel failure into a pre-flooded cavity can occur in these sequences. The results of the Level 2 analysis, depicted in Figure 19D.5-3 indicate that SBRC sequences with failure of the vessel (no IV) are an extremely small percentage of all core damage sequences. The Class IV ATWS sequences were treated very conservatively in the containment event trees. All of these sequences were presumed to lead to core damage with high releases. However, as indicated by the analysis in Subsection 19E.2.2.8, Class IV sequences do not necessarily lead to core damage. Several hours are available for the operator to take appropriate actions to terminate the event. A conservative factor is applied to Class IV events to estimate the frequency of sequences with core damage and a pre-flooded lower drywell. These sequences are an extremely small percentage of all core damage sequences.

The passive flooder is designed to open when the temperature in the lower drywell airspace reaches 533 K (500°F). This temperature is slightly less than the temperature of the steam in the vessel under normal operating conditions. However, any potential break flow would cool by flashing as it reaches the lower drywell. Therefore, the passive flooder will not open until after vessel failure.

A LOCA in the bottom head of the vessel is also a source of water which could be present in the lower drywell at the time of vessel failure. All of the penetrations in the lower head are small, and any loss of coolant accident through them is classified as a small break LOCA. A conservative estimate of the core damage frequency for events initiated by LOCAs in the bottom head is the frequency of all small break LOCAs which lead to core damage for the ABWR. Examining Table 19D.4-1, the fraction of all core damage events initiated by a small LOCA is extremely small.

The potential for a fuel coolant interaction which could threaten the containment may be bounded by summing the frequencies of the sequences with water in the lower drywell at the time of vessel failure. Three sequences were identified above. The total frequency of these sequences is an extremely small percentage of all core damage sequences. Because this value is very small, it is judged that fuel coolant interactions will not have a significant impact on risk.

19EB.2 Applicability of Experiments

A large number of experiments have been performed to better understand FCI. Most of these experiments have been performed at bench scale with simulant materials. Freon-Water and Liquid Nitrogen / Water Systems are often used. While these experiments are necessary to understand the underlying physics of FCI, they are not directly applicable to the reactor condition. However, there are also several experiments performed with metal and oxides which provide insight to the potential for energetic FCI in a severe accident.

Other experiments, performed for different reasons, also yield some insights to FCI. Some experiments performed for debris coolability and core concrete interaction studies added water to the debris. With one notable exception, these experiments did not result in an energetic FCI. Finally, one experiment was performed to examine the impact of a water solid reactor cavity on direct containment heating. In the following subsection each of these experiments is examined for the insights into FCI and applicability to the ABWR.

19EB.2.1 Fuel Coolant Interaction Tests

A wide variety of experiments have been performed to investigate steam explosions. This subsection discusses results from selected experiments. Most of the experiments are prototypic of the reactor condition wherein debris falls into a pre-existing pool of water. The implications of these experiments on the potential for large, energetic FCI in the ABWR are also discussed.

Investigations into energetic fuel coolant interactions and steam explosions date back to 1950. Early experiments, including those by Long (References 19EB-1 and 19EB-2) and Higgins (Reference 19EB-3), identified the requirements for considerable mixing of the molten debris and water. Higgins and Lemmon (Reference 19EB-4) noted that the debris must be superheated and that the violence of the explosion increased with the melt temperature. Unfortunately, the triggers used in many of these experiments were very large. Thus, information about the propagation and energetics of these experiments is not applicable to reactor conditions.

One of the important parameters in determining the potential challenge to the containment from a steam explosion is the duration of the pressure pulse. Buxton and Benedick (Reference 19EB-5) performed a large series of experiments using iron-alumina thermite. The pressure traces for these experiments indicate an explosive pressure pulse of about 5 milliseconds.

The final, intermediate scale test performed at Sandia (Reference 19EB-6) used a corium thermite mass to simulate the materials which might be typical of a severe accident. As in the Buxton and Benedick experiment, the duration of the pressure pulse in these experiments was about 5 milliseconds. Three shakedown tests were performed using iron-alumina thermite with water in a crucible. In all of the tests spontaneous, self-triggered explosions occurred. In contrast, all four of the corium tests were externally triggered which resulted in one run with a “weak explosion” and one with a “mild explosion”. Two hypotheses were proposed to explain these results:

(1) The non-condensable gasses generated by oxidation stabilized the film boiling blanket, making it less susceptible to triggering;

(2) The UO_2 and ZrO_2 superheat was only about 300 K. It is possible that the debris froze before the trigger was initiated. This would prevent fine fragmentation of the debris.

Both these hypotheses have important implications for application to the severe accidents. Presuming a BWRSAR-type melt progression, the early pour of debris from the vessel would be metallic. In this case stabilization of the gas film around the debris could prevent a large mass of molten material from participating in a steam explosion. On the other hand, the superheat associated with a large oxidic melt is typically less than a few hundred degrees. Therefore, it is likely that the surface of the debris droplets would freeze. This would slow the heat transfer to the coolant and a steam explosion would not occur.

19EB.2.2 Experiments With a Stratified System

In some of the recent experiments performed to examine core concrete interaction, water has been added to the debris. As discussed in Subsection 19EB.1.1, the probability of a large amount of water in the lower drywell at the time of vessel failure is very small. After core debris is introduced to the lower drywell, it is flooded either by active systems or the Passive Lower Drywell Flooding System. Therefore, this is the most probable initial configuration for an FCI event in the ABWR.

Far fewer experiments have been performed in this stratified geometry than in the configuration of debris poured into water. Work by Bang and Corradini (Reference 19EB-7) used triggered Freon / Water and Liquid Nitrogen / Water Systems. In these studies the interaction zone for the vapor explosion is less than 1 cm thick.

Assuming this depth is representative of reactor material, this would lead to the conclusion that less 3% of the ABWR core inventory could participate in an FCI event.

Prototypic materials have been used in a few core-concrete interaction experiments in which water is added to molten debris. The MACE and WETCOR tests added water to a pre-existing pool of debris. These tests involved fairly large masses of molten simulant to which water was added. Thus, the initial condition is a stratified pool in which water lies over the core debris. The materials and masses of the experiments are summarized in Table 19EB-1. No energetic fuel coolant interactions were observed to occur in the stratified configuration. The experiments typically indicated an early heat transfer phase in which the heat fluxes were on the order of 1.5 to 2 MW/m^2. Later, presumably after the formation of a crust above the molten debris pool, the heat fluxes decreased. These heat fluxes are considered in Subsection 19EB.6.2 in bounding the non-explosive steam generation rates.

19EB.2.3 BETA V6.1

Recently, an energetic FCI occurred in the BETA facility. Experiment V6.1 was intended to represent the Bibilus reactors. These reactors have an annular pool of water around the pedestal cavity. BETA V6.1 was designed to determine the impact of these water pools on corium concrete interaction. The configuration of V6.1 is shown in Figure 19EB-1. The system consisted of a concrete crucible with an annular water pool which was vented back to the inner crucible via a small path. Molten iron alumina thermite was introduced into the cavity which was then allowed to ablate.

The debris eroded the concrete in the approximate shape shown in Figure 19EB-1. The superheat of the melt was very high since there was no water on the debris. Eventually, the sideward erosion caused the debris to reach the annular water pool at one local point. Instants later an explosion occurred. The bottom of the crucible was sheared off. There was severe damage to the facility. All of the instrumentation was destroyed and the melt injector was thrown several meters up, damaging the ceiling.

The energy required to do the damage has not yet been determined. However, the structure surrounding the test facility was fairly weak, unprotected sheet metal. Although the doors were blown open they were not damaged. Therefore, it is believed that the pressure spike may not have been very large.

The symmetry of the damage to the crucible indicates that the explosion was very symmetric. There was very little irregularity in the shearing of the bottom of the crucible. Thus, it is difficult to believe that the explosion began on one side of the crucible and propagated sideward. An alternate hypothesis has been proposed (Reference 19EB-8). When the debris penetrated to the annular pool, the steam generation rate increased. Since the annular compartment vents back to the center of the crucible via a small line, the pressure increased and water was forced back into the debris. The debris was still highly superheated at this time. The confinement of the system allowed for intermixing of the debris and water and prevented the pressure from being relieved. Thus, the damage caused to the system was not a result of a shock wave, but rather due to simple pressurization of a confined region.

The steam explosion observed in the BETA facility is not applicable to the ABWR system. Although suppression pool and vent system of the ABWR is located in an annulus around the lower drywell, there is adequate vent area to relieve the pressure in the wetwell drywell connecting vents. In fact, the BETA configuration is also much more restrictive than the Bibilus reactor it was intended to represent. This restrictive condition resulted in ingression of water into the melt. Since the ABWR configuration has much more vent area, water ingression will not occur.

Additionally, there was no water on top of the debris before penetration into the annulus. Thus, the molten debris in V6.1 was highly superheated. This is contrasted to the situation in the ABWR. The ability to use active systems, such as the Firewater Addition System, and the presence of the passive lower drywell flooder virtually ensure that there will be water above the debris in the ABWR. The area of the ABWR lower drywell is also very large which enhances coolability. The uncertainty analysis of Attachment 19EC indicates there is a low probability that significant core concrete attack will occur. Therefore, the initial contact mode observed in V6.1 is unlikely.

Even if CCI occurs and the pedestal is eroded to the wetwell drywell connecting vents. The presence of water above the debris will cause a crust to form. The temperature on the lower surface of the crust will be at the melt point of the debris. Within any molten region, the debris temperature will be nearly equal to the melt temperature due to convection in the debris pool. Thus, the addition of any water to the molten pool will cause the debris to freeze and a steam explosion will not occur.

The conditions which led to the explosion at the BETA facility are not prototypic of the ABWR. Due to operation of the flooder there is a small likelihood that the debris will ablate the side wall and enter the wetwell drywell connecting vents. This is examined in Attachment 19EC. Even if the debris does penetrate the pedestal to the connecting vents, the vent area in the ABWR is sufficient to relieve the steam generation caused by the initial contact of water and debris. Thus, water would not be forced into the melt as occurred at BETA. Finally, the superheat of the melt at the BETA facility was very high, whereas the superheat of any debris which contacted water in the ABWR would be low. Thus, debris would be easily solidified, reducing the heat transfer to the water and preventing rapid steam generation. Thus, the explosion in V6.1 does not indicate that containment damage will occur in the ABWR as a result of FCI.

19EB.2.4 High-pressure Melt Ejection Experiments

Sandia performed a series of experiments to examine the influence of water pools on the behavior of high-pressure melts in a Zion-like cavity (Reference 19EB-9). Two configurations were examined. In the SPIT-15 test debris was injected into a closed acrylic box. This allowed for visualization of the phenomena. In the SPIT-17 and HIPS experiments a Zion-like cavity was constructed. The basic configuration of the SPIT-17 and HIPS experiments is shown in Figure 19EB-2. The SPIT-17 cavity was made of aluminum while the HIPS experiments used reinforced concrete cavities.

In all of the experiments water was present in the cavity at the time of melt ejection. The inertia of the water prevented venting of the cavity. Thus, the steam generation in the cavity forced the region to pressurize and the structures were destroyed before gas flow from the end of the structure could relieve the pressure in the cavity.

It is interesting to compare these experiments to BETA V6.1. In both instances it appears that large pressure spikes were created when the debris and water were tightly confined. This early confinement keeps the water and debris in close contact, and seems to lead to the fragmentation of the hot molten material which is a necessary precondition for steam explosions.

The results of this experiment are not applicable to the ABWR configuration. The lower drywell is not initially full of water and there is ample venting of the region. The extreme damage observed in these experiments appear to be consistent with that in BETA V6.1, both in the mode and magnitude of the damage to the facilities.

19EB.3 Explosive Steam Generation

This subsection presents a bounding analysis of the maximum steam generation rate which can occur for a given mass of corium interacting with water.

19EB.3.1 Phenomenology

Corium interactions with water can result in rapid steam generation. The rate of steam generation can be limited by the amount of corium or water present. Maximum generation for a given amount of corium occurs when enough water is present to completely quench the corium. Corium mass, surface area, temperature and heat transfer coefficient dictate the maximum rate when ample water is available.

Two configurations are possible for quenching in the ABWR. First, corium can exit the vessel when the lower drywell contains significant amounts of water. Corium exit from the vessel can be either by a slow pour (small vessel breach) or by a sudden drop (catastrophic failure of lower vessel head). Second, corium can enter a dry lower drywell and form a pool. Subsequently, the lower drywell is flooded with water and the debris is quenched. This situation, commonly referred to as a stratified geometry steam explosion, is the expected configuration for any large FCI in the ABWR.

Molten core debris is expected to be discharged from the vessel close to its liquidus temperature, 2600 K. Therefore, the maximum temperature in either the pour or stratified geometries will be 2600 K. The actual temperature will be lower due to heat loss by the debris prior to interaction with water. In the pour case, corium will transfer heat to the air surrounding the vessel as it falls. Any residual water in the lower drywell, as well as concrete beneath and air above the debris pool will absorb heat in the stratified geometry.

For rapid steam generation to occur in either situation, the ejected corium must break up into small particles. The analysis presented in Subsection 19E.2.3.1.4 demonstrated that corium breakup in the ABWR will be driven by Taylor instabilities. The smallest particles formed will be approximately 2.5 mm based on the Taylor critical wavelength. Debris breakup in the stratified geometry will also be governed by Taylor instabilities.

Crust formation will hinder debris breakup. Since corium is expected to exit the vessel near its liquidus temperature, any heat loss should contribute to crust formation. Furthermore, the outer debris surface will freeze rapidly after encountering water. Freezing will hinder further droplet division because more energy will be required to fracture the outer crust than it does to overcome the liquid surface tension. This, in part, explains why self-triggering can be observed with some highly superheated metals, but is much less likely with molten core debris.

19EB.3.2 Bounding Analysis

Moody, et al., (Reference 19EB-10) determined the maximum steam generation rate during FCI based on a simplified thermal-hydraulic methodology. The steam formation rate from a single corium droplet assuming heat transfer to saturated water is:

$$\dot{m}_{g,\,d} = \frac{HA_d(T_{ci} - T_{\infty})}{h_{fg}} e^{-t/\tau_h} \qquad (19EB\text{-}1)$$

where:

$\dot{m}_{g,\,d}$ = steam formation rate,

H = heat transfer coefficient,

A_d = surface area of a corium droplet,

T_{ci} = droplet surface temperature,

T_∞ = saturation temperature of water at the ambient pressure,

h_{fg} = latent heat of vaporization for water,

t = time from beginning of interaction,

τ_h = thermal response time.

Heat transfer from the droplet to the surrounding is dominated by convection and radiation. The heat transfer coefficient is:

$$\begin{aligned} H &= H_c + H_r \\ &= H_c + \frac{\sigma\left(T_{ci}^4 - T_\infty^4\right)}{(T_{ci} - T_\infty)}\varepsilon \end{aligned} \tag{19EB-2}$$

where:

H_c = convective heat transfer coefficient,

H_r = radiative heat transfer coefficient,

σ = Stefan-Boltzmann constant,

ε = emissivity of the droplet.

Due to the high temperature of corium, convective heat transfer from the surface of the particle will be in film boiling regime. The maximum convective heat transfer coefficient that can be expected is that of enhanced film boiling, which is 390 W/m^2K. The emissivity suggested for use in MAAP (Reference 19EB-11) for corium is 0.85. This value will be used for this analysis.

If a mass of corium, M_c, interacts with water and breaks up into droplets of average radius, r, the number of droplets, N, will be given by:

$$N\left(\frac{4}{3}\pi r^3\right) = \frac{M_c}{\rho_c} \tag{19EB-3}$$

where:

ρ_c = density of corium.

The total steam generation rate of N corium droplets is:

$$\dot{m}_g = N\dot{m}_{g,\,d} = \dot{m}_{g,\,Max}e^{-t/\tau_h} \tag{19EB-4}$$

where the maximum generation rate is:

$$\dot{m}_{g,\,Max} = \frac{3M_cH(T_{ci} - T_\infty)}{\rho_c h_{fg} r} \tag{19EB-5}$$

This is the maximum steam generation rate that can occur for a given amount of corium broken up into small droplets in a large body of saturated water.

19EB.4 Impulse Loads

Rapid steam generation can produce a shock wave which imparts impulse loads to containment structures. Energetic FCIs, however unlikely, may occur in the lower drywell of the ABWR. Water in the lower drywell, which must be present for rapid steam generation, can transmit shock waves from the site of FCI to the walls of the pedestal. Shock waves which pass into the gas space above the water will be rapidly damped due to gas compressibility and will not represent any threat to containment integrity. If the impulse load is large enough, the pedestal will fail causing the vessel to tip. Tipping of the vessel would most likely lead to tearing of the containment penetrations. The scoping analysis presented in this subsection estimates the amount of corium which can participate in a FCI without exceeding the impulse load capability of the pedestal.

19EB.4.1 Maximum Impulse Pressure

Moody, et. al., (Reference 19EB-10) determined the maximum pressure increase at the site of an FCI based on the steam generation rate given in Equation 19EB-5. His analysis applied the Rayleigh bubble equation to a single steam bubble with an equivalent volume of the many bubbles formed during interaction with N corium droplets of radius, r. Because the volume varies as r^3, this results in overestimation of the rate of bubble expansion. The bubble expansion rate dictates the pressure rise. Therefore, this analysis bounds the pressure generated by the maximum steam generation during FCI.

The maximum pressure increase of a single submerged steam bubble above the ambient pressure during its formation at the generation rate given in Equation 19EB-5 is:

$$\Delta P_{Max} = 0.178\left[\rho_1\frac{\left(R_gT_\infty\dot{m}_{g,\,Max}\right)^2}{R_o^4}\right]^{1/3} \tag{19EB-6}$$

where:

ρ_1	=	density of saturated water at the ambient pressure,
R_g	=	gas constant for steam,
R_o	=	starting radius for steam bubble growth.

The starting radius for bubble growth can be estimated by a spherical volume equal to the corium volume plus the total volume of water it vaporizes which in equation form is:

$$\frac{4}{3}\pi R_o^3 = \frac{M_c}{\rho_c} + \frac{M_c c_c (T_{ci} - T_\infty)}{h_{fg}\rho_1} \tag{19EB-7}$$

where:

ρ_c	=	density of corium,
c_c	=	specific heat of corium.

The maximum pressure predicted by Equation 19EB-6 is shown in Figure 19EB-3 for participating corium masses from 0 to 30,000 Kg. The required corium properties were taken from Table 19E.2-17. The steam and water properties are saturated conditions at two atmospheres. Two atmospheres is a likely containment pressure at vessel failure for the ABWR.

The peak pressure during impulse loading of the ABWR pedestal resulting from fuel coolant interactions should be bounded by the pressure shown in Figure 19EB-3. The pressure predicted by Equation 19EB-6 is conservative because of the assumptions which went into its creation. Furthermore, this is the pressure at the site of FCI. The pressure experienced by the pedestal wall will be reduced because the shock wave has to pass through some amount of water before it impinges on the wall. The pressure will decay as r^{-2} as it moves away from the source (Reference 19EB-12).

19EB.4.1.1 Impact of FMCRD Platform Grating (on FCI)

The FMCRD platform grating is located in the lower drywell at the elevation of the access tunnel. This rotating platform is circular and mounted on the rotating rail under the reactor vessel. There is an opening area at the center of the platform which is provided with a traveling rail for the CRD handling device. Gratings will be installed on both sides of the rail for maintenance personnel. Typically, the grating consists of 2.54 cm (1-inch) by 0.95 cm (3/8-inch) metal slats mounted edge-wise to form a grid with a grid size on the order of 2.54 cm (1 inch) by 5.08 cm (2 inch).

The presence of the grating could provide some increased fragmentation of the debris as the leading edge of the debris enters a pre-existing water pool. This will tend to increase the voiding of the pool. Because the structure of the grating is very open, there will be no significant limitations on the venting of steam generated below the grating. Increased voiding in the water pool will reduce the impulse loading from an FCI. This in turn will decrease the potential for early containment failure from FCI.

The grating will be ablated as the debris passes through it, in the same manner as the ablation of the bottom head. Therefore, the grating will have no impact on the severe accident performance after the initial debris relocation. Any late debris relocation would be a slow drip-like relocation which would fall straight through the ablated region of the platform.

19EB.4.2 Impulse Duration

The main difference between energetic fuel coolant interactions (steam explosions) and non-energetic interactions is the time in which the energy stored in the corium is transferred to the coolant. Short transfer times, on the order of milliseconds, indicate explosive reactions. Longer times are indicative of non-energetic interactions. Several fuel coolant interaction experiments involving corium simulants were reviewed in Subsection 19EB.2.2. Pulse widths were observed to be of the order 5 milliseconds or less for FCI.

19EB.4.3 Pedestal Capability

Detailed calculations of the capability of the ABWR pedestal to withstand impulse loading have not been performed. However, a simple elastic-plastic calculation can provide a capability which can be used for scoping analysis. This estimate can be compared to the maximum pressure expected during a FCI for a given amount of participating corium and the impulse duration. The pedestal in Grand Gulf (MARK III containment) was analyzed in NUREG-1150 (Reference 19EB-13) with regards to its ability to withstand pressure spikes generated by steam explosions. Since the ABWR pedestal is expected to be at least as strong as that of a MARK III, the impulse capability of the Grand Gulf pedestal can also be used for comparison.

19EB.4.3.1 Elastic-Plastic Calculation

A failure limit estimate based on a simple elastic-plastic calculation has been performed by Corradini (Reference 19EB-12). The assumptions made in this analysis are:

(1) The pedestal wall is thin compared to its diameter,

(2) The pressure loading is uniform both spatially and temporally,

(3) Failure is based on a strain criteria of μ (failure strain / yield strain) equal to 10,

(4) The pedestal wall is considered to be free standing.

The resistance to deformation, R_m, of the pedestal is:

$$R_m = \frac{\sigma_y \Delta_w}{R_w} \tag{19EB-8}$$

where:

σ_y	=	yield stress of the pedestal wall,
Δ_w	=	thickness of the pedestal wall,
R_w	=	radius of curvature of the wall.

The natural period of the pedestal, T, can be calculated from:

$$T = 2\pi\sqrt{\frac{\rho_w R_w^2}{E_w}} \tag{19EB-9}$$

where:

ρ_w	=	wall density,
E_w	=	Young's Modulus of the pedestal.

Since the pedestal is a composite structure, the determination of each of these parameters can be quite complicated. A conservative estimate of the resistance to deformation and the natural period can be obtained by using the following parameters:

σ_y	=	175 MPa (value for the A441 steel plates which define the boundaries of the pedestal),
Δ_w	=	6 cm (total thickness of the two A441 steel plates which define the boundaries of the pedestal, ignores steel webs and concrete fill),
R_w	=	6.15 m (average radius of the pedestal),
ρ_w	=	2,400 kg/m^3 (density of concrete fill between steel plates),

E_w = 200 GPa (typical value of steel).

Using these parameters yields: R_m = 1.7 MPa and T = 4.2 milliseconds.

The maximum response of elastic-plastic one-degree systems (undamped) due to rectangular load pulses is shown in Figure 19EB-4. The ratio of pulse duration, t_d, to natural period is the horizontal axis. The strain criteria, μ, forms the vertical axis. The relationship between these two axis parameters is given by a series of curves defined by the ratio of resistance to deformation, R_m, to the average pressure of an impulse, F_1. The amplitude of the square pulse can be conservatively estimated by the maximum pressure rise expected during a FCI, ΔP_{Max}, which is calculated in Subsection 19EB.4.1.

As discussed previously, the impulse duration of a FCI is expected to be approximately 5 milliseconds (Subsection 19EB.2.1). The ratio of t_d/T for this duration is 1.2. Using this ratio and a strain criteria of 10 yields a R_m/F_1 of approximately 1.0. This implies that the pedestal can withstand a ΔP_{Max} of 1.7 MPa.

The maximum ratio of R_m/F_1 in Figure 19EB-4 is 2.0. Using this ratio, the lower limit of the pedestal capability is estimated to be 0.85 MPa. The uncertainty in pulse duration (assumed to be 5 milliseconds) is irrelevant for the maximum ratio of R_m/F_1 because it is obtained for pulse durations much greater than the natural period of the pedestal.

This simple elastic-plastic calculation predicts that the pedestal can withstand a maximum pressure during a fuel coolant interaction of 1.7 MPa and that the conservative lower limit of the pedestal capability is 0.85 MPa. The amount of corium which must participate in a FCI to achieve this lower limit can be obtained from the analysis presented in Subsection 19EB.4.1 and summarized in Figure 19EB-3. The amount is 22,400 kg. The ABWR contains 235,000 kg of corium. Therefore, the ABWR pedestal can withstand a FCI involving 9.5% of the corium inventory.

19EB.4.3.2 Comparison to NUREG-1150 Grand Gulf Pedestal

The ability of the Grand Gulf pedestal to withstand steam explosions was considered in NUREG-1150 (Reference 19EB-13). The smallest impulse load expected to fail the pedestal was reported to be 0.024 MPa•s. This limit can be used for comparison to the ABWR because the ABWR pedestal is expected to be sturdier than that of a MARK III. For a pulse duration of 5 milliseconds, this impulse corresponds to a square wave pressure of 4.8 MPa. This value is significantly higher than the pressure predicted by the elastic-plastic scoping analysis. Alternatively, the lower pressure limit predicted by the elastic-plastic analysis (0.85 MPa) can be applied for 28 milliseconds before an impulse load of 0.024 MPa•s is exceeded. Both of these comparisons imply that the elastic-plastic analysis bounds the impulse load required to fail the pedestal.

19EB.4.4 Capability of the ABWR to Withstand Pressure Impulse

The ABWR pedestal has been shown in this scoping analysis to be capable of withstanding a peak pressure of at least 0.85 MPa during a steam explosion. The amount of corium required to produce this pressure impulse during a fuel coolant interaction was shown to be 22,400 kg. This represents 9.5% of the ABWR corium inventory. This is more than three times the maximum amount of debris which could participate in an FCI event based on the observations discussed in Subsection 19EB.2.2. Therefore, the ABWR pedestal is very resistant to the impulse loading which could occur in a severe accident. This failure mechanism need not be considered further in the containment event trees or the uncertainty analysis.

19EB.5 Water Missiles

Submerged steam formation resulting from fuel coolant interactions can be rapid enough to propel an overlying liquid mass. Impact loads can be imparted to containment structures if the liquid mass (water missile) is ejected from the water pool with a great enough velocity. Although a prediction of impact by a water missile does not imply damage, additional analysis would be needed to assess the structural response. The maximum height to which a water missile can rise will be determined in this subsection for a given amount of participating corium. The rise height will be compared to the distance between the expected water surface of a pre-flooded lower drywell and the bottom of the reactor vessel to determine if damage to the containment could occur. No other structures are considered because damage to them will not lead to containment failure.

19EB.5.1 Maximum Rise Height

Moody, et. al., (Reference 19EB-10) used the steam generation rate determined in Subsection 19EB.3.2 to predict the upward propulsion velocity and elevation characteristic of a water missile. The maximum velocity that a water missile can obtain is the maximum radial expansion rate of the steam bubble formed during FCI. This expansion rate is:

$$\dot{R}_\infty = \frac{3}{5}\left[\frac{5}{2}\frac{R_g T_\infty \dot{m}_{g,\,Max}}{4\pi\rho_1 R_\infty^2}\right]^{1/3} \qquad (19EB\text{-}10)$$

where:

R_∞ = equilibrium steam bubble radius.

R_∞ is equal to:

$$R_{\infty} = \left[\frac{3}{4\pi}\frac{M_c c_c (T_{ci} - T_{\infty})}{h_{fg}\rho_g}\right]^{1/3} \quad (19EB\text{-}11)$$

where:

ρ_g = vapor density.

Balancing the kinetic and potential energies of a water missile yields:

$$\Delta y_{Max} = \frac{\dot{R}_{\infty}^2}{2g} \quad (19EB\text{-}12)$$

where:

Δy_{Max} = maximum rise height a missile will rise above the water surface,

g = acceleration of gravity.

Maximum missile rise heights are presented in Figure 19EB-5 for participating corium masses of 0 to 30,000 kg.

19EB.5.2 Available Rise Height

The water level in the lower drywell will not be greater than suppression pool water level during a severe accident. The normal water level of the suppression pool is 6.10 meters below the bottom of the reactor vessel. Consequently, a water missile can rise approximately six meters before encountering any structure the damage of which could lead to containment failure.

19EB.5.3 Capability of ABWR to Withstand Water Missiles

The amount of corium which can participate in a FCI in the ABWR and not generate a pressure impulse which is expected to fail the containment is 22.4 Mg. This amount of corium will produce a water missile which will rise 1.75 meters (Figure 19EB-5). This rise height is significantly lower than the available rise height of 6 meters. Therefore, the pedestal will fail from impulse loading before the required amount of corium participates to elevate a water missile even to the bottom of the reactor vessel. For this reason, water missiles are not expected to play a role in determining if the ABWR containment fails due to fuel coolant interactions.

19EB.6 Containment Overpressurization

The final element of this study focuses on the pressurization of the containment which may occur during periods of rapid steam generation which may occur when corium is being quenched. In the highly unlikely event of an ABWR core melt which leads to vessel failure, the corium will fall into the lower drywell. There are ten connecting vents which join the lower drywell, the upper drywell and the wetwell, as shown in Figure 19EB-6. The pressure suppression containment prevents large increases in containment pressure by sparging the steam through the connecting vents to the suppression pool which condenses the steam. However, if the pressure rise is extremely rapid, the vents may not be able to clear before the containment is damaged. At even higher steam generation rates, the area from the lower drywell to the upper drywell could be too small and a pressure difference between the drywell regions could occur, failing the lower drywell. This analysis determines the steam generation rates for different limits on FCI. The maximum rate is then compared to the containment pressure capability to assess the potential for containment damage as a result of overpressure during an FCI event.

19EB.6.1 Methodology

This calculation compares the pressurization due to rapid quenching of corium to the pressure capability of the containment. Two non-explosive steam generation limits are considered. If there is a sufficiently large water mass, then the quenching of corium will provide the steam generation limit. If the mass of water limits the steam spike then the steam generation will be less than, or equal to, the water flow into the lower drywell. The impulse pressure limited mass, calculated in Subsection 19EB.3.1, is also considered.

If there is no water in the lower drywell at the time of vessel failure, then the maximum rate of steam generation at some later point in time is the rate at which water is introduced into the lower drywell. If there is still water in the lower plenum at the time of vessel failure, as predicted by MAAP-ABWR, then this source of water could react with the corium in the lower drywell. Water addition could also occur via the passive flooder, the use of the Firewater Addition System or by means of ECCS recovery. Each of these possibilities will be examined to determine the maximum rate at which water could be added to the lower drywell.

For most of the core melt sequences in the ABWR PRA there will not be water in the lower drywell at the time of vessel failure (Subsection 19EB.1.1). Nonetheless, an evaluation will be performed assuming that corium falls into a pre-existing pool of water and is quenched instantaneously. This will provide a limit on the peak containment pressure which could result from quenching of debris as it falls into the lower drywell. For the ABWR, the majority of sequences with vessel failure occur at low pressure. Therefore, gravity is the driving force for the flow of corium from the lower head of the vessel to the lower drywell. Both MELCOR and MAAP predict that the vessel fails at the penetrations for low pressure melts. After the initial hole is formed, the hole ablates due to the flow of hot corium. In order to determine the sensitivity of the ABWR containment to rapid steam generation 40% of the total UO_2 mass is assumed to be molten at the time of vessel failure. This value is consistent with the upper limit for molten debris used in the uncertainty analyses for direct containment heating (Subsection 19EA.2.1.4).

Two potential limits for pressurization due to steam generation are considered. First, the pressurization of the lower drywell is determined considering the limit of the vent area from the lower drywell to the upper drywell. This determines any limits for the assumption that the upper and lower drywell regions have good communication and will respond similarly to the pressurization. Second, the response of the Pressure Suppression System is evaluated. Drywell pressurization rates are used to determine the vent clearing response which is in turn used to determine the peak containment pressure as a function of the pressurization rate.

19EB.6.2 Maximum Steam Generation Rates

The first step in determining the peak pressures that may result from fuel coolant interactions is to determine the maximum steam generation rates. The steam generation can be limited either by the available water or the available corium. Both of these possibilities will be considered separately.

19EB.6.2.1 Water Added to Debris

There are four potential sources of water addition to the lower drywell. First, in a MAAP-type core melt progression, there may be water in the lower plenum at the time of vessel failure. After the corium falls into the lower drywell, the water will follow through the ablated hole in the lower plenum. Second, the lower drywell passive flooder opens when its fusible material melts. Water from the wetwell is then driven by gravity into the lower drywell. Third, the Firewater System may be used to add water to either the vessel or the upper drywell. In either case, water will eventually flow into the lower drywell at the firewater injection rate. Finally, if the ECCS is recovered, these systems could be used to inject water into the vessel which again will flow into the lower drywell.

19EB.6.2.1.1 Water Inventory from Lower Plenum

If there is water in the lower plenum at the time of vessel failure, then it will fall into the lower drywell after the corium. Under these conditions, the flow will be driven by gravity through the ablated vessel failure. The expected failure mode for a BWR is penetration failure (Reference 19EB-14). A parametric study was performed to determine the final, ablated area resulting from different numbers of CRD penetrations. The study was conducted by varying the number of vessel penetrations presumed to open at the time of vessel failure. Since this affects the initial area of the vessel failure, multiple penetration failures have higher initial debris pour rates. As seen in Figure 19EB-8, the final area varied from 0.06 m^2 for 10 penetrations failed to 0.08 m^2 for one penetration. The final area is smaller for cases with multiple penetration openings because the duration of the debris pour is shorter. In order to bound the flow of water into the lower plenum, a value of 0.1 m^2 is used which results in a maximum mass flow rate of 1020 kg/s.

19EB.6.2.1.2 Passive Flooder Flow

The passive flooder is composed of ten pipes connecting the lower drywell to the suppression pool with fusible material at the lower drywell end which opens when it reaches a specified temperature. This is shown schematically in Figure 19EB-7.

The flow from the wetwell into the lower drywell is driven by the difference in the water height, h, between the connecting vents and the flooder. The flow rate is given by:

$$\dot{m} = \rho A \sqrt{2gh} \qquad (19EB\text{-}13)$$

where:

$\dot{m}$	=	water mass flow into the lower drywell (kg/s),
ρ	=	density of water (kg/m^3),
A	=	total area of passive flooders (m^3),
g	=	acceleration of gravity (9.81 m/s^2),
h	=	driving head of water (m).

The maximum flow through the passive flooder would occur when the pressure difference between the wetwell and the drywell was sufficient to open the vacuum breakers, and the suppression pool is cold. Assuming a suppression pool temperature of 303.2 K (30°C), ρ = 996 kg/m^3. The total area of the passive flooders is A = 0.081 m^2. Assuming that the pool is at the high water level and the pressure difference between the wetwell and drywell is at the full open setpoint of the vacuum breakers, the height of water above the passive flooder is h = 4.75, which yields a maximum flow rate of $\dot{m}$ = 780 kg/s. The flow rate will typically be less than this maximum because the drywell pressure is greater than wetwell and the first row of vents will be clear.

19EB.6.2.1.3 ECCS and Firewater Flow

The ECCS and Firewater System are both capable of adding water to the vessel which would flow into the lower drywell. The Firewater System has a direct-drive diesel pump which does not rely on AC power, so it is available even during a station blackout event. The ECCS is dependent on AC power; and, thus, will not be available during station blackout but could inject water during recovery late in a severe accident. The ECCS System has a flow rate far greater than the Firewater System. Therefore, no determination of the firewater flow is necessary. The maximum ECCS flow will be bounded by the runout flow of the ECCS pumps. The actual flow will be somewhat smaller due to the flow losses at higher velocities when all of the pumps are operating simultaneously.

There are two HPCF Systems, each with a runout flow of 230 kg/s (3800 gpm), and three LPFL Systems with flow of 265 kg/s (4200 gpm). The RCIC System is not considered since the vessel will be depressurized. The total water addition rate to the lower drywell is 1250 kg/s.

19EB.6.2.2 Steam Generation Rate for Pre-flooded Lower Drywell

For the ABWR, it is very unlikely that there is water in the lower drywell at the time of vessel failure. Thus, steam generation is usually limited by the availability of water. However, there may be sequences for which there is ample water, and the limitation on the steam generation rate is the energy of the quenching corium. Thus, it is prudent to determine the maximum steam generation from this limit if there were a large water supply available. A large mass of water is assumed to be present in the lower drywell for this portion of the analysis.

A wide number of analyses have been performed to determine the mode of vessel failure. While there are still some uncertainties in the details of the analysis, the work performed to date provides overwhelming indication that a BWR vessel fails at the penetrations (References 19EB-15 and 19EB-16). Once there is some flow through a penetration, the molten material will begin to ablate the hole. Neglecting the change in the driving force for the flow of molten material, the maximum flow rate will occur when the hole size is maximized as the mass is exhausted.

In some MELCOR-type analyses, the corium quenches in the lower plenum of the vessel. It subsequently heats up and causes vessel failure. Therefore, there is little corium molten at the time of vessel failure. The flow rate of corium from the vessel is limited by the rate at which the corium melts in the vessel. Conversely, using a MAAP-type analysis, the corium does not quench in the lower plenum. Thus, there is a large molten mass at the time of vessel failure. Since this will result in larger flow rates than the MELCOR-type model, the MAAP results will be used to determine the corium flow rate for this analysis.

MAAP-ABWR (as well as MELCOR) uses the Pilch model for the ablation of the penetration (Reference 19EB-11). The velocity of the corium through the vessel failure is approximately constant; therefore, the ablation rate of the failure is linear. A series of MAAP-ABWR runs were performed which examined the flow rate of molten debris and vessel failure area as a function of the number of failed penetrations. The results of these calculations are shown in Figures 19EB-8 and 19EB-9. The maximum rate of debris ejection from the vessel is about 6000 kg/s. Assuming this material quenches as it is ejected, the steam generation rate is about 2800 kg/s.

The experimental heat flux observed when molten core debris simulants are poured into water is on the order of 1.5 to 2.0 MW/m^2 based on the floor area. Using the upper bound on the experimental observations, the maximum steam generation rate for the ABWR is 80 kg/s. This is far below the value determined above for the instantaneous quenching of debris for a bounding debris pour rate.

19EB.6.2.3 Explosive Steam Generation Rates

Based on the examination of the impulse loading calculation of Subsection 19EB.4.3.1, the ABWR can withstand the shock wave which corresponds to 22.4E3 kg of core debris. The maximum steam generation rate associated with this amount of debris is 4100 kg/s (Subsection 19EB.3.2).

19EB.6.2.4 Maximum Steam Generation

The maximum steam generation rates for each of the mechanisms described above are summarized in Table 19EB-2. Based on these results, the limiting scenario is the maximum steam explosion from the scoping study. Therefore, even though this event is far larger than the expected steam generation rate, the containment pressurization will be estimated using this value.

19EB.6.3 Containment Pressurization

The containment peak pressures may be calculated based on the flow rates determined above. The results given below are for the most restrictive pressurization rate. Three limits are considered. The first condition is the flow rate of steam from the lower drywell to the upper drywell. Second, the time period before the suppression pool vents open must be considered. Finally, the quasi-steady condition of flow from the drywell to the wetwell through the suppression pool is considered.

19EB.6.3.1 Drywell Connecting Vent Flow

Consideration of the flow through the drywell/wetwell connecting vents is important to ensure that there is adequate vent area to allow the upper and lower drywells to communicate freely. If the flow is restricted a significant pressure difference could exist between the upper and lower drywell regions. This could potentially result in lower drywell region failure. Using the maximum steam generation rate and an effective area of about 11.25 m^2 in the drywell/wetwell connecting vents, the pressure difference between the upper and lower drywell regions is less than 0.15 MPa.

19EB.6.3.2 Vent Clearing

If the drywell pressure is higher than the wetwell pressure at the time of the FCI, then steam flow to the wetwell can begin immediately. However, if the vents are not open, the pressure must accelerate the water in the vents to allow steam flow. During this interval the pressure in the drywell will rise quickly.

Assuming that the initial drywell and wetwell are at equal pressures maximizes the time for vent clearing. The time to vent clearing is calculated based on analysis by Moody (Reference 19EB-17). This model requires the pressurization rate for the drywell. The pressurization rate is determined by assuming a steam generation rate and using the ideal gas relationship for steam. The pressure rise in the drywell due to steam generation is then calculated using the pressurization rate and the time to vent clearing. Using the maximum steam flow rate, a pressure rise of 0.17 MPa is calculated.

19EB.6.3.3 Horizontal Vent Flow

After the vents have cleared, steam will begin to flow from the drywell to the suppression pool. The drywell pressure during this time is equal to the wetwell pressure plus the flow and water heads. Using conservative assumptions and the maximum steam flow rate, the drywell wetwell pressure difference is found to be 0.16 MPa.

19EB.6.4 Summary of Overpressurization Limits

Based on the calculations presented above, the maximum pressure rise in the lower drywell due to fuel coolant interactions occurs just before the wetwell / drywell connecting vents clear. At this time a pressure spike in the lower drywell of 0.17 MPa may occur. FCI events of the magnitude considered here occur when there is a large mass of unquenched debris which comes into sudden contact with water. In the ABWR this only occurs early in the course of a severe accident when the wetwell pressure is well below the COPS setpoint, typically at about 0.2 MPa. Even if the wetwell pressure were near the COPS setpoint of 0.72 MPa, the pressure difference between the drywell and wetwell would be equal to the design pressure of 0.27 MPa. There will be ample margin to the ultimate capability. Therefore, FCI leading to overpressurization failure of the lower drywell is not a credible event.

Concerning the upper drywell region, a conservative calculation based on the maximum steam generation rate given in Table 19EB-2 indicates that the maximum pressure in the upper drywell is the wetwell pressure plus 0.172 MPa. Again, considering that FCI events of the magnitude considered here occur when there is a large mass of unquenched debris which comes into sudden contact with water, the drywell will be well below even the service level C pressure of 0.72 MPa. Therefore, one would not expect upper drywell failure as a result of FCI.

The only FCI event one could hypothesize to occur late in the accident is the recovery of ECCS just before containment failure. However, in the ABWR design the passive flooder ensures that there is water above the debris. The addition of ECCS water will not cause increased heat transfer from the molten debris. Therefore, FCI leading to containment failure late in a severe accident has been ruled out by design.

The rapid steam generation rates which can occur due to bounding fuel coolant interactions do not lead to failure of the containment structure or opening of the rupture disk in the ABWR. Therefore, no further consideration of steam generation rates is required.

19EB.7 References

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Table 19EB-1 Core Concrete Interaction Tests with Water Addition to Debris

Experiment	Simulant	Debris Mass (kg)	Water Addition
MACE M0	UO_2 - ZrO_2 - Zr	130	Flooded after attack started
MACE M1	UO_2 - ZrO_2 - Zr	400	Flooded after attack started, upper crust was not fully molten
MACE M1B	UO_2 - ZrO_2 - Zr	400	Flooded after attack started, no crust above debris
WETCOR	Al_2O_3 - CaO	34	Water added at 1 liter/s

Table 19EB-2 Maximum Steam Generation for Steam Spikes

Water Limited Cases	
Flow from lower plenum at the time of vessel failure	1020 kg/s
Passive flooder	780 kg/s
Recovered ECCS	1250 kg/s
Debris Limited Case	
Debris falling into cavity is quenched instantaneously	2800 kg/s
Experimentally observed limit for debris poured into water	80 kg/s
Explosive Steam Generation	
Scoping result for shock wave capability	4100 kg/s

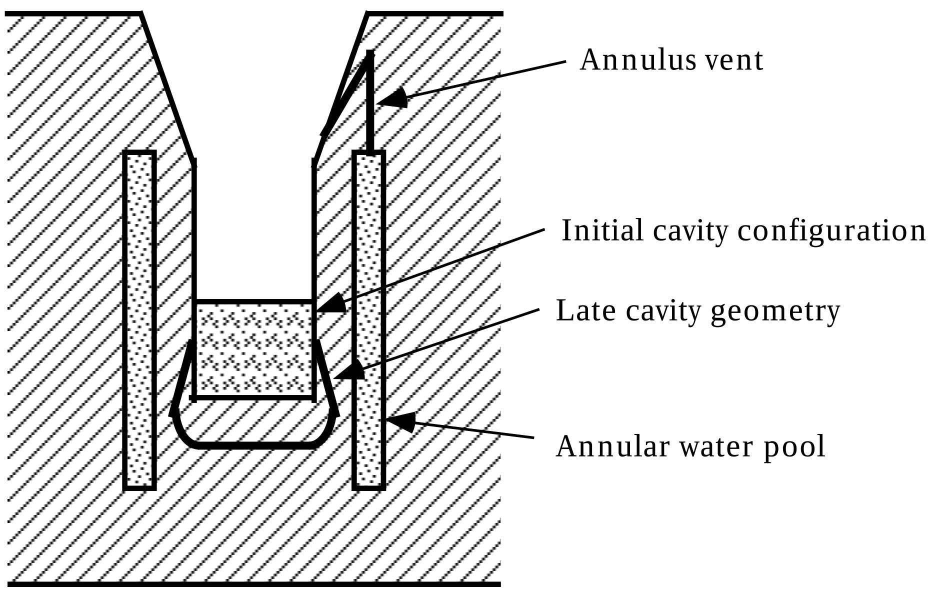

Figure 19EB-1 BETA V6.1 Configuration

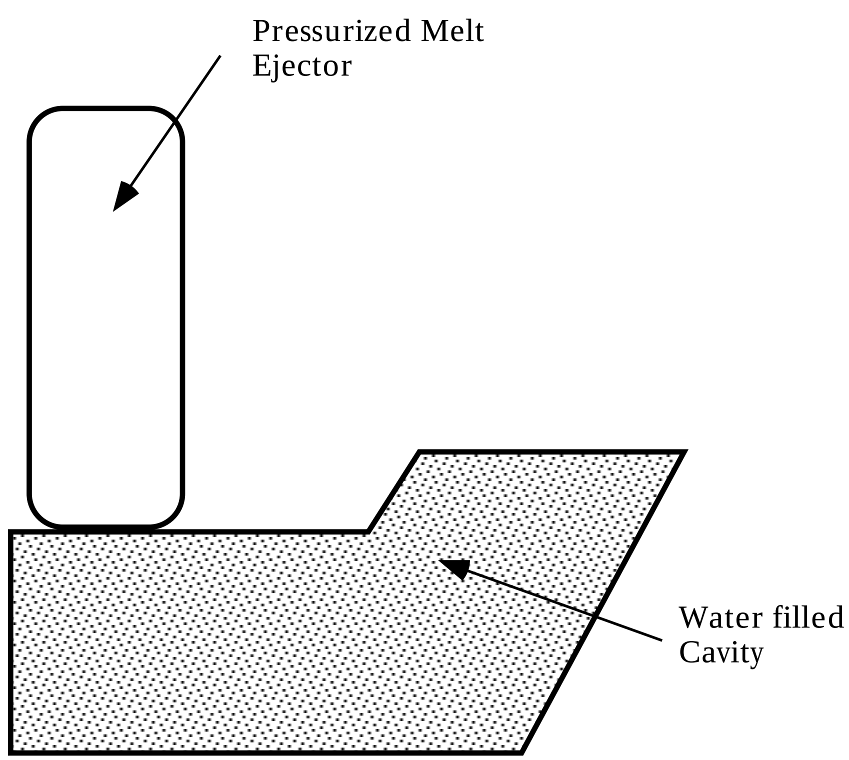

Figure 19EB-2 HIPS Experimental Configuration

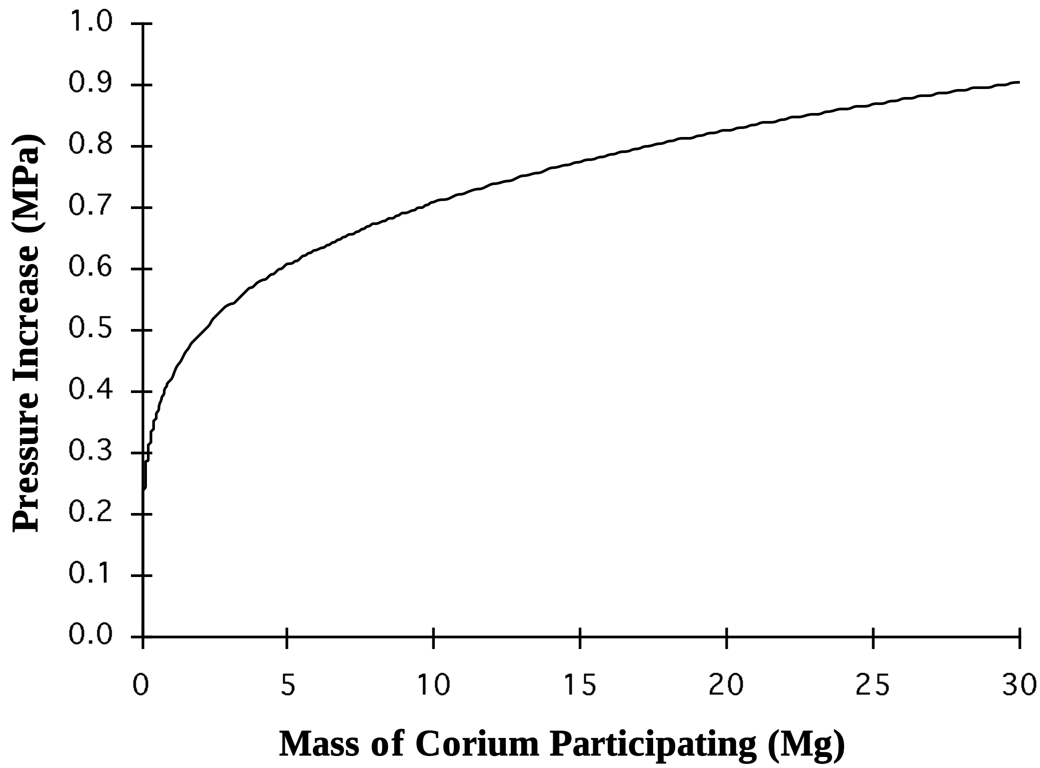

Figure 19EB-3 Peak Impulse Pressure from FCI

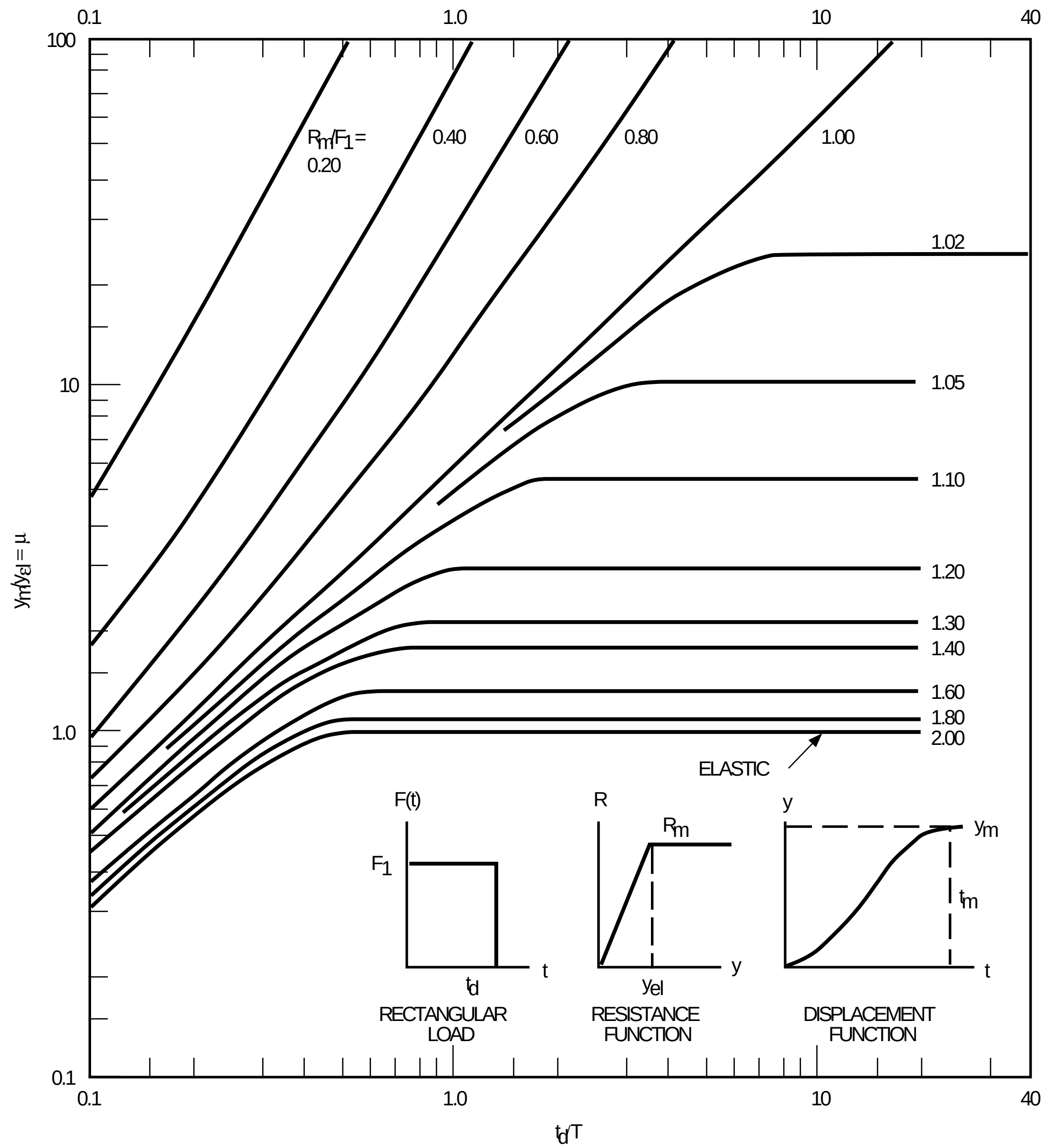

Figure 19EB-4 Maximum Response of Elastic-plastic One-degree Systems (Undamped) Due to Rectangular Load Pulses (Reference 19EB-18)

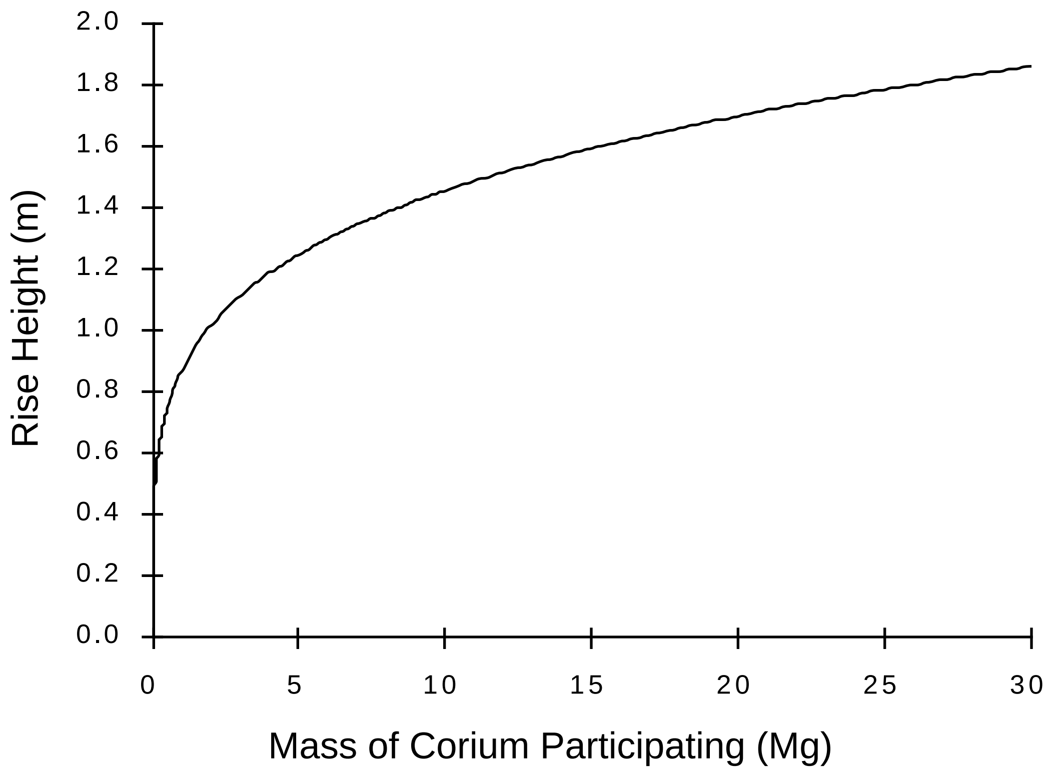

Figure 19EB-5 Rise Height of Water Missile

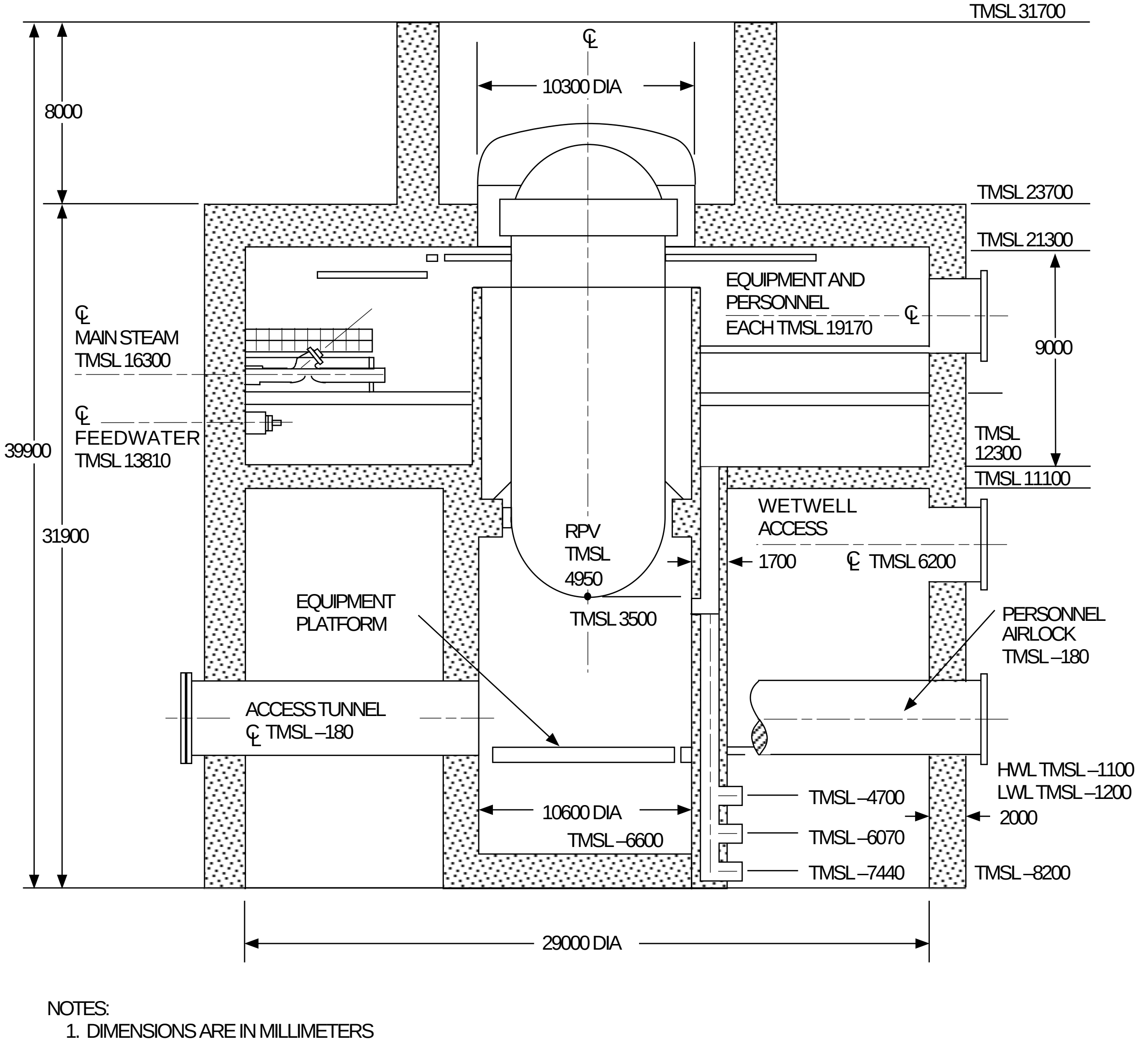

Figure 19EB-6 ABWR Containment Configuration

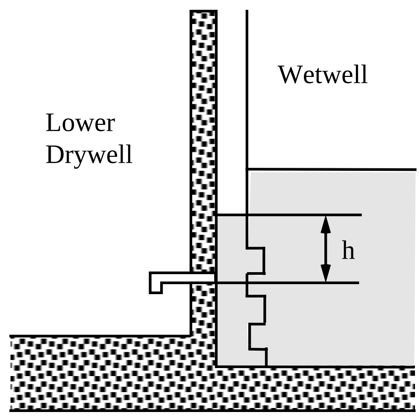

Figure 19EB-7 Pressure Head for Lower Drywell Flooder Flow

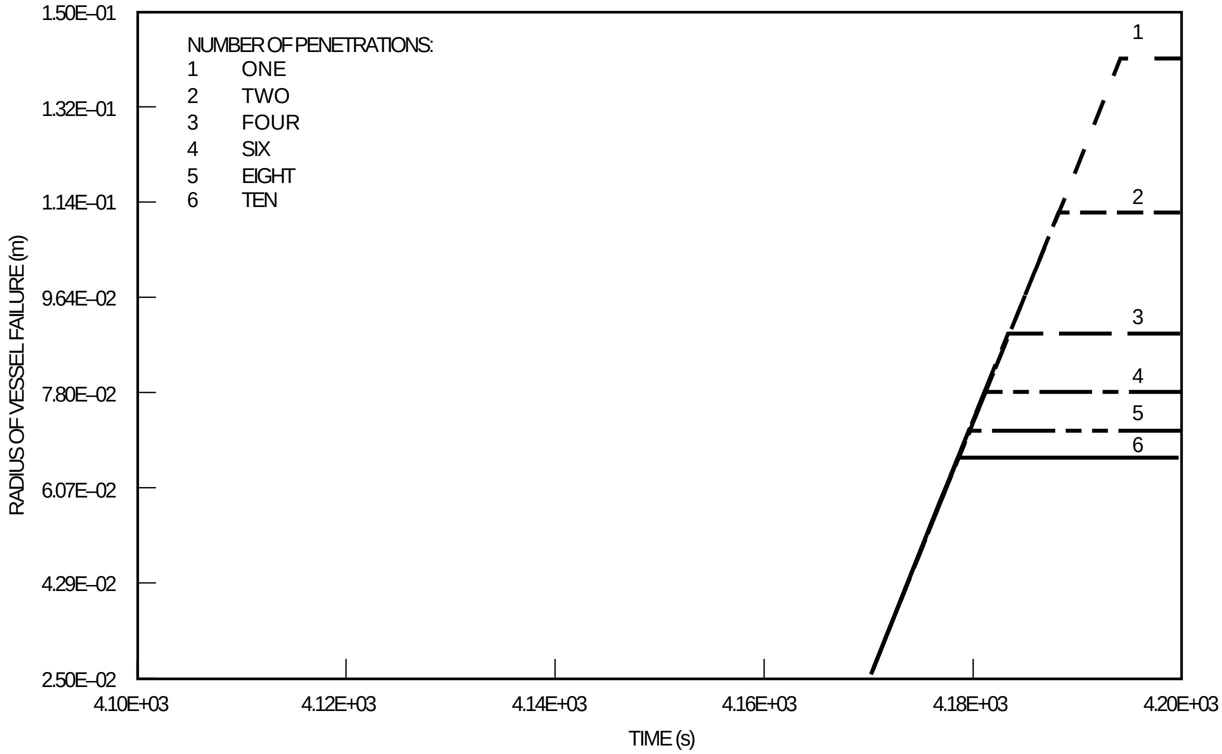

Figure 19EB-8 Ablated Radius of Vessel Failure

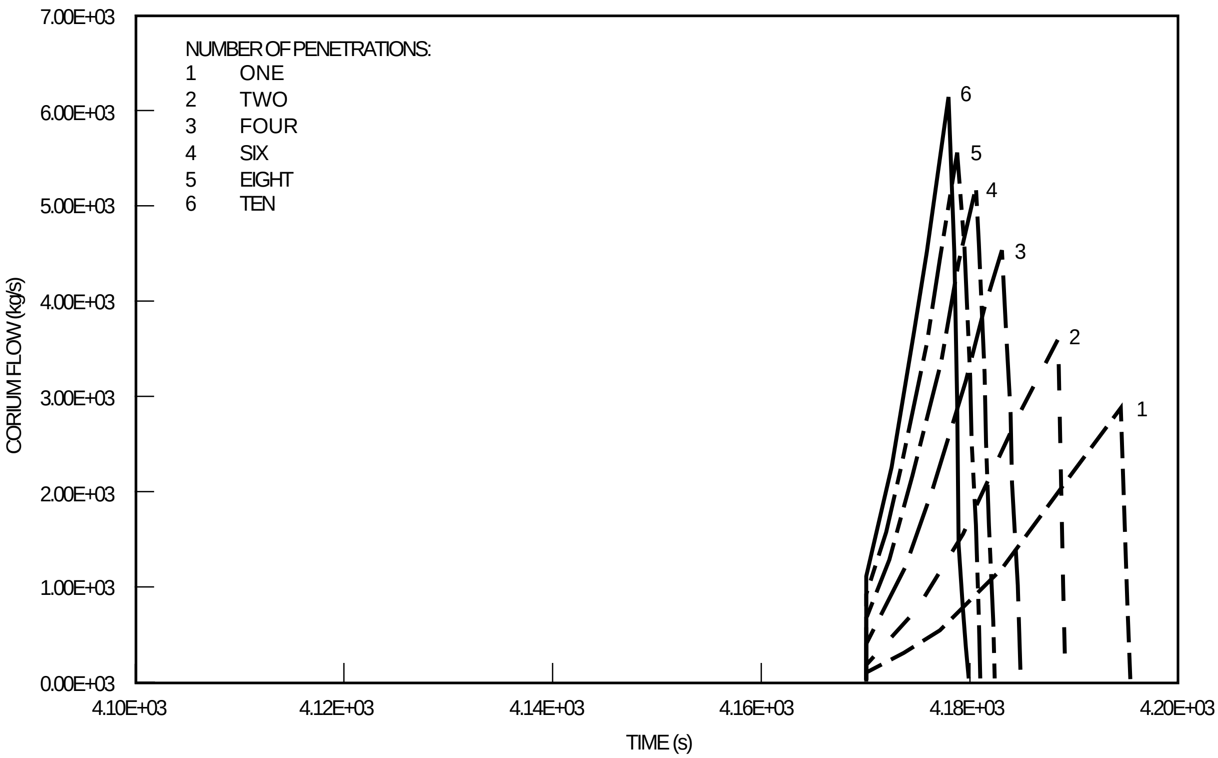

Figure 19EB-9 Mass Flow of Core Debris Through Vessel Failure

19EC Debris Coolability and Core Concrete Interaction

Appendix 19E of the ABWR PRA discusses core concrete interaction. In particular, in Subsection 19E.2.1.3.6, it is stated that the core debris will be quenched preventing substantial concrete ablation due to operation of the passive flooder. Even if the flooder was assumed to fail, water from the suppression pool would flood the lower drywell after 20.32 cm (8 inches) of radial ablation had occurred. This conclusion was based on available experimental information and the work performed in IDCOR Subtask 15.2 (Reference 19EC-1).

Since the original ABWR PRA was submitted there has been continued research in the areas of debris coolability and core concrete interaction. Recent experiments performed at Argonne as part of the MACE program have indicated that, due to crust formation, debris cooling may be limited. This section will investigate the uncertainties associated with debris coolability in the lower drywell of the ABWR. The investigation will begin with a look at applicable experimental data. Next, the issue of debris coolability will be decomposed into the controlling parameters and followed by the development of a decomposition event tree (DET). After creation of the DET, deterministic evaluations will be made to quantify the end points of the tree. Finally, sensitivities to key assumptions will be investigated.

19EC.1 Applicability of Experiments to ABWR

Several experiments have been carried out to investigate the influence of an overlying water pool on debris coolability. The critical parameter that appears to dominate the behavior in several of the experiments is the formation of a stable crust. This crust is found to prevent substantial water ingression and, therefore, debris cooling. The major criticism of these experiments is that, due to their small scale, a stable crust is preferentially formed. This limitation makes it quite difficult to extrapolate the results to a large reactor cavity. The MACE tests at Argonne have attempted to address this weakness by investigating larger cavity designs.

The following provides a brief summary of several debris coolability experiments.

(1) Theofanous and Saito - 1980 (Reference 19EC-2)

Experiments were performed with liquid nitrogen and water and liquid nitrogen and Freon 11. Crust formation was observed at low gas velocities but found to become unstable at high sparging rates. It was observed that as the gas velocity increased to a magnitude typical of core-concrete interaction, the heat transfer rate increased by a factor of ten. The heat transfer rates were found to approach those associated with critical heat flux.

(2) Greene - 1988 (Reference 19EC-3)

Tests were run with liquid metals with water and Freon R11. Gases were injected in the melt. It was observed that the water/melt interactions were generally unstable and that the upward heat transfer increased with gas velocity. The typical upward heat transfer rates were found to be 6 times greater than the classical Berenson correlation.

(3) FRAG (Reference 19EC-4)

This series of tests performed at Sandia National Laboratories (SNL) used 3 mm diameter steel spheres heated and placed in a 20 cm diameter concrete crucible. Tests were performed both with and without water addition. Both limestone and basaltic concrete types were investigated. The limestone tests showed that a stable crust made of concrete and steel formed that kept the water from penetrating the rest of the debris bed. The basaltic concrete allowed for some water penetration. The conclusion from these tests was that core-concrete attack continued even in the presence of water and that a substantial amount of steel oxidation took place.

(4) SWISS (Reference 19EC-5)

These tests, also performed at SNL, involved the interaction of molten steel on limestone concrete. The steel was heated at approximately five times the expected reactor decay heat levels. There appeared to be no violent melt-water interactions and the melt did not quench. There was a stable crust that was found to attach to the MgO sidewall. Typical upward heat flux was 800 kW/m^2. There was also information from the experiment that the overlying water pool provided substantial aerosol scrubbing (DFs of 10-30).

(5) Mark I Shell Failure Experiments (Reference 19EC-6)

Several experiments were carried out Fauske and Associates, Inc. to investigate the influence of water on debris coolability and specifically to observe drywell shell heatup. Iron-alumina thermite was discharged onto a concrete slab pre-flooded with water. The initial heat transfer was found to be quite high (20 times CHF) and leveled off at about 800 kW/m^2 later.

(6) MACE (Reference 19EC-7)

A series of large-scale experiments are being performed at Argonne National Laboratory investigating the coolability of molten-corium by water during its interaction with concrete. The MACE program has attempted to

(a) Employ prototypic corium melt materials

(b) Employ prototypic concrete types

(c) Obtain realistic melt temperatures

(d) Obtain realistic MCCI initial conditions

(e) Include prototypic chemical and internal heating, and by the increased size

(f) Ensure applicability to reactor cavities

In the scoping test, a high initial heat removal was observed. The crust that was formed was found to be supported by the electrodes. There were periodic melt eruptions through the crust that lead to substantial melt quenching. However, the melt did not completely cool and continued to erode concrete. One of the major difficulties with the test was that there were larger than prototypic heating rates.

The next test, M1, was performed on November 25, 1991. The major difficulty with this test was that not all of the material melted initially and the sintered region on the top kept the water from penetrating the melt. Low melt-water heat transfer rates were observed. Concrete attack continued with the debris not cooled. This sintered crust configuration is not prototypical of the ABWR.

The most recent test, M1B, corrected the problems encountered with M1. The melt temperature was observed to decrease steadily to near the concrete liquidus temperature after the water was introduced. Concrete ablation was found to continue but at a reduced rate (a few mm/h). The post-test examination showed that there were large holes in the top surface.

The experiments described above are insufficient to enable a full understanding of debris coolability in the lower drywell of the ABWR. Some insights can, however, be extracted. The following shows the observed upward heat flux for three of the tests.

SWISS	- 800 kW/m^2
Mark I Shell Test	- 800 kW/m^2
MACE Scoping	- 600 kW/m^2

One of the major reasons why these tests are not prototypic is that, due to their small scale, they promote a stable crust formation. The larger scale MACE tests should generate some useful insights.

19EC.2 Description of Event Tree Analysis

19EC.2.1 Debris Coolability

A decomposition event tree (DET), shown in Figure 19EC-1, was developed to assess the likelihood of debris coolability. This subsection describes the branch points and the quantification of this DET.

19EC.2.1.1 Fraction of Debris in Lower Drywell Early (COR_DW_E)

This event assesses the initial debris mass which relocates to the lower drywell soon after vessel failure. The amount of debris which enters the lower drywell early is dependent on the amount of debris molten in the lower RPV head at the time of RPV failure and on the amount of entrainment of the debris from the lower drywell. However, for simplicity, debris entrainment to the upper drywell was conservatively neglected in this analysis. For consistency with the DCH analysis, two regimes are considered for the fraction of the core inventory which is molten in the RPV at the time of RPV failure (Subsection 19EA.2.1.4). These regimes are:

Low 0 - 20% (nominal 10%)*

High 20 - 60% (nominal 40%)

*Probabilites not part of DCD (Refer to SSAR).

19EC.2.1.2 Amount of Initial Debris Superheat (SUP_HEAT)

This event is used to represent the initial debris temperature when the debris first contacts the lower drywell floor. It is also used as a surrogate to represent the additional metal/water reaction heat production associated with a high metal to oxide ratio in the debris. Superheated debris or debris with a high metal content is expected to be more difficult to quench initially and to experience faster initial concrete erosion. In the deterministic CCI analysis discussed in Subsection 19EC.3, the low superheat cases are represented by (molten) debris at the U-Zr-O eutectic melting temperature (approximately 2500 K). High superheat was taken to be temperatures in the range 300-500 K above the melting temperature. This was represented in the deterministic analysis by increasing the amount of steel added to the melt prior to vessel breach.

Two cases were considered in the DET analysis. The first case represents sequences with a small amount (10% of core inventory) of molten debris in the lower plenum at vessel rupture and the second case represents large amounts of debris (40%).

- Case 1—Small Debris Mass in Lower Drywell Early

For the case of a small debris mass in the lower RPV, it is likely that either

(1) Vessel failure occurred fairly quickly after core slump into the lower plenum (MAAP-type failure model), or that

(2) The debris in the lower plenum was initially quenched by residual water in the lower plenum and that RPV failure occurred later, after the water was boiled away and the debris started to reheat (BWRSAR-type failure model).

For these situations it is judged likely that the debris temperature will be at, or near, its melting point.

For the case of a large amount of molten debris it could be expected that this resulted from a delayed failure of the RPV allowing more debris to flow into the lower plenum (MAAP model) or for melting and heating of quenched debris already relocated to the lower plenum (BWRSAR model). For both situations the extended time to vessel failure could result in higher molten debris temperatures at RPV failure. It is unclear what the actual debris temperature would be for this case. Hence, equal probabilities are assigned to each branch to represent this large uncertainty.

19EC.2.1.3 Debris Quenched Early (QUENCH_E)

The probability that long term debris cooling will be established is greatly increased if the initial debris pour is quenched soon after being expelled from the vessel. Initial quenching of the debris implies either that the debris has been fragmented to sizes which allow cooling, or if the debris is a continuous “pool” that it is sufficiently shallow to allow cooling by conduction through the layer of solid debris.

The ABWR design makes it extremely unlikely that water will be in the lower drywell prior to RPV failure. Most of the core damage frequency is the initiated by a transient. This type of sequence would not result in water in the lower drywell at the time of vessel failure. Only a LOCA in the RPV bottom drain line or an accident in which a large mass of water is added to the containment before core damage would result in water entering the drywell. All other LOCAs blow down into the upper drywell (which drains directly to the suppression pool). Hence, water which enters the lower drywell coincident with the expelled debris must come from residual RPV inventory or from in-vessel injection systems which are operating at (or are initiated at) RPV failure. For a MAAP-type melt progression the lower plenum is nearly full of water at the time of vessel failure. Thus, 70,000 kg of water is available to quench the debris.

In addition, water may enter the lower drywell at the time of vessel failure via the passive flooder. If water from the vessel does not enter the cavity, the debris will rapidly heat the lower drywell, and the flooder will open quickly. For a BWRSAR type melt progression model, there will not be water in the lower plenum at the time of vessel failure. In this case, the lower drywell will heat up quickly and the passive flooder will open. A calculation was performed with a modified version of MAAP-ABWR, which simulates the BWRSAR melt progression model, described in Subsection 19EC.6. Case LATE indicates that the flooder will open about 30 minutes after vessel failure for this case. Thus, it is very likely that water will be available to quench the initial debris expelled from the vessel.

The major parameters judged to impact the probability of initial debris quenching are

(1) The mass of debris in the lower drywell following RPV failure

(2) The availability of water in the lower drywell

(3) The initial temperature of the debris

The mass of debris retained in the lower drywell is determined in a preceding event. The initial debris temperature is also determined in a prior event. The source of water depends on the presumed core melt progression model as described above. In a MAAP-type melt progression, the initial availability of water is assured. For a BWRSAR model, the water comes either from injection systems which begin to inject at vessel failure or from the operation of the flooder which is considered in the next node. Since no credit will be taken for early quenching if a significant amount of debris enters the cavity before lower drywell flooding occurs, the order of this question and the late cavity flooding question (CAVWAT_L) question is not important for the BWRSAR case.

Four cases were defined in the DET. These cases are:

- Case 1—Small Debris Mass and Low Superheat

  For this case, approximately 24000 kg of molten debris are released from the RPV at vessel failure. Since the debris has a low superheat and the debris depth is very shallow (< 5 cm) it is highly likely that the debris would be initially quenched.

- Case 2—Small Debris Mass and High Superheat

  As for Case 1, approximately 24000 kg of molten debris are released from the RPV at vessel failure. In this case, the debris has a high superheat and, although the debris depth is very shallow (< 5 cm), it is somewhat less likely that the debris would be initially quenched when the lower drywell is flooded for this case than for Case 1.

- Case 3—Large Debris Mass and Low Superheat

  For this case, approximately 94000 kg of molten debris are released from the RPV at vessel failure. The debris depth in this case would be relatively shallow (< 15 cm). Since the debris pool is relatively shallow and the debris superheat is low, it is judged that it is likely to be initially quenched when the lower drywell is flooded.

- Case 4—Large Debris Mass and Low Superheat

  As for Case 3, approximately 94000 kg of molten debris are released from the RPV at vessel failure and the debris depth would be relatively shallow (< 15 cm). However, the debris superheat is high and it is judged to be indeterminate whether or not the debris will be quenched by residual RPV coolant inventory.

19EC.2.1.4 Water Enters Cavity Late (CAVWAT_L)

This parameter is used to represent the longer term addition of water to the lower drywell. The lower drywell water addition systems which are considered are the direct drive diesel firewater system, any vessel injection which is available late in the accident and the passive flooder. Initiation of the firewater addition system is the most likely means of late water addition to the lower drywell. If the firewater system is not started, the passive flooder system will begin to inject water when the fusible material, located at the ends of the pipes near the drywell floor, melts. The fusible material on the passive flooder system is assumed to open when the lower drywell gas temperature reaches 533 K (500°F). Assuming a BWRSAR melt progression model, the fusible valves on the passive flooder system would open in approximately 30 minutes. For a MAAP-type melt progression model, the water in the lower drywell is first boiled off. The debris then begins to heat up. If the debris is quenched during the early boil-off phase the debris must reheat resulting in approximately 2 hours to flooder actuation. If the debris was not quenched early, the flooder opens about 30 minutes after the debris bed dries out. This event is a sorting type event, quantified (either 0 or 1) based on prior branch decisions in the CET.

19EC.2.1.5 Time Remaining Core Debris Falls Into Cavity (COREDROP)

This event assesses the timing of the entry of the remaining debris into the lower drywell relative to the timing of the addition of water (i.e., from the passive flooder or firewater system). If the majority of the debris is held up in the vessel until after water addition begins, then debris cooling is substantially more likely than if the bulk of the RPV debris enters the lower drywell prior to water addition. MAAP calculations indicate that the residual RPV debris will melt and fall into the lower drywell very slowly after vessel failure. This behavior is also typical of BWRSAR-type calculations (Reference 19EC-8).

Two cases are considered in the quantification of the event. The timing of residual RPV debris entry into the lower drywell is considered to be sensitive to the extent of the accident progression in-vessel at the time of vessel failure. For the case of a small amount of molten debris in the lower RPV plenum at RPV failure (Event 1 in this DET), it is inferred that RPV failure has occurred relatively “early” in the in-vessel accident progression process. Conversely, for a large amount of molten debris in the lower RPV plenum at RPV failure, it is more likely that the in-vessel accident progression is further advanced at the time of RPV failure. Consequently, it would be expected that for the case of small initial debris pours the timing between vessel failure and later debris pours would be delayed relative to the case of large initial debris pours. Based on insights from ABWR specific MAAP analyses and from a review of BWRSAR calculations for other BWR sequences branch probabilities were estimated.

19EC.2.1.6 Heat Transfer Rate to Overlying Water (HT_UPWARD)

This event assesses the longer term steady state heat transfer rate which characterizes upwards heat transfer from the debris. Three regimes are considered

(1) Heat transfer limited by hydrodynamics in an overlying water pool (CHF limit)

(2) Heat transfer limited by film boiling to an overlying water pool

(3) Heat transfer limited by conduction through a debris crust on the upper debris surface

Nominal values of the heat transfer rate used in the deterministic CCI model to characterize these three heat transfer regimes are 900, 300 and 100 kW/m^2, respectively.

The conduction limit represents conditions where a crust forms on the surface of the debris and water cannot penetrate into the debris bed. The use of a 100 kW/m^2 heat flux is believed to be very conservative. If the debris is not quenched and core concrete interaction occurs, the upper crust will thin to a condition where the upward and downward heat fluxes are nearly equal. This will lead to a heat flux much higher than 100 kW/m^2. Therefore, this value will lead to very aggressive core concrete interaction. Further discussion of the upward heat flux for the conduction limited configuration is given in Subsection 19EC.3.1.

The hydrodynamic limit represents cases where water can penetrate into the debris bed allowing a much greater effective debris/coolant heat transfer area. Under these conditions the heat transfer rate is limited by the ability of the water to penetrate the debris bed. The use of 900 kW/m^2 is much lower than the typical heat fluxes observed in the experiments performed to date.

The film boiling regime is selected to represent an intermediate heat transfer rate where, for example, the crust is unstable allowing water to penetrate the debris bed in a limited fashion. The early phase of the experiments indicate a heat flux well in excess of 300 kW/m^2 before the formation of a crust.

Four cases were identified for quantification. These cases are described below.

- Case 1—Large Debris Mass in Lower Drywell Early, Debris Initially Quenched and Residual Core Debris Enters Lower Drywell After Flooding

  This case is considered the most favorable set of conditions for establishment of a particulated debris bed which would be conducive to water ingression and coolability. The initial phase of the interaction is characterized by a large amount of debris which is initially quenched in the lower drywell. Prior to the entry of the residual RPV debris, the lower drywell is flooded resulting in the residual debris pouring into a pool of water which is likely to lead to fragmentation, quenching and the establishment of a particle bed. Consequently, a high probability is assigned under these conditions to an upwards heat flux characteristic of a particle bed with water ingression.

- Case 2—Small Debris Mass in Lower Drywell Early, Debris Initially Quenched and Residual Core Debris Enters Lower Drywell After Flooding

  This case is considered to represent nearly as favorable a set of conditions for establishment of a particulated debris bed as was Case 1. In contrast to Case 1 however, the initial phase of the interaction is characterized by only a small amount of debris which is quenched in the lower drywell. Hence, a larger amount of debris enters the lower drywell after RPV failure than for Case 1. Prior to the entry of the residual RPV debris, the lower drywell is flooded resulting in the residual debris pouring into a pool of water which is likely to lead to fragmentation, quenching and the establishment of a particle bed. Consequently, as for Case 1, a relatively high probability is assigned under these conditions to an upwards heat flux characteristic of a particle bed with water ingression.

- Case 3—No Initial Debris Quench and Residual Core Debris Enters Lower Drywell After Flooding

  This case is considered less favorable for establishment of a particulated debris bed which would be conducive to water ingression and coolability. The initial phase of the interaction is characterized by failure to quench the debris soon after RPV failure. However, prior to the entry of the residual RPV debris, the lower drywell is flooded resulting in the residual debris pouring into a pool of water which is likely to lead to fragmentation of this debris. However, since the initial debris pour was not quenched, long term establishment of a coolable particulated debris bed is somewhat uncertain. Consequently, a lower probability has been assigned for the most favorable debris bed configuration compared with Cases 1 and 2.

- Case 4—Residual Core Debris Enters Lower Drywell Prior to Flooding

  This is considered the least favorable set of conditions for establishment of a particulated debris bed which would be conducive to water ingression and coolability. For this case the bulk of the residual core debris enters the lower drywell prior to lower drywell flooding. This could lead to formation of a molten pool undergoing concrete attack. Later water addition, instead of particulating the debris may lead to crust formation, limiting the ability of water to penetrate into the debris.

19EC.2.1.7 Core Debris Concrete Attack (CCI)

This event characterizes the nature of the debris concrete attack. Three branches are considered. The No CCI branch represents cases where the little or no debris concrete attack would be expected. Wet CCI represents cases where CCI occurs in the presence of an overlying water pool and Dry CCI is for cases where the lower drywell was not flooded.

- Case 1—Lower Drywell Not Flooded

  The Dry CCI case occurs for all sequences where both active injection and the passive flooder fail to supply water to the lower drywell after vessel failure. Under these conditions Dry CCI is assured.

- Case 2—Lower Drywell Flooded, Upward Heat Transfer Limited by CHF

  For cases where the lower drywell is flooded, MAAP analysis and supplemental hand calculations indicate that if the upward heat transfer is above about 300-400 kW/m^2 then the debris bed will be coolable.

- Case 3—Lower Drywell Flooded, Upward Heat Transfer Limited by Film Boiling

  For cases where the lower drywell is flooded, MAAP analysis and supplemental hand calculations indicate that if the upward heat transfer is in the range of about 300 kW/m^2 then the debris bed should be coolable. Since this case represents a range of upward heat transfer regimes (200-400 kW/m^2) and the lower part of this range may not in all cases be coolable and probabilities were assigned accordingly.

- Case 4—Lower Drywell Flooded, Upward Heat Transfer Limited by Conduction

  For cases where the lower drywell is flooded, MAAP analysis and supplemental hand calculations indicate that if the upward heat transfer is below about 200 kW/m^2 then the debris bed will not be coolable.

19EC.2.2 Pedestal Resistance to CCI

This subsection describes the decomposition event tree (DET) analysis used to assess the probability of pedestal failure as a result of radial core concrete (CCI) attack in the lower drywell after reactor vessel failure. The DET is shown in Figure 19EC-2. Pedestal wall failure is considered to be sensitive to

(1) The nature of the CCI (i.e. whether wet or dry)

(2) Whether the debris spreads from lower drywell into the suppression pool following radial penetration through the pedestal wall to the wetwell/drywell connecting vents

(3) The extent of radial erosion compared to downward erosion

The lower drywell will be flooded in most cases as a result of either active injection systems such as the firewater addition system or via passive injection through the lower drywell flooder.

19EC.2.2.1 Core Concrete Attack (CCI)

This event characterizes the nature of core concrete attack. Three branches are considered.

- No CCI
- Wet CCI
- Dry CCI

The No CCI branch represents cases where there is little or no concrete attack. Wet CCI represents cases where CCI occurs in the presence of an overlying water pool. Dry CCI is for cases where the lower drywell was not flooded. The rate of CCI is higher for cases with dry CCI.

This event is a sorting type event which assigns a probability of 0 or 1 depending on the final branch taken in the CCI CET.

19EC.2.2.2 Suppression Pool Water Floods Lower Drywell after Downcomer Penetration (SP_INGRESS)

This event assesses if suppression pool water will flood into the lower drywell after the erosion front reaches the wetwell/drywell connecting vents. The vents are imbedded in the pedestal. If 25 cm of the pedestal concrete is eroded, the ablation front will reach the inner surface of the connecting vents. It is considered quite likely that this will result in water ingression and flooding of the lower drywell.

This event is only significant for Dry CCI sequences where the lower drywell is not flooded by either active injection or the passive flooder. The probabilities are assigned based on judgement.

19EC.2.2.3 Debris Flows From Lower Drywell to Suppression Pool after Downcomer Penetration (WW_DEB)

This event assesses whether a significant amount of the molten debris will flow from the lower drywell into the suppression pool following penetration of the wetwell/drywell connecting vents. After 25 cm of radial erosion the ablation front will reach the inner surface of the downcomers. The floor of the lower drywell is above the bottom of the connecting vents, which, in turn, are above the floor of the wetwell. Thus, once the downcomers are breached, a flowpath exists from the lower drywell into the suppression pool. Flow of a significant portion of the molten debris into the suppression pool will increase the debris surface area in contact with water and decrease the debris depth in the lower drywell. Although there is a great deal of uncertainty in this behavior, it is considered fairly likely that the debris will flow into the suppression pool.

19EC.2.2.4 Ratio of Radial to Axial Erosion (RAD_EROS)

Given that CCI is occurring, this event assesses the ratio of the radial concrete erosion to the downward erosion. Three branches are considered 1/5, 1/3 and 1/1. CCI experiments have generally demonstrated significantly more downward concrete penetration than radial penetration. It is hypothesized that radial erosion is limited because the concrete decomposition gasses establish a gas film between the debris pool and the concrete walls. This gas film acts to insulate the concrete sidewalls, and to convect debris heat upwards. This limits the heat transfer to, and ablation of, the concrete sidewalls. Conversely, the gas film at the bottom surface of the pool would be unstable due to the heavier overlying debris pool. The density difference would cause the lower gas film to collapse, allowing contact of the debris with the concrete. This difference in gas film behavior would limit the sideward heat transfer compared to the downward heat transfer.

In the BETA series of debris concrete experiments conducted at the KfK research center in Germany, downward erosion rates exceeded sideward erosion rates by a factor from 3 to greater than 5. For example, in the high power CCI experiment BETA V1.8, the downward erosion was measured to be approximately 40 cm and the sideward erosion was only about 2 cm (1/20 sideward to downward erosion ratio). For the low power experiment V6.1, the downward erosion was 35 cm and the sideward erosion was 10 cm (1/3.5 ratio).

Based on the CCI experiments, and the generally accepted model described above, it seems appropriate to assume that downward erosion is strongly favored over sideward erosion. Consequently, larger probabilities are assigned to the 1/5 and 1/3 branches than for the 1/1 branch. However, since some residual uncertainty remains as to the appropriate assumption for the extent of radial erosion for large reactor scale situations, a small probability of is assigned to the 1/1 erosion branch.

19EC.2.2.5 Pedestal Failure (PED)

This branch assesses the probability of pedestal failure as a result of excessive radial concrete erosion of the lower drywell pedestal wall.

Structural analysis of the pedestal indicates that the loads can be supported without yielding if only the outer shell and 15 cm of the steel webbing remains intact. Thus, for a total wall thickness of 1.7 m, the lower limit for the amount of radial erosion which can be sustained without pedestal structural failure is 1.55 m. However, since the total depth of the pedestal is 1.7 m, erosion to the full 1.7 m depth will obviously result in pedestal failure. Additional discussion of the pedestal strength under radial concrete erosion is presented in Subsection 19EC.4.

Analyses were performed to estimate the extent of concrete erosion in the lower drywell under a variety of conditions. The results of these analyses are summarized in Subsection 19EC.5. Four cases were considered in the DET for quantification of pedestal wall failure. These cases are described below.

- Case 1—Debris Flows into Suppression Pool after Downcomer Penetration

  This case represents sequences where a substantial amount of the core debris relocates into the suppression pool after downcomer penetration. This is represented by deterministic calculations FMX100, FMXCSP and NFlood. The calculations indicate that the increase in the pool surface area results in either a coolable debris configuration, or greatly reduced radial erosion rates. Consequently, the likelihood of sufficient radial penetration to fail the pedestal in this case is considered to be remote.

- Case 2—Wet CCI With No Debris Flow into The Suppression Pool after Downcomer Penetration

  For sequences where CCI was predicted to occur in the presence of an overlying water pool with no debris relocation to the suppression pool, the maximum amount of downward concrete erosion at 50 hours was 1.55 m (Case FMX1P). Using this value for the amount of axial erosion, the radial erosion depth is estimated for the three cases. Comparing this value to the pedestal capability of 1.55 m, estimates were made for the probability of pedestal failure.

- Case 3—Dry CCI With No Debris Flow into Suppression Pool and No Late Suppression Pool Water Ingression into The Lower Drywell

  This case represents case DRY in the deterministic analysis. In this case the debris is assumed to remain dry for the entire duration of the accident. No flow of either water or debris through the wetwell / drywell connecting vents is presumed to occur when the ablation front reaches the vents. For this case the axial ablation depth at 50 hours was calculated to be 2.5 m. Using this value to estimate the radial erosion depth for the three radial to axial erosion ratios, split fractions are assigned based on the pedestal capability.

- Case 4—Dry CCI With No Debris Flow into The Suppression Pool and Late Suppression Pool Water Ingression into The Lower Drywell

  The case in which the debris is initially dry, but becomes flooded with water after the ablation front reaches the wetwell / drywell connecting vents is considered to be slightly better than Case 3. In this case the debris is assumed to remain in the lower drywell throughout the period of CCI.

19EC.3 Deterministic Model for Core Concrete Interaction

As described above, several key parameters influence the potential for concrete erosion in the presence of an overlying water pool. An analytical tool was selected to investigate the impact that these parameters have on CCI, containment pressurization, opening of the overpressure protection system, and possible fission product release. MAAP-ABWR was selected since, with a few minor code modifications, it was capable of investigating the key parameters identified in the DET. MAAP-ABWR allowed the impact of parameter variations to be carried out through containment pressurization and fission product release.

A few simple code modifications were made to allow the user to control the debris coolability and to simplify the specification of the severe accident scenario. These changes are summarized below.

(1) Subroutine PLSTM was modified to allow the user to specify the upward heat flux. Model parameter, FCHF, was redefined to be the upward heat flux in Watts / m^2. All other debris-to-water heat transfer mechanisms were disabled in PLSTM.

(2) The following actions were added to the MAIN routine:

  (a) If lower drywell gas temperature exceeds 533 K - open passive flooder

  (b) If radial erosion exceeds 25 cm - allow debris to spread to wetwell and allow water to flood the lower drywell

  (c) If radial erosion exceeds 50 cm (note that this limit is very conservative) - fail drywell with an area of ADWLEK (user input)

  (d) If upper drywell wall surface temperature exceeds 533 K - begin to leak out of the upper drywell as specified in Subsection 19F.3.2.2

The major assumptions included in the MAAP analysis are described below:

(1) CCI experiments have generally demonstrated significantly more downward concrete penetration than radial penetration. It is hypothesized that radial erosion is limited because the concrete decomposition gasses establish a gas film between the debris pool and the concrete walls. This gas film acts to

insulate the concrete sidewalls, and to convect debris heat upwards. This limits the heat transfer to, and ablation of the concrete sidewalls. Conversely, the gas film at the bottom surface of the pool would be unstable due to the heavier overlying debris pool. The density difference would cause the lower gas film to collapse, allowing contact of the debris with the concrete. This difference in gas film behavior would limit the sideward heat transfer compared to the downward heat transfer.

In the BETA series of debris concrete experiments conducted at the KfK research center in Germany, downward erosion rates exceeded sideward erosion rates by a factor from 3 to greater than 5. For example, in the high power CCI experiment BETA V1.8, the downward erosion was measured to be approximately 40 cm and the sideward erosion was only about 2 cm (1/20 sideward to downward erosion ratio). For the low power experiment V6.1, the downward erosion was 35 cm and the sideward erosion was 10 cm (1/3.5 ratio).

Based on the CCI experiments, and the generally accepted model described above, it seems appropriate to assume that the ratio of radial to axial attack is 1/5. However, this parameter is included as a parameter in the DET for pedestal erosion since the ratio is still uncertain.

Since MAAP assumed that radial and axial penetration were identical, the axial ablation numbers were multiplied by 1/5 to obtain an estimate on the radial attack depth.

(2) The heat transfer from the debris to the water was assumed to be equal to the user specified value throughout the transient.

Other than the changes described above, the standard MAAP-ABWR code was used to quantify the CCI decomposition event tree.

19EC.3.1 Minimum Heat Flux

The most critical element in determining the potential for core concrete interaction, and the containment response if it should occur is the minimum heat flux. The heat transfer between the water and the debris can be limited by:

(1) Conduction within the debris

(2) Critical heat flux

(3) Film boiling

The last is of concern if the debris surface temperature remains so hot that the water cannot wet the surface, i.e. if an insulating blanket of steam forms. Film boiling has been observed in well controlled laboratory environments using polished surfaces. However, it has also been observed that the smallest of surface imperfections or contaminants quickly results in a transition to nucleate boiling. It seems highly unlikely that the irregular surface of the debris would be able to maintain itself in film boiling. Therefore, film boiling is not likely to limit upward heat transfer.

Critical heat flux is sufficiently high that it would not impose a practical limit on debris coolability. Therefore, a lower limit on the upward heat flux may be obtained by consideration of the conduction limit. The biggest unknown is whether the debris remains in an intact slab-like configuration, an intact configuration with irregularities which increase the heat transfer area and act as fins, or if the debris develops cracks which allow water to ingress. The presence of cracks would increase the heat flux. Therefore, let us consider the worst situation (intact slab).

The temperature distribution in steady state, assuming one dimensional heat transfer and a homogeneous debris mixture, is given by:

$$k\frac{\partial^2 T}{\partial x^2} + q''' = 0 \qquad \text{(19EC-1)}$$

where:

k	=	thermal conductivity (3.5 W/mK for oxide debris)
q'''	=	volumetric heat generation

It is sufficient for our purposes to consider the case of 1% decay power. For a total debris mass of about 235,000 kg, this implies an average initial volumetric heat generation rate:

$$q''' = (1.5)\frac{MW}{m^3}$$

In a one-dimensional flat geometry, integrating Equation 19EC-1 twice yields:

$$T = \frac{-q'''x^2}{2k} + C_1 x + C_2 \qquad \text{(19EC-2)}$$

If we

(1) Assume nucleate boiling is maintained at the surface

(2) Conservatively assume that the bottom of the debris in contact with concrete is adiabatic

(3) Impose the condition that the debris not ablate concrete and the temperature at the debris-concrete interface is 1550 K.

we obtain:

$$C_1 = 0$$

$$C_2 = 1550 \text{ K}$$

$$T(\delta_{lim}) = 450 \text{ K}$$

where:

δ_{lim} = debris thickness

Substituting into Equation , we have for the limiting debris thickness for coolability:

$$\delta_{lim} = 0.07 \text{ m}$$

This means that if we are in nucleate boiling at the surface, we can just remove decay heat purely by conduction through the debris slab at a thickness of 7 cm. The surface heat flux is:

$$q'' = q'''\delta_{lim} = 100\text{kW}/\text{m}^2$$

The heat flux which would result from critical heat flux would be substantially higher than this value. Thus, one could view this as the lowest possible upward heat transfer given the boundary conditions. A higher temperature at the bottom of the crust or heat transfer into the slab would both increase the debris-to-water heat transfer.

This rather low heat transfer would be increased if the surface was of non-uniform thickness (fin effects) or especially if the surface cracked sufficiently to allow water to ingress.

19EC.4 Pedestal Strength

The configuration of the ABWR pedestal is shown in Figure 1.2-13e. The width of the pedestal is 1.7 m. The design consists of two concentric steel cylinders connected by steel web stiffeners. Ten wetwell-drywell connecting vents run through the annular region between the cylinders. The remainder of the space is filled with concrete. If significant core concrete attack occurs, the strength of the pedestal could be compromised as the pedestal is eroded. The strength of the pedestal after it has undergone erosion is examined to determine the maximum erosion depth allowable to ensure that the pedestal does not collapse.

The pedestal is designed based on the maximum stress obtained in the steel plates. The strength of the concrete is neglected. The allowable stress in the steel plates is 0.6 times the yield strength, neglecting temperature. The calculated stress without seismic loads in the ABWR pedestal is 0.4 times the yield strength.

For design analysis the largest single load is the accident temperature. If core concrete interaction were to take place as a result of a severe accident, the inner plate of the pedestal would melt. Without a continuous inner plate the moment induced by the differential temperature disappears. It is expected that any temperature induced moments acting along the stiffeners will be strain limited. Therefore, they will not reduce the capability of the outer plate.

In order to estimate the allowable ablation depth, the seismic and thermal loads are removed and the remaining loads are calculated. No attempt was made to take credit for the relocation of fuel from the vessel onto the floor of the drywell. The strength of the remaining concrete is neglected. The loads are compared to the yield strength of the remaining pedestal steel. Therefore, this calculation corresponds roughly to a service level C type of calculation.

The results of the calculation show that the outer shell of the pedestal plus 15 cm of the web stiffeners are required to maintain the pedestal loads below 90% of yield. This limit is used as a conservative estimate of the pedestal ultimate capability after erosion. The total pedestal width is 1.7 m. Therefore, pedestal integrity is ensured for ablation depths up to 1.55 m.

19EC.5 Application of CCI Model to ABWR

The deterministic code used for investigating core-concrete interaction in the ABWR was described in Subsection 19EC.3. This subsection will describe the evaluations that were made to support the quantification of the CCI decomposition event tree.

19EC.5.1 Sequence Selection

The MAAP-ABWR code, as modified for this application, allowed for a great amount of flexibility in analyzing the impact of key parameter variations on core-concrete attack. The following lists the key parameter variations that were investigated:

(1) Upward heat transfer to overlying water pool

(2) Mass of debris discharged from vessel

(3) Mode of fission product release from containment

(4) Flooding of lower drywell resulting from radial penetration of vertical connecting vents

(5) Debris spreading related to radial penetration of vertical connecting vents

The base case sequence selected to investigate core concrete interaction was the low pressure loss of injection scenario. This event was initiated by a transient with the assumption that all injection was unavailable. The RPV was depressurized manually when the core level dropped below 2/3 core height. Without coolant injection, the core melts and slumps into the lower vessel head. Local penetration failure occurs and the debris is discharged into the lower drywell. A radial to axial ablation rate of 1/5 is assumed in all sequences. Table 19EC-1 provides a chronology of the events up until the vessel is failed.

Table 19EC-2 defines each of the sequences analyzed and provides a summary of the results. The first column gives the case designation along with reference to specific notes. Columns two through four provide the relevant sequence definition information. For purposes of demonstration, all cases were executed for the dominant sequence, a low pressure loss of injection sequence with a containment pressure at the time of vessel breach of approximately 1 atm. The upward heat flux was varied between 100, 300, and 900 kW/m^2. A value of 100 kW/m^2 was selected to approximate the heat transfer associated with a stable crust formation where the upward loss is controlled by conduction of heat through the crust. A value of 300 kW/m^2 was selected to represent limited water ingression into the debris bed with the upward heat transfer being controlled by film boiling. The largest value used represents the critical heat flux limit for debris cooling. Further discussion of these values is included in Subsection 19EC.2.1.6

As run in its standard manner MAAP-ABWR calculates that 60% of the total core inventory was released from the vessel. The remaining 40% was calculated to be held up in the core with the decay heat being radiated to the vessel wall and convected into the upper drywell. The 40% remaining behind is typically the outer peripheral bundles which have low decay heat. To support the DET quantification, additional cases were run assuming that 100% of the core was discharged from the reactor vessel. This has two major influences on the containment behavior. Without the peripheral bundles in the core, the drywell heatup is reduced. Second, the added core mass on the lower drywell floor will influence the calculation of core-concrete attack, debris coolability and containment pressurization.

19EC.5.2 Summary of Results

Table 19EC-2 summarizes the results of the deterministic analyses for the ABWR. The following general conclusions are indicated by these results:

(1) For all sequences with successful operation of the flooder, radial concrete erosion was less than the structural limit described in Subsection 19EC.4. Radial attack does not pose a significant challenge to containment.

(2) For sequences with operation of the containment overpressure protection system, due to suppression pool scrubbing, the fission product release is dominated by noble gas.

(3) Release times for cases with the passive flooder are on the order of 20 hours after the initiation of core damage (defined as onset of melting).

(4) The extended time period between vessel breach and rupture disk actuation (or containment failure) provides for a substantial reduction in the amount of fission product released from containment.

(5) Using experimentally-based values for the upward heat transfer (Subsection 19EC.1) would result in debris cooling in the ABWR and early termination of the core concrete attack. Therefore, the lower bound for upward heat transfer is conservatively assumed to be 100 kW/m^2. This is done in order to obtain substantial concrete erosion and demonstrate the robustness of the containment design if the debris is not quenched.

(6) For the dominant scenarios with successful operation of firewater to provide water to the debris, the time from onset of melting to fission product release is 24 hours from the beginning of the accident for all upward heat transfer rates.

A set of plots for case FMX100 case are included in Figures 19EC-3 through 19EC-7. This case demonstrates long-term core-concrete interaction, but is otherwise typical of the conditions analyzed. The depletion of zirconium in this case occurs at about 20,000 seconds, coincident with the onset of CO production. The hydrogen gas generation is not equivalent to the amount which would be generated from a 100% metal water reaction because of a competing reaction between the zirconium and CO_2.

19EC.5.3 Initial Concrete Attack due to Impinging Corium Jet

At vessel failure, core material is discharged from the RPV onto the floor of the lower drywell. At low RPV pressures, the discharge rate of the debris is controlled by gravity and the vessel breach area in the lower head. From analyses performed for FCI calculations, Subsection 19EB.6.2.2, it is assumed that ten penetrations failed. This results in a maximum corium discharge rate of 6000 kg/s. The total failure area is 0.145 m^2. Assuming a density for corium of 8000 kg/m^3, a discharge of 6000 kg/s corresponds to a corium velocity of 5 m/s. The following calculation estimates the initial concrete attack depth resulting from this impinging corium jet.

The model from the MAAP subroutine JET (Reference 19EC-9) was used to compute the concrete attack from an impinging jet of corium. The stagnation point heat transfer coefficient between the corium jet and the concrete is approximated by the expression,

$$Nu = \frac{hD}{k} = 1.14\sqrt{Re} \tag{19EC-3}$$

or

$$h = 1.14 k_{cm}\sqrt{\frac{\rho_{cm} u_c}{\mu_{cm} D_{jet}}} \tag{19EC-4}$$

where:

k_{cm}	=	Corium thermal conductivity
μ_{cm}	=	Corium viscosity
u_c	=	Velocity of the corium stream impinging on the floor
D_{jet}	=	Diameter of the jet
h	=	Heat transfer coefficient
ρ_{cm}	=	Corium density
Nu	=	Nusselt number
Re	=	Reynolds number

The corium velocity at the cavity floor is given by,

$$u_c = u_0 + g t_{fall} \tag{19EC-5}$$

where:

u_0	=	Velocity of the corium expelled from the reactor vessel
g	=	Acceleration of gravity

and t_{fall} is defined by

$$'u_0 't_{fall} + \frac{1}{2} g t_{fall}^2 = z_v \tag{19EC-6}$$

where:

z_v	=	the elevation of the reactor vessel above the lower drywell floor.

A crust of frozen corium forms on the concrete and the ablation process is the same as at the reactor vessel penetration. Thus, the concrete ablation velocity is given by

$$u_{cn} = \frac{'h\,('T_{cm} - 'T_{cnp})}{\rho_{cn}['c_{pcn}('T_{cnp} - 'T_0) + \gamma_{cn}]} \tag{19EC-7}$$

where:

T_{cm}	=	Bulk corium temperature
ρ_{cn}	=	Concrete density
c_{pcn}	=	Concrete specific heat
γ_{cn}	=	Concrete latent heat
T_{cnp}	=	Concrete melting temperature
T_0	=	Initial concrete temperature

Substituting the corium velocity and the ABWR specific geometrical parameters into the above equations, results in an ablation rate of approximately 1 cm/s. With the debris being discharged over 5 seconds, the resulting ablation depth is 5 cm. This would only occur in the central portion of the lower drywell, and would in no way threaten the integrity of the structures.

19EC.6 Sensitivity to Various Parameters

Also included in Table 19EC-2 are other analyses that address possible sensitivities to modeling assumptions. These results are described below.

- Case DRY

  This case was run assuming that the passive flooder did not open and that, even after radial penetration of the vertical vent pipes, water was not introduced into the lower drywell. The drywell began to leak as a result of high temperature at about 20 hours and resulted in a slow, low magnitude, release of fission products.

- Case DWFAIL

This case is identical to case FMX300 except that the drywell was assumed to fail at the COPS set point. Due to the long time between vessel breach and containment failure, the fission products settle out very effectively and the result is a low magnitude release.

- Case FMX1P

This case was identical to case FMX100 except that the debris is assumed to not spread into the wetwell after penetrating the vertical connecting vents. The results indicate little sensitivity to this assumption. The radial attack at 50 hours is 31 cm for a ratio of radial to axial attack of one to five.

- Case NFLOOD

This case was identical to case ABWR100 except that the firewater addition system and passive flooder were not operational. Therefore, the debris was initially dry. After 25 cm of radial erosion, the debris was assumed to spread into the wetwell and water from the suppression was introduced into the lower drywell. The results indicate more concrete erosion with the COPS actuating at 17.4 hours compared to 19.1 hours.

- Case FIRE

This sequence was identical to FMX100 except that the firewater system was used to add water to the debris. Due to the addition of cold water, the pressurization of containment due to steam was reduced and the COPS was not predicted to open until 24.6 hours as compared to 17.6 hours for the case with passive flooder operation.

- Case LATE

This sequence was identical to case DRY except for a delayed vessel failure. The RPV was assumed to fail after all of the water in the lower plenum had boiled away and the debris heated up to the eutectic melting point (2501 K). Vessel failure occurred at 5.3 hours into the sequence as compared to 1.5 hours for the base case. Since there was no water discharged with the core debris at vessel failure, the gas temperature quickly increased to above the flooder actuation temperature. The flooder was assumed not to work for this case. The purpose of the run was to obtain an estimate of the time period between vessel failure and flooder actuation. The MAAP analysis conservatively assumes that the gas must reach 533 K before the flooder can open. In this case it took about one hour before the gas reached 533 K. Factoring in the difference between the wall surface and the gas temperature, the flooder would be expected to open within 30 minutes after discharge of the core debris. All other aspects of this run were similar to the DRY case.

The overall conclusions from the sensitivity analyses are that the ABWR containment design is quite insensitive to the uncertainties associated with core concrete interaction. The concrete erosion rates are consistent with other published results (Reference 19EC-8) and do not pose a serious threat to containment integrity. Operation of the COPS provides for a scrubbed release of the fission products and greatly limits the risk to the public.

19EC.6.1 Impact of Pedestal Concrete Selection

The pedestal of the ABWR is defined as the sidewalls of the lower drywell. This structure supports the vessel and the wetwell/upper drywell diaphragm floor. The type of concrete to be used in the pedestal is not specified. Concrete with low gas generation potential is required for the floor of the lower drywell.

Basaltic concrete was used for the lower drywell in determining the response of the containment to core concrete attack. This type of concrete is often used in the United States. The other type of concrete which is frequently used is limestone-common sand. Basaltic concrete is more rapidly eroded during core concrete interaction than is limestone-common sand concrete. Therefore, one would expect that if limestone-common sand concrete were used in the ABWR pedestal (i.e. the side walls), the sideward erosion would be slower than that presented in Table 19EC-2. Therefore, the estimates in that analysis for the times at which pedestal integrity could be threatened are expected to be conservative if non-basaltic concrete is used in the pedestal.

The other key impact of the type of concrete is the production of non-condensable gas. Limestone-common sand concrete produces more non-condensable gas than does basaltic concrete. However, this will not have a significant impact on this analysis because the surface area of the sidewall will be only 10-15% of the floor area if core concrete attack should occur. Furthermore, the shape of the debris pool will be pancake-like. The gas generated at the side wall will not be able to reach into the debris pool and cause more rapid metal water reaction in the debris pool. Rather, it will bypass the debris. Therefore, there will be little impact of the gas generation on the rate of attack due to any enhanced metal water reaction.

In summary, the type of concrete to be used in the pedestal side wall is not specified. If non-basaltic concrete is used in the pedestal the rate of sideward ablation may be somewhat reduced as compared to the analysis presented here. The rate of non-condensable gas generation may be slightly higher. However, because of the relative areas of the sidewall and the floor the impact will be small. The conclusions of the uncertainty analysis will not be affected by a different choice of concrete.

19EC.6.2 Impact of FMCRD Platform Grating

The FMCRD platform grating is located in the lower drywell at the elevation of the access tunnel. This rotating platform is circular and mounted on the rotating rail under the reactor vessel. There is an opening area at the center of the platform which is provided with traveling rail for the CRD handling device. Gratings will be installed on both sides of the rail for maintenance personnel. Typically, the grating consists of 2.54 cm (1 inch) by 0.95 cm (3/8 inch) metal slats mounted edge-wise to form a grid with a grid size on the order of 2.54 cm (1 inch) by 5.08 cm (2 inch).

However, it is expected that the grating will quickly ablate due to the flow of debris. This is much the same as the ablation of the vessel bottom head as the debris leaves the vessel. Any late debris relocation would be a slow, drip-like movement which would fall straight through the ablated region of the platform. Therefore, debris will not be retained above the platform and there will be no impact on containment performance.

19EC.7 Impact on Offsite Dose

The effect of the maximum core concrete interaction source term on a release with operation of the rupture disk is shown in Figure 19EC-8. The cases with rupture disk are the only risk significant release categories which would be impacted by core concrete interaction (The other sequences are cases with early containment failure due to DCH.) The figure shows the probability of dose exceedence assuming a release. Case 1 represents a sequence with no significant CCI, while Case 2 is a sequence with ongoing CCI. As the Figure 19EC-7 clearly shows, CCI does not have significant impact on the offsite dose.

19EC.8 Conclusions

This attachment investigated the impact of core-concrete interaction on the ABWR containment response. First, detailed DETs were developed to address all of the key parameters that influence CCI. Then, several deterministic analyses were carried out to support quantification of the trees. The following summarizes the important conclusions of the CCI investigation:

(1) For the dominant core melt sequences that release core material into the containment, most result in no significant CCI. Virtually no sequences have dry CCI.

(2) Even for the low frequency cases with significant CCI, radial erosion remains below the structural limit.

(3) The fission product release mode for a sequence with CCI is dominated by operation of the containment overpressure protection system. The release, which occurs at about 24 hours, is not distinguishable from a case with no CCI.

(4) Experimental results indicate that sufficient upward heat transfer to an overlying water pool would exist in the ABWR lower drywell to cool the debris.

19EC.9 References

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19EC-2 "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution", to be published as NUREG/CR, Draft 1991.

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19EC-6 R. Henry, "Experiments Relating to Drywell Shell - Core Debris Interaction", BWR Mark I Containment Workshop, Baltimore, MD, February 24-26, 1988. See also B. Malinovic, R. Henry, and B. Sehgal, "Experiments Relating to BWR Mark I Drywell Shell - Core Debris Interactions", ASME/AIChE National Heat Transfer Conference, Philadelphia, August 1989.

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Table 19EC-1 Summary of Timing for Core Concrete Interaction Base Case

Time (s)	Event
0.0	Loss of all injection
4.2	Reactor scrammed
1097.0	Core uncovered
1138.0	Manual depressurization
3451.0	Onset of core melt
5364.0	Slump into lower head
5382.0	Vessel failure

Table 19EC-2 Summary of CCI Deterministic Analysis for ABWR

Case #	Containment Press. at Vessel Failure (atmosphere)	Upward Heat Trans. (kw/m^2)	Debris Mass at Vess. Fail (Frac. of Tot. Inventory)	Radial Attack at 50 h (meters)	H2 Generated at 50 h (kg)	Time of FP Release (hours)	Mode of Release	Fission Product Release Fraction from Containment		
								NG	CsI	Sr
ABWR100	1	100	0.6	0.22	1813	19.1	COPS	1.0	2E-06	3E-09
ABWR300	1	300	0.6	9E-07	122	23.3	COPS	1.0	2E-10	2E-12
ABWR900	1	900	0.6	7E-06	122	23.2	COPS	1.0	3E-11	2E-12
FMX100	1	100	1.0	0.25	2130	17.6	COPS	1.0	1E-06	1E-08
FMX300	1	300	1.0	7E-03	154	19.3	COPS	1.0	1E-08	3E-15
FMX900	1	900	1.0	7E-04	111	19.1	COPS	1.0	1E-08	2E-14
FMXCSP*	1	100	1.0	0.25	2126	15.7	COPS	1.0	4E-07	3E-10
SENSITIVITY RUNS										
DRY	1	N/A	0.6	0.50	4990	19.8	DWT	0.34	4E-03	1E-05
DWFAIL	1	300	1.0	7E-03	154	19.3	DWF	1.0	8E-04	2E-10
FIRE	1	100	1.0	0.25	2131	24.6	COPS	1.0	5E-06	4E-10
FMX1P†	1	100	1.0	0.31	2762	17.6	COPS	1.0	1E-06	1E-08
NFlood‡	1	100	0.6	0.25	2127	17.4	COPS	1.0	8E-07	5E-10
LATEƒ	1	N/A	1.0	0.31	2697	20.0	DWT	0.23	6E-03	9E-09

* FMX100 Run with five times steel mass.

† Penetration into connecting vents does not cause debris spread.

‡ Flooder not operational; radial attack results in penetration to WW and debris spread.

ƒ Vessel failure assumed to occur after lower plenum water boiled away and debris reheats.

COPS = Containment Overpressure Protection System

DWT = Drywell Leakage occurs through penetrations.

DWF = Drywell Failure (0.0973 m^2)

Figure 19EC-1 Core Debris Concrete Attack DET

Not Part of DCD (Refer to SSAR)

Figure 19EC-2 Containment Event Evaluation DET for Pedestal Failure

Not Part of DCD (Refer to SSAR)

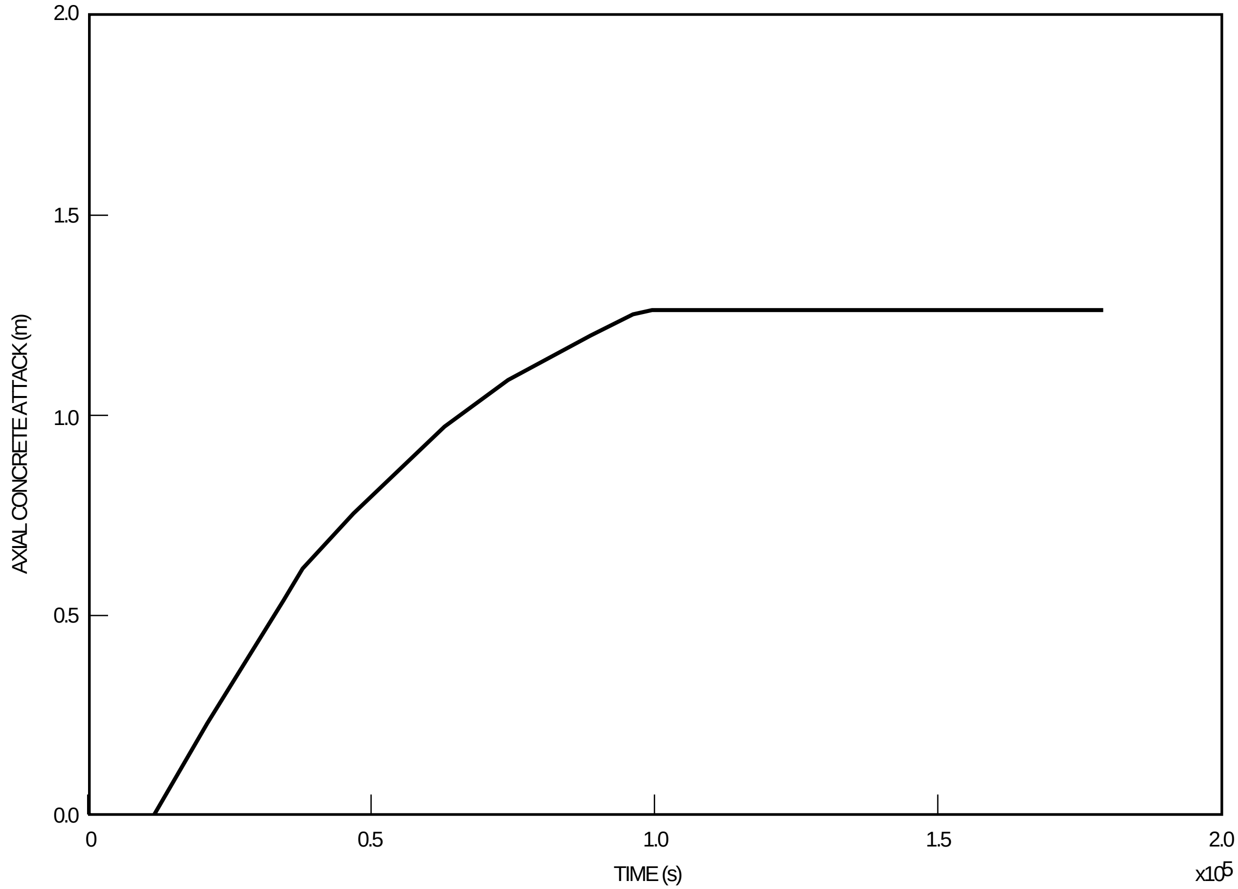

Figure 19EC-3 Sample Calculation for CCI Upward Heat Flux 100 kW/m^2: Axial Concrete Attack

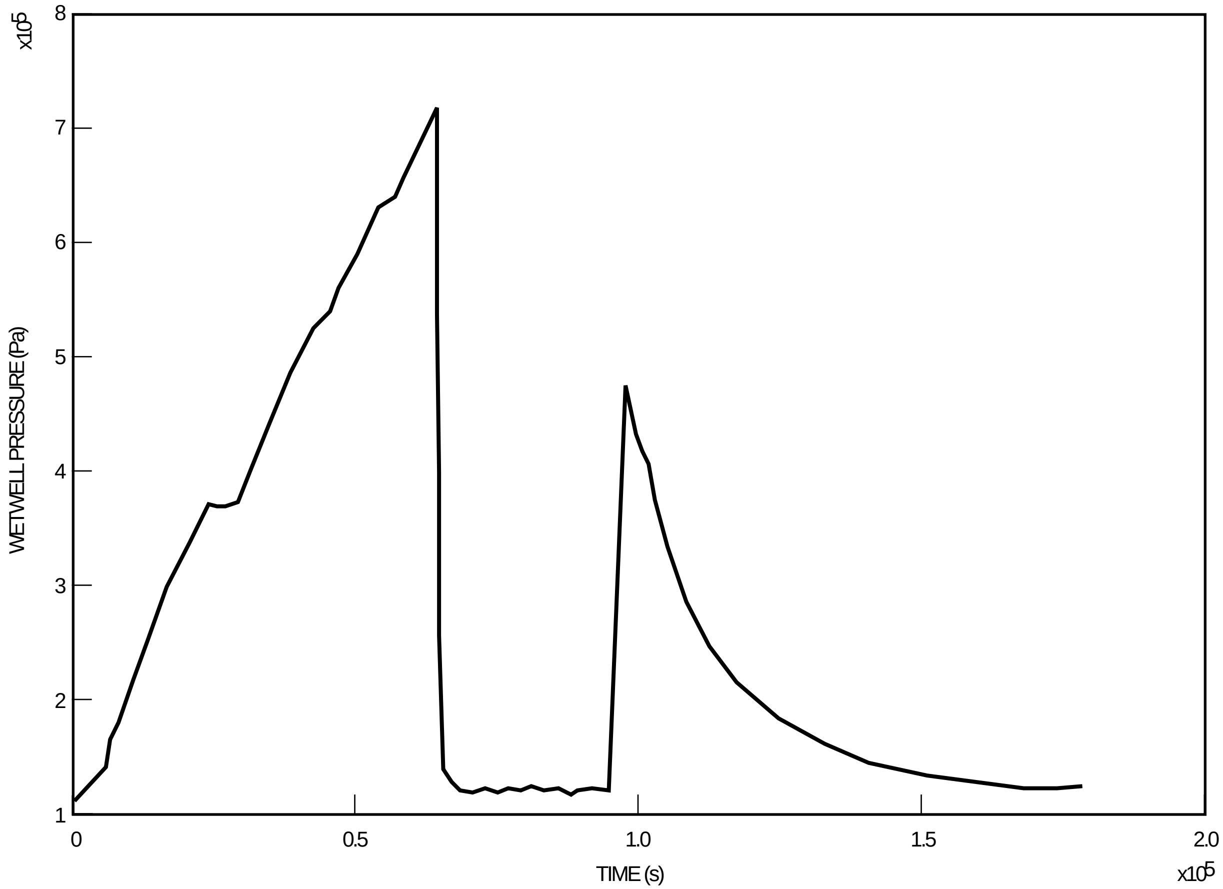

Figure 19EC-4 Sample Calculation for CCI Upward Heat Flux 100 kW/m^2: Wetwell Pressure

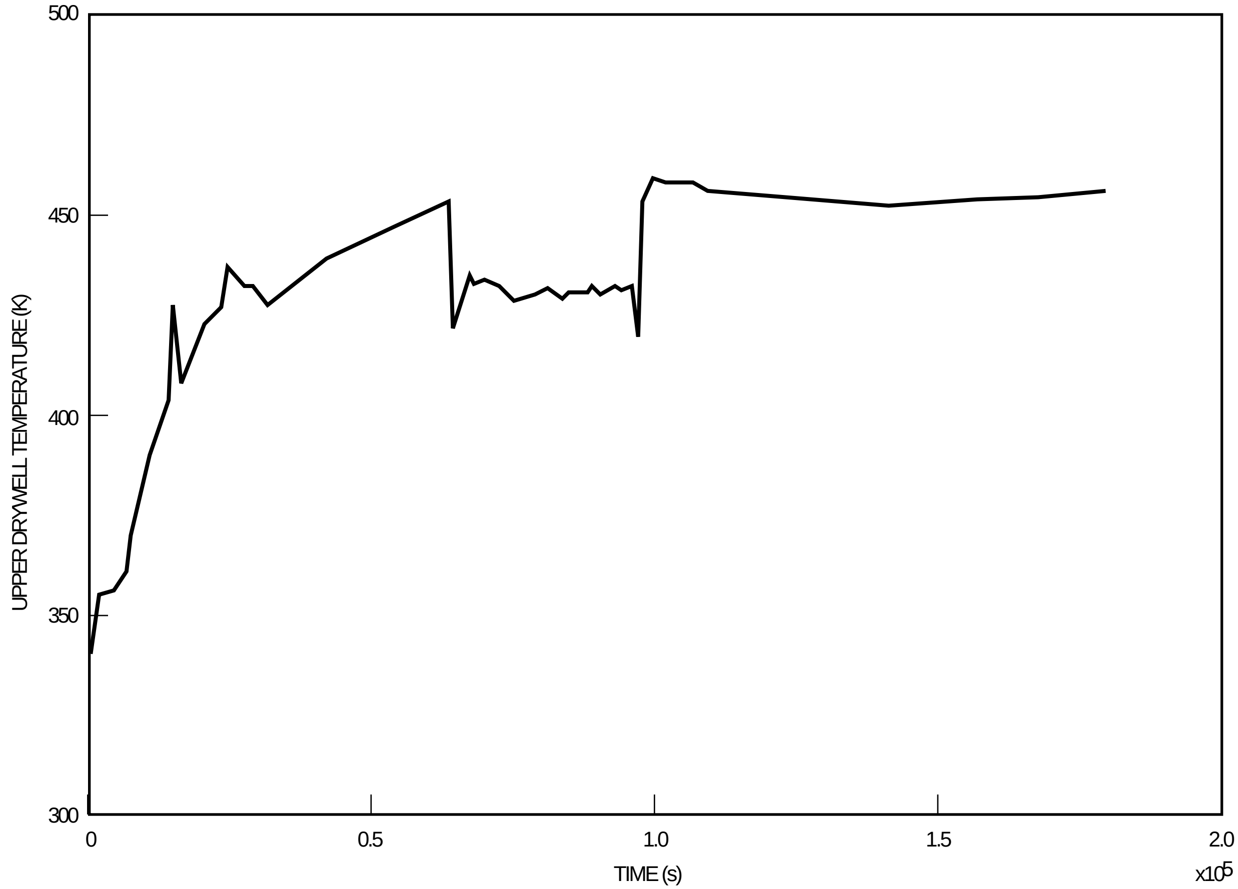

Figure 19EC-5 Sample Calculation for CCI Upward Heat Flux 100 kW/m^2: Upper Drywell Temperature

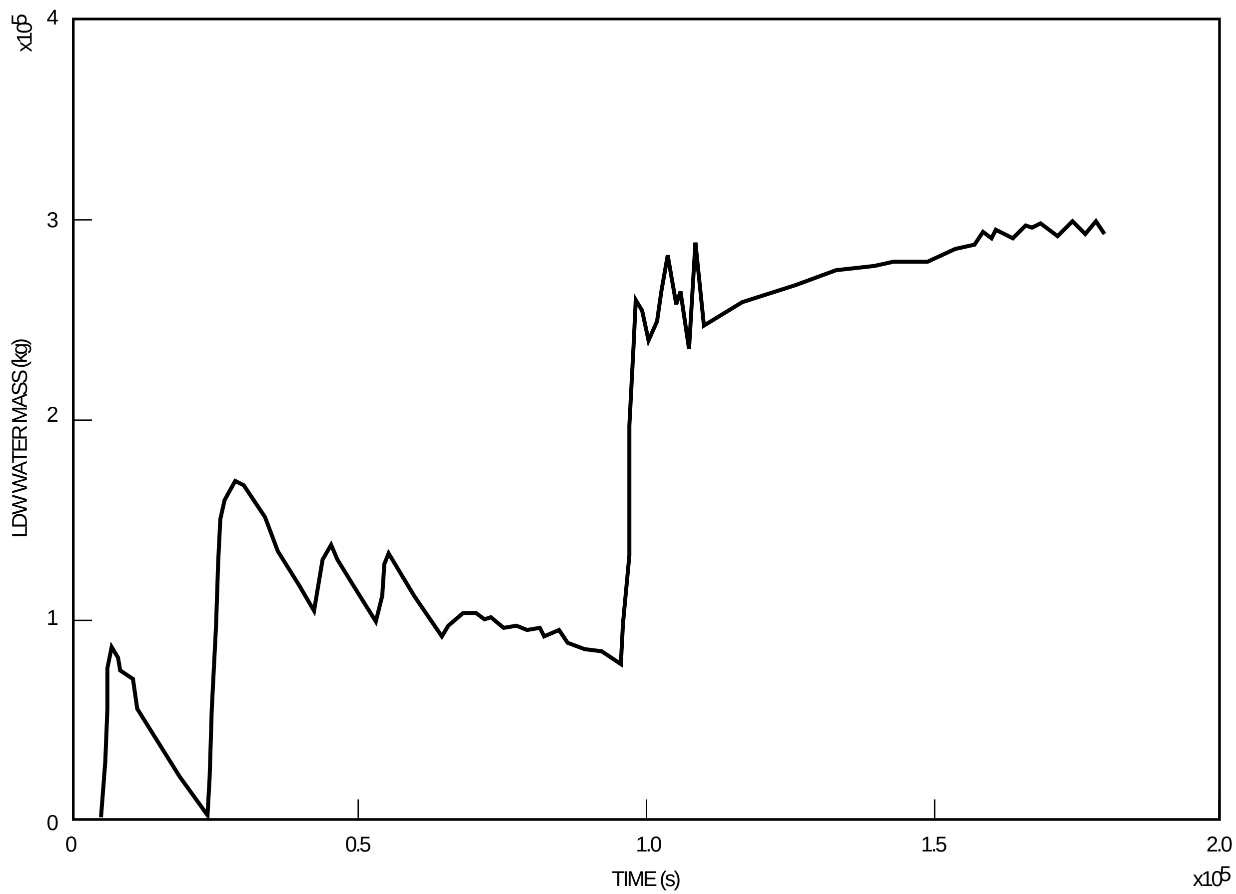

Figure 19EC-6 Sample Calculation for CCI Upward Heat Flux 100 kW/m^2: LDW Water Mass

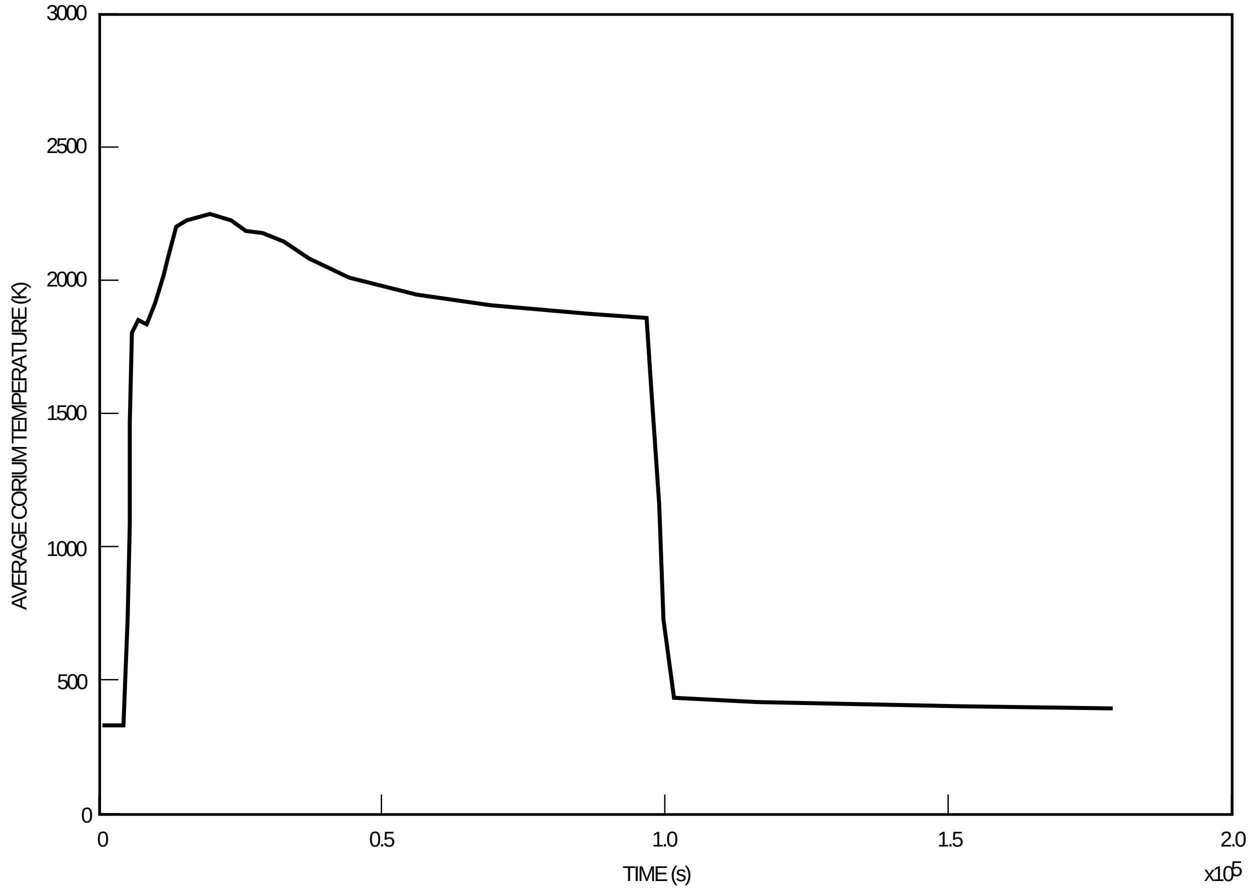

Figure 19EC-7 Sample Calculation for CCI Upward Heat Flux 100 kW/m^2: Average Corium Temperature

Figure 19EC-8 Whole Body Dose at 805 m (0.5 Mile) as a Probability of Exceedence
Not Part of DCD (Refer to SSAR)

19ED Corium Shield

19ED.1 Issue

During a hypothetical severe accident in the ABWR, molten core debris may be present on the lower drywell (LD) floor. The EPRI ALWR Requirements Document specifies a floor area of at least 0.02 m^2/MWth to promote debris coolability. This has been interpreted in the ABWR design as a requirement for an unrestricted LD floor area of 79 m^2.

The ABWR has two drain sumps in the periphery of the LD floor which could collect core debris during a severe accident if ingression is not prevented. If ingression occurs, a debris bed will form in the sump which has the potential to be deeper than the bed on the LD floor. Debris coolability becomes more uncertain as the depth of a debris bed increases.

The two drain sumps have different design objectives. One, the floor drain sump, is designed to collect any water which falls on the LD floor. The other, the equipment drain sump, collects water leaking from valves and piping. Both sumps have pumps and instrumentation which allow the plant operators to determine water leakage rates from various sources. Plant shutdown is required when leakage rate limits are exceeded for a certain amount of time. A more complete discussion on the water collection system can be found in Subsection 5.2.5.

19ED.2 Design Description

A protective layer of refractory bricks—a corium shield—will be built around the sumps to prevent corium ingression. The shield for the equipment drain sump (EDS) will be solid except for the inlet and outlet piping which will go through its roof. The shield for the floor drain sump (FDS) will be similar except that it must have channels at floor level to allow water which falls onto the LD floor to flow into the sump. The channels will be long enough that any molten debris which reaches the inlet will freeze before it exists and spills into the sump. The width and number of the channels will be selected so that the required water flow rate during normal reactor operation is achievable. A sketch of the FDS shield is shown in Figure 19ED-1.

The walls of the EDS shield and the walls of the FDS shield without channels only have to be thick enough to withstand ablation, if any is expected to occur, for the chosen wall material. The walls of the FDS shield containing channels must be thick enough that molten debris flowing through the channels has sufficient residence time to ensure debris solidification.

Both shields extend above the LD floor to an elevation greater than the expected maximum height of the core debris bed. Thus, no significant amount of debris will collect on the shield roofs. The walls of both shields extend below the LD floor to prevent debris from tunneling under the walls and entering the sumps.

Both shields have provisions in their roofs to allow water to flow into the sumps when the lower drywell is flooded. The provisions are located next to the pedestal wall so that the debris which relocates from the vessel can not directly enter the sumps due to geometrical constraints. Additionally, the provision for the roof of the EDS shield will not affect the normal water collection capabilities of the EDS.

To prevent the debris which falls on the lower drywell floor from directly entering the FDS shield, the channels in the FDS shield are in the walls which face away from the center of the lower drywell. The FDS shield wall which faces the center of the lower drywell is solid and does not contain any channels.

The analyses presented in Subsections 19ED.4 and 19ED.5 provide a basis for sizing the FDS shield walls with channels. The sizing of the shield walls without channels is presented in Subsection 19ED.5.3.

19ED.3 Success Criteria

The shield walls must satisfy the following requirements:

(1) Melting Point of Shield Material Above Initial Contact Temperature

The shield wall material will have a melting temperature that is greater than the interface temperature between the debris and the shield wall. Specifying alumina as the shield wall material satisfies this requirement.

(2) Channel Length

The length of the channels in the FDS shield must be long enough to ensure that a plug forms in the channel before debris spills into the sump. The freezing process is expected to take on the order of seconds or less to complete. A channel length of 0.5 meters satisfies this requirement.

(3) Shield Height Above Lower Drywell Floor

The shield height above the lower drywell floor shall be chosen to ensure long term debris solidification and to prevent debris from collecting on top of the shield. The freezing process will be complete during the time frame when the shield walls are behaving as semi-infinite solids. A height of 0.4 meters satisfies this requirement.

(4) Shield Depth Below Lower Drywell Floor

The walls of the FDS and EDS shields extend to the floors of the sumps to prevent tunneling of debris into the sumps.

(5) Channel Flow Area

The total flow area of the channels in the FDS shield shall be great enough to allow water flow rates stated in the Technical Specifications without causing excessive water pool formation in the lower drywell.

(6) Chemical Resistance of Shield Walls

The wall material chosen for the corium shields must have good chemical resistance to siliceous slags and reducing environments. Resistance can be determined to a first degree by comparing the Gibb's free energy of the oxides which make up the shield wall and the oxides present in core debris. Specifying alumina as the shield wall material satisfies this requirement.

(7) Seismic Adequacy

The seismic adequacy of the corium shields will be determined in the detailed design phase. Adequacy should be easily met because the shields are at the lowest point in the containment. Missile generation is not an issue because the shields are not near any vital equipment.

(8) Channel Height

The channel height shall be small enough to ensure freezing. The current analysis is based on a channel height of 1 cm which satisfies this requirement.

19ED.4 Channel Length Analysis

Heat transfer and phase change analyses are presented in this subsection to determine the FDS channel length which prevents molten debris ingression into the sump. A freeze front analysis is performed for early times (on the order of seconds) after vessel failure to determine the time required to form a plug. The freeze front analysis is evaluated for three debris scenarios which envelope the expected debris parameters.

19ED.4.1 Assumptions

The major assumptions invoked in the analysis and their bases follow:

(1) Molten debris enters the channel with negligible superheat.

Molten debris interacts with structural material (steel, concrete, etc.) and the lower drywell environment as it passes from the vessel, contacts the LD floor and spreads to the shield. This interaction depletes the molten debris of any superheat. To account for uncertainties, the impact of superheat will be considered in the sensitivity study contained in Subsection 19ED.4.5.2.

(2) During the freezing process, the temperature profile of the solidified debris rapidly obtains its steady state value.

This assumption introduces little inaccuracy because:

(a) The thermal conductivity of the debris is larger than that of the shield material for most debris scenarios, see Subsection 19ED.4.4.

(b) The depth of the debris is only 1 cm; thus, the thermal time constant of the debris in the channel is small compared to the freezing times. This assumption can be checked by comparing the freeze times calculated considering thermal gradients within the debris and the lumped mass analysis used in the superheat study.

(3) Heat transfer within the channel and shield is one-dimensional during plug formation.

The height of each channel is much less than its length. The heat transfer in the shield material is low enough that any heat transferred from debris contacting the shield wall outside of the channel does not affect the temperature along the channel until long after a plug has formed. Any heat transfer to the shield material between adjacent channels enhances the debris freezing process.

(4) The shield wall acts as a semi-infinite slab with an initial temperature of 330 K during the initial freezing process.

The shield material has been selected such that it is a poor conductor of heat. The penetration depth during the short duration of the freezing process is on the order of ten millimeters. The small increases in LD temperature prior to the presence of core debris does not significantly alter the shield temperature from its value during normal plant operation.

(5) Core debris is not expected to enter the LD until about two hours after accident initiation at which time the decay heat level is approximately 1% of rated power.

Core debris will not enter the lower drywell before about two hours for any credible severe accident (Subsection 19E.2.2).

(6) The decay heat generation in the debris is negligible compared to the rate of latent heat generation during the freezing process.

This assumption was verified during the analysis.

(7) The thermal conductivity and thermal diffusivity of debris in solid and liquid phases are the same.

(8) The contact resistance between the bricks was assumed to be negligible. Contact resistance will be controlled in the detailed design by varying the thickness of the bricks or by using a high temperature binder between the bricks. Thicker bricks tend to minimize overall contact resistance by reducing the number of contact points. Some contact resistance may be acceptable in the final design if the composite thermal conductivity is high enough that the shields provide short- and long-term debris solidification.

(9) The corium shields were assumed to be structurally stable. Structural stability is only an issue during the initial onslaught of debris into the lower drywell. After debris comes into contact with the shields, a crust will form and it will tend to grow in time. Crust formation eliminates buoyancy forces and will hold the individual bricks into place.

19ED.4.2 Initial Freezing of Molten Debris in Channel

If the floor drain sump shield fulfills its design objective, a debris plug will form in the channel before a significant amount of molten corium has a chance to traverse the channel and reach the sump. Molten debris enters the channel at a significantly elevated temperature (1800 K to 2500 K) compared to the shield wall (~ 330 K). The walls absorb heat from the debris because of the large temperature difference. Since the debris contains negligible superheat, any heat loss by the debris results in freezing. Freeze fronts start at the channel walls and move toward the center of the channel. The freezing process is symmetric about the centerline of the channel because the same amount of heat is transferred through each wall while they are behaving as semi-infinite slabs. The channel walls behave as semi-infinite slabs during the freezing process because the heat conduction rate through the wall material is low compared to the release rate of latent heat. A sketch of the freezing process is shown in Figure 19ED-2.

(1) Freezing Time

The temperature profile in the crust (Reference 19ED-1), assuming it quickly reaches its steady state shape, is:

$$T_c(x) = \frac{\dot{q}L_c^2}{2k_f}\left(1 - \frac{x^2}{L_c^2}\right) + \frac{T_s - T_{f,m}}{2}\frac{x}{L_c} + \frac{T_s + T_{f,m}}{2} \qquad (19ED\text{-}1)$$

where:

$T_c(x)$	=	temperature within the crust
x	=	vertical coordinate measured from the crust centerline
$\dot{q}$	=	heat density of the debris
L_c	=	half thickness of the crust
k_f	=	thermal conductivity of debris
T_s	=	interface temperature between the wall and debris
$T_{f,m}$	=	melting temperature of debris

The energy balance at the freeze front is:

$$q''_{lh} = -k_f \left.\frac{dT_c}{dx}\right|_{x = -L_c} \qquad (19ED\text{-}2)$$

where:

q''_{lh} = the latent heat flux

The latent heat flux is:

$$q''_{lh} = \frac{dx_c}{dt}\rho_{cm}h_{lh} \qquad (19ED\text{-}3)$$

where:

x_c	=	crust thickness
t	=	time
ρ_{cm}	=	density of debris
h_{lh}	=	debris latent heat of fusion

Combining these two equations, evaluating the temperature gradient and rearranging yields:

$$\frac{dx_c}{dt} = \frac{1}{\rho_{cm} h_{lh}}\left[\frac{k_f}{x_c}(T_{f,m} - T_s) - \dot{q}\frac{x_c}{2}\right] \qquad (19ED\text{-}4)$$

This is a non-linear, non-homogeneous, first-order differential equation. Before effort is expended to solve it, the relative magnitudes of the terms containing the crust thickness will be determined to see if either one dominates.

The initial interface temperature between the wall of the channel and the debris can be approximated by assuming both the debris and the shield wall behave as semi-infinite solids. The resulting temperature will be somewhat less than the actual interface temperature because the freezing process will force the crust to stay closer to its initial temperature than it would if it were a semi-infinite solid body only experiencing conduction. The contact temperature between the debris and the channel wall (Reference 19ED-2), assuming semi-infinite bodies, is:

$$T_s = \frac{T_{f,m}\sqrt{(k\rho c)_{cm}} + T_i\sqrt{(k\rho c)_w}}{\sqrt{(k\rho c)_{cm}} + \sqrt{(k\rho c)_w}} \qquad (19ED\text{-}5)$$

where:

T_i = initial temperature of shield wall (assumed to be 330K)

c = specific heat

cm = debris material properties

w = wall material properties

Using the material properties for the wall and the debris contained in Tables 19ED-1 and 19ED-2, respectively, the contact temperature is estimated to be between 1475 and 2070K.

The debris energy generation density can be found by assuming a decay heat level and a total amount of corium. The density is:

$$\dot{q} = \frac{Q_{dh}\rho_{cm}}{m_{cm}} \qquad (19ED\text{-}6)$$

where:

Q_{dh} = decay heat level

m_{cm} = total mass of corium, 235 Mg.

Assuming the decay heat level is approximately 1% of rated power yields:

$$\dot{q} = 1.5\ \mathrm{MW/m^3}$$

The two terms inside the brackets in Equation 19ED-4 can now be evaluated. For a channel height of 1 cm ($x_{c,Max}$ = 0.5 cm) and melt Scenario I (the scenario with the limiting terms) these values are:

$$\frac{k_f}{x_c}(T_{f,m} - T_s) = 9.72 \times 10^5\ \mathrm{W/m^2}$$

$$\dot{q}\frac{x_c}{2} = 3.8 \times 10^3\ \mathrm{W/m^2}$$

Therefore, the term containing the temperature difference across the crust is much larger than the one containing the heat generation rate. The temperature profile in the channel system ignoring energy generation in the debris is shown in Figure 19ED-2. Equation 19ED-4 can be simplified to:

$$\frac{dx_c}{dt} = \frac{k_f}{\rho_{cm} h_{lh} x_c}(T_{f,m} - T_s) \qquad (19ED-7)$$

Solving this equation with the initial condition that $x_c(t = 0) = 0$, reveals:

$$x_c = \sqrt{\frac{2k_f(T_{f,m} - T_s)t}{\rho_{cm} h_{lh}}} \qquad (19ED-8)$$

This equation can be rearranged to determine the time required to freeze debris in a channel of height H_o. The freezing time is:

$$t_{freeze} = \frac{H_o^2 \rho_{cm} h_{lh}}{8k_f(T_{f,m} - T_s)} \qquad (19ED-9)$$

(2) Interface Temperature, T_s

The interface temperature between the debris and the channel wall can be determined by equating the heat flux from the crust to that which the crust can absorb. The heat flux from the crust is:

$$q''_{crust} = -k_f \left.\frac{dT_c}{dx}\right|_{x = x_c/2} \tag{19ED-10}$$

From Equation 19ED-1, this evaluates to:

$$q''_{crust} = \frac{\dot{q} x_c}{2} + \frac{k_f}{x_c}(T_{f,\,m} - T_s) \tag{19ED-11}$$

As shown previously, the temperature-difference term dominates the energy-generation term in this equation for small channel heights. Therefore, the crust heat flux can be simplified to:

$$q''_{crust} = \frac{k_f}{x_c}(T_{f,\,m} - T_s) \tag{19ED-12}$$

Inserting the expression for x_c in Equation 19ED-8 and rearranging yields:

$$q''_{crust} = \sqrt{\frac{k_f \rho_{cm} h_{lh}(T_{f,\,m} - T_s)}{2t}} \tag{19ED-13}$$

The heat flux (Reference 19ED-3) absorbed by the channel wall can be approximated by that which a semi-infinite solid body can absorb. This flux is:

$$q''_w = \frac{k_w(T_s - T_i)}{\sqrt{\pi \alpha_w t}} \tag{19ED-14}$$

where:

α_w = thermal diffusivity of the wall material

Combining Equation 19ED-13 and Equation 19ED-14 produces a relationship governing the interface temperature. It is:

$$\frac{T_s - T_i}{\sqrt{T_{f,\,m} - T_s}} = \left(\frac{\pi k_f \rho_{cm} h_{lh} \alpha_w}{2k_w^2}\right)^{1/2} \tag{19ED-15}$$

Solving this equation for T_s using the quadratic formula yields:

$$T_s = \frac{-(c_o - 2T_i) \pm \sqrt{(c_o - 2T_i)^2 - 4\left(T_i^2 - c_o T_{f,m}\right)}}{2} \qquad \text{(19ED-16)}$$

where:

c_o = the square of the right hand side of Equation 19ED-15

Negative solutions of this equation have no physical meaning. Using the material properties for the wall and the debris contained in Tables 19ED-1 and 19ED-2, respectively, the interface temperature is between 1580K and 1900K.

Since this temperature range is within the range determined for two semi-infinite bodies coming into contact, the dominance of the temperature term in Equations 19ED-4 and 19ED-11 is still valid.

19ED.4.3 Required Channel Length to Insure Freezing

The propagation rate of the freeze front was determined in form a freeze plug Subsection 19ED.4.2. This allows the determination of the time to form a freeze plug in a channel of specified height. A simple approximation of the channel length, required to provide this residence time, is the product of the initial molten debris velocity and the freezing time. This approximation would predict shield dimensions considerably larger than actually required. A more realistic channel length can be obtained by considering the reduction in channel flow area as debris freezes. In the remainder of this subsection, the following parameters will be determined:

- debris velocity at channel entrance,
- channel area decrease resulting from debris freezing,
- average channel debris velocity, and
- the required channel length to insure plug formation at the channel entrance before corium ingression into the sump.

(1) Debris Velocity at Channel Entrance

The possibility exists that molten debris will not even enter the channel after it has come into contact with the shield wall. Debris which is spreading across the lower drywell floor will have at least a thin crust formed on its leading edge. If the flow energy of the advancing debris front is not great enough to break this crust and overcome surface tension on the length scale of the channel height, debris will not enter the channel. Unfortunately, the physics of crust formation is not currently understood well enough to support this argument without a great deal of uncertainty.

Since the channels are in shield walls which are not facing the center of the lower drywell, the entrance velocity is governed by the height of corium outside of the channel. Assuming that the debris spreads uniformly across the lower drywell floor, the height of debris can be obtained by integrating the volumetric expulsion rate of corium from the vessel divided by the floor area of the lower drywell. The expulsion rates for three scenarios will be considered in Subsection 19ED.4.4. The three scenarios cover the spectrum of core melt phenomena and debris properties.

The velocity at the channel entrance can be conservatively over predicted by ignoring frictional effects. Frictional effects should actually be quite large because the viscosity of the debris will increase dramatically as it freezes. This velocity is:

$$v_e(t) = \sqrt{2g\Delta z(t)} \tag{19ED-17}$$

where:

v_e	=	velocity at the entrance of the channel
g	=	gravitational acceleration constant
Δz	=	height of debris in the lower drywell

Expanding debris height yields:

$$v_e(t) = \sqrt{\frac{2g\dot{m}_{ves}t}{\rho_{cm}A_{ld}}} \tag{19ED-18}$$

where:

$\dot{m}_{ves}$	=	ejection rate of corium from a failed vessel
A_{ld}	=	floor area of the lower drywell, 79 m^2 minimum

(2) Channel Area Decrease Resulting From Debris Freezing

The mass flow rate of corium in the channel decreases in time due to the area reduction resulting from debris freezing. A conceptual picture of this area reduction process is shown in Figure 19ED-3. The mass flow rate at the entrance of the channel and at the location downstream where the debris front has just arrived is:

$$\dot{m}_i(t) = \rho_{cm} v_e(t) H_i(t) = \rho_{cm} v_o(t) H_o + \dot{m}_{fr} \qquad (19ED\text{-}19)$$

where:

$\dot{m}_i$	=	time varying mass flow rate per unit width at the entrance of the channel
H_i	=	time varying entrance flow height of the channel
v_o	=	time varying velocity at the downstream location in the channel where molten debris has just arrived
H_o	=	unobstructed height of the channel
$\dot{m}_{fr}$	=	mass freezing rate of debris per unit width in the channel

This equation requires that:

$$v_o(t) = \frac{v_e(t)}{H_o} H_i(t) - \frac{\dot{m}_{fr}}{\rho_{cm} H_o} \qquad (19ED\text{-}20)$$

The entrance flow height is:

$$H_i(t) = H_o - 2x_c(t) \qquad (19ED\text{-}21)$$

Inserting the relationship for x_c found in Equation 19ED-8 into this expression yields:

$$H_i(t) = H_o - \sqrt{\frac{8k_f(T_{f,m} - T_s)t}{\rho_{cm} h_{lh}}} \qquad (19ED\text{-}22)$$

The product of this equation and the width of the shield channel describes the reduction of channel inlet flow area with time.

(3) Average Channel Debris Velocity

The velocity of the leading edge of molten debris in the channel can be obtained by combining Equations 19ED-18, 19ED-20 and 19ED-22. It is:

$$v_o(t) = a_o\sqrt{t} - \frac{2a_ob_o}{H_o}t - \frac{\dot{m}_{fr}}{\rho_{cm}H_o} \qquad \text{(19ED-23)}$$

where:

$$a_o = \sqrt{\frac{2g\dot{m}_{ves}}{\rho_{cm}A_{ld}}}$$

$$b_o = \sqrt{\frac{2k_f(T_{f,m} - T_s)}{\rho_{cm}h_{lh}}}$$

(19ED-24)

The average velocity of debris between the entrance of the channel and the leading edge of molten corium is:

$$\bar{v}(t) = \frac{\int_0^t v_o(t)\,dt}{\int_0^t dt} \qquad \text{(19ED-25)}$$

Evaluating this integral yields:

$$\bar{v}(t) = \frac{2}{3}a_o\sqrt{t} - \frac{a_ob_o}{H_o}t - \frac{1}{t}\int_0^t \frac{\dot{m}_{fr}}{\rho_{cm}H_o}dt \qquad \text{(19ED-26)}$$

This is the time averaged velocity of the molten debris in the shield channel.

(4) Mass of Debris Frozen in Channel

The time varying mass of debris freezing in the channel per unit width can be approximated by

$$\dot{m}_{fr} = \frac{d}{dt}(A'\rho_{cm}) \qquad \text{(19ED-27)}$$

where:

A' = cross sectional area of frozen debris

The cross sectional area can be related to the crust thickness by modifying Equation 19ED-8 to account for the variable residency time of the debris at various vertical locations in the channel. This process yields

$$A' = 2b_o \int_o^L \sqrt{t - \frac{y}{\bar{v}(t)}}\,dy \tag{19ED-28}$$

where:

L = length from the channel entrance to the leading edge of the debris front

y = vertical coordinate measured from the entrance of the channel

Evaluating this integral yields:

$$A' = \frac{4}{3} b_o \bar{v}(t)\, t^{3/2} \tag{19ED-29}$$

Combining this result with Equations 19ED-25 and 19ED-26 yields:

$$\bar{v}(t) = \frac{2}{3} a_o \sqrt{t} - \frac{a_o b_o}{H_o} - \frac{4 b_o \bar{v}(t)}{3 H_o} \sqrt{t}$$

$$= \frac{\frac{2}{3} a_o \sqrt{t} - \frac{a_o b_o}{H_o} t}{1 + \frac{4 b_o}{3 H_o} \sqrt{t}} \tag{19ED-30}$$

(5) Required Channel Length to Insure Freezing

The channel length, required to ensure a plug forms at the channel entrance before debris spills into the sumps, is:

$$L_{freeze} = \bar{v}(t_{freeze})\, t_{freeze} \tag{19ED-31}$$

where t_{freeze} is given by Equation 19ED-9.

19ED.4.4 Channel Lengths for Different Melt Scenarios

The analysis to determine the channel length required to ensure that a plug forms in the channel prior to debris entering the sump is contained in Subsection 19ED.4.3. The analysis will be executed in this subsection for three different melt scenarios which cover the range of expected core melt conditions. The scenarios differ in debris composition, debris material properties, initial temperature of the debris and the ejection rate of debris from the failed vessel. The impact of debris superheat will be considered in a sensitivity study contained in Subsection 19ED.4.5.2.

Scenarios I and II were taken from NUREG/CR-5423 (Reference 19ED-7). These scenarios represent predominantly oxidic and metallic melts, respectively. The Peach Bottom Atomic Power Station was used for specifics in the NUREG/CR-5423 calculations. However, the similarities between Peach Bottom and the ABWR in core composition and vessel geometry allow the NUREG/CR-5423 core melt parameters to be applied in this analysis.

Scenario I is based on MAAP calculations which predict that there is a significant amount of molten debris available for relocation at the time of vessel failure. The debris release in this scenario is consistent with the core composition, corresponding to approximately 30% by weight of zirconium. Further, 20% of the available zirconium was assumed to be oxidized.

Scenario II is based on BWRSAR calculations which predict that debris which initially relocates to the lower plenum of the vessel is quenched. Subsequent water depletion results in the remelting of the debris and local failure of the vessel. Since the metallic constituents of the debris remelt before the oxidic constituents, the initial debris pour from the vessel is primarily metallic.

For both scenarios, only the initial molten material relocation is important in determining the required channel length because the channels will plug in a relatively short period of time (less than 10 seconds). Only the maximum initial debris relocation rates were utilized so that the calculated channel lengths would be conservatively over-estimated. Subsequent molten material releases from the vessel will go into filling the lower drywell with debris and have no bearing on plug formation. Thus, the long-term debris relocation parameters discussed in NUREG/CR-5423 are of no consequence to this analysis.

The third scenario considered, Scenario M, represents the MAAP-ABWR analysis that was used to predict a bounding debris ejection rate from a failed oxidic vessel, see Subsection 19EB.6.2.2. MAAP-ABWR predicts that the mass flow rate from the vessel jumps to approximately 1000 kg/s at vessel failure and then increases to 6000 kg/s in eight seconds, see Figure 19EB-9. The debris release is essentially complete (flow rate 0 kg/s) in ten seconds. In order to avoid determining the debris entrance velocity into the channel for this complicated flow rate profile, the maximum debris relocation rate was assumed to prevail throughout the plug formation process. This assumption will lead to a conservative over-estimation of the required channel length.

Another melt scenario commonly considered in calculations involving ex-vessel core debris is the formation of eutectics as a result of core-concrete interactions. However, consideration of eutectics is not required in the analysis of plug formation. In order for the debris properties to be changed significantly, core-concrete interactions must increase the debris mass by at least 10%. The time required for this to happen is longer than the time required for plug formation for a quickly spreading debris front. Alternatively, if the debris front is spreading slowly enough to allow a significant amount of core-concrete interaction, the leading edge of the debris will have a thick crust and have a height greater than the channel height. Thus, debris will not even enter the channel.

The parameters describing each of the three scenarios considered is contained in Table 19ED-2. The debris material properties for Scenarios I and II were determined using the constituent material properties contained in Table 19ED-3. Material properties for Scenario M were taken from MAAP-ABWR.

Inserting the scenario parameters contained in Table 19ED-2 into the analysis contained in Subsection 19ED.4.3 results in the channel lengths contained in Table 19ED-4. All of the required channel lengths are less than 0.5 meters.

19ED.4.5 Sensitivity to Melt Parameters

The three scenarios considered in Subsection 19ED.4.4 were chosen to represent a wide range of possible debris characteristics. The calculation of required channel length depends on the debris flow rate, the debris temperature, the channel height and several material properties of the debris and wall material. This subsection will evaluate the sensitivity of the calculation to these parameters. The final part of this subsection will evaluate the impact of debris superheat.

19ED.4.5.1 Material Properties

The channel length required to ensure freezing is dependent on both debris and channel wall material properties. The relevant debris material properties are density, latent heat of fusion and thermal conductivity. For the channel wall, the relevant properties are thermal conductivity and thermal diffusivity. The sensitivity of the channel length calculation to these material properties will be determined in this subsection. Additionally, the sensitivity to debris temperature and debris flow rate from the vessel will also be determined.

The sensitivity of the channel length calculation to these parameters will be estimated by varying each parameter, except debris temperature, by ± 20%. Debris temperature will be varied by ± 200K. The wide variations in material properties of the three base scenarios take into account deviations outside of this range and combinations of deviations.

The results of varying these parameters are contained in Table 19ED-5. The following discussion will describe the effect on channel length due to increasing each parameter. The reverse of the effect described holds true for decreasing the parameter.

Increasing debris density, latent heat of fusion or flow rate increases the amount of energy that must be transferred to the channel walls before a plug forms, and, as a result, increases the required channel length. Increasing the debris thermal conductivity increases the rate at which the debris can transfer its energy, and decreases the required channel length. Increasing the debris temperature increases the debris to channel wall temperature difference and, as a result, the rate of heat loss by the debris. Since this analysis assumes that the debris enters the channel with negligible superheat, increasing the debris temperature reduces the required channel length.

The impact on channel length due to variations in wall material properties is a direct result of the change in the interface temperature between the debris and the channel wall. Increasing the wall thermal conductivity decreases the interface temperature and results in a shorter channel length. Conversely, increasing the wall thermal diffusivity increases the interface temperature and results in a longer channel length.

Decreasing the interface temperature increases the temperature difference driving heat flow from the debris, and results in more rapid freezing. Alternatively, increasing the interface temperature decreases the interface temperature, and results in a longer freezing length. The interface temperature is decreased by increasing wall conductivity and increased by increasing wall thermal diffusivity. Thus, increasing the wall thermal conductivity decreases the required channel length, while the opposite is true for increasing wall thermal diffusivity.

The three parameters which have the largest impact on channel length are debris density, debris latent heat of fusion and wall thermal conductivity. The impact of the debris properties are not surprising because they directly impact the amount of energy that must be removed from the debris in order for plug formation to occur. The impact of decreasing the thermal conductivity of alumina does not need to be considered because a lower bound of alumina thermal conductivity was used in the base analysis. Thus, a ten-percent decrease in the wall thermal conductivity is physically unrealistic.

As can be seen in Table 19ED-5, none of the parameter variations for Scenarios I and II resulted in required channel lengths greater than 0.5 meters. Only two of the variations for Scenario M resulted channel lengths greater than 0.5 meters, and these channel lengths were only slightly greater. If the channels are 0.5 meters long and two meters wide, the amount of debris which could enter the sump for a Scenario M melt with these two parameter variations is approximately 0.001 m^3 (average depth 0.03 cm). This amount of debris entering the sump does not pose a threat to containment integrity Therefore, a channel length of 0.5 meters supplies enough margin to account for uncertainties in material properties.

19ED.4.5.2 Impact of Superheat

The channel length analysis contained in Subsection 19ED.4.3 assumes that the debris enters the channel with negligible superheat. The analysis contained in this subsection considers the effects of superheat on the corium freezing process. First, the credible amount of superheat for each melt scenario is discussed. Then, the change in energy due to including superheat will be calculated. Next, the impact on freezing time will be determined. The analysis will conclude with the determination of channel length when superheat is present.

According to NUREG/CR-5423 (Reference 19ED-7), the initial superheat of Scenario I is expected to be negligible “because of several concurrent reasons: (a) convective heat transfer to boundaries--typically less than 100 °C can be sustained at decay heat levels, (b) continuous melting and incorporation of boundary material, and (c) heat losses to water and control rod guides during the relocation through the lower plenum.” The upper bound of reasonable superheat that the debris could obtain at the time of vessel failure was specified to be 125 °C. The most probable superheat was limited to less than 50 °C. Due to similarities in Scenario I and Scenario M (quick penetration failure after delayed core plate failure and mostly oxidic melt), the superheat of the two cases can be assumed to be the same.

Scenario II is expected in NUREG/CR-5423 to have more superheat than Scenario I because the molten mass does not have as much opportunity to lose heat to the lower head. The upper bound of reasonable superheat that the Scenario II debris could obtain at the time of vessel failure was specified to be 150 °C. The most probable superheat was limited to less than 100 °C.

After the debris exits the vessel in any of the scenarios, the debris will lose some of its heat before it reaches the corium shields. The heat loss will be by radiation and convection to the lower drywell environment and structures. Additionally the debris pool on the lower drywell floor will conduct heat to the lower drywell floor. These heat losses will remove some, if not all, of the debris superheat. This analysis conservatively assumes that the debris enters the shield wall channels with the same superheat it had when exiting the vessel.

19ED.4.5.2.1 Change in Energy

The fractional change in debris energy due to adding superheat can be obtained by comparing the energy content of the debris with superheat to the energy content without superheat. This yields

$$E_{change} = \frac{mh_{lh} + mc_p\Delta T}{mh_{lh}} = \frac{c_p\Delta T}{h_{lh}} \tag{19ED-32}$$

where:

m	=	mass of debris to be frozen
c_p	=	heat capacity of debris
ΔT	=	amount of superheat

Table 19ED-6 shows the percent change in energy which results from assuming different amounts of superheat for the three scenarios. The energy comparison indicates that a significant amount of superheat (greater than 25 °C) could impact the channel length. However, for superheats on the order of a few degrees, there is negligible impact.

19ED.4.5.2.2 Channel Length with Superheat

A simplified analysis of the channel length required to ensure plug formation before debris ingression into the sumps will be presented in this subsection. The superheat temperature will be assumed to decrease the fusion point of the melt, not increasing the temperature of the melt exiting the vessel. This will conservatively result in longer required channel lengths, as indicated by Subsection 19ED.4.5.1.

The alternative manner of accounting for superheat is increasing the temperature of the debris. This results in a shorter channel length because the interface temperature increases compared to the case in which the fusion point is decreased. The interface temperature for all three scenarios resulting from adding the upper bounds of superheat are less than 1975K. This temperature is less than the melting temperature of alumina. Therefore, adding superheat to the debris temperature results in a shorter freezing length without thermally degrading the channel walls. The remainder of this analysis will conservatively account for superheat by decreasing the solidus temperature.

Balancing the energy required to be removed from the debris to freeze and the energy to be absorbed by the shield wall, assuming that the debris in the channel behaves as a lumped mass, yields

$$t_{freeze} = \frac{e''_f}{\bar{q}''_w}$$

where e''_f = energy required to be removed from the debris in order for freezing to occur

$$= \frac{H_o \rho_{cm}}{2}(h_{lh} + c_p \Delta T) \tag{19ED-33}$$

$\bar{q}''_w$ = time averaged heat flux into either the upper or lower channel wall (semi-infinite body), see Equation 19ED-14 for the instantaneous heat flux

$$= \frac{2k_w(T_s - T_i)}{\sqrt{\pi \alpha_w t}} \tag{19ED-34}$$

Combining all the portions of this equation yields

$$t_{freeze} = \left[\frac{H_o \rho_{cm}(h_{lh} + c_p \Delta T)\sqrt{\pi \alpha_w}}{4k_w(T_s - T_i)}\right]^2 \tag{19ED-35}$$

The plug formation time for the three scenarios is presented in Table 19ED-7 for various amounts of superheat. Note that this equation produces the same result as Equation 19ED-9 if there is no superheat.

The average velocity in the channel can be obtained by following the same methodology used in Subsection 19ED.4.3 for the case without superheat. Performing this analysis yields

$$\bar{v} = \frac{\frac{2}{3}a_o\sqrt{t} - \frac{a_o b'_o}{H_o}t}{1 + \frac{4b'_o}{3H_o}\sqrt{t}}$$

$$\text{where} \qquad b'_o = \frac{2k_w(T_s - T_i)}{\rho_{cm}(h_{1h} + c_p\Delta T)\sqrt{\pi\alpha_w}} \qquad (19ED\text{-}36)$$

Note that this equation and Equation 19ED-29 are identical except for the definition of the parameter b_o which describes the ratio of heat removal capability to the energy that must be removed for freezing to occur. The channel length required to ensure freezing is simply the product of the average velocity and the freezing time.

The required channel lengths for each of the three scenarios is shown in Table 19ED-8 for various amounts of superheat. All of the Scenario I and II required channel lengths are less than 0.5 meters. The required channel lengths for Scenario M exceed 0.5 meters for superheats greater than approximately 70 °C. However, for a channel length of 0.5 meters, the amount of debris that can enter the sump under these conditions is small. The average velocity of the debris in the channel for the Scenario M is less than approximately 0.1 m/s. Assuming that the total channel width is 2 meters, the amount of debris that can enter the sump is less than 0.006 m^3(0.2 cm average depth) for superheats up to 125 °C. This amount of debris entering the sump does not pose a threat to containment integrity.

19ED.4.5.2.3 Conclusion of Superheat Impact

A channel length of 0.5 meters will prevent significant debris ingression into the sumps for credible superheats in all three debris scenarios. No debris is expected to enter the sump for either the Scenario I or II melts, while a small amount may enter for Scenario M melts for superheats in excess of 70 °C. Based on the methodology of NUREG/CR-5423, the superheat in Scenario M type melt is probably limited to less than 50 °C. For Scenario M melts with superheats outside the probable range, only a small amount of debris will enter the sump and the average depth will be limited to approximately 0.2 cm. Therefore, the corium shield for the floor drain sump will perform its function even if the core debris exiting the vessel is superheated.

19ED.4.6 Conclusion of Channel Length Analysis

A channel length of 0.5 meters provides adequate assurance that molten debris that enters the floor drain sump corium shield will form a plug prior to debris spilling into the sump. Three debris melt scenarios were considered which bound the credible melt compositions and the credible debris ejection rates from the vessel. Two of the scenarios represented the oxidic and metallic melts used in the Mark I liner failure analysis, NUREG/CR-5423. The third scenario was developed with MAAP-ABWR to provide a gross over-estimation of the maximum debris ejection rate from a failed vessel. A sensitivity analysis demonstrated that a channel length of 0.5 meters provides enough margin to account for uncertainties in material properties. Additionally, the impact of superheat was shown to be minimal.

19ED.5 Long-Term Capability of the Shield Walls

Initial debris solidification was considered in Subsection 19ED.4. The requirements for keeping the debris in the channel frozen for an extended period of time (at least 24 hours) will be determined in this subsection. The height of the upper shield walls (above the lower drywell floor) with channels and depth of the lower shield walls (below the lower drywell floor) with channels will be specified. Additionally, the thickness of the shield walls without channels will be specified.

19ED.5.1 Upper Shield Wall (Above Lower Drywell Floor) with Channels

To help assure the integrity of the roof of the corium shield, the upper shield wall should be tall enough to prevent the debris that is collecting on the lower drywell floor from flowing on top of the shield. Debris falling directly on the roof during relocation from the vessel does not pose a threat to roof integrity because the amount of debris that can fall on the roof is small. This is a result of the CRD and lower drywell configurations. The length and density of the CRDs in the lower drywell prevents the debris from exiting the CRD grid with any significant horizontal velocity component. The sumps are on the periphery of the lower drywell floor. Thus, debris which relocates from the vessel will not fall directly on the sump roofs.

To prevent any debris from flowing on top of the shield roof, the shield should be taller than the maximum possible debris pool depth in the lower drywell. This requirement is given by:

$$H_{uw} \geq \frac{m_{cm,\,tot}}{\rho_{cm} A_{ld,\,min}} \tag{19ED-37}$$

where:

$m_{cm,tot}$ = maximum amount of corium, 235 Mg

$A_{ld,min}$ = minimum floor area of the lower drywell, 79 m^2

Evaluating this expression yields:

$$H_{uw} \geq 0.33 \text{ m} \tag{19ED-38}$$

After debris relocation into the lower drywell, the lower drywell will almost always be flooded with water either by active systems (e.g., the firewater addition system) or by the passive flooder. The probability that the lower drywell will not be flooded after debris relocation is extremely small and is low enough that the case of a non-flooded lower drywell can be excluded from consideration.

The water in the lower drywell will provide long-term cooling to the debris on the floor, to any debris that is on the roof of the corium shield, and to the corium shield walls. Additionally, since the roof of the corium shield allows water flow into the sump when the lower drywell is flooded, the inner walls of the shield will be cooled. The heat transfer from the shield walls to the water is effective in preventing the debris frozen in the channels from remelting. Therefore, if the lower drywell is flooded, long-term solidification of the debris in the channels is assured and debris ingression into the sump is prevented.

To meet the requirements set forth in this subsection, the upper shield wall is specified to be 0.4 meters.

19ED.5.2 Lower Shield Wall (Below Lower Drywell Floor) with Channels

As stated in Subsection 19ED.5.1, long-term solidification of the debris in the channels is assured due to heat transfer from the shield walls to the water which has filled the lower drywell due to initiation of an active system or the passive flooder. Therefore, the only requirements for the lower shield wall are to absorb the initial energy released by the debris during the freezing process and to prevent tunneling of the debris beneath the shield when significant core-concrete interaction has not already occurred.

During the freezing process, the channel walls behave as semi-infinite bodies with a temperature penetration depth less than a centimeter. Thus, the lower shield wall needs to have a depth of at least one centimeter to meet the requirement for initial debris freezing.

If core-concrete interaction is occurring, the potential exists that the lower drywell floor will be eroded to a depth below the lower shield wall. If this occurs, the debris could tunnel into the sump. This concern is eliminated by specifying that the shield wall extends down to the floor of the sump.

19ED.5.3 Shield Walls without Channels

The discussion in most of this attachment has focused on the shield walls with channels. This subsection will address the requirements for the shield walls without channels. The thickness, height and depth will be specified.

The corium shield walls without channels only need to be thick enough to provide a long-term, stable interface between the debris on the lower drywell floor and the interior of the sumps. As discussed in Subsection 19ED.5.1, only the long-term scenario with a flooded lower drywell needs to be considered.

The wall thickness needed to transfer a given heat flux under steady-state conditions is

$$\Delta w = \frac{kw\,(T_{w,\,o} - T_{w,\,i})}{q''_d} \tag{19ED-39}$$

where:

$T_{w,o}$ = surface temperature of wall in contact with debris

$T_{w,i}$ = surface temperature of wall on inside of shield, and

q''_d = heat flux from the debris.

Assuming that the water in the sump is at three atmospheres, the inner shield wall will achieve a temperature of approximately 410 K to allow nucleate boiling. The MAAP-ABWR analysis contained in Subsection 19E2.2 demonstrate that the typical drywell pressure at the time of vessel failure is three atmospheres. To avoid ablation, the wall surface temperature in contact with debris must be less than the melting temperature of the shield wall material (approximately 2280 K for alumina).

The heat flux from the debris bed in the lower drywell can be approximated by assuming that the decay heat level is one-percent and that all the surfaces of the bed have the same heat flux. This approximation yields:

$$q''_d = 250\ \mathrm{kW/m^2}. \tag{19ED-40}$$

The actual heat flux to the wall may be significantly lower due to enhanced heat transfer from the debris bed to the overlying pool of water or separation of the bed from the wall.

Evaluating the wall thickness for these conditions yields

$$\Delta x = 3\ \mathrm{cm}. \tag{19ED-41}$$

This wall thickness provides a stable interface between the debris bed and a water filled sump. If the wall is thicker, it will ablate to this thickness and then establish a stable interface. To provide margin for any erosion due to initial debris contact with the wall, the thickness of the shield walls without channels is specified to be 10 cm.

The reasoning contained in Subsections 19ED.5.1 and 19ED.5.2 regarding the height and depth of the shield walls with channels also applies to the walls without channels. The height of the shield walls should be 0.4 meters which is greater than the maximum height of the debris bed in the lower drywell. The shield walls should extend to the floor of the sump to prevent debris tunneling.

19ED.6 Related Experimental and Analytical Work

The freezing of molten fuel in narrow channels and tubes has been studied previously in regards to core disruptive accidents in liquid metal fast breeder reactors (References 19ED-10, 19ED-11, 19ED-12 and 19ED-13) and in regards to ceramic core retention devices (Reference 19ED-13). This subsection will review these works for application to the channel freezing analysis contained in Subsection 19ED.4.

Cheung and Baker (Reference 19ED-10) analytically and experimentally studied the transient flow and freezing of molten core debris in coolant channels of a liquid-metal fast breeder reactor. Their data analysis determined the impact of several parameters on the penetration depth of the fuel into the channel. The derived variations are in general agreement with the trends shown in the sensitivity study contained in Subsection 19ED.4.5. This work culminated in the determination of penetration depths for several coolant channel diameters. The material properties of Scenario M compare somewhat favorably to the material properties used by Cheung and Baker. However, they used a channel flow velocity of 100 cm/s. Modifying their results to account for velocity differences yields penetration distances from 20 to 66 cm for channel diameters between 0.64 and 1.27 cm and a debris temperature of 2770 °C. This result compares well to the results determined for Scenario M—a penetration depth of 30 cm for a 1 cm channel.

Fieg, et. al., (Reference 19ED-12) performed channel plugging experiments at the Karlsruhe THEFIS facility in Germany using alumina and alumina-iron melts as fuel simulants. The results indicated that the conduction-limited crust growth is an adequate hypothesis for modeling the penetration and freezing of molten fuel. The basis of the this crust-growth model is that a stable crust forms at the channel boundary and then grows continually until it clogs the channel. This is also the basis of the model developed in Subsection 19ED.4. The experimental results presented by Fieg cannot readily be compared to the corium shield because the experimental velocities are so much higher (2.2 m/s to 4.4 m/s compared to 0.01 m/s to 0.1 m/s). However, Fieg's findings that increasing wall temperatures and/or driving pressures increases penetration depth are consistent with the model developed in Subsection 19ED.4.

Soussan, et. al., (Reference 19ED-11) compared the results of experiments performed at AAE Winfrith and CEN Grenoble with the BUCOGEL code developed at Cadarache. The comparison revealed that penetration depths are over predicted using the conduction freezing model and under predicted using the bulk freezing model. The freezing of UO_2 was shown to be consistent with the conduction freezing mode. Alternatively, freezing of molybdenum was determined to undergo bulk freezing. This would tend to indicate that the analysis in Subsection 19ED.4 over predicts the freezing of metallic melts such as Scenario II. However, the overall impact to the analysis is negligible because Scenario II is not limiting.

PLUGM (Reference 19ED-13) was developed to analyze freezing in a variety of geometries including the gaps between the ceramic bricks of a core retention device. The model in 19ED.4 is similar to the PLUGM model for “Thin Slit Geometry - No Crust with a Nonmelting, Infinitely-Thick Wall”. The primary difference is that 19ED.4 is somewhat more simplified to allow a closed-form analytical solution, whereas PLUGM must be solved numerically. Unfortunately, the example contained in Reference 19ED-13 for a thin slit geometry does not lend itself to comparison with the corium shield analysis because the example models a vertical channel with a high entrance velocity.

The past investigations into the freezing of molten fuel in narrow channels tend to support the modeling and results of the channel length analysis contained in Subsection 19ED.4.

19ED.7 References

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19ED-2 Glen E. Myers, “Analytical Methods in Conduction Heat Transfer”, Genium Publishing Corp., Schenectady, NY, 1987, p. 202.

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19ED-7 T.G. Theofanous, W.H. Amarasooriya, H. Yan, and U. Ratnam, “The Probability of Liner Failure in a Mark-I Containment”, NUREG/CR-5423, August 1991.

19ED-8 “MAAP-3.0B Computer Code Manual”, EPRI NP-7071-CCML, November 1990.

19ED-9 “CRC Handbook of Chemistry and Physics”, 62nd Ed. CRC Press, Boca Raton, Florida, 1981.

19ED-10 F.B. Chueng and L. Baker, “Transient Freezing of Liquids in Tube Flow, Nuclear Science and Engineering”, 60, pp. 1-9, 1976.

19ED-11 P. Soussan, M. Schwartz, D. Maxon, and B. Berthet, “Propagation and Freezing of Molten Material Interpretation of Experimental Results”, Proceedings of the 1990 International Fast Reactor Safety Meeting, Snowbird, Utah, August 12-16, 1990.

19ED-12 G. Fieg, M. Möschke, I. Schub and H. Werle, “Penetration and Freezing Phenomena of Ceramic Melts Into Pin-Bundles”, Proceeding of the 1990 International Fast Reactor Safety Meeting, Snowbird, Utah, August 12-16, 1990.

19ED-13 M. Pilch and P.K. Mast, “PLUGM: A Coupled Thermal-hydraulic Computer Model for Freezing Melt Flow in a Channel”, NUREG/CR-3190, SAND82-1580, Sandia National Laboratories, Albuquerque, NM 1982.

Table 19ED-1 Material Properties

Property	Alumina (Reference 19ED-6)	Concrete
Melting Temperature (K)	2280	1450
Density (kg/m^3)	2700	2300
Thermal Conductivity (W/m•K)	4-8	1.3
Specific Heat (J/kg•K)	1000	800
Thermal Diffusivity (m^2/s)	1.48×10^{-6}	7.5×10^{-7}

Table 19ED-2 Scenario Parameters

	Scenario I*	Scenario II	Scenario M†
Flow Rate (m^3/min)	4	0.7	42
Debris Temperature (K)	2850	1800	2500
Composition (w/o):			
UO_2	70	4	61
ZrO_2	10	0	3
Zr	20	47	24
Fe	0	35	~
Cr	0	8	~
Ni	0	6	~
Carbon steel‡	~	~	12
Material Properties:			
Density (kg/m^3)	8900	7300	8500
Specific heat (J/kgK)	960	705	800
Thermal conductivity (W/mK)	6	26	12
Heat of fusion (MJ/kg)	0.31	0.26	0.28

* Scenarios I and II correspond to the Scenarios I and II defined in NUREG/CR-5423 (Reference 19ED-7).

† Scenario M corresponds to the MAAP-ABWR case run to bound debris ejection rate, see Subsection 19EB.6.2.2.

‡ MAAP uses carbon steel instead of its constituents Fe, Cr and Ni.

Table 19ED-3 Constituent Material Properties*

	Density (kg/m^3)	Specific Heat (J/kgK)	Thermal Conductivity (W/mK)	Heat of Fusion (MJ/kg)
UO_2	10100	1000	3.3	0.27
ZrO_2	5600	991	3	0.71
Zr	6500	780	18	0.25
Fe	7800	570	35	0.27
Cr	7200	781	35	0.26
Ni	8900	609	35	0.30
carbon steel†	8000	795	35	0.25

* Material properties from NUREG/CR-5423 (Reference 19ED-7), MAAP User's Manual (Reference 19ED-8) and the CRC Handbook (Reference 19ED-9)

† MAAP uses carbon steel instead of its constituents Fe, Cr and Ni.

Table 19ED-4 Results of Channel Length Calculation

	Scenario I	Scenario II	Scenario M
Interface Temperature (K)	1880	1580	1900
Freeze Time (s)	5.7	4.2	4.2
Channel Velocity (m/s)	0.03	0.01	0.07
Required Channel Length (m)	0.18	0.05	0.35

Table 19ED-5 Effect of Parameter Variations

	Channel Length (m)			
	Scn I	Scn II	Scn M	Average Effect
Base Case	0.18	0.05	0.35	
Debris:				
Density + 20%	0.26	0.07	0.50	+33%
Density – 20%	0.11	0.03	0.24	–63%
Latent heat of fusion + 20%	0.26	0.07	0.55	+35%
Latent heat of fusion – 20%	0.11	0.03	0.21	–69%
Flow rate + 20%	0.19	0.05	0.39	+9%
Flow rate – 20%	0.16	0.04	0.32	–12%
Thermal conductivity + 20%	0.15	0.04	0.32	–12%
Thermal conductivity – 20%	0.21	0.05	0.41	+13%
Temperature + 200K	0.15	0.03	0.29	–27%
Temperature – 200K	0.21	0.07	0.45	+23%
Wall:				
Thermal conductivity + 20%	0.14	0.03	0.26	–36%
Thermal conductivity – 20%	0.25	0.08	0.53	+34%
Thermal diffusivity + 20%	0.20	0.06	0.42	+15%
Thermal diffusivity – 20%	0.15	0.04	0.30	–21%

Table 19ED-6 Change in Energy due to Superheat

	Change in Energy (%)		
Superheat (°C)	Scenario I	Scenario II	Scenario M
0	0	0	0
5	2	1	1
10	3	3	3
25	8	7	7
50	15	14	14
100	31	27	29
125	39	34	36
150	~	41	~

Table 19ED-7 Plug Formation Times with Superheat

	Plug Formation Time (sec)		
Superheat (°C)	Scenario I	Scenario II	Scenario M
0	5.7	4.2	4.2
5	5.9	4.3	4.3
10	6.1	4.4	4.4
25	6.7	4.8	4.8
50	7.7	5.4	5.4
100	9.9	6.7	6.9
125	11.1	7.5	7.7
150	~	8.3	~

Table 19ED-8 Required Channel Lengths with Superheat

	Required Channel Length (m)		
Superheat (°C)	Scenario I	Scenario II	Scenario M
0	0.18	0.05	0.35
5	0.19	0.05	0.37
10	0.19	0.05	0.39
25	0.22	0.06	0.44
50	0.27	0.07	0.53
100	0.40	0.09	0.75
125	0.47	0.11	0.89
150	~	0.13	~

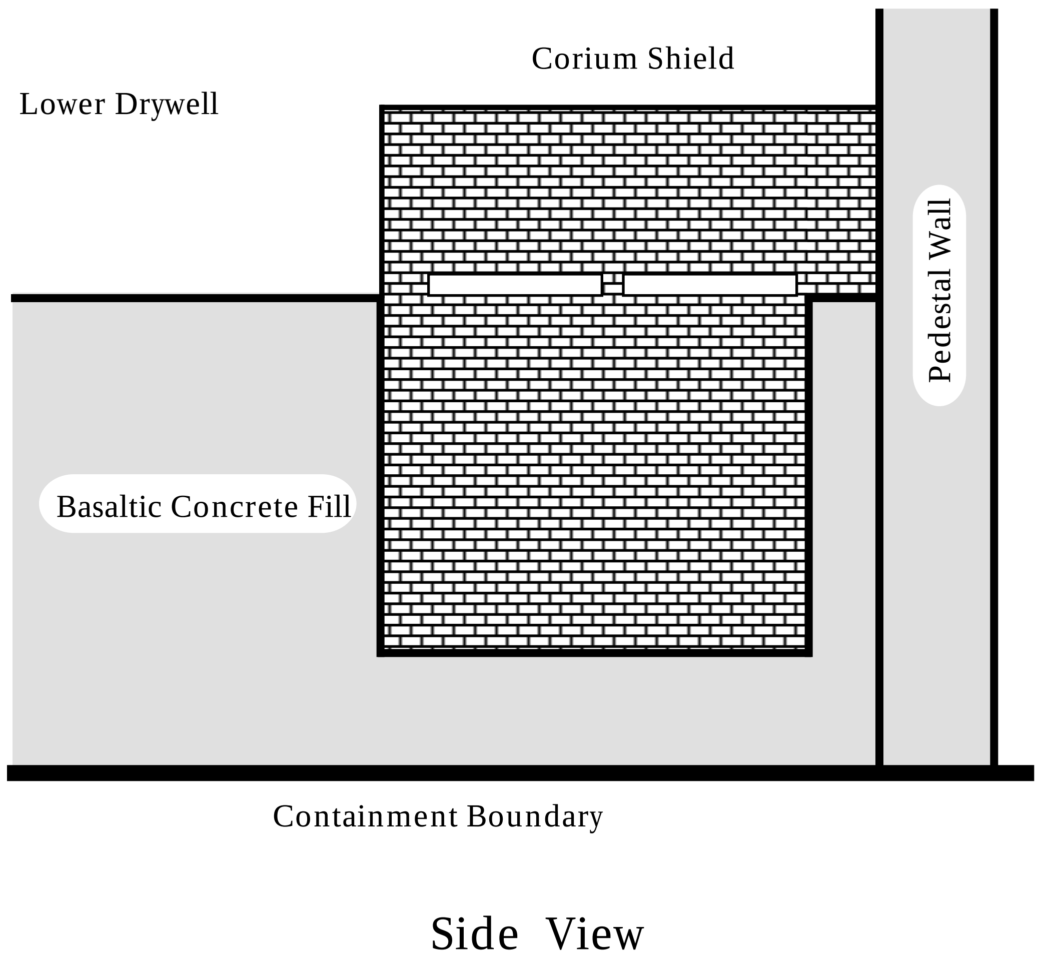

Figure 19ED-1 Conceptual Design of Lower Drywell Floor Drain Sump Shield

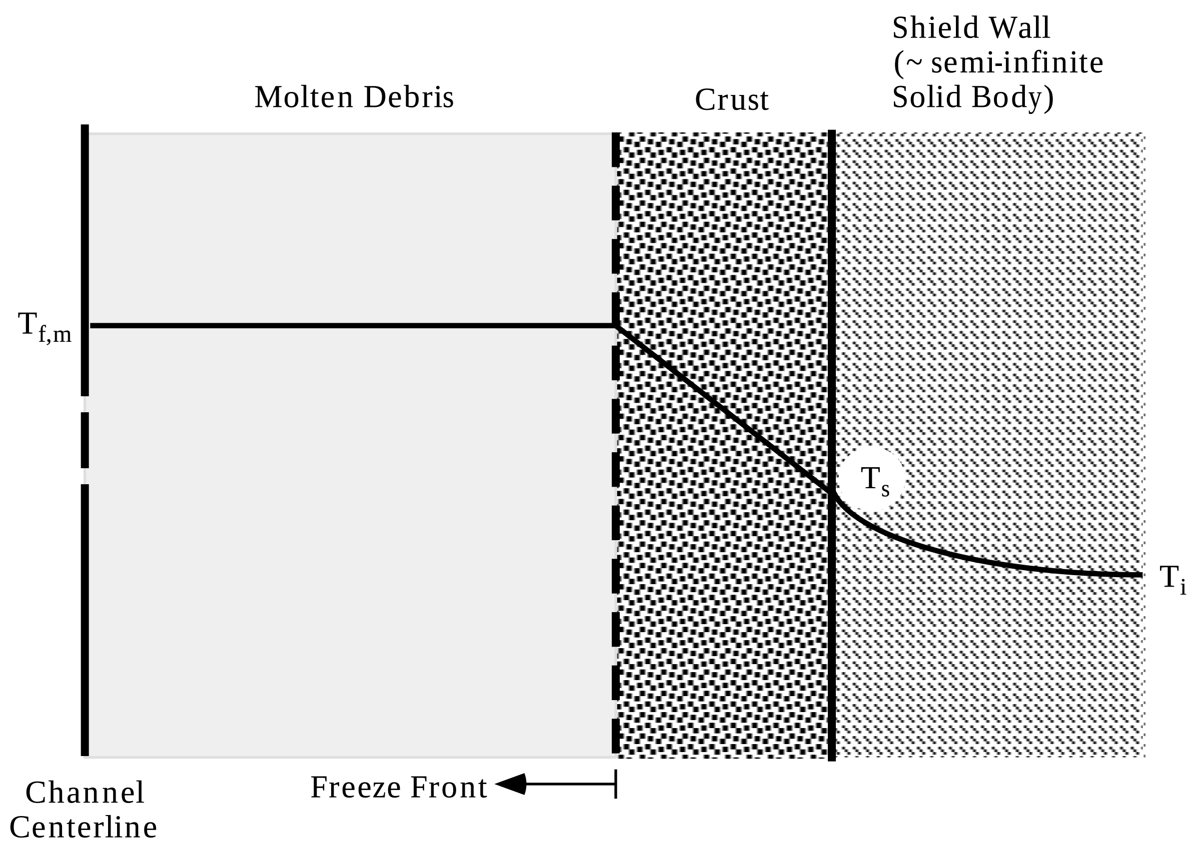

Figure 19ED-2 Temperature Profile in Channel Region

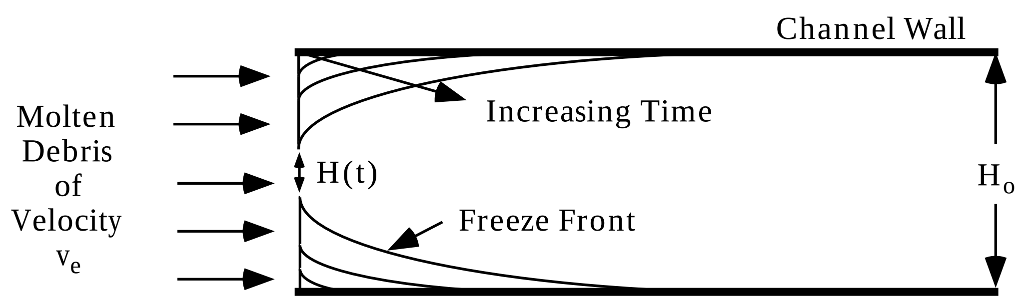

Figure 19ED-3 Channel Flow Height Reduction During Freeze Process

19EE Suppression Pool Bypass

19EE.1 Suppression Pool Bypass

As shown in Subsection 19E.2.3.3.3(4), the only mode of suppression pool bypass that presents any significant risk during a severe accident is vacuum breaker leakage. Vacuum breaker leakage is the passage of gas from the drywell into the wetwell air space. Vapor suppression and fission product scrubbing by the suppression pool are not available to the gas and vapor which pass through the vacuum breakers.

The ABWR contains eight vacuum breakers. ABWR vacuum breakers are swing check valves which begin to open passively when wetwell pressure exceeds drywell pressure by 0.0014 MPa and are fully open at 0.0035 MPa. When the pressure differential is less than this, or drywell pressure exceeds wetwell pressure, the vacuum breakers should be completely seated and no flow should be passing through them. A large drywell to wetwell pressure differential will produce a large force tending to close the vacuum breaker valves. A pressure differential of +0.048 MPa is typical in a severe accident after core damage occurs and the passive flooder opens. This pressure differential produces a closing force of 9810 N (2200 lbf) on the valves. For severe accident scenarios in which the firewater system is actuated, the pressure differential is about +0.096 MPa which produces a closing force of 19600 N (4400 lbf) on the valves. These large closing forces, as well as routine inspection, maintenance, and testing, ensure the probability of vacuum breaker leakage after the actuation of the passive flooder or the drywell spray system is extremely low.

Large amounts of leakage can occur as a result of catastrophic failure of valve components or a valve sticking open. Lesser amounts of leakage can result from normal wear and tear including degradation of the valve seating surfaces or retaining magnets. For sufficiently large amounts of leakage during a severe accident, the time to rupture disk opening or containment failure can be reduced and the amount of fission products released can be increased.

A study utilizing decomposition event trees and deterministic modeling was performed to assess the impact of vacuum breaker leakage on the performance of the ABWR during a severe accident. The event tree analysis is contained in Subsection 19EE.2. Subsection 19EE.3 contains the deterministic evaluation.

19EE.2 Description of Decomposition Event Tree Analysis

The suppression pool bypass decomposition event tree analysis consists of one decomposition event tree (DET), Figure 19EE-1. The DET considers the major phenomena which influence accident consequences. The first two events on the DET sort out vacuum breaker leakage area. Plugging of vacuum breaker leakage pathways by aerosols is considered in the third event. If leakage exists but the pathway is not very large, aerosol plugging can significantly diminish the consequences of suppression pool bypass through the vacuum breakers. The last event assesses the amount of suppression pool bypass.

The probabilities for each sequence pathway with similar end states were summed and these results transferred as the branch probabilities of the main containment event tree.

19EE.2.1 Vacuum Breaker Stuck Open (VB)

When a vacuum breaker sticks open or fails catastrophically, a large pathway is established between the drywell and wetwell. The deterministic analysis described in Subsection 19EE.3 demonstrates that pathway areas greater than 41 cm^2 (opening widths greater than 0.9 cm) can significantly affect accident consequences.

The suppression pool bypass scoping analysis presented in Subsection 19E.2.3.3 assumed a failure probability for vacuum breaker full reverse flow based on pre-1970 U.S. BWR operating history of general check valves. This failure rate is highly conservative because:

(1) The ABWR vacuum breaker design is based on current knowledge which is substantially improved over earlier check valve designs.

(2) The ABWR vacuum breaker environment is significantly less severe than general check valves—the working fluid is gas rather than liquid and the ABWR vacuum breakers will not experience chugging loads.

The failure probability used in this analysis was based on BWR operating experience from April 1981 to March 1991 as contained in a database of Licensing Event Reports. The database was queried for abnormal wetwell-to-drywell vacuum breaker operation. Information about the valves connecting the containment and reactor building were not included because some of these valves are not swing, check valves. The database query provided a short narrative of each abnormal operation as well as the total component operating time.

The database query included BWR Mark I, II and III containments. The vacuum breakers in these containments are similar in design to the ABWR vacuum breakers (passive, flapper-type valves attached to horizontal piping). The ABWR vacuum breakers will be slightly different in size than some of those currently in operation, but this does not undermine the applicability of the data. Only flapper-type vacuum breakers were represented in the data. The motor-operated valves (MOVs) used in the vacuum relief systems of Mark III containments were not considered.

The failures were culled to exclude failures other than those that could lead to a vacuum breaker sticking open or catastrophically failing. Failures to open were excluded because mechanical binding was never the root cause. Most failures to open (10 out of

12) were attributed to either the setpoint drift or worn retaining magnets. Neither of these conditions would prevent the vacuum breaker from closing once it had opened, albeit at a differential pressure outside the normal range. The remaining failures were due to:

(1) a loose set screw on the flapper pivot pin, and

(2) excessive clearance between the valve shaft and disk.

Both of these conditions led to opening forces greater than technical specification limits and greater than the forces required to open the other vacuum breakers tested in the same sequence. In the ABWR design, only seven of the eight vacuum breakers are required to accommodate the most rapid drywell depressurization. Therefore, if either of the these two failure conditions existed during an accident, the affected valves would probably not open because the other vacuum breakers would open and relieve high differential wetwell pressure before the force required to open the affected valves was achieved.

Failures to pass leak rate tests during refueling and maintenance outages when the vacuum breaker proximity switch indicated “closed” were also excluded because they represent small leakage paths. These failures were included in the probability for VB_LEAK as described in Subsection 19EE.2.2. A “closed” indication will be given only when the vacuum breaker disk is seated or very nearly so. Failures to close were included, as were cases in which excessive force was required to cycle a vacuum breaker during stroke capability testing.

The database query provided the following results:

Abnormal operation which could lead to failure to close	18 (N_{close})
Cumulative vacuum breaker operating time	2.66E7 hours (T_{close})

The ability of vacuum breakers to open and close in current plants is demonstrated monthly during stroke capability tests (T_{stroke} = 720 hours). Therefore, the probability that one of the eight ABWR vacuum breakers will fail to close on demand and a large leakage path will be established between the wetwell and drywell can be approximated.

This failure probability conservatively over-estimates the probability that one of the ABWR vacuum breakers will fail to close during accident conditions because the closure forces during an accident will be at least an order of magnitude greater than those present during testing and normal operation. Additional closure force will enhance sealing and overcome some, if not all of the closing resistance.

The vacuum breakers in the ABWR will not be stroke tested every month as are those in current operation. This is expected to improve vacuum breaker reliability because the monthly stroking increases wear, increases galling potential, imparts impact loads to the valve components, loads the valves in a non-uniform manner, and decreases the sealing ability of the soft seats. Reliability will also be increased by improvements made possible by the operational experience of vacuum breakers currently in BWRs with Mark I, II and III containments. These improvements will include material selection, valve assembly techniques and maintenance procedures. Corrosion on ABWR vacuum breaker load bearing components will be negligible because of material selection and operating environment (nearly pure nitrogen). Since reliability is improved and corrosion will be negligible, the failure probability determined during monthly testing of current vacuum breakers provides a conservative over-estimation of ABWR vacuum breaker reliability.

19EE.2.2 Vacuum Breaker Leaks (VB_LEAK)

The consequences of small leakage paths between the drywell and wetwell are less severe than those for a vacuum breaker sticking open. The small leakage area cutoff was determined to be 41 cm^2 in the sensitivity study contained in Subsection 19EE.3. The BWR operating history described in Subsection 19EE.2.1 was also used to determine the probability of small leakage.

BWRs with Mark I containments have a single passive, flapper-type valve attached to the end of each vacuum breaker line. Mark II containments have two passive, flapper-type valves in series in each vacuum breaker line. Mark III containments have a single, flapper-type valve in series with a motor operated valve (MOV) in each line. All of the valves are attached to horizontal piping in the wetwell air space. Since the ABWR has a single, flapper-type valve on the end of each line in the wetwell air space, the operating experience of BWRs with Mark I containments provides the best indication of ABWR vacuum breaker leakage. Actual ABWR vacuum breakers will perform better than those in Mark I containments because:

(1) The ABWR vacuum breaker materials—especially those of the seating surfaces—will be improved because they will be based on the many years accumulated vacuum breaker experience of current BWRs.

(2) The ABWR vacuum breakers will not experience chugging loads.

(3) The ABWR vacuum breakers will not be cycled every month.

The ability of vacuum breakers to remain leak tight is demonstrated during wetwell-to-drywell leakage tests performed as part of each refueling and maintenance outage. During these tests, the drywell is pressurized with respect to the wetwell and the pressure decay rate measured. If the pressure differential decreases too rapidly indicating excessive leakage, the root cause is found and corrected. The instances when a vacuum breaker was found to be the leakage pathway are reported in Licensing Event Reports and included in the operating experience database. The pressurization rate used in the leakage tests are generally slower than those experienced during accident conditions. Increased pressurization rates improve the sealing capability of soft seats and reduce leakage.

All failures reported in the selected operating history of wetwell-to-drywell vacuum breakers in Mark I containments, except failures to open and those used to determine vacuum breaker stuck open, were included in the determination of small leakage probability. The database query provided the following results:

Number of Mark I wetwell-to-drywell vacuum breaker abnormal operations which could lead to small leakage	42 (N_{leak})
Cumulative Mark I vacuum breaker operating time	2.37E7 hours (T_{leak})

The actual amount of leakage was not reported in the database and is generally not available. However, the vacuum breaker leakage area can be roughly characterized. Currently, wetwell-to-drywell vacuum breakers are verified to be closed by indication lights in the control room every seven days. Position is determined by proximity switches which are generally accurate to within the 0.9 cm disk opening, corresponding to the 41 cm^2 cutoff area. The proximity switches used in conjunction with the ABWR vacuum breakers will have even closer tolerances because of the increased importance placed on bypass leakage. None of the leakage failures included failure of “closed” indication. Therefore, leakage was occurring when the valve was open less than the cutoff amount.

During the operating period selected in the database query, refueling and maintenance outages were conducted every twelve to eighteen months. Thus, taking the test time to be eighteen months (T_{test} = 13,140 hours) is conservative. The probability that one of the eight ABWR vacuum breakers develops a small leakage path can be approximated.

This probability is a conservative over-estimation since wetwell-to-drywell leakage test are conducted at differential pressures much lower than those expected during accident conditions. The additional differential pressures will greatly enhance sealing.

19EE.2.3 Aerosols Plug Leakage Path (LEAK_PLUG)

The consequences of leakage pathways between the drywell and wetwell can be greatly diminished if aerosols plug the path. The Vaughan aerosol plugging model (Reference 19EE-1) was used with MAAP-ABWR to determine if and at what time plugging occurred. A full description of this methodology can be found in Subsection 19EE.3.1.

The sensitivity study contained in Subsection 19EE.3.2 predicts that if plugging is allowed to occur in small leakage paths (opening widths ≤ 0.9 cm), accident consequences are not effected by the presence of leakage paths. Even though plugging may reduce the consequences of larger opening widths, no credit was taken in the DET. The sensitivity study predicted plugging for opening widths up to 1.25 cm. Therefore, a high probability was given to plugging of opening widths up to 0.9 cm.

19EE.2.4 Suppression Pool Bypass (POOL_BP)

This heading on the DET summarizes the amount of suppression pool bypass. “No Pool Bypass” indicates that either no leakage, an insignificant amount of leakage, or a plugged leakage pathway exists. The consequences of a particular accident scenario will be unaffected by pool bypass for this condition. “Small Leak” indicates that a small amount of pool bypass is present. Small amounts of bypass will have marginal impact on accident consequences. Large amounts of pool bypass are indicated by “Large Leakage”. Accident consequences will increase in severity when large amounts of pool bypass exist.

19EE.3 Deterministic Analysis

A sensitivity study was performed with MAAP-ABWR to access the impact of suppression pool bypass during severe accident conditions.

19EE.3.1 Method

The dominant severe accident sequence [Loss of all core Cooling with vessel failure occurring at Low Pressure (LCLP)] was chosen to evaluate plant performance. MAAP-ABWR runs were made with effective vacuum breaker area, $A/\sqrt{K}$, varying from 0 to 2030 cm^2 (315 in^2). The upper bound corresponds to one fully open vacuum breaker with no flow resistance. Five variations were analyzed. In each case the overpressure relief rupture disk opened when the wetwell pressure reached 0.72 MPa. The five scenarios were:

(1) Bypass leakage begins after passive flooder activation, aerosol plugging is neglected.

(2) Bypass leakage is present from the beginning of the accident, aerosol plugging is neglected.

(3) Bypass leakage begins after passive flooder activation, aerosol plugging of the vacuum breaker opening is considered.

(4) Bypass leakage is present from the beginning of the accident, aerosol plugging of the vacuum breaker opening is considered.

(5) Bypass leakage is present from the beginning of the accident and the operator initiates the firewater spray system.

MAAP-ABWR uses the MAAP 3.0B aerosol plugging model developed by E.U. Vaughan (Reference 19EE-1). The model predicts the mass of aerosol required to flow through the leak path in order to form a plug as a function of the size of the opening. MAAP conservatively assumes that the flow rate through the vacuum breaker opening is not affected by the growing aerosol plug until the aerosol mass required to plug the leak completely has passed through the opening. For a circular opening, the mass is proportional to the cube of the diameter; and, for a rectangular opening, the mass is proportional to the product of the length and the square of its width. The proportionality constant has been experimentally determined to range from 10,000 to 50,000 kg/m^3 (130 to 640 lbm/ft^3), and varies with aerosol size, aerosol mass flow rate, and leak path geometry. The MAAP-ABWR runs for scenarios 3 and 4 used a conservative proportionality constant of 50,000 kg/m^3 (640 lbm/ft^3).

Although the Vaughan aerosol plugging model does not suggest an upper bound on the size of leak paths which can be plugged, there is some question about the applicability of the model for leak paths greater than 1 cm (0.39 in) in diameter. In NRC/IDCOR Technical Issue 13A (Reference 19EE-2), the NRC asserted that the data cited by Morewitz (Reference 19EE-3) in support of the Vaughan plugging model for pathways greater than 1 cm diameter does not adequately simulate severe accident conditions. The experiments cited with pathways greater than 1 cm (0.39 in) in diameter involved straight ducts with lengths greater than 10 meters (32.8 ft). Therefore, due to the lack of appropriate experimental data, the NRC has accepted the Vaughan aerosol plugging model only for leak pathways smaller than 1 cm (0.39 in). The NRC's position on this issue is stated in the resolution of NRC/IDCOR Technical Issue 13A (Reference 19EE-2).

The applicability of the Vaughn Plugging Model to the conditions of the vacuum breakers was examined by consideration of various test data provided by Morewitz (Reference 19EE-3). The data surveyed includes a variety of experiments involving orifices as well as pipes. The data for orifice plugging indicates that the plugging coefficient is comparable to that for small piping.

Morewitz also discusses the impact of steam on the plugging of leakpaths. He indicates test data which indicated that "leak paths quickly plugged when steam was introduced in the containment atmosphere". He also notes that densification effects such as condensation of water on hygroscopic deposits could increase the rate of plug formation. In the ABWR, hygroscopic CsOH particles form a significant fraction of the aerosol. A large portion of the aerosol mass is expected to be made up of tin (Sn) which is released during the core degradation phase of the accident. Tin is insoluble in water (Reference 19EE-4) and therefore, any plug created with tin would not be expected to be affected by the presence of steam. If continued core-concrete interaction is predicted, a major contributor to the aerosol mass would be SiO2 which is also insoluble in water (Reference 19EE-4).

Most of the experimental evidence cited by Morowitz involves systems with very high pressure differences across the plug. For example, in the orifice test data noted above, the differential pressures ranged from 0.21 MPa to 6.9 MPa. Morowitz indicates that "either solid or porous plugs formed" in these experiments. Reference 19EE-3 also describes a test of a small, concrete, tilt-up-panel building at Atomics International in the early 1960's. The building was overpressurized and cracked so that it leaked badly. In order to plug the leaks, a sodium fire was lit inside the building and observers were stationed around the outside. No smoke was seen issuing from the building. Upon pressure testing of the building, no gas leaks could be detected. Reference 19EE-3 describes several other situations with lower pressure differences in which termination (or significant reduction) in gas flow rates was observed. In the ABWR the maximum pressure difference across the plug will be limited to the head of water above the first row of horizontal vents [about 0.02 MPa assuming normal water level]. Therefore a complete blockage is expected. Any small gas leakage would have an insignificant affect on the wetwell pressurization.

In order to accurately simulate aerosol flow through open vacuum breaker valves in the ABWR, experiments should be conducted with ducts of less than 2 cm (0.79 in) in length. However, the trends of the experimental data do not suggest that the Vaughan plugging model is invalid for openings only slightly larger than 1 cm. Unfortunately, no definitive conclusions can be reached regarding the applicability limit without additional experimental data. For this reason, studies were performed with and without plugging for vacuum breaker bypass widths up to 1.6 cm (0.63 in) corresponding to an effective area of 75 cm^2 (11.6 in^2). This information is used to indicated the conservatisms which may exist in the analysis.

The opening of a stuck-open vacuum breaker is neither circular nor rectangular. Rather it is a crescent shape formed by two circular disks separating while remaining hinged at one point. The leak path width used for the Vaughan plugging model is conservatively assumed to be the maximum crack width. The length of opening is approximated as the effective area divided by the width. For vacuum breaker opening widths of up to 1 cm (0.39 in), corresponding to bypass effective areas of up to 46 cm^2 (7.1 in^2), use of the plugging model provides the best estimate of containment response. As discussed above, additional calculations were run for widths up to 1.6 cm (0.63 in).

19EE.3.2 Results

A series of bypass flow areas was analyzed using MAAP-ABWR for each of the assumed scenarios. A summary of the time and magnitude of fission product releases for each scenario is presented in Table 19EE-1. It was not necessary to run all of the variations in bypass area for each of the five scenarios for this analysis. Thus, Table 19EE-1 contains some blanks. The characteristics of each scenario is discussed below.

19EE.3.2.1 Late Suppression Pool Bypass with no Plugging

For the scenario 1 accident sequence, the passive flooder opens [based on the gas temperature in the lower drywell reaching 533 K (500°F)] at 5.5 hours. The pressure in the drywell decreases as cold water floods into the suppression pool from the lower drywell. Fifteen minutes later, the drywell starts to repressurize and the suppression pool bypass is presumed to begin. If there is no bypass leakage, the elapsed time before rupture disk opening and fission product release is about 20 hours. MAAP predicts that the time to rupture disk opening is not affected for effective vacuum breaker bypass areas of up to 5 cm^2 (0.78 in^2). As the effective area increases from 5 to 50 cm^2 (0.775 to 7.75 in^2), the time to rupture disk opening steadily decreases to about 10 hours. Above 50 cm^2 (7.75 in^2), the time asymptotically approaches 9 hours and remains at 9 hours even for a fully open vacuum breaker valve.

As expected, fission product releases are much higher for cases with bypass leakage than for the case without bypass leakage. For non-bypass cases, the release fraction of CsI at 72 hours is less than 1E-7. The release fractions of CsI at 24 and 72 hours approach asymptotes as the effective bypass area increases. For cases with effective areas greater than 400 cm^2, the 24 hour CsI release fractions are about 6% and the 72-hour release fractions are about 17%. Most of the releases occur late in the sequences as fission products revaporize from the vessel surfaces.

19EE.3.2.2 Pre-existing Suppression Pool Bypass with no Plugging

Bypass leakage was assumed to be present from the beginning of the accident sequence for the cases in scenario 2. As with the scenario 1 cases, the elapsed time before rupture disk opening is not affected by effective bypass areas smaller than 5 cm^2 (0.775 in^2). Unlike the scenario 1 cases, however, the elapsed time did not reach a 9-hour asymptote. Instead, the elapsed time continued to decrease to a value of 2.2 hours for a fully open vacuum breaker valve.

The 24- and 72-hour CsI release fractions asymptotically approached a maximum value for large effective areas. The CsI release fractions for the scenario 2 cases are very similar to those for the cases of scenario 1. The variations in release are caused by changes in revaporization behavior due the slight differences in thermal hydraulic performance.

19EE.3.2.3 Late Suppression Pool Bypass with Plugging

For the scenario 3 cases, bypass leakage was assumed to begin after the actuation of the passive flooder. Plugging of the vacuum breaker opening before the wetwell pressure reached the rupture disk setpoint was predicted for all widths below 1.25 cm. After the leak plugs, all flow from the drywell is directed through the drywell connecting vents into the suppression pool. There is then a period in which little steam is generated in the wetwell vapor space. The wetwell gas temperature decreases during this time due to condensation on the walls. This in turn causes the containment pressure to decrease for a short time. Steam generation in the drywell eventually causes the suppression pool to heat up and the containment pressure increases again. For cases with vacuum breaker opening widths up to 1 cm (0.39 in), the elapsed time to rupture disk actuation is about 20 hours, the same as for the case with no bypass leakage. MAAP-ABWR predicts CsI releases of less than 1E-7 at 72 hours for all of the opening widths less than 1 cm.

The maximum vacuum breaker opening width for which MAAP predicts that the leak path will plug before the rupture disk opens was determined to be 1.25 cm (0.49 in). Even if the rupture disk opens before an aerosol plug forms, reductions in source term can be observed. After the rupture disk opens, aerosols will continue to flow through the vacuum breaker opening and can eventually form a plug. This essentially terminates fission product release. The CsI release fractions at 72 hours for cases with late bypass and credit for aerosol plugging are significantly less than for the cases in which no plugging is assumed.

19EE.3.2.4 Pre-existing Suppression Pool Bypass with Plugging

The scenario 4 cases, in which suppression pool bypass flow was present from the beginning of the accident, show similar results to those of the scenario 3 cases. For cases with vacuum breaker opening widths up to 0.9 cm (0.35 in), the bypass leak plugged before the rupture disk opened and the elapsed time to fission product release was the same as the case with no bypass (about 20 hours). Also, the fission product release for these cases at 72 hours was less than 1E-7, as in the case with no bypass.

The case with an effective bypass area of 46 cm^2 (7.1 in^2), opening width of 1 cm (0.39 in), exhibited a different response. The mass of aerosol passing through the opening was not sufficient to plug the leak before the wetwell pressure reached 0.72 MPa and the rupture disk opened. However, the leak did plug about 30 minutes after the rupture disk opened which reduced the amount of fission products that was released to the environment. MAAP predicts a CsI release fraction of 0.04% at 72 hours for this case, which is about two orders of magnitude less than the corresponding case in which no plugging is assumed. The same behavior was observed for the slightly larger 50 cm^2 case.

19EE.3.2.5 Suppression Pool Bypass with Drywell Spray

The last scenario examined the effects of the drywell spray on cases with bypass leakage present from the beginning of the accident. The firewater addition system was used for these cases since its flow rate is smaller than the drywell spray function of the RHR system. Assuming the operator initiates the firewater spray within 2 hours of the start of the accident, the elapsed time to rupture disk opening can be delayed to nearly 30 hours. This time is comparable to the base case, LCLP-FS-R-N, with no bypass leakage (Subsection 19E.2.2.1).

The fission product releases for all bypass areas analyzed are on the same order of magnitude as the releases for the cases of scenarios 1 and 2 (with no plugging or firewater addition), but the elapsed time to release is much longer. The long times to release allow for a great deal of fission product decay which leads to a substantial reduction in risk as compared to cases in which the drywell spray is not actuated.

19EE.3.3 Conclusions of Deterministic Analysis

Suppression pool bypass can lead to a significant increase in fission product release. Releases can be on the order of 10% for a fully stuck-open vacuum breaker. For sequences in which the firewater addition system is used in spray mode, the time to release is not significantly affected by bypass. However, for sequences without sprays, the time from the beginning of the accident until the onset of the release can be significantly reduced. The use of the Morewitz blockage model results in a significant improvement in the calculated risk associated with suppression pool bypass. Nonetheless, there is a substantial increase in consequences associated with large bypass areas.

19EE.4 Summary of Results

19EE.4.1 Quantification of DET

The event tree is shown in Figure 19EE-1. The probabilities for different leakage areas are transferred to containment event trees.

19EE.4.2 Impact of Release Fractions

MAAP-ABWR predicts the release fraction of CsI for the LCLP case without bypass leakage is less than 1E-7. The effect of leakage on the CsI release fraction (f) is shown below.

Amount of Leakage	Release Fraction of CsI
None	f < 1E-7
Small	1% < f < 10%
Large	f > 10%

19EE.4.3 Impact on Time to Rupture Disk Opening

The sensitivity study contained in Subsection 19EE.3 focused on the Loss of all core Cooling with vessel failure occurring at Low Pressure (LCLP) accident sequence. This is the dominant sequence and its response to suppression pool bypass should be typical of the other accident sequences.

Without suppression pool bypass, rupture disk opening is predicted to occur at approximately 20 hours into the accident for cases with passive flooder operation. The effect of leakage on time to rupture disk opening, *t*, is summarized below.

Amount of Leakage	Time to Rupture Disk Opening
None	~20 hours
Small	$6 < t < 16$ hours
Large	$t < 6$ hours

19EE.4.4 Impact on Aerosol Plugging on Integrated Offsite Risk

The quantification of the environmental source term presented in the decomposition event trees considers a substantial reduction in the frequency of bypass release due to plugging of the release path as described in Subsection 19EE.3.1. There is some uncertainty associated with the gas permeability of the plug under these conditions which may affect the source term timing and magnitude. In order to assess the effect of this uncertainty, a sensitivity study has been performed.

The impact of aerosol plugging on offsite risk can be assessed by eliminating the credit taken for plugging in the suppression pool bypass DET (Figure 19EE-1) and recalculating the source term category frequencies in Figure 19D.5-3. The change in source term category frequencies is propagated into the consequence analysis as described in Subsection 19E.3. The results of eliminating the credit taken for aerosol plugging in the suppression pool bypass DET are included.

The results of recalculating the frequencies for the affected STC#s are shown in Table 19EE-2. Events with extremely low frequencies do not significantly contribute to the offsite risk. Therefore, only STC#s 6, 8, 18, 19 and 30 are risk significant.

The significant STC#s are binned in Table 19D.5-7 into Case 1, Case 7 and Case 8 for the consequence analysis performed in Subsection 19E.3. The total frequencies of these two cases must be re-evaluated using the frequency changes in Table 19EE-2. The base case frequencies appear in Table 19E.3-6.

The impact of neglecting credit for aerosol plugging on offsite risk is presented in Figure 19EE-1. While aerosol plugging can significantly reduce the amount of fission products released from the containment given by bypass event, the phenomenological assumption has negligible impact on offsite risk.

19EE.5 Conclusions

Suppression pool bypass (the passage of gas and vapor from the drywell directly into the wetwell air space) can lead to increased fission product releases. As shown in Subsection 19E.2.3.3.3(4), the only mode of suppression pool bypass that has the possibility of significantly increasing risk is vacuum breaker leakage. This attachment determined the probabilities and consequences for vacuum breaker leakage areas from zero to that corresponding to one vacuum breaker stuck fully open.

Fission product release fractions were determined with MAAP-ABWR using the dominant accident sequence [Loss of all core Cooling with vessel failure occurring a Low Pressure (LCLP)] modified to include a path between the drywell and the wetwell air space. Plugging of leakage paths by fission products was considered for small pathways. Leakage probabilities were determined by reviewing recent operating experience of wetwell to drywell vacuum breakers in BWRs with Mark I, II and III containments.

Suppression pool bypass does not significantly add to the risk associated with the ABWR because the bypass areas resulting in increased releases are offset by low probabilities of occurrence. No leakage and, correspondingly, no impact on plant risk is expected to occur for almost all of the accident demands. Small amounts of leakage have a small probability, and can result in medium volatile fission product releases (one to 10% of initial inventory). Volatile fission product releases on the order of 10–20% of initial inventory can result when large amounts of suppression pool bypass are present. However, the impact on plant risk is still negligible because the probability of large leakage is very small.

19EE.6 References

19EE-1 Vaughan, E.U., “Simple Model for Plugging of Ducts by Aerosol Deposits”, Trans. Am. Nuclear. Soc., 28, 507, 1978.

19EE-2 “Technical Support for Issue Resolution”, IDCOR Technical Report 85.2, July 1985.

19EE-3 Morewitz, H.A, “Leakage of Aerosols from Containment Buildings”, Health Physics, Vol. 42, No. 2, 1982, pp. 195-207.

19EE-4 “Handbook of Chemistry and Physics”, 53rd Edition, CRC Press, 1972-1973.

Table 19EE-1 Summary of Volatile Fission Product Releases for Severe Accidents with Suppression Pool Bypass Leakage through Vacuum Breaker Valves

Eff. Area (cm^2)	0	5	20	41	46	50	58	75	100	400	2030
Leak Width (cm)	0	0.11	0.44	0.90	1.00	1.09	1.25	1.63	2.17	8.70	*
Scenario	**Time to Fission Product Release (h)**										
1	19.9	19.8	15.4	*	*	9.9	*	9.1	9.1	9.0	9.0
2	19.9	20.0	13.1	*	*	5.5	*	4.0	3.5	2.7	2.2
3	19.9	20.2	20.2	*	*	20.3	20.4	9.2	†	†	†
4	19.9	20.2	20.2	20.4	5.9	5.6	*	*	†	†	†
5	31.1	*	*	*	*	29.7	*	*	*	*	28.9
Scenario	**CsI Release Fraction at 72 Hours**										
1	< 1E-7	0.38%	1.6%	*	*	3.6%	*	6.3%	8.5%	18%	17%
2	< 1E-7	0.55%	1.7%	*	*	4.2%	*	6.5%	8.5%	16%	18%
3	< 1E-7	< 1E-7	< 1E-7	*	*	< 1E-7	< 1E-7	0.06%	†	†	†
4	< 1E-7	< 1E-7	< 1E-7	< 1E-7	0.04%	0.06%	*	*	†	†	†
5	< 1E-7	*	*	*	*	4.8%	*	*	*	*	14%

* Not calculated

Table 19EE-2 Effect of Eliminating Aerosol Plugging Credit on Source Term Category Frequencies

STC#*	Frequency (w/ plugging credit)†	Frequency (w/o plugging credit)†
6		
7		
8		
18		
19		
20		
30		
31		
32		
42		
43		
44		

* See source term category grouping diagram, Figure 19D.5-3

† Probabilities not part of DCD (Refer to SSAR).

Figure 19EE-1 Containment Event Evaluation DET for Suppression Pool Bypass

Not Part of DCD (Refer to SSAR)

Figure 19EE-2 Impact of Aerosol Plugging Credit on Offsite Risk Measured by Whole Body Dose at 805 m (0.5 Mile) as Probability of Exceedence

Not Part of DCD (Refer toSSAR)

19F Containment Ultimate Strength

19F.1 Introduction and Summary

This appendix describes analysis and judgement used to estimate the containment internal pressure capability and associated failure mode and location. The ultimate pressure capability of the containment structure is limited by the drywell head whose failure mode is plastic yield of the torispherical dome. The pressure capability is 1.025 MPa at 533 K (500°F) (typical temperature for most severe accident sequences), or 0.921 MPa at 644 K (700°F) [representative for those accidents in which the temperature exceeds 533 K (500°F)]. The containment is conservatively assumed to depressurize rapidly when the pressure capability is reached. No significant leakage through penetrations is anticipated before the capability pressure is reached. However, for the purpose of source term calculations, leakage in terms of leak areas is conservatively estimated for pressures below the capability pressure.

The primary function of the containment structure is to serve as the principal barrier to control potential fission product releases. The design basis event for this function is a postulated loss-of-coolant accident (LOCA). Based on this functional requirement, the containment pressure vessel is designed to withstand the maximum pressure and temperature conditions which would occur during a postulated LOCA. The ABWR containment system employs pressure suppression which allows a design pressure of 0.411 MPa and a design temperature of 444 K (340°F) for the primary containment pressure vessel. In addition the suppression pool retains fission products which could be released in the event of an accident. In this appendix the capability of the containment structural system of the ABWR standard plant to resist potentially higher internal pressures and temperatures associated with severe accidents is evaluated.

The primary containment vessel, also referred to as the RCCV for reinforced concrete containment vessel, is a right, cylindrical structure of the steel-lined, reinforced concrete design. The containment is integrated with the reactor building (RB) walls from the basemat up to the elevation of the containment top slab. The top slab, together with pool girders and building walls, form the spent fuel and equipment storage pools. The elevation view of the reactor building/containment structural system along 0°–180° direction is shown in Figure 19F-1. The containment is divided by the diaphragm floor and the reactor pressure vessel (RPV) pedestal into a drywell chamber and a suppression chamber. The drywell chamber above the diaphragm floor is called the upper drywell (U/D). The drywell chamber enclosed by the RPV support pedestal beneath the RPV is called the lower drywell (L/D). The primary containment and internal structures are shown in Figure 19F-2. The major penetrations in the containment wall include:

(1) The upper drywell equipment and personnel hatches at azimuth 130° and 230°

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

(3) The suppression chamber airspace access hatch

(4) The main steam and feedwater pipe penetrations

Detailed descriptions of the containment design are included in Subsection 3.8.1.

The pressure boundary of the containment structure consists of the reinforced concrete containment vessel (RCCV) and the steel drywell head. The structural integrity of the RCCV is investigated for its global strength under internal pressure beyond the design basis using the FINEL computer program which is based on the nonlinear finite element method of analysis for axisymmetric reinforced concrete structures. The pressure capability of the steel drywell head is evaluated using approximate methods. During various severe accident conditions, the ABWR containment could also be challenged by high temperatures with a typical temperature of 533 K (500°F) for most accident sequences and a representative temperature of 644 K (700°F) for those accidents in which the temperature exceeds 533 K (500°F). At typical accident temperature of 533 K (500°F), the drywell head is found to have a ultimate pressure capability of 1.025 MPa and a service level C pressure capability of 0.77 MPa. For the RCCV the pressure capability at ambient temperature is governed by the top slab and its level C value (corresponding to ASME-III, Div. 2, CC-3420 and CC-3720 limits for factored load category) is 1.23 MPa. This is much higher than the ultimate pressure capability of the drywell head. On the basis of a recent Argone National Laboratory (ANL) study for the Sandia's 1/6-scale containment model, it is expected that with thermal effects included the RCCV pressure capability will not be reduced below the drywell head capacity for the range of temperatures considered. The drywell head is the controlling component for the structural strength of the containment structure.

In order to evaluate liner response to over- pressurization, liner plates are included in the FINEL analysis. The analysis results show that the liner strains are much smaller than the code allowables when the internal pressure is as high as 1.34 MPa. A separate evaluation further demonstrates that at the governing containment failure pressure of 1.025 MPa at 533 K (500°F), the liner and anchor system will maintain its structural integrity and no liner tearing will occur.

The leakage potential through large operable penetrations such as the drywell head, equipment hatches, and personnel airlocks is evaluated. Assuming no sealing action from degraded seals at temperatures above 533 K (500°F), the total leak areas before the capability pressure is reached are conservatively estimated to be:

Pressure	Leak Area	
MPa	Cm^2	In^2
0.100	0	0.00
0.412 (design)	0	0.00
0.460 (SIT)	0	0.00
0.512	7.94	1.23
0.584	17.9	2.77
0.653	27.8	4.31
0.722	37.7	5.85
0.790	47.6	7.39
0.860	57.6	8.93
0.929 (capability)	67.5	10.47

At and below the Structural Integrity Test (SIT) pressure of 0.460 MPa, leakage is within the design limit and the equivalent leak area is negligible.

In conclusion, the ultimate pressure capability is limited by the drywell head. The postulated failure mechanism is the plastic yield of the torispherical dome. The pressure capability is 1.025 MPa at 533 K (500°F), and it reduces to 0.929 MPa when the containment temperature reaches 644 K (700°F). The governing service level C (for steel portions not backed by concrete) / factored load category (for concrete portions including steel liner) pressure capability of the containment structure is 0.770 MPa at 533 K (500°F) which is the internal pressure required to cause the maximum stress intensity in the steel drywell head to reach general membrane yielding according to service level C limits of ASME-III, Division 1, Subsubarticle NE-3220.

The pressure capability evaluation described above is based on the deterministic approach. The uncertainties associated with the failure pressure are assessed in Attachment 19FA.

19F.2 RCCV Nonlinear Analysis

This subsection describes the non-linear analysis performed for the reinforced concrete containment vessel (RCCV) (excluding the drywell head) of the ABWR Standard Plant. Computer code “FINEL” was used for evaluation of the axisymmetrical components of the RCCV.

19F.2.1 Finite Element (FE) Model Description

The containment and the containment internal structures are axisymmetric while the RCCV top slab together with the reinforced concrete girders even though not axisymmetric, are idealized and included in the axisymmetrical model. Solid elements are used to represent the girders at the top of the RCCV, approximating the stiffness of the actual structure.

For simplicity, the reactor pressure vessel (RPV), the reactor building outside of the RCCV and superstructure above the operating floor, are not modeled. To represent the restraining effects of the floors outside the containment, horizontal restraining elements are used with pseudo material properties. The model includes concrete elements, the reinforcing steel, the steel liner plate of the drywell, and the wetwell and the diaphragm floor structures, and the structural steel elements used for the pedestal.

The model consists of 868 nodal points and 1280 elements. There are 448 elements used with unidirectional stiffness representing rebar, whereas 832 elements are isotropic, representing steel, concrete, and soil. The soil below the foundation mat was modeled to a depth of 50.0m and to a radius of 76.0m. See Figure 19F-3 for the model.

The FINEL computer program permits the specification of bi-linear, brittle or ductile material properties. The concrete and soil elements are specified to have brittle properties such that they are strong in compression and weak in tension. The steel plate elements and the rebar elements are specified to have ductile material properties with the same strength in tension and compression. The capability of the FINEL program to accommodate ductile and brittle material behaviors permits both concrete cracking and yielding of steel and rebar. This allows the program to consider redistribution of forces throughout the structure due to the non-linear behavior.

19F.2.2 Analysis

The FE model was run for three different load conditions shown in Table 19F-1.

(1) Structural Integrity Test 1 (SIT-1), with 0.46 MPa pressure in the drywell and wetwell (RCCV).

(2) Structural Integrity Test 2 (SIT-2), with 0.411 MPa pressure in the drywell and 0.18 MPa in the wetwell.

(3) Four times design pressure, with 1.34 MPa pressure in the RCCV.

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

19F.2.3 Results

Table 19F-1 summarizes analytical results for various loading conditions. The results are shown in terms of maximum rebar stresses, concrete stresses, liner strains and structural deformations.

Based on the FINEL analysis, it can be concluded that the axisymmetric components of the RCCV, as designed based on ASME Section III Division 2 code requirements, can withstand an internal pressure of 1.34 MPa i.e, four times the design pressure, with stresses and strains in the rebar, liner plate and concrete within code allowable limits. The strength is governed by wetwell wall. The strength of the non-axisymmetric top slab region is evaluated by extrapolation of the elastic analysis results using a 3D finite element model as discussed in Subsection 19F.3.1.1.

19F.3 Prediction of Containment Ultimate Strength

19F.3.1 Structural Capability

19F.3.1.1 Concrete Shell

The structural integrity of the RCCV axisymmetric components has been demonstrated for an internal pressure of 1.34 MPa from the FINEL analysis. Based on extrapolation of analysis results, estimate of the level C pressure capability and the ultimate pressure capability is made and discussed in this subsection. The level C pressure capability is defined to be the pressure value at which the ASME-III, Div. 2, CC-3420 and CC-3720 limits for factored load category are reached. The ultimate pressure capability is assumed reached when rebars at both faces of a cross section reaching yield stress. The estimated pressure capabilities of the various components of the RCCV are shown in Table 19F-2. It should be noted that the extrapolation of results gives only approximate values beyond the analyzed values. The level C pressure value of 1.23 MPa for the top slab shown in Table 19F-2 is based on extrapolation of elastic 3D STRARDYNE analysis results and it is governed by the strength of the supporting pool girders. It should be recognized that this value could be somewhat different from inelastic analysis results. The ultimate pressure capability of the top slab is not made since its level C value is much higher than the drywell head ultimate pressure discussed in Subsection 19F.3.1.2.

The analysis performed was static analysis. Dynamic effect on structural response becomes significant only when the rate of applied loading is within the range of natural periods of the structure. The fundamental natural period of the integrated RB and RCCV structures is less than one second. Therefore, containment loading resulting from severe accident conditions can be treated statically when the rate of loading buildup is longer than one second. For all accidents sequences considered in this PRA, except hydrogen detonation, the pressure buildup rate within the containment is longer than one second. Thus, the results based on static analysis can be applied. Since the ABWR containment is inerted, hydrogen detonation is of no concern.

During various severe accident conditions, the ABWR containment could be challenged by high temperatures with a typical temperature about 533 K (500°F). The effect of elevated temperature on containment pressure capability has been investigated recently by Argonne National Laboratory (ANL) (Reference 19F-1). The ANL study concluded that for temperatures up to 644 K (700°F), the failure mode and location did not change from the case of internal pressure alone, and the failure pressure was reduced slightly (11% maximum) from that predicted for the internal pressure alone case. On the basis of the ANL study it is expected that with thermal effects included the RCCV pressure capability will be not reduced below the drywell head capability for the range of temperatures considered.

19F.3.1.2 Drywell Head

This subsection presents an evaluation of the structural capability of the drywell head under internal pressure and temperature loading. The leakage potential of the head closure is discussed in Subsection 19F.3.2.2.

The drywell head which covers the 10.29 m (33 ft-9 in.) in diameter opening in the upper drywell top slab is a steel torispherical dome assembly. Figure 19F-4 shows the major dimensions of the design. Under internal pressure loading, the most critical location of this type of configuration is the knuckle (or torus) region of the torispherical dome which may fail by plastic yield or buckling.

For torispherical pressure vessel heads, an approximate formula for the limit pressure at which significant plastic deformation occurs was developed by Shield and Drucker (Reference 19F-2) based on the upper and lower bound theorems of limit analysis, and it is

$$P_c = S_y \left\{ (0.33 + 5.5r/D)\frac{t}{L} + 28\,(1-2.2r/D)\left(\frac{t}{L}\right)^2 - 0.0006 \right\} \qquad (19F\text{-}1)$$

where:

P_c = limit pressure

S_y	=	yield strength of the material
t	=	uniform thickness of the head
r	=	radius of the knuckle shell
D	=	diameter of the cylindrical shell
L	=	radius of the spherical cap

Substituting the dimensions shown in Figure 19F-4 into Equation 19F-1 gives

$$P_c = 0.003965 \cdot S_y$$

The material yield strength depends on temperature. The actual strength of as-built material is generally higher than the specified minimum value used in design. To have a more realistic estimate of the structural strength, the minimum yield strength of material SA-516, Gr. 70 as specified in Appendix I of ASME Section III is increased by 10%. The calculated limit pressure as a function of temperature is shown in Figure 19F-5. As shown in the figure, the limit pressure is 1.025 MPa at 533 K (500°F), and reduces to 0.929 MPa at 644 K (700°F) which is a representative temperature for those accidents in which the temperature exceeds 533 K (500°F). From a linear elastic finite element analysis, it is found that a pressure of 0.770 MPa at 533 K (500°F) is required to cause the maximum stress intensity in the head to reach general membrane yielding according to service level C limits of ASME-III, Division 1, Subsubarticle NE-3220.

Buckling is another potential failure mode of the torispherical head under internal pressure since the knuckle is subjected to compressive stress in the hoop direction. Galletly recently (Reference 19F-3) proposed a design equation for preventing buckling in fabricated torispherical shells under internal pressure.

$$P_d = \frac{80 S_y \left(\frac{r}{D}\right)^{0.825}}{\left(\frac{D}{t}\right)^{1.5} \left(\frac{L}{D}\right)^{1.15}} \qquad (19F\text{-}2)$$

This equation is based on his previous studies (References 19F-4 and 19F-5) and is formulated for design use with knock-down (capacity reduction) factors included. As compared to all known test results (43 in total), the ratios of the actual buckling pressure to the allowable buckling pressure predicted by this equation were found to range from 1.51 to 4.01. Hence, a minimum factor of safety of 1.5 is ensured by this equation.

The test data presented in Reference 19F-3 (excluding the test performed by Blenkin since no buckling was observed at the maximum test pressure) are summarized graphically in Figure 19F-6, showing the relationship between the test and predicted pressures. The predicted pressures, as can be seen, are at least 1.5 times lower than the test results. In order to gain more insight about the data variability, statistical analyses are performed and the results are given in Figure 19F-7. The PDF (probability density function) of the data shown by solid lines is the histogram of 42 data points expressed in terms of the ratio of test to predicted pressure. It is observed that the data can be reasonably approximated by the lognormal distribution. The medium value of the test to predicted pressure ratios in the data set is 2.27 and the logarithmic standard deviation is 0.293. The resulting lognormal density and cumulative functions are shown in Figure 19F-7. The cumulative probability is 8% for the ratio up to 1.5. It means that the probability of the ratio of actual to predicted pressure being less than 1.5 is 8%. In other words, there is 92% confidence that the margin of safety against buckling is at least 1.5 when Equation 19F-2 is used. The 1.5 factor of safety corresponding to 92th percentile is deemed sufficient for the assurance of no buckling failure against severe accident loadings of very low probabilities of occurrence. Equation 19F-2 can be therefore used for the determination of level C buckling pressure of torispherical heads.

Substituting the dimensions shown in Figure 19F-4 and the material properties specified in Appendix I of ASME Section III for SA516, Gr. 70 into Equation 19F-2, the level C buckling pressure for the ABWR drywell head is calculated to be 0.860 MPa at 533 K (500°F). It is higher than 0.770 MPa associated with general membrane yielding per level C stress intensity limits. Therefore, the governing level C pressure is 0.770 MPa.

As mentioned earlier, Equation 19F-2 has a factor of safety of 1.5 as compared to the lower bound of all known test results. From a statistical study of these test results, the medium buckling pressure is estimated to be 2.27 times the value predicted by Equation 19F-2. Subsequently, the critical buckling pressure of the ABWR drywell head are:

Lower bound = 1.5 x 0.860 MPa = 1.246 MPa,

Best estimate = 2.27 x 0.860 MPa = 1.839 MPa.

A comparison with the plastic yield limit pressure P_C calculated above indicates that plastic yield will occur before buckling and is the governing failure mode of the drywell head. The capability pressure is 1.025 MPa at 533 K (500°F), and reduces to 0.929 MPa when the containment temperature reaches 644 K (700°F).

19F.3.2 Leakage Potential

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

19F.3.2.1 Liner Plate

As discussed earlier, the containment liner plates were included in the FINEL model. The maximum liner strains are found to be well within the code allowables when the internal pressure is as high as 1.34 MPa. Therefore, liner tearing is not expected to occur before the capability pressure of the drywell head is reached. This is confirmed by a separate evaluation as follows.

The ABWR containment liner plate system consists of 6.35-mm (1/4-in) thick plate made of ASME SA-516 Grade 70 steel. The liner plate is anchored into the concrete containment through WT 4 X 7.5 made of ASTM A-36 steel and welded to the liner plate. The following are the pertinent data of the ABWR containment liner plate system used for the evaluation.

- Concrete Containment:

  Inside Radius = 14.5 m (571 in)

  Thickness = 2.0 m (78.7 in)

  Hoop Rebar = 4.57 cm^2/cm (1.8 in^2/in) total on both faces

  Vertical Rebar = 7.05 cm^2/cm (1.666 in^2/in) total on both faces

- Steel Liner Plate:

  6.35-mm (1/4-in) thick ASME SA 516 Grade 70

  Yield stress at 294 K (70°F) = 262 MPa

  Yield stress at 533 K (500°F) = 212 MPa

  Ultimate Uniaxial Strain at Fracture = 21%

  [Based on ASTM 516 Grade 70 minimum guaranteed elongation in 51 mm (2")]

- Liner Anchor:

  WT 4 X 7.5 spaced at 500 mm on center

  Yield stress at 294 K (70°F) = 248 MPa (A36 steel)

The following severe accident conditions are used for the evaluation of the ABWR containment liner plate.

- Containment Internal Pressure = 1.025 MPa
- Steam Jet Temperature = 533 K (500°F)
- Ambient Temperature = 311 K (100°F)
- Rise in Temperature = 477 K (400°F)

Due to the internal pressure, the liner plate has a tendency to elongate by virtue of hoop tension whereas it has a tendency to shorten due to compression on the inside face of the containment for elevated internal temperature. Thus, the combination of internal pressure and temperature loads on the liner plate has compensatory effects. The friction and the physical bond between the liner plate and concrete wall are conservatively neglected for the evaluation. The corresponding shrinkage strains, being small, are neglected, and the concrete is assumed to have zero tensile strength.

(1) Evaluation for Internal Pressure Loading

The hoop force due to the internal pressure of 1.025 MPa on an internal radius of 14.5 m (571 in) is computed to be 13690 kg/cm (76.5 kips per inch) height of the containment. Assuming that this hoop force is resisted by the total hoop steel of 11.6 cm^2 (1.8 in^2) and liner plate area of 1.61 cm^2/cm (0.25 in^2 per inch) height of the containment, the hoop stress is computed to be 257 MPa which gives the value of hoop strain of 0.13%.

This compares very closely with the strain values obtained from the “FINEL” analysis for 1.025 MPa internal pressure which gave maximum strain of 0.126%. Assuming a very conservative estimate of strain concentration factor of 33 at the discontinuities on the Sandia Containment Test results based on internal pressure of 1.11 MPa (Reference 19F-6), the maximum liner plate strain due to the internal pressure is estimated to be 0.13 X 33 = 4.3%. This strain is still far lower (by a factor of almost 5) than the ultimate fracture strain of 21% for the liner plate material. The internal pressure results in uniform tension in the liner but does not produce any load on the liner anchors. Thus, it can be inferred that the liner plate will not tear for the severe accident pressure of 1.025 MPa.

(2) Evaluation for Thermal Loading/Jet Impingement

The internal temperature causes compressive forces on the liner resulting in potential buckling of the plate, as illustrated in Figure 19F-8. The thermal strain resulting from a 477 K (400°F) temperature rise is 0.002744 cm/cm

(0.002744 in/in). The resulting hoop membrane force for the 6.35-mm (1/4-in) thick liner plate is 4625 kN/m (26.4 k/in). This is beyond the elastic limit and a plastic solution is sought using the procedure suggested in Bechtel Topic Report BC-TOP-1 (Reference 19F-7). For the buckled configuration shown in Figure 19F-8, the following spring constants are found:

K_c = 890 kN/m/m (200 k/in/in) for the anchor

K_{BPL} = 441 kN/m/m (99 k/in/in) for the bent plate

K_{RPL} = 1522 kN/m/m (342 k/in/in) for plate relaxation

The resulting total force N_T on the system is found to be 6244 kN/m (35.64 k/in) and the corresponding deflection δ is 1.9 mm (0.0748 in). From the energy balance approach, the safety factor expressed in terms of the ratio of the total energy capacity of the anchor to the energy required for the equilibrium is calculated to be 1.9. The forces exerted on the liner plate and anchor are 897 kN/m (5.12 k/in) and 865 kN/m (4.94 k/in) which are within their respective yielding capacity of 1356 kN/m (7.74 k/in) and 929 kN/m (5.3 k/in). Therefore, the anchor system is safe under applied loads.

(3) Conclusions

Based on the above discussions it is concluded that the ABWR containment liner plate and the liner anchorage will maintain its structural integrity even when subjected to severe accident pressure of 1.025 MPa at 533 K (500°F). In this evaluation, a very conservative strain concentration factor of 33, observed at the discontinuities in the Sandia containment tests, has been used. It is demonstrated that there are adequate margins over the maximum conceivable strains to the ultimate fracture strains available to preclude the type of tearing failures observed in the Sandia containment tests. It should also be noted that there are major differences between the liner plate system used in the Sandia tests and the ABWR containment. The significant difference is the use of intermittent stud type liner anchor in the Sandia tests as opposed to the welded WT 4 anchors for the ABWR containment. This should result in a more uniform distribution of the strain for the ABWR containment and in much lower strain concentration factors compared to those observed in the Sandia test. This will further improve the margins of safety over the ones computed.

19F.3.2.2 Penetrations

An ANL study (Reference 19F-8) assigned high priority to the study of large operable penetrations such as the drywell head closure, equipment hatches, and personnel airlocks since they are expected to have high potential for leakage under severe accident conditions. Leakage from fixed penetrations (both electrical and mechanical) appears to be less likely based on the results of experiments conducted to date by Sandia National Labratories (SNL) and its contractors (Reference 19F-8). In fact, according to the same reference, no leakage was detected from any of the three current electrical penetration assemblies (EPAs) during the severe accident testing (steam environments). Depending on the EPA type the highest temperature loading ranged from 456 K (361°F) to 644 K (700°F), and the highest pressure loading ranged from 0.517 MPa to 1.069 MPa. The EPAs used in the ABWR containment will be capable of maintaining leak tightness up to the containment pressure of 1.025 MPag and temperature of 644 K (700°F)(Subsection 8.3.3.7). The leakage estimate in this study therefore concentrates on large operable penetrations.

The leakage potential of operable penetrations depends on both the relative position of the sealing surfaces and the performance of the seal material. The position of the sealing surfaces depends on the initial conditions (metal-to-metal contact is maintained under design conditions for most penetrations) and on the deformations induced by accident pressure and temperature. The seal performance depends mainly on temperature as well as the effect of thermal and radiation aging. The recent SNL tests of seals for mechanical penetrations, Reference 19F-8, indicated that

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

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

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

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

Sandia (Reference 19F-8) has proposed the following equations for “available gasket springback”, S_p, for evaluating the leakage potential as a function of the compression set retention and the degradation temperature:

$$S_p = (1 - C_B) S_q h_i \text{ for } (T < T_d) \tag{19F-3}$$

$$S_p = 0.127 \text{mm } (0.005 \text{ inch}) \text{ for } (T > T_d) \tag{19F-4}$$

where:

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

S_q = the squeeze as illustrated in Figure 19F-9 (a dimensionless measure of the gasket deformation under normal operation conditions),

h_i = the initial seal height, and

T_d = the degradation temperature of the gasket material.

Equation 19F-3 is based on the assumption that significant leakage can be prevented as long as positive compression of the gasket is maintained. Equation 19F-4 is empirical based on test results that even a degraded gasket can effectively prevent leakage if the separation of the sealing surfaces is equal to or less than 0.127 mm (0.005 in).

For the pressure-unseating drywell head closure and equipment hatches, the pressure required to separate the sealing surfaces is a function of the bolt preload, axial stiffness of the bolts and the compression flanges, and the differential thermal expansion between the bolts and the compression flanges. The separation pressure for operable penetrations typically ranges from 1.1 to 1.5 times design pressure (Reference 19F-8). In this study, the separation pressure is conservatively assumed to be 0.460 MPa which is the Structural Integrity Test (SIT) pressure (1.15 times design pressure). At and below this pressure, a metal-to-metal contact is maintained and no leakage other than design allowable leak rate is anticipated, even if the seal degradation temperature of about 533 K (500°F) has reached. Additional pressure in excess of the separation pressure is carried entirely by the bolts. The separation displacement between the sealing surface after the separation pressure is reached is:

$$s = \frac{\pi r^2 (p - p_s)}{K_b} \qquad (19F\text{-}5)$$

where:

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

p_s = the separation pressure, and

K_b = the total bolt axial stiffness.

The above expression neglects the flexibility due to axial deflection of the compression flanges caused by the Poisson effect which contributes little to the total flexibility of the bolts. This approach for predicting leakage is based on the consideration of structural deformations in terms of separation of connecting flanges of pressure unseating equipment hatches and drywell head. The adequacy of this approach has been recently confirmed by the Sandia hatch leakage tests (Reference 19F-9) in that the predicted leakage onset pressures were in favorable agreement with the test results. The drywell head anchorage to the top slab has a pressure capability higher than the drywell head shell and the leakage path of the drywell head assembly before the failure pressure is reached is through the flanges.

The drywell head is a 10.3-m diameter closure with double seal. One hundred twenty 68-mm diameter bolts hold the head in place. There are 3 equipment hatches in the containment wall. The largest of them has twenty 36-mm diameter bolts with double seal, and has a diameter of 2.6 m. According to Equation 19F-5, the separation displacement at 0.929 MPa capability pressure is calculated to be about 0.0838 mm (0.0033 in) for the drywell head and 0.140 mm (0.0055 in) for the most flexible equipment hatch. The equipment hatch separation displacement is slightly larger than 0.127 mm (0.005 in). However, the resulting gap of 0.0127 mm (0.0005 in.) is small and no significant leakage is expected before the capability pressure is reached.

For equipment hatches, another potential leakage mechanism is ovalization of the sleeve which causes the sleeve to slide relative to the tensioning ring (or the cover flange). An initiation of leakage due to sleeve ovalization, however, requires significant deformations of the containment shell around the equipment hatch. The average circumferential membrane strain in the shell that is needed to result in the initiation of leakage from ovalization for equipment hatches identified in the ANL survey (Reference 19F-8) was found to range from 2.5% to 7.3% by SNL (Reference 19F-8). For the equipment hatches under consideration, the ovalization leakage onset strain which is the ratio of the sleeve wall thickness at the sealing surface to the sleeve radius ranges from about 1.2% to 5.8%. At a pressure of 1.34 MPa, the maximum radial deflection of the wetwell wall was calculated to be 25.0 mm (0.983 in.) from the FINEL analysis (Table 19F-1). The corresponding hoop membrane strain is 0.15%. It is less than 1.2% and no leakage from sleeve ovalization of the equipment hatches will occur before the capability pressure is reached.

The leakage rate which should be small for the separation displacements of the drywell head and equipment hatches defined above cannot be quantified based on current capabilities. To facilitate source term calculations, leak areas as a function of pressure are conservatively taken to be the product of the separation displacements and the seal length for the drywell head and equipment hatches. It should be noted that this approach results in a very conservative leak area estimate since

(1) the seal is assumed lost at 533 K (500°F), and

(2) no credit is taken for the springback capability of 0.127 mm (0.005 in.) for degraded seals.

The personnel airlocks are the pressure-seating type. Although separation between the sealing surfaces at the door corners may still be possible at high pressures, the amount of separation is nevertheless expected to be less than that of the pressure-unseating drywell head and equipment hatches at same pressures. Therefore, no calculations are made to predict separation displacements for the airlocks. Instead, the leak area through the airlocks is assumed to be 10% of the sum of the leak areas estimated for the drywell head and equipment hatches. This assumption is realistic since the sealing length of the airlocks is less than 20% of the total sealing length of the drywell head and equipment hatches, and the separation of airlock sealing surfaces is expected to be smaller.

The total estimate of pressure-dependent leak areas attributed to the drywell head, equipment hatches and personnel airlocks is shown below. At and below the separation pressure [0.460 MPa], leakage is within the design limit and the equivalent leak area is negligible.

Pressure	Leak Area	
MPa	Cm^2	In^2
0.100	0	0.00
0.411	0	0.00
0.460	0	0.00
0.515	7.93	1.23
0.584	17.9	2.77
0.653	27.8	4.31
0.72	37.7	5.85
0.791	47.6	7.39
0.860	57.6	8.93
0.929	67.5	10.47

19F.3.3 Summary

The ultimate pressure capability of the containment structure is limited by the drywell head whose failure mode is plastic yield of the torispherical dome. The pressure capability is 1.025 MPa at 533 K (500°F) and reduces to 0.929 MPa at 644 K (700°F). The governing service level C (for steel portions not backed by concrete) / factored load category (for concrete portions including steel liner) pressure capability of the containment structure is 0.770 MPa at 533 K (500°F) which is the internal pressure required to cause the maximum stress intensity in the steel drywell head to reach general membrane yielding according to service level C limits of ASME-III, Division 1, Subsubarticle NE-3220.

No liner leakage will occur before the capability pressure is reached. Leakage through fixed (mechanical and electrical) penetrations is negligible compared to leakage through large operable penetrations such as the drywell head, equipment hatches, and personnel airlocks. The total pressure-dependent leak areas attributed to those operable penetrations are conservatively estimated in Subsection 19F.3.2.2, assuming no sealing action from degraded seals at temperatures above 533 K (500°F).

19F.4 References

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19F-6 Clauss, D.B., "Round Robin Analysis of the Behavior of a 1:6 scale reinforced concrete containment model pressurized to Failure: Post-test Evaluations", NUREG/CR-5341 SAND 89-0349, Sandia National Laboratories, October 1989.

19F-7 "Containment Building Liner Plate Design Report", BC-TOP-1, Rev. 1, Bechtel, December 1972.

19F-8 Clauss, D. B., von Riesemann, W. A., and Parks, M. B., "Containment Penetrations", SAND88-0331C.

19F-9 Parks, M.B., Walther, H.P., and Lambert, L.D., "Evaluation of the Leakage Behavior of Pressure-Unseating Equipment Hatches and Drywell Heads", SAND90-180C, 18th Water Reactor Safety Meeting, October 1990.

Table 19F-1 Summary of Stresses and Strains

Loading Case				Maximum Rebar Stress/ Allow. Stress		Liner Strain		Concrete Comp. Stress/ Allow. Str (MPa)	Component Rebar Stresses / Allowable Stresses (MPa)								Max. Radial Defl. @ Wet-well mm (In.)
									Wetwell		Drywell		Basemat		Diaphrg.		
No.	Title	P.D. MPa	P.W. MPa	Merid. MPa	Hoop MPa	Tens. mm/mm	Comp. mm/mm		Mer.	Hoop	Mer.	Hoop	Rad.	Hoop	Rad.	Hoop	
1	SIT-1	0.359	0.359	79.3 310.3	82.7 310.3	.00052	–.00011	–3.86 –16.55	79.3 310.3	82.7 310.3	42.7 310.3	35.2 310.3	27.6 310.3	30.3 310.3	75.2 310.3	42.7 310.3	6.25 (.246)
2	SIT-2	0.310	0.138	60.7 310.8	31.0 310.8	.00035	–.00007	–3.86 –16.55	41.6 310.3	31.0 310.3	20.0 310.3	22.8 310.3	26.2 310.3	27.6 310.3	60.7 310.3	26.2 310.3	2.29 (.090)
3	4Pa.	1.24	1.24	277.9 413.7	337.8 413.7	.00185	–.00016	–4.69 –23.4	277.9 413.7	337.8 413.7	200.6 413.7	93.8 413.7	84.1 413.7	74.5 413.7	230.3 413.7	128.2 413.7	24.9 (.983)

Table 19F-2 Summary of Pressure Capabilities of Various Components of the RCCV

Structural Component	Pressure Capability MPa Categories (Criteria)	
	Level C MPa	Ultimate** MPa
Wetwell	1.467	1.818
Upper Drywell	2.404	>2.659
Basemat	4.500	6.203
Top Slab*	1.232	- - -

Notes:

* The pressure capability shown for the RCCV top slab which is a non-axisymmetric portion of the RCCV, is calculated based on extrapolation of elastic STARDYNE analysis results. Pressure value 1.232 MPa is governed by the pool girders, pressure capacity of the reinforcing of the top slab is 1.329 MPa.

** Ultimate capability has been calculated based on rebars at both faces of a cross section reaching yield stress.

\> (Greater than) sign means that rebar on only one face of the section reached yield, and the ultimate capacity will be higher than the value indicated.

Figure 19F-1 ABWR Reactor Building/ Primary Containment (0° - 180° Section View)

(Refer to Figure 1.2-2)

Figure 19F-2 Primary Containment Configuration

(Refer to Figure 6.2-26)

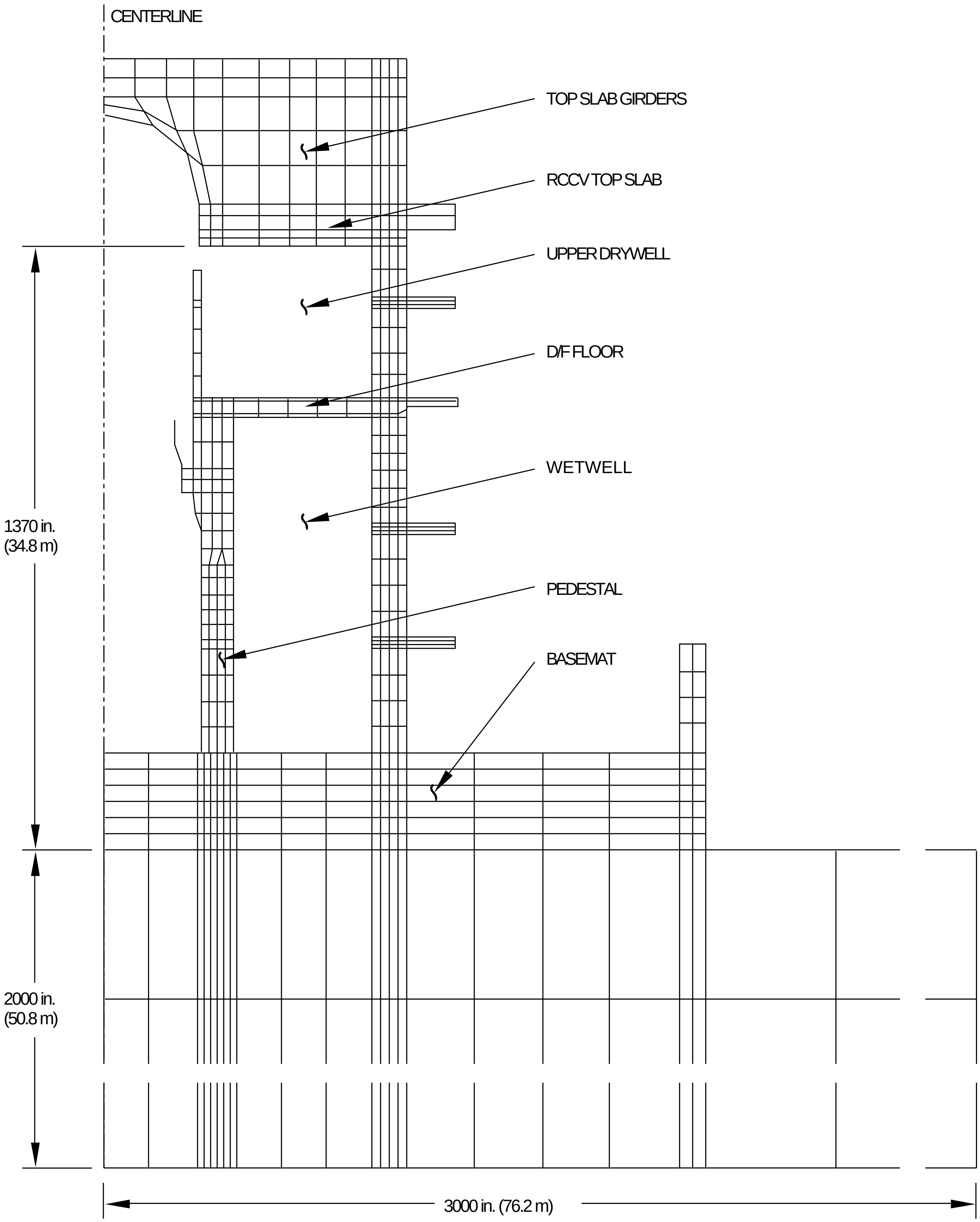


Figure 19F-3 FINEL Model

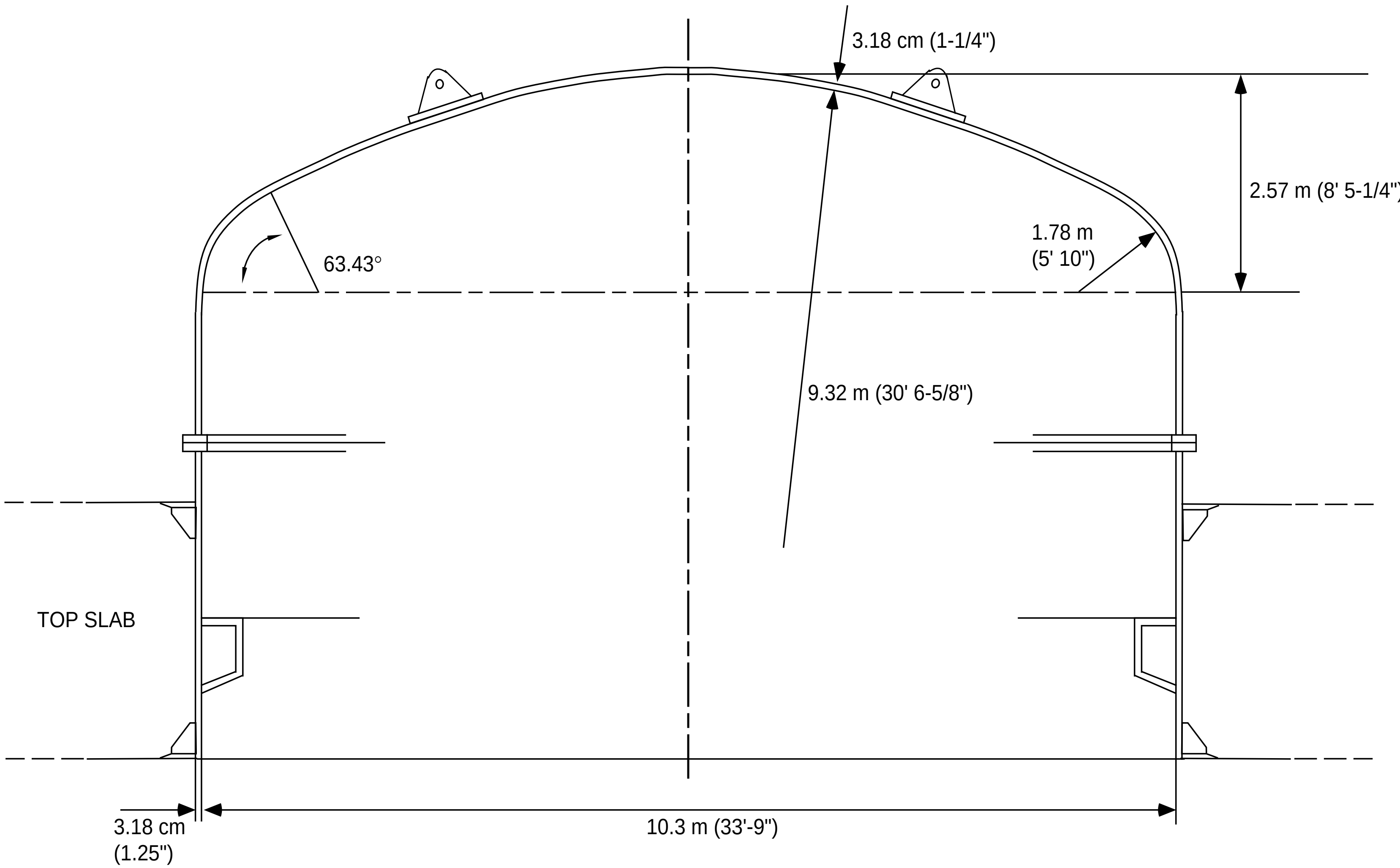


Figure 19F-4 Drywell Head

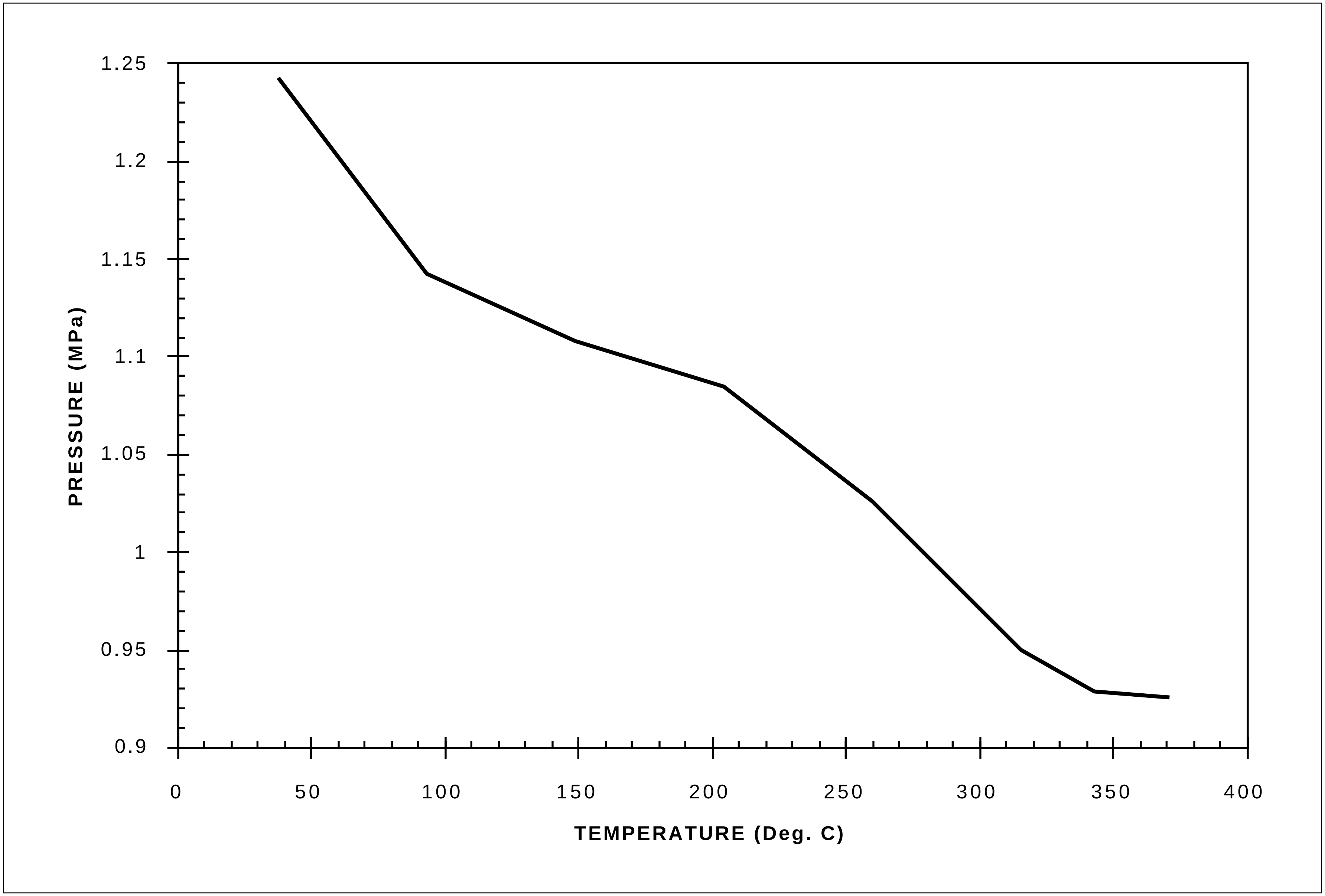


Figure 19F-5 Drywell Head Pressure Capability vs Temperature

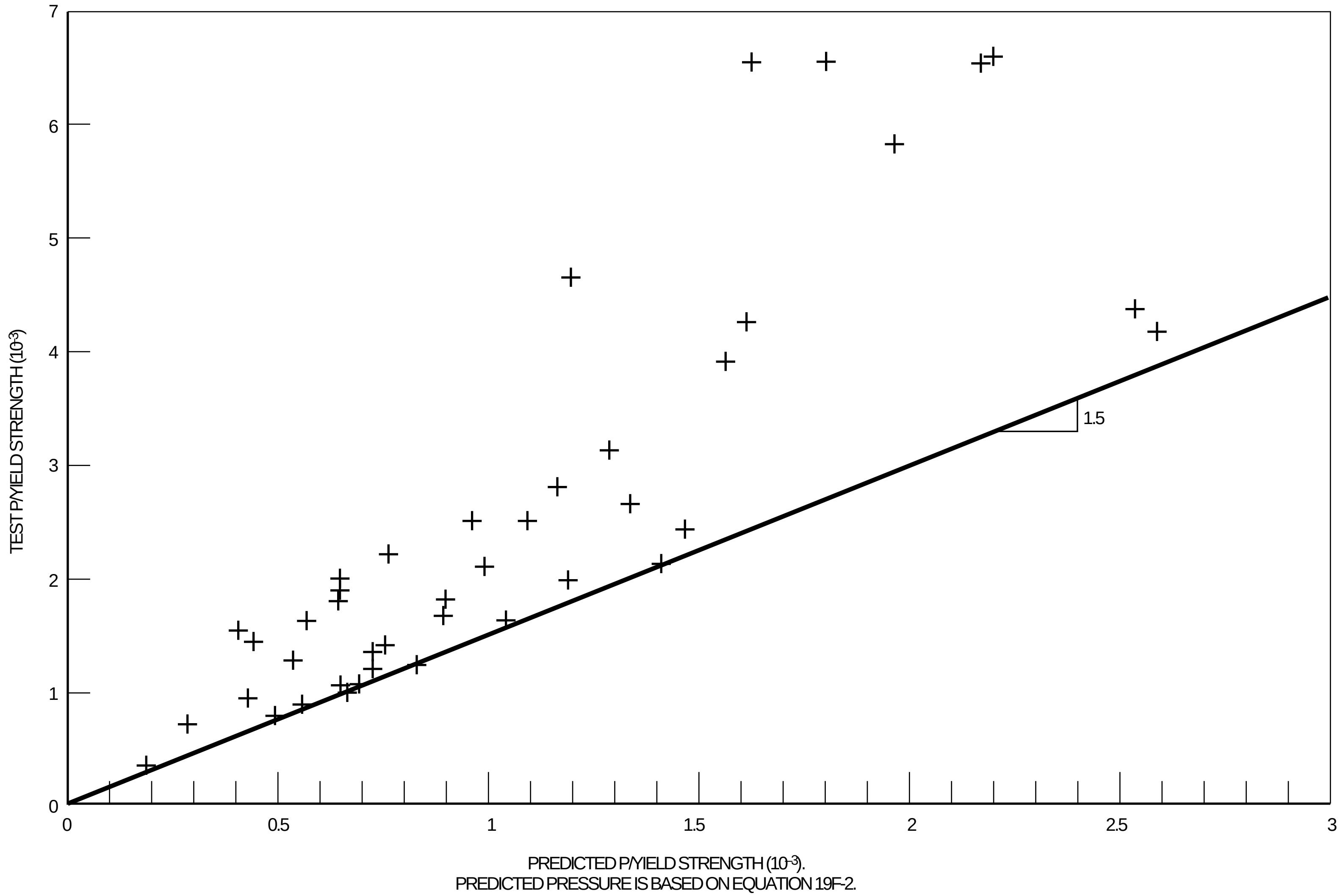


Figure 19F-6 Torispherical Head Buckling Test Data

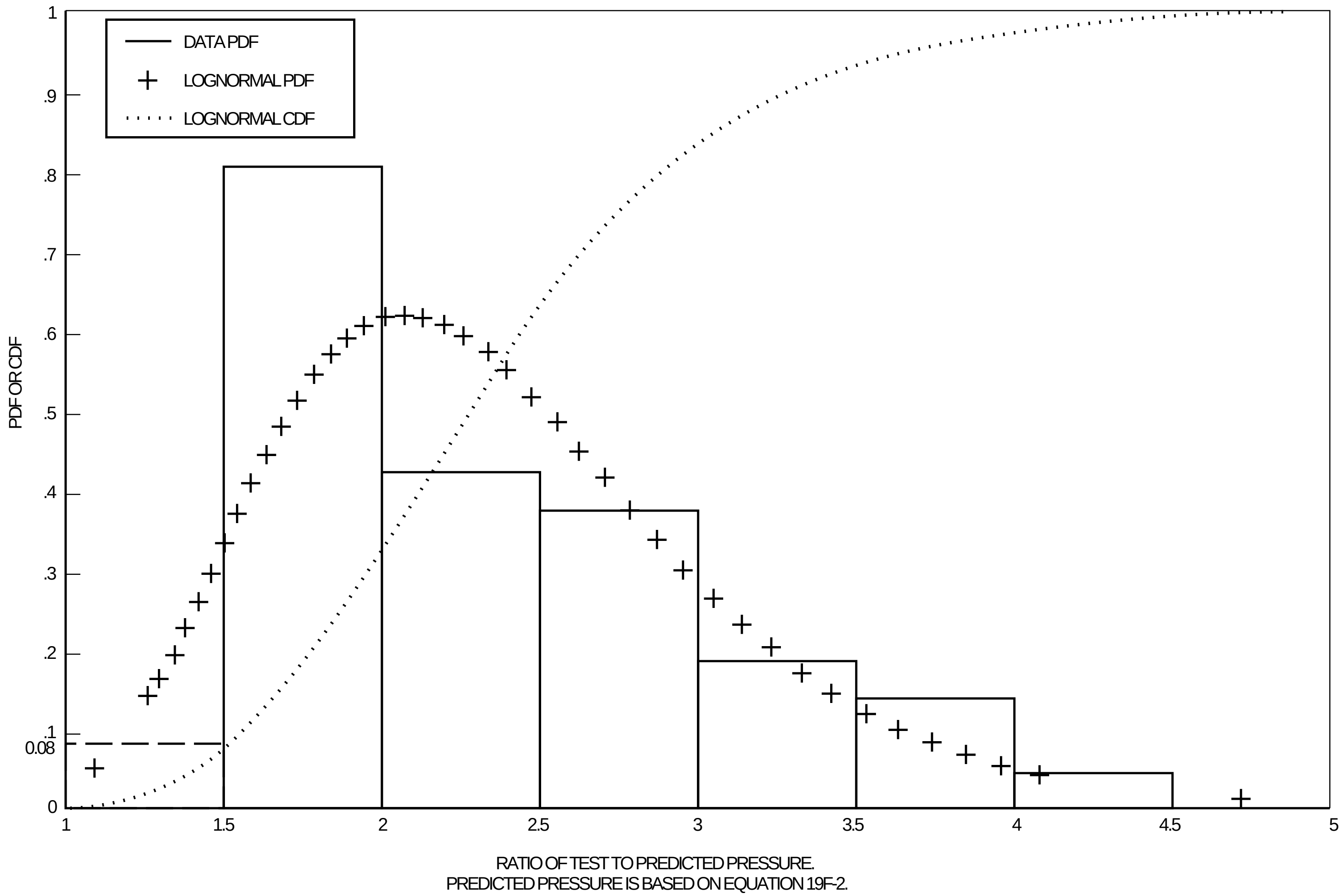


Figure 19F-7 Torispherical Head Buckling Test Data Statistical Distribution

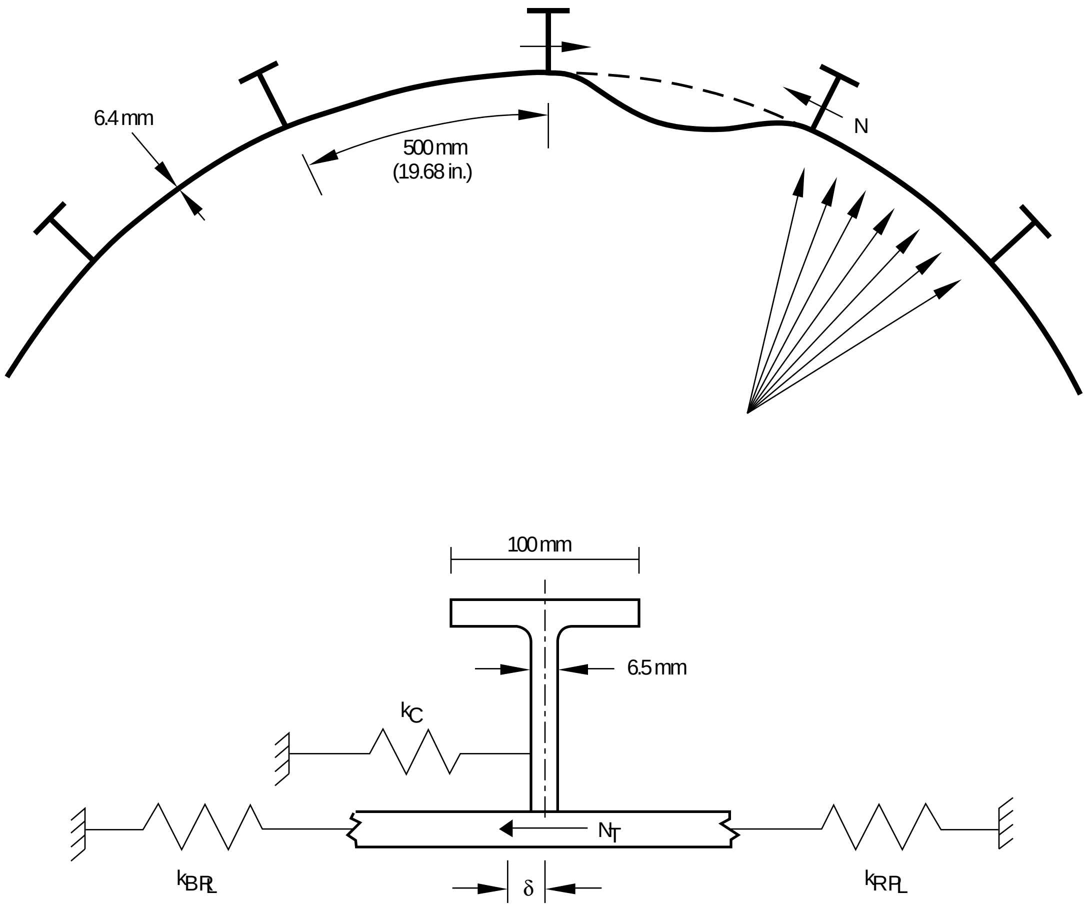
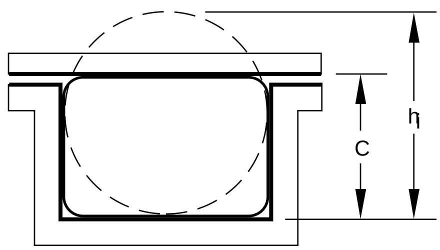


Figure 19F-8 Containment Liner Buckling



Sq = $(h_i - C)/h_i$

WHERE

h_i = INITIAL SEAL HEIGHT

C = COMPRESSED SEAL HEIGHT IN NORMAL OPERATIONS

Figure 19F-9 Definition of Squeeze for Seals

19FA Containment Ultimate Strength

This Attachment is not part of the DCD (Refer to SSAR)

19G Not Used

19H Seismic Capacity Analysis

19H.1 Introduction

This subsection presents seismic capacities for selected structures and components that have been identified as potentially important to the seismic risk analysis of the ABWR standard plant. The seismic capabilities in terms of seismic fragilities are first estimated, from which the high confidence low probability of failure (HCLPF) capacities are then derived. The HCLPF capacities serve as input to the system analysis following the seismic margins approach.

The peak ground acceleration of the design earthquakes is 0.3g for the Safe Shutdown Earthquake (SSE). Extensive seismic soil-structure interaction analyses of the reactor building and control building complex were performed for a wide range of generic site conditions under a 0.3g SSE. The analysis results in terms of site-envelope SSE loads are presented in Appendix 3A. The standard plant designed to these site-envelope seismic loads may result in significant design margins when it is situated at a specific site, particularly a soft soil site. Thus, the seismic capacities estimated from the site-envelope design requirements may be very conservative for certain sites.

For the seismic category I structures and components for which seismic design information is available, the seismic fragilities are evaluated using the factor of safety approach, which is called the Zion method in NUREG/CR-2300, PRA Procedures Guide (Reference 19H-1). This approach identifies various conservatisms and associated uncertainties introduced in the seismic design process and provides a probabilistic estimate of the earthquake level required to fail a structure or component in a postulated failure mode by linear extrapolation of the design information supplemented by judgement.

For certain safety-related components such as pumps, valves, and electrical equipment whose design details are not currently available, the generic seismic fragilities recommended in the EPRI ALWR Requirements Document, Appendix A PRA Key Assumptions and Groundrules (Reference 19H-2) or other data sources are used as appropriate. Those generic fragilities were chosen based on a review of prior PRAs and fragility data. They are considered achievable for the ABWRs with an evolutionary improvement in the seismic capacities of the components designed to a 0.3g SSE.

19H.2 Fragility Formulation

Seismic fragility of a structure or component is defined herein to be the cumulative conditional probability of its failure as a function of the mean peak ground acceleration (i.e., the average of the peak of the two horizontal components).

The probability model adopted for fragility description is the lognormal distribution. Using the lognormal distribution assumption, an entire family of fragility curves can be fully described in terms of the median ground acceleration and two random variables as:

$$A = A_m \varepsilon_\gamma \varepsilon_\mu \tag{19H-1}$$

where:

A_m	=	median peak ground acceleration corresponding to 50% failure probability.
ε_γ	=	a lognormally distributed random variable accounting for inherent randomness about the median. It is characterized by unit median and logarithmic standard deviation β_γ.
ε_μ	=	a lognormally distributed random variable accounting for uncertainty in the median value. It is characterized by unit median and logarithmic standard deviation β_μ.

With known values of A_m, β_γ, and β_μ, the failure probability P_f at acceleration less than or equal to a given acceleration a can be computed using the following equation for any nonexceedance probability (NEP) level Q.

$$P_f(A \le a|Q) = \phi\left[\frac{1}{\beta_\gamma}\ln\left(\frac{a}{A_m}\right) + \frac{\beta_\mu}{\beta_\gamma}\phi^{-1}(Q)\right] \tag{19H-2}$$

where $\phi(.)$ is the standard Gaussian cumulative distribution function. Figure 19H-1 shows a typical family of fragility curves for various NEP levels. The center solid curve represents the median fragility curve at 50% NEP level. The logarithmic standard deviation of the randomness component β_γ determines the curve slope. The logarithmic standard deviation of the uncertainty component β_μ is a measure of the spread from the median curve. The 95th percentile and 5th percentile curves in Figure 19H-1 are the upper and lower bounds of the failure probability for a given acceleration, corresponding to 95% and 5% NEP levels, respectively.

When only the point estimate is of interest, which is the case for this analysis, the total variability about the median value is taken to be the square root of the sum of the squares (SRSS) of the randomness and uncertainty components.

$$\beta_c = \sqrt{\beta_\gamma^2 + \beta_\mu^2} \tag{19H-3}$$

The fragility curve corresponding to the median value A_m with associated composite logarithmic standard deviation can be computed by the following equation:

$$P_f(A \le a) = \phi\left[\frac{1}{\beta_c}\ln\left(\frac{a}{A_m}\right)\right] \tag{19H-4}$$

This composite fragility curve is also called the mean fragility curve and is shown as the dashed curve in Figure 19H-1 for illustration. It represents the best estimate fragility description.

In estimating the median ground acceleration capacity and the associated variability, an intermediate variable defined as safety factor F is utilized. The safety factor is related to the median ground acceleration capacity by the following relationship.

$$A_m = FA_d \tag{19H-5}$$

where A_d is the ground acceleration of the reference design earthquake to which the structure or component is designed. A key step in the seismic fragility estimate thus involves the evaluation of the factor of safety associated with the design for each important potential failure mode. The design margins inherent in the component capacity and the dynamic response to the specific acceleration are the two basic considerations. Each of the capacity and response margins involves several variables, and each variable has a median factor of safety and variability associated with it. The overall factor of safety F is the product of the factor of safety for each variable F_i.

$$F = \prod_i F_i \tag{19H-6}$$

The overall composite logarithmic standard deviation is SRSS of the composite logarithmic standard deviations in the individual factors of safety.

$$\beta_c = \sqrt{\sum_i \beta_{ci}^2} \tag{19H-7}$$

Knowing the median peak ground acceleration (A_m) and associated logarithmic standard deviation (β_c), the HCLPF capacity is obtained using the equation below.

$$HCLPF = A_m \exp(-2.326\beta_c) \tag{19H-7a}$$

19H.3 Structural Fragility

19H.3.1 General

The plant structures are divided into two categories according to their function and the degree of integrity required to protect the public during a seismic event. These categories are seismic category I and non-category I. Seismic category I includes those structures whose failure might cause or increase the severity of an accident which would endanger the public health and safety. The reactor building and control building structures are in this category. The non-category I structures are those structures which are important to reactor operation, but are not essential for preventing an accident which would endanger the public health and safety, and are not essential for the mitigation of the consequences of these accidents. One example is the turbine building structure.

For the purpose of this study, structures are considered to fail functionally when inelastic deformations of the structure under seismic load increase to the extent that the operability of the safety-related components attached to the structure cannot be assured. The ductility limits chosen for structures are estimated as corresponding to the onset of significant structural damage. For many potential modes of failure, this is believed to represent a conservative bound on the level of inelastic structural deformation which might interfere with the function of the system housed within the structure.

The potential of seismic-induced soil failure such as liquefaction, differential settlement, or slope instability is highly site dependent and cannot be assessed for generic site conditions. It is assumed in this analysis that there is no soil failure potential in the range of ground motions considered.

Building-to-building impact due to differential building displacements under strong earthquakes is deemed incredible since adjacent buildings are separated by more than 182 cm (6 feet). Differential building displacements of sufficient magnitude could, however, potentially result in damage to interconnecting piping, depending on system configuration and sliding resistance of building foundation. Detailed evaluation of seismic capacities of interconnecting systems against differential building displacement cannot be made due to lack of design details and specific site conditions. It is assumed that the mode of failure due to differential building displacement has a capacity no less than the generic piping fragility.

19H.3.2 Reactor Building Complex Structures

Detailed fragility evaluations were made for the following structures in the reactor building complex:

- Reactor building shear walls
- Containment
- Reactor pressure vessel pedestal

Those structures were evaluated according to the approach outlined previously and using various safety factors as presented below.

The factor of safety for a structure against a specific failure mode is the product of the capacity factor F_c and structural response factor F_{rs};

$$F = F_c F_{rs} \tag{19H-8}$$

The individual factors in the capacity and response factors are presented in the following subsections.

19H.3.2.1 Capacity Factor (F_c)

The capacity factor represents the capability of a structure to withstand seismic excitation in excess of the design earthquake. This factor is composed of two parts:

$$F_c = F_s F_u \tag{19H-9}$$

where:

F_s = the ultimate structural strength margin above the design SSE load, and

F_u = the inelastic energy absorption factor accounting for additional capacity of the structure to undergo inelastic deformations beyond yield.

The capacity estimated by this approach is the elastic capacity equivalent to the actual nonlinear behavior under strong motion earthquakes.

(1) Strength Factor (F_s)

The strength factor associated with seismic load can be calculated using the following equation.

$$F_s = \frac{P_u - P_n}{P_s} \tag{19H-10}$$

where:

P_u = the actual ultimate strength,

P_n = the normal operating and operation transient (i.e., SRV) loads, and

P_s = the design SSE load.

The earthquake-resistant structural elements of the reactor building are reinforced concrete shear walls which are integrated with the reinforced concrete cylindrical containment through concrete floor slabs. The reactor pressure vessel pedestal is of a composite steel-concrete construction consisting of two concentric steel shells filled with concrete in the annulus. In addition, stiffeners are welded to the steel shells. The specified compressive strength of concrete is 27.5 MPa. The specified yield strength of reinforcing steel of ASTM A615, Grade 60 is 414 MPa. The structural steel material for the pedestal shells and stiffeners is A572, Gr. 50, for which the specified yield strength is 345 MPa. These are design values; the actual material strengths are higher.

Concrete compressive strength used for design is normally specified as a value at a specific time after mixing (28 or 90 days). This value is verified by laboratory testing of mix samples. The strength must meet specified values, allowing a finite number of failures per number of trials. There are two major factors which affect the actual strength:

(a) To meet the design specifications, the contractor attempts to create a mix that has an “average” strength somewhat above the design strength, and

(b) As concrete ages, it increases in strength.

Taking those two elements into consideration, the actual compressive strength of aged concrete is commonly 1.3 times the design strength (Reference 19H-3). The total logarithmic standard deviation about the median strength is about 0.13.

According to the same reference, the ratio of the median yield strength to the specified strength of reinforcing steel is taken to be 1.2 with logarithmic standard deviation of 0.12.

The median yield strength of steel plates is typically 1.25 times the code specified strength with logarithmic standard deviation of 0.14 (References 19H-3 and 19H-4).

The reactor building shear wall is chosen as an example for the discussion of the strength factor evaluation. For reinforced concrete shear walls the ultimate shear strength can be computed using the following equation (Reference 19H-5).

$$
\begin{aligned}
v_u &= v_c + v_s \\
&= 8.3\sqrt{f_c} - 3.4\sqrt{f'_c}\left(\frac{h}{w} - \frac{1}{2}\right) + \frac{N}{4wt} + \rho_{se} f_y
\end{aligned}
\tag{19H-11}
$$

where:

v_c	=	shear strength provided by concrete
v_s	=	shear strength provided by reinforcing steel
f'_c	=	concrete compressive strength
h	=	wall height
w	=	wall length
N	=	bearing load
f_y	=	yield strength of reinforcing steel
t	=	wall thickness
ρ_{se}	=	$A\rho_v + B\rho_h$
ρ_h	=	horizontal steel reinforcement ratio
ρ_v	=	vertical steel reinforcement ratio
A & B	=	constants depending on h/w:

	A	**B**
$h/w < 0.5$	1	0
$0.5 < h/w < 1.0$	$2(1 - h/w)$	$2h/w - 1$
$1.0 < h/w$	0	1

In computing ultimate shear strength with this equation, the median material strengths of the concrete and reinforcing steel defined above are used and the wall bearing load is conservatively neglected.

The strength factor F_s is then calculated using Equation 19H-10 for each of the levels of the reactor building shear walls. The operating loads do not result in lateral shear force and horizontal loads induced by SRV actuations are found to be negligible compared to the SSE-induced horizontal loads. Therefore, the strength factor is the ratio of the median shear strength to the design SSE shear. The least strength factor is found to be 3.32. The associated logarithmic standard deviation is calculated to be 0.09 using the second moment approximation (Reference 19H-5) accounting for both concrete and reinforcing steel material strength variabilities. There is also an uncertainty associated with Equation 19H-11 since it is an approximate model fit to data. The modeling uncertainty is 0.15 expressed in terms of logarithmic standard deviation (Reference 19H-5). The total composite logarithmic standard deviation in the median strength factor is 0.17, which is the SRSS value of 0.09 for the material strength uncertainty and 0.15 for the equation uncertainty.

(2) Inelastic Energy Absorption Factor (F_u)

The inelastic energy absorption factor (F_u) accounts for the fact that an earthquake represents a limited energy source and many structures are capable of absorbing substantial amounts of energy beyond yield without loss of function. The parameter commonly used to measure the energy absorption capacity in the inelastic range is the ductility ratio, μ. It is defined as the ratio of the maximum displacement to the displacement at yield. Newmark, Reference 19H-6, has shown that in the amplified acceleration range (approximately 2 to 8 Hz) the inelastic energy absorption factor F_u can be estimated by

$$F_u = \varepsilon\sqrt{2\mu - 1} \qquad (19H\text{-}12)$$

where ε is an error variable to account for the uncertainty associated with the use of this equation. This error variable is assumed to be lognormally distributed with a median of unity and a logarithmic standard deviation ranging from 0.02 to 0.1 (Reference 19H-7). For rigid structures (fundamental frequency above 20 Hz), the following equation given by Reference 19H-7 may be used.

$$F_u = \varepsilon\mu^{0.13} \qquad (19H\text{-}13)$$

Again, ε is an error variable of unit median and logarithmic standard deviation ranging from 0.02 to 0.1. For intermediate frequencies, the F_u factor can be interpolated from Eqs. 19H-12 and 19H-13.

According to Reference 19H-3, the system ductility ratio for reinforced concrete shear walls failing in shear is 2.5. The integrated building/containment system responds in multiple modes with predominant modes up to 10 Hz. The corresponding inelastic energy absorption factor is thus about 2.0 according to Equation 19H-12. The associated logarithmic standard deviation is 0.25 (Reference 19H-3). Flexural failures tend to be more ductile than shear failures. A ductility ratio of 4.0 is estimated and the corresponding F_u is 2.65 with logarithmic standard deviation of 0.25.

Steel structures are typically more ductile than concrete structures. When local buckling is prevented, the allowable ductility ratio is 5 (Reference 19H-8) for which the corresponding F_u is 3. The F_u factor is taken as unity when the failure mode is of a brittle type such as buckling or failure of high strength anchor bolts.

19H.3.2.2 Structural Response Factor (F_{rs})

The structural response factor (F_{rs}) consists of a number of factors or parameters introduced in the calculation of structural response in the seismic dynamic analysis. Response calculations performed in the design analysis utilized conservative deterministic parameters. The actual response may differ significantly from the calculated response for a given peak ground acceleration level since many of these parameters are random. The structural response factor is evaluated as the product of the following factors that are considered to have the most influence on the structural response.

$$F_{rs} = F_{sa} F_d F_{ssi} F_m F_{mc} F_{ecc} \qquad (19H\text{-}14)$$

where:

F_{sa} = spectral shape factor accounting for the margin of the design ground response spectra with respect to the median centered spectra,

F_d = damping factor accounting for the variability in response due to difference in expected damping at failure and damping used in the analysis,

F_{ssi} = soil-structure interaction factor accounting for the variability associated with SSI effects on structural response,

F_m = structural modeling factor accounting for the variability in response due to modeling assumptions,

F_{mc} = modal response combination factor accounting for the variability in response due to the method used in combining modal responses,

F_{ecc} = earthquake component combination factor accounting for the variability in response due to the method used in combining the earthquake components.

(1) Spectral Shape Factor (F_{sa})

The ground response spectrum considered in the seismic design is the site-independent spectrum from Regulatory Guide (RG) 1.60, normalized to the design ground acceleration. To facilitate dynamic analysis using the time history method, artificial acceleration time histories of three directional components were generated so that the resulting spectra envelop the design spectra for the damping ratios of interest.

For the purpose of seismic risk assessment, the median ground spectrum given in NUREG/CR-0098 (Reference 19H-9) is considered to be the realistic input ground motion definition. The differences between the design spectra and median spectra are the margins in the ground motion input.

The spectral shape factor (F_{sa}) is defined to be the ratio of the amplification factor of the design spectrum to that of the median spectrum at the same frequency and damping level.

$$F_{sa} = AF_d / AF_m \tag{19H-15}$$

In constructing the median spectrum, the competent soil condition is conservatively assumed since it results in higher maximum ground velocity and displacement amplitudes than the rock condition for a same maximum ground acceleration. The design spectrum and median spectrum are compared at the 5% damping level for the maximum ground acceleration of 1g. The average spectral shape factors in representative frequency ranges are approximately

Frequency Range (Hz)	Average F_{sa}
2 to 10	1.34
10 to 20	1.20

Frequency Range (Hz)	Average F_{sa}
20 to 33	1.07
above 33	1.00

The logarithmic standard deviation in the spectral shape factor is the variability in the median spectra which is 0.2 according to Reference 19H-2. No variability exists for frequencies above 33 Hz.

(2) Damping Factor (F_d)

The SSE loads were calculated using the SSE damping ratios specified in RG 1.61. The RG 1.61 damping values are considered to be quite conservative, particularly at response levels near failure. More realistic damping values are specified in Reference 19H-9.

For reinforced concrete structures the damping ratio considered in the SSE analysis is 7%. The realistic values at or near yield range from 7 to 10% (Reference 19H-9). The upper bound value is considered to be median and the lower bound corresponds to the 84th percentile level.

The RG 1.60 design ground spectra are used to evaluate the margin in response due to difference in actual damping at failure and design damping. The damping factor F_d can be calculated to be the ratio of the amplification factor at design damping (AF_{dd}) to the amplification factor at median damping (AF_{md}) at the same frequency.

$$F_d = AF_{dd}/AF_{md} \tag{19H-16}$$

The associated logarithmic standard deviation can be calculated to be the natural log of the ratio of the amplification factor at 84th percentile damping (AF_{bd}) to the amplification factor at median damping (A_{md}) at the same frequency.

$$\beta_c = \ln(AF_{bd}/AF_{md}) \tag{19H-17}$$

For reinforced concrete structures the average damping factors and associated logarithmic standard deviations in representative frequency ranges are approximately

Frequency Range (Hz)	Average F_d	Average β_c
2 to 10	1.19	0.18
10 to 20	1.12	0.11
20 to 33	1.02	0.02
above 33	1.00	0.0

(3) Soil-Structure Interaction Factor (F_{ssi})

Seismic soil-structure interaction (SSI) analyses for the SSE were performed for the reactor building complex situated in a wide range of generic site conditions as described in Appendix 3A. The design seismic loads were established to be the site-envelope loads calculated by the SSI analyses. The site-envelope loads may have margins for a given site. The margin may be substantial if the specific site is a soft soil site. Since the ABWR standard plant is designed for generic site conditions, no credit is taken for site margins. Thus, the F_{ssi} factor is taken as 1.0. The associated logarithmic standard deviation is estimated to be 0.1.

(4) Modeling Factor (F_m)

The reactor building complex structural model considered in the seismic design analysis is a multi-degree-of-freedom system constructed according to common modeling techniques and the Standard Review Plan (SRP) requirements in terms of number of degrees of freedom and subsystem decoupling. The model is thus considered to be the best estimate and the resulting dynamic characteristics are median centered. The modeling factor is thus unity. A relatively large logarithmic standard deviation of 0.15 is estimated to account for the complexity of the integrated reactor building and the containment design.

(5) Modal Combination Factor (F_{mc})

The analysis method used in the seismic response analysis is the time history method solved in the frequency domain. The phasing between individual modal responses are known and the total response is the algebraic sum of all modes of interest. The maximum response is thus precise and the modal combination factor (F_{mc}) is unity. The associated uncertainties should be less than the uncertainties associated with the response spectrum method, in which the maximum modal responses are combined by the SRSS method. Therefore, a relatively small logarithmic standard deviation of 0.05 is estimated.

(6) Earthquake Component Combination Factor (F_{ecc})

The effects of multi-directional earthquake excitation on structural response depend on the geometry, dynamic response characteristics, and relative magnitudes of the two horizontal and the vertical earthquake components. The design method is SRSS, according to RG 1.92, which is considered to result in median-centered response. The earthquake component combination factor is 1.0.

The reactor building walls are designed to resist in-plane loads. The torsional effects were found to be small and the walls mainly respond to the horizontal motion parallel to the walls. The vertical loads on the walls due to the vertical excitation are typically less significant in contributing to the total stresses and there is an equal probability of acting upward or downward. The earthquake component combination effect on the wall design is thus not significant and a small logarithmic standard deviation of 0.05 is estimated.

Other major structures inside the reactor building such as the containment and the pedestal are cylindrical structures. The responses to the three orthogonal excitation components are essentially uncoupled. The logarithmic standard deviation is estimated to be 0.05.

19H.3.2.3 Reactor Building Complex Summary

The median values of individual factors and associated logarithmic standard deviations are summarized in Tables 19H-2 through 19H-4 for the critical failure modes of the reactor building walls, the containment, and the reactor pressure vessel pedestal. The overall factor is the product of all individual factors. The total logarithmic standard deviation is the SRSS value of individual logarithmic standard deviations. The seismic fragility in terms of median ground acceleration is the product of the overall factor and the SSE design ground acceleration of 0.3 g.

19H.3.3 Other Seismic Category I Structures

Seismic category I structures other than the reactor building structures in the ABWR standard plant include the control building, and the radwaste building substructures.

The control building fragility is evaluated using the same procedure described above for the reactor building. The controlling mode of failure is shear of shear walls. Table 19H-5 shows the margin in each of the strength and response factors.

The radwaste building does not contain safety-related equipment and its failure will not lead to core damage. Consequently, an estimate of the radwaste building fragility is not required.

19H.4 Component Fragility

19H.4.1 General

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

(1) ABWR specific components whose fragility evaluation is made according to existing design information.

(2) Generic components whose fragilities are based on the data recommended in Reference 19H-2 or other data sources as appropriate.

19H.4.2 ABWR Specific Components

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

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

The design seismic loads for these components were calculated directly using a coupled building structures and RPV/internals model. Consequently, no subsystem dynamic analyses using input motions at support points were required. Therefore, the fragility evaluation procedures used for the reactor building structures as presented previously are also applicable to these specific components.

Reactor Pressure Vessel (RPV)

The failure of the RPV due to an earthquake results in a sequence similar to a large break loss-of-coolant accident, with the exception that there may be no means to provide makeup (i.e., injection or cooling) to the core. The ABWR RPV is supported by a conical skirt which is anchored to the pedestal with 120-68 mm minimum diameter high-strength anchor bolts. At an upper elevation, the RPV is laterally restrained by stabilizers which are connected to the reactor shield wall.

Failure of the RPV support system would result in excessive RPV deflection which could induce failure of the connecting pipes. The ultimate capacity of the support system is provided by both the skirt and the stabilizers. In this analysis, the resistance capacity of the support system is conservatively limited to the yielding capacity of the stabilizers or the skirt, whichever is smaller.

The critical failure mode is found to be stabilizer yielding.

RPV Internal Components

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

Tables 19H-7 through 19H-10 show the failure modes and associated median ground acceleration capacities of those components. The contributing factors are also shown in these tables.

As noted, the fuel assemblies are found to have the lowest seismic capacity among the RPV internal components. The failure mode is excessive deflection of the fuel channel. The maximum deflection that the channel can undergo without collapse is limited by the amount that would inhibit the control rod from insertion to achieve reactor scram. The scram limited deflection is larger than the channel deflection at yield. To assess the seismic capacity of the channel, the moment-deflection resistance function is conservatively assumed to be of perfect elasto-plastic. The strength margin is taken to be the ratio of the yielding moment to the SSE induced moment. The additional capacity due to inelastic deformation is accounted for with a ductility ratio equal to the scram-limited deflection divided by the yielding deflection.

19H.4.3 Generic Components

Detailed fragility evaluations for safety-related components other than those specific components presented above cannot be made at this stage of certification due to lack of design details.

The ABWR generic components of interest for this seismic risk analysis are the following:

- Cable trays
- Large flat-bottom storage tanks
- Air-operated valves

- Heat exchangers
- Off-site Power (transformers and ceramic insulators)
- Batteries and battery racks
- Electric equipment (chatter failure mode)
- Switchgear / Motor control centers
- Transformers (480V)
- Diesel generators and support systems
- Turbine-driven pumps
- Motor-driven pumps
- Diesel-driven pumps
- Small tanks (e.g., standby liquid control tank)
- Motor-operated valves
- Safety relief, manual, and check valves
- Hydraulic control units
- Heating, ventilation, and air conditioning ducting
- Air handling units / room air conditioners
- Piping
- Service water pump house

Their seismic fragilities and corresponding HCLPF values are summarized in Table 19H-1. These generic seismic capacities are selected from a review of ALWR recommendation (Reference 19H-2) and other PRA studies (References 19H-10 and 19H-11).

19H.5 COL License Information

19H.5.1 Seismic Capacity

The COL applicant shall determine the HCLPF values for the plant-specific / as-designed components corresponding to those generic components defined in Subsection 19H.4.3. The values should be compared to their assumed HCLPF values given in Table 19H-1 (or Table 19I-1 on system basis). It should be noted that only the capacities of important contributors (Section 19.8) need to be determined and compared. These important contributions are hereafter referred to as SMA SSCs for systems, structures, and components needed for consideration in the seismic margins assessment.

An explicit evaluation of HCLPF values of only the important contributors (Section 19.8) need to be performed. However, prior to the HCLPF evaluation it is essential to verify that the quality of construction of structures and installation of equipment and systems are in conformance with the commitments in the SSAR and that the as-built structures systems and components meet all the applicable ITACC requirements. These important components are hereafter referred to as SMA SSCs for systems, structures, and components needed for consideration in seismic margins assessment.

The HCLPF calculations can be made using fragility analysis or the conservative deterministic failure margin (CDFM) approach. The location effects should be taken into account in determining the limiting capacity of the same component on different locations.

For structures, equipment and systems other than the important items metioned above, it is only necessary to verify that the site-dependent conditions are within the site envelope parameters in accordance with the procedure described in Subsection 2.3.1.2 or that site-specific SSE responses are bounded by those considered in the standard design, provided that the as-biult structures, systems and components are verified to be designed, constructed, installed and tested in accordance with Tier 2 and Tier 1 commitments. Otherwise, site-specific HCPLF capacities for these structures and components need to be established.

It is not necessary that in each case the HCLPF equal or exceed the value assumed in the margins analysis of the standardized design, especially since the NRC has judged that HCLPF=0.5 is acceptable. However, depending on the degree of difference and the significance of the component in accident sequences, an evaluation of the site-specific plant level HCLPF capacity may be needed. The level of acceptable seismic margin for the plant should be established in a manner consistent with that used in existing nuclear power plants.

The site should also be investigated for the potential of seismic-induced soil failure (liquefaction, differential settlement, or slope stability) at 1.67 times the site-specific SSE.

In order to increase confidence that the as-designed seismic capacities of the SMA SSCs are realized in the final constructed plant, a seismic walkdown shall be performed by the COL applicant according to the process as follows:

- Step 1—Preparation for Plant Walkdown
- Step 2—Plant Seismic Logic Model Walkdown
- Step 3—Assessment of As-Built SMA SSC HCLPF Values

- Step 4—Seismic Plant Walkdown
- Step 5—Plant Damage State and Plant Level HCLPF Calculations

These steps are discussed in detail in the remainder of this subsection.

Step 1—Preparation for Plant Walkdown

The SMA presented in Appendix 19I contains seismic logic models for the plant. These models include the seismic-induced failures that were considered necessary to be evaluated as part of the SMA. These failures, and the associated HCLPF values of the SMA SSCs shall be reviewed. In preparing for the plant walkdown, all appropriate information regarding these failures should be gathered. These include, but are not necessarily limited to:

- Piping and instrumentation drawings,
- Electrical one-line diagrams,
- Plant arrangement drawings,
- Detailed design drawings,
- Procurement specifications,
- Construction drawings (especially those concentrating on seismic detailing and load paths),
- Quality assurance records,
- Seismic analysis used for defining floor response spectra,
- Floor spectra used as required response spectra by vendors,
- Engineering analyses of seismic performance (especially for representative seismic anchorages), and
- Equipment qualification data / material test data.

Step 2—Plant Seismic Logic Model Walkdown

The walkdown will concentrate on the identification of potential systems interactions that could impact the performance of the front-line and support SSCs included in the models. The original SMA model considered in Appendix 19I included the most significant systems interactions (e.g., collapse of major buildings). However, it is necessary to assure that no other interactions exist in the as-built plant that were not included in the SMA model. The walkdown should include a thorough examination of the SSCs included in the SMA, including piping runs, cable trays, etc. During the walkdown process, the team should identify the presence of any SSCs whose failure could impact the performance of the SMA SSCs. Based on a review of the seismic event trees (Figures 19I-1,-2 and -3) it was considered appropriate to add the following systems to SMA SSCs for this step; RCIC one HPCF train, one LPFL train, and SLC.These could include such things as:

- Non-load bearing walls adjacent to SMA SSCs
- Non-safety components above or adjacent to SMA SSCs
- Hard surfaces within deflection range of SMA SSCs
- Flooding/deluge sources in the vicinity of SMA SSCs.

All such potential interactions should be identified, along with the failure mode that could impact the performance of the SMA SSCs. These are new failure modes based on as-built plant conditions. This must be done for 100 percent of the SSCs included in the event and fault tree models. These new failure modes should be added as basic events on the SMA fault/event trees as appropriate and be added to the list of SMA SSCs. In addition, the design information specified in Step 1 should be assembled for these new failures. Note that all future reference to SMA SSCs is intended to refer to the expanded list, including the newly added system interactions.

Step 3—Assessment of As-Built SMA SSC HCLPF Values

For each SMA SSC, a compilation of the design characteristics that control the HCLPF value should be prepared. These design characteristics can be one of two things: either they directly contribute to the dominant failure mode(s) or to failure modes that are close to being dominant. The dominant failure modes(s) is defined as the failure modes(s), from the list of all potential failure modes that will cause the SSC to be unable to perform its safety function, whose HCLPF value is the lowest (or equal to the lowest). Thus, the reduction of the HCLPF value of this failure mode would result in a corresponding reduction in the HCLPF of the SSC. This being the case, the design characteristics that would be compiled would include all of the specific design conditions that directly contribute to the dominant SSC failure mode(s). Another way to express this is that any change in any one of these design conditions that results in a reduction in seismic capacity will directly cause a reduction in the SSC HCLPF value. In addition, they would also include all such conditions that directly contribute to SSC failure mode(s), if any, that could become the dominant failure mode if it were to have a “somewhat” lower HCLPF value. For the purpose of this review, “somewhat” is defined as about a 10 percent to 20 percent HCLPF reduction. Thus, these failure modes are those whose calculated HCLPF value is only on the order of 10 percent to 20 percent higher than the dominant failure mode.

The characteristics that would be identified could include such things as:

- Size, type and number of anchor bolts,
- Size, type and orientation of support members,

- Distance between rigid pipe supports (allowance for differential motion),
- Distance between components.

The specification of these characteristics should be quite definitive (i.e., numerical where possible).

Step 4—Seismic Plant Walkdown

Final determination of the as-built plant design characteristics affecting HCLPF values is required. This should take the form of a final plant walkdown of the SMA SSCs, and RCIC, one HPCF train, one LPFL train and SLC as noted in step 2. As a product of Step 3, a compilation of key design characteristics (those that control or could control the HCLPF value of the SMA SSCs) was prepared. The plant walkdown is intended to determine the extent to which these design characteristics exist in the plant. Each SSC should be inspected and the as-build condition compared with the key design characteristics.

It is not required to perform a detailed walkdown inspection of 100 percent of the SMA SSCs. A 100 percent “walk by” is sufficient. The “walk by” is intended to assure that there is a reasonable basis for the assumption that the HCLPF of broad classes of SSC are essentially the same (i.e., that the SSCs are of similar design and manufacture and are similarly anchored). For each group of SSCs for which this condition of similarity can reasonably be established by the “walk by”, it will then be necessary to select one representative SSC from each group to be subjected to a more rigorous inspection. This inspection will be conducted in such a manner as to determine if the representative SSC is in agreement with the assumed design characteristics compiled in Step 3.

It is understood that it will not always be possible to visually determine the existence of all the key characteristics, since some of them may be embedded within walls or in other inaccessible places. In such cases, it will be acceptable to use the construction QA records as adequate demonstration that the as-build SSC has the design characteristics required. In all cases, the result of the seismic plant walkdown should be fully documented.

Step 5—Plant Damage State and Plant Level HCLPF Calculations

The final step in the process is to determine HCLPF values for each event sequence, each plant damage state and for the overall plant. This should be done using both the min-max and convolution approaches and reported in the same form as in the SMA in Appendix 19I.

19H.6 References

19H-1 “PRA Procedures Guide”, NUREG/CR-2300, January 1983.

19H-2 “ALWR Utility Requirements Document, Volume II, Chapter 1, Appendix A PRA Key Assumptions and Groundrules”, EPRI, October 1988.

19H-3 “Report on Quantification of Uncertainties, Report of Seismic Analysis Main Committee”, ASCE, March 15 1983.

19H-4 “Severe Accident Risk Assessment—Limerick Generating Station”, NUS Report, April 1983.

19H-5 “Handbook of Nuclear Power Plant Seismic Fragilities, Seismic Safety Margins Research Program”, NUREG/CR-3558, June 1985.

19H-6 Newmark, N. M., “Inelastic Design of Nuclear Reactor Structures and Its Implication on Design of Critical Equipment”, SMIRT Paper K4/1, 1977 SMIRT Conference, San Francisco, 1978.

19H-7 Kennedy, R. P., and Ravindra, M. K., “Seismic Fragilities for Nuclear Power Plant Risk Studies, Nuclear Engineering and Design”, PP47-68, (79) 1984.

19H-8 “Structural Analysis and Design of Nuclear Plant Facilities, Manual and Reports on Engineering Practice”, No. 58, ASCE, 1980.

19H-9 “Development of Criteria for Seismic Review of Selected Nuclear Power Plants”, NUREG/CR-0098, May 1978.

19H-10 Harrison, S. W., Esfandiari, S., Pandya, D., and Ahmed, R., “Seismic Fragility Curves for Evaluation of Generic Electrical Conduit Supports, to be presented in the ASME PVP Annual Meetings”, Honolulu, Hawaii, July 22-24, 1989.

19H-11 Campbell, R. D., Ravindra, M. K., and Bhatia, A., “Compilation of Fragility Information from Available Probabilistic Risk Assessments”, LLNL, September 1985.

Table 19H-1 Seismic Capacity Summary

Structure/Component	Failure Mode	Fragility[1] Capacity[2] Am (g)	Combined[3] Uncertainty	HCPLF (g)
Reactor Building	Wall Shear			
Containment	Shear			
RPV Pedestal	Flexural			
Control building	Shear			
Service Water Pump House	Structural			
Reactor pressure vessel	Support			
Shroud support	Buckling			
CRD guide tubes	Buckling			
CRD housing	Plastic yielding			
Fuel Assemblies	Channel deflection			
Hydraulic Control Unit	LOF			
Cable trays	Support			
Large flat-bottom storage tanks[4]	Anchorage			
Air-operated valves	Stem binding/Air line			
Heat Exchanger	Anchorage			
Off-site power	Ceramic insulators			
Batteries and battery racks	Anchorage/LOF			
Electric equipment (chatter)				
function req'd during event	Relay chattering[5]			
function req'd after event	Relay chattering[4]			
Switchgear/Motor control centers	Functional/Structural[4]			
Transformers	Functional/Structural			
Diesel generators & support systems	Support			
Turbine-driven pumps	Anchorage			
Motor-driven pumps	Anchorage/Impeller deflection			

Table 19H-1 Seismic Capacity Summary (Continued)

Structure/Component	Failure Mode	Fragility[1] Capacity[2] Am (g)	Combined[3] Uncertainty	HCPLF (g)
Small tanks	Anchorage			
Motor-operated valves	Operator distortion			
Safety relief & check valves	Internal damage			
Manual valves[3]	Internal damage			
HVAC ducting	Support			
Air handling units/Room A.C.	Blade rubbing			
Piping[3]	Support			
Diesel-driven pumps[3]	Support			

1 Fragility not part of DCD. Refer to SSAR.

2 Capacities are in terms of median peak ground acceleration.

3 Combined uncertainties are composite logarithmic standard deviations of uncertainty and randomness components.

4 Except for ACIWA (firewater) components (Table 19I-1).

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

Switchgear and motor control centers do include relays whose failure could prevent safety actions after the seismic event. It was assumed that the indicated capacity of this equipment was more representative than the specific relay chatter value since switchgear and motor control centers are normally qualified with the auxiliary relays in place. Also, the type of auxiliary relays used tend to be the most rugged of relay types and would have a higher capacity. The multiplexer output devices for ECCS and RHR operation have been assumed to be solid state devices (rather than relays), so the relay chatter failure mode does not apply.

Table 19H-2 Seismic Fragility For Reactor Building

Component:		Shear Walls		
Failure Mode:		Shear		
		Factor of Safety	**Median Value**[1]	**$ß_c$**[1]
F_c	F_s	: strength margin		
	F_u	: inelastic energy absorption		
F_{rs}	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

Table 19H-3 Seismic Fragility For Containment

Component:	Containment			
Failure Mode:	Shear			
		Factor of Safety	**Median Value[1]**	**$ß_c$[1]**
	F_s	: strength margin		
F_c				
	F_u	: inelastic energy absorption		
	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
F_{rs}	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

Table 19H-4 Seismic Fragility For RPV Pedestal

Component:		RPV Pedestal		
Failure Mode:		Flexural		
		Factor of Safety	**Median Value[1]**	**$ß_c$[1]**
	F_s	: strength margin		
F_c				
	F_u	: inelastic energy absorption		
	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
F_{rs}	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

Table 19H-5 Seismic Fragility For Control Building

Component:		Shear Walls		
Failure Mode:		Shear		
		Factor of Safety	**Median Value[1]**	**$ß_c$[1]**
	F_s	: strength margin		
F_c				
	F_u	: inelastic energy absorption		
	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
F_{rs}	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

Table 19H-6 Seismic Fragility For Reactor Pressure Vessel

Component: RPV

Failure Mode: Support

		Factor of Safety	Median Value[1]	$ß_c$[1]
	F_s	: strength margin		
F_c		:		
	F_u	inelastic energy absorption		
	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
F_{rs}	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

Table 19H-7 Seismic Fragility For Shroud Support

Component:	Shroud Support			
Failure Mode:	Buckling			
		Factor of Safety	**Median Value[1]**	**$ß_c$[1]**
	F_s	: strength margin		
F_c				
	F_u	: inelastic energy absorption		
	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
F_{rs}	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

Table 19H-8 Seismic Fragility For CRD Guide Tubes

Component:	CRD Guide Tubes			
Failure Mode:	Buckling			
		Factor of Safety	**Median Value**[1]	**$ß_c$**[1]
	F_s	: strength margin		
F_c				
	F_u	: inelastic energy absorption		
	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
F_{rs}	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

Table 19H-9 Seismic Fragility For CRD Housings

Component:	CRD Housings			
Failure Mode:	Plastic Yielding			
		Factor of Safety	**Median Value**[1]	**$ß_c$**[1]
	F_s	: strength margin		
F_c				
	F_u	: inelastic energy absorption		
	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
F_{rs}	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

Table 19H-10 Seismic Fragility For Fuel Assemblies

Component:	Fuel Assemblies			
Failure Mode:	Channel Excessive Deflection			
		Factor of Safety	**Median Value[1]**	**$ß_c$[1]**
	F_s	: strength margin		
F_c				
	F_u	: inelastic energy absorption		
	F_{sa}	: spectral shape margin		
	F_d	: damping margin		
F_{rs}	F_{ssi}	: soil-structure interaction		
	F_m	: modeling factor		
	F_{mc}	: modal combination		
	F_{ecc}	: earthquake component combination		
	F	**: overall factor**		

1 Fragility not part of DCD. Refer to SSAR.

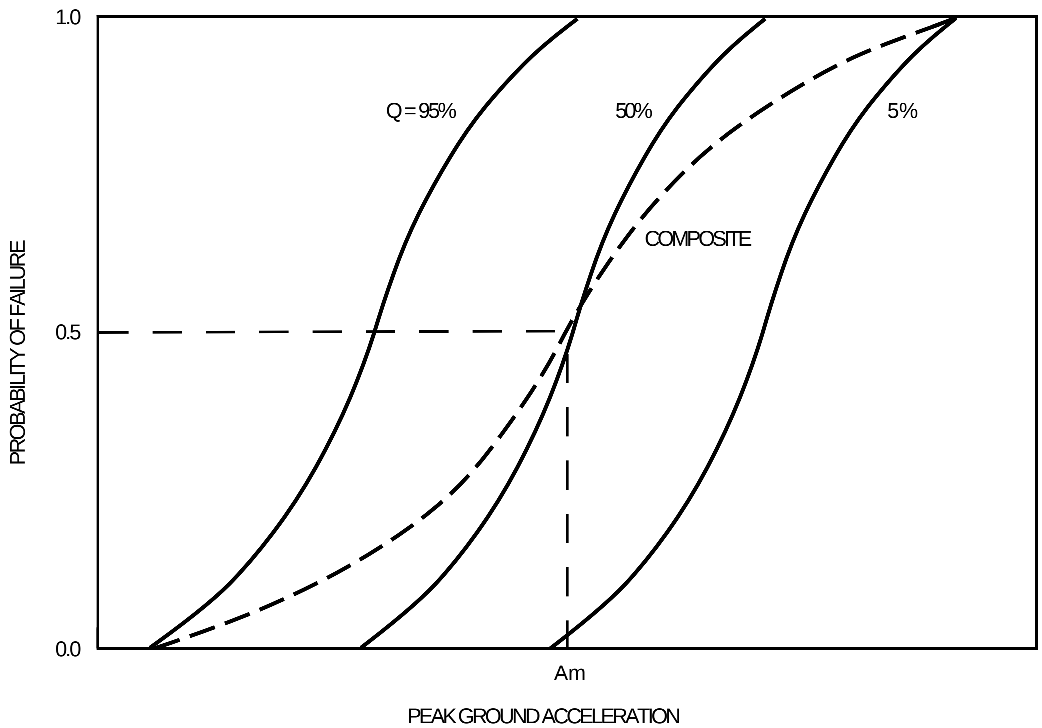


Figure 19H-1 Typical Fragility Curves

19I Seismic Margins Analysis

19I.1 Introduction

A seismic margins analysis (SMA) has been conducted for the ABWR using a modification of the Fragility Analysis method of Reference 19I-1 to calculate high confidence low probability of failure (HCLPF) accelerations for important accident sequences and accident classes. HCLPF values were calculated for components and structures using the relationship

$$HCLPF = A_m * \exp(-2.326 * \beta_c)$$

where:

A_m	=	the median peak ground acceleration corresponding to 50% failure probability,
β_c	=	the logarithmic standard deviation of the component or structure fragility.

The resulting HCLPF acceleration corresponds essentially to the 95th percent confidence level that at that acceleration the failure probability of a particular structure or component is less than 0.05 (5%). HCLPFs for accident sequences were evaluated through use of event trees, and seismic system analysis was performed with fault trees to determine HCLPFs of systems.

The seismic margins analysis evaluates the capability of the plant and equipment to withstand a large earthquake of 2 times the safe shutdown earthquake (2*SSE). In this analysis, two alternative methods were used to evaluate the seismic accident sequences—a “convolution” method and a “min-max” method.

In the convolution method, accident sequences are evaluated by combining input fragility curves according to the Boolean expression for each sequence. Seismic and random/human failure probabilities are calculated and combined (convolved) for discrete intervals of ground acceleration, and then integrated over the range of interest.

In the min-max method, input fragilities are combined by using the lowest (minimum) HCLPF value of a group of inputs operating in an OR logic, and by using the highest (maximum) HCLPF value of a group of inputs operating in an AND logic. Random/human failure probabilities are reported in combination with HCLPFs for each accident sequence.

Analysis of the effects beyond core damage (Level 2 PRA analysis) was not a part of this seismic margins analysis. However, event trees were constructed to examine the possibility of loss of containment isolation resulting in a large release given the earthquake and a resulting core damaging accident.

Because of the inclusion of a rupture disk in the ABWR design as an ultimate means of containment heat removal, and because an earthquake would not prevent rupture of the disk, failure of containment heat removal is not modeled in the seismic margins analysis. (There are no Class II sequences in the analysis.) There are two valves in line with the rupture disk; however, these valves are left in an open position, and the earthquake would not cause these valves to close.

There are several operator actions included in the seismic margins analysis. These operator actions are discussed in Subsection 19D.7.4.

19I.2 Component and Structure Fragility - A_M, $ß_C$

Component and structure fragility values have been established for selected structures and components that have been identified as potentially important to the seismic margins analysis. The fragility values used in the analysis are shown in Table 19I-1, together with the calculated component/structure HCLPFs. For more information regarding the development of these fragilities and capacities, refer to Appendix 19H.

19I.3 Event Tree Analysis

The event trees used in the ABWR Level 1 seismic margins analysis are shown on Figures 19I-1 through 19I-3. The individual paths through the event trees represent the accident sequences which are input to the HCLPF analysis. There is essentially only one seismic event tree, but it is presented on three figures representing transfers from Figure 19I-1 to Figures 19I-2 and 19I-3.

The event trees show large random failure probabilities and min-max HCLPFs for each top event. Human error probabilities are included in the random failure probabilities.

19I.3.1 Support State Event Tree

The seismic event tree of Figure 19I-1 starts with the spectrum of seismic events, considers whether or not there is a structural failure (node SI), whether or not offsite power is lost (node LOP) and continues from there. Because of the ground rules of the analysis and the relative values of seismic fragilities, loss of structural integrity results in core damage, and survival of offsite power results in successful event termination. Thus, all remaining accident sequences on Figure 19I-1 are for cases of no structural failure, but always with loss of offsite power.

The success or failure of emergency DC power (station batteries) (node DP), and the emergency AC power and/or service water (node APW) are taken into consideration in Figure 19I-1 to account for support system dependencies. Failure of all DC power results in a high-pressure core melt since all control is lost, the high-pressure systems fail, and the reactor cannot be depressurized. The condition of successful emergency DC and AC power and successful scram is indicated by the ET transfer and is described in detail in Figure 19I-2. The condition of successful emergency DC and AC power, but with failure to scram is indicated by the ATWS transfer, and is described in Figure 19I-3.

The condition of successful emergency DC and failure of emergency AC continues on Figure 19I-1. The next question is whether or not there is a failure to scram (node C). Failure to scram is considered as a Class IV core melt. With successful scram, RCIC (node UR) and firewater (node FA) are the only available means of water injection into the RPV since all AC power is lost. Since station batteries will eventually discharge resulting in loss of RCIC, or if RCIC fails, the reactor must then be depressurized (node X) to allow firewater injection. The loss of emergency DC power (station batteries) results in a high-pressure core melt as shown in Figure 19I-1.

The firewater system has a diesel driven pump and all needed valves can be accessed and operated manually. No support systems are required for firewater operation. See Subsection 19.9.21 for COL license information pertaining to housing of ACIWA equipment. The random failure probability of firewater is dominated by operator failure to initiate the system. For the upper branch, where RCIC is successful, the operator has 8 hours before the station batteries expire and RCIC trips. The human error probability (HEP) for this case is very small. For the lower branch, where RCIC fails, the operator has only 30 minutes in which to depressurize the reactor and initiate firewater injection. For this case, the HEP is moderate. In the event that the firewater diesel fails to start, the operator could make use of a fire truck, but this was not modeled.

If the RHR heat exchanger fails (node HX) due to the earthquake, it is presumed that the failure could include a pipe break that could partially drain the suppression pool into the RHR pump room. These core damage sequences are identified with a “P” (e.g., IB2-P). Fission product scrubbing would still be effective in preventing a large release. The effects of possible flooding on equipment operation beyond the RHR room were considered and found to be relatively insignificant because of the relatively high HCLPF of the heat exchangers, the ability of the operator to isolate the break, and the presence of the independent ACIWA (firewater) system.

19I.3.2 LOOP with Emergency Power and Scram Event Tree

In the event tree of Figure 19I-2 (ET transfer), there are two similar divisions depending on whether or not there is a stuck-open relief valve (node PC). If there is a stuck-open valve, the reactor will eventually depressurize causing loss of RCIC steam supply. The probability of having a stuck-open valve is based on operating experience. If both high-pressure injection systems fail, the reactor must be depressurized rapidly for low-pressure system use (LPFL -V1).

19I.3.3 ATWS Event Tree

Figure 19I-3 (ATWS transfer) represents failure to scram, and requires standby liquid control (automatic) and operator action to control reactor water level with the injection system(s) that are available. The HEP for this action is small. In this ATWS analysis, if high-pressure systems fail, core damage results. No credit is given to low-pressure injection. For an ATWS, the probability of a stuck-open SRV was conservatively increased on the basis of increased SRV activity.

19I.4 System Analysis

The fault trees used in the seismic system analysis are shown on Figures 19I-4 through 19I-15. The seismic system analysis calculates the probability of seismic failure and corresponding system HCLPFs of each of the important systems throughout the seismic ground acceleration spectrum. The system HCLPFs are then input to the event trees and combined with random system failure probabilities and human errors. The seismic fault trees contain only those components that might be subject to seismic failure. Random system failure probabilities are taken from the internal events analysis and include all other components. One of the important ground rules of the seismic margins analysis is that all like components in a system always fail together.

The reactor protection system, control rod drive system, and alternate rod insertion system were not modeled since the failure of control rods to insert is dominated by the relatively low seismic fragility of the fuel assemblies, control rod guide tubes, and housings. A seismic fault tree for reactivity control is shown on Figure 19I-13. The fuel assemblies are the most fragile component.

A seismic fault tree for the standby liquid control system is shown on Figure 19I-14. Failure of the standby liquid control system is dominated by failure of two components: the pump and boron supply tank.

Since the most fragile essential component in the plant is the ceramic insulator in the switchyard, the loss of offsite power dominates the analysis and the availability of emergency power becomes very important. The loss-of-power fault tree (Figure 19I-10) is for emergency AC power. In the loss of emergency AC power fault tree, the more fragile components are the diesel generator, transformers, motor control centers, inverter and circuit breaker. The DC power fault tree (Figure 19I-11) has two elements: batteries and cable tray.

Systems and equipment which require offsite power, such as the feedwater system and condensate injection system, are not modeled since offsite power is presumed to be not available for the core damage sequences.

Essential service water is as important as emergency power, and its loss would have much the same effect as the loss of emergency AC power. The loss-of-service-water fault tree is shown on Figure 19I-12. The more fragile components in this system are the service water pump, heat exchanger, and room air conditioning unit. The service water pump house is also included in this fault tree.

Structure failures that could contribute to seismic core damage are shown on Figure 19I-9. In this analysis, any one or more of these structural failures are conservatively presumed to result in core damage. The structures having the lowest seismic capacity are the reactor building and control building.

The remainder of the fault trees are for core cooling (Figures 19I-4 through 19I-8). The more fragile components in these systems are the pumps, heat exchangers, and the firewater supply tank. The condensate storage tank (CST) is not modeled since the ECCS systems that take suction from the CST have automatic switchover to the suppression pool if CST level is low. Valves for the switchover are included in the fault trees.

The ACIWA (firewater) system (Figure 19I-8) is designed to inject water into the reactor if the ECCS systems are not available. It is also the only means of water injection in case of a station blackout beyond 8 hours. Although firewater is not a Class 1E safety system, because of the safety function described above, the firewater diesel-driven pump, the firewater tank, valves, and related piping will have seismic margin above the SSE.

Because of the importance of RCIC in station blackout sequences, differences between the seismic RCIC fault tree and the internal events fault tree are explained below:

(1) The internal events fault tree contains basic events that would not be affected by an earthquake, e.g., test and maintenance unavailability. These events contribute to the random failure probability during the seismic event and are included in the random failure part of the seismic analysis. They are deleted from the RCIC seismic fault tree.

(2) The internal events fault tree contains common-cause failure events. These are deleted from the RCIC seismic fault tree since a basic rule of the seismic analysis is that all like components within a system fail together.

(3) The internal events RCIC fault tree contains separate events for the turbine and for the pump. The seismic fault tree uses a combined event, “turbine-driven pump”, since that is the assembly for which there is a seismic capacity.

19I.5 Accident Sequence HCLPF Analysis

Seismic fragility of a structure or component is defined as the conditional probability of its failure as a function of peak ground acceleration. The probability model adopted for each component fragility is the log-normal distribution. The density function for the component fragility, f(g), can be written

$$f(g) = \frac{1}{\sqrt{2\pi}*\beta_c*g}\exp\left(-\frac{1}{2}\left[\frac{\ln\left(\frac{g}{A_m}\right)}{\beta_c}\right]^2\right) \text{for } g > 0$$

where:

A_m	=	median capacity of the component,
β_c	=	logarithmic standard deviation of the fragility function,
g	=	peak ground acceleration.

The cumulative distribution of the component fragility, F(g), will then be

$$F(g) = \int_0^g \frac{1}{\sqrt{2\Pi}*\beta_c*g_1}\exp\left\{-\frac{1}{2}\left[\frac{\ln\left(\frac{g_1}{A_m}\right)}{\beta_c}\right]^2\right\} dg_1$$

19I.5.1 Convolution Analysis

If a system, S, (or sequence) contains two components (A, B) operating in OR logic, the failure of either component will fail the system (S = A + B), and the cumulative fragility distribution of the system is one minus the product of their complementary cumulative fragility distributions:

$$F_s(g) = 1 - (1 - F_A(g)) * (1 - F_B(g))$$

On the other hand, if two elements operate in AND logic, only the failure of both components will fail the system (S = A * B), and the cumulative fragility distribution of the system is the product of their cumulative fragility distributions:

$$F_s(g) = F_A(g) * F_B(g)$$

Using the two principles above, the distribution function of each system fragility is obtained by combining its component fragility functions based on its Boolean expression derived from the system fault tree.

Then the OR logic methodology is used to convolve the seismic and random/human failure probability of the systems. The combined cumulative fragility distribution of a system, $F_c(g)$, is the OR logic combination of the cumulative seismic fragility distribution, $F_s(g)$, and the cumulative random/human failure distribution, F_r, as follows:

$$F_c(g) = 1 - (1 - F_s(g)) * (1 - F_r)$$

Similarly, the distribution for each accident sequence is derived from the combined system fragility functions by using the Boolean expression obtained from the seismic accident sequence event trees. The fifth and fiftieth percentiles of the combined cumulative distribution of each accident sequence are used to obtain the A_m and β_c for the corresponding sequence. Then, the HCLPF of each accident sequence is obtained by using the formula presented in Subsection 19I.1 as follows:

$$HCLPF = A_m * \exp(-2.326 * \beta_c)$$

where the parameters A_m and β_c are the median capacity and logarithmic standard deviation of the lognormal distribution of the accident sequence.

19I.5.2 Min-Max Analysis

If a system, S, (or sequence) contains two components (A,B) operating in OR logic, the failure of any component will fail the system $(S = A + B)$, and the cumulative fragility distribution of the system is governed by the fragility distribution of the weakest component. This principle is applied to the system fault trees, which generally are made up of OR gates.

If two elements operate in AND logic, only the failure of both components will fail the system $(S = A * B)$, and the cumulative fragility distribution of the system is governed by the fragility distribution of the strongest component. This principle is applied to accident sequences, which are composed of ANDed elements.

Significant random/human failure probabilities are combined with HCLPFs for elements in an accident sequence as follows:

$$(HCLPF1 + RHP1) * (HCLPF2 + RHP2) =$$

$$HCLPF1 * HCLPF2,$$

HCLPF1*RHP2,

HCLPF2*RHP1,

RHP1*RHP2,

where:

HCLPF1	=	the HCLPF of one event,
RHP1	=	the random/human failure probability of that event,
HCLPF2	=	the HCLPF of a second event, and
RHP2	=	the random/human failure probability of the second event.

The resulting combinations are reduced according to min-max rules.

19I.6 Results of the Analyses

The results of the convolution analysis are shown on the event trees and in Table 19I-2 in terms of HCLPF values for the accident sequences, with and without the inclusion of random failures. The results of the convolution analysis in terms of accident classes are shown in Table 19I-3. The combination of HCLPF and random failure probabilities of accident sequences are described in Table 19I-4.

19I.7 Containment Isolation and Bypass Analysis

In the seismic margins analysis there were no cutsets leading to core damage with low HCLPF values. A supplemental analysis was conducted to evaluate the HCLPF values for containment isolation for events that could cause containment bypass as a result of an earthquake, with potential for large releases to the environment.

Based on the results of the bypass analysis discussed in Subsection 19E.2.3.3 and shown on Figure 19I-16 through 19I-25, the events selected for evaluation in this analysis are:

(1) Main steam lines (Figure 19E.2-19a)

(2) Feedwater or SLC injection lines (Figure 19E.2-19b)

(3) Reactor instrument, CUW instrument, LDS instrument/sample or containment atmosphere monitoring lines (Figures 19E.2-19d, 19E.2-19e, and 19E.2-19f, respectively)

(4) RCIC steam supply or CUW suction lines (Figure 19E.2-19e)

(5) Post accident sampling lines (Figure 19E.2-19j)

(6) Drywell sump drain line (Figure 19E.2-19j)

(7) SRV discharge lines (Figure 19E.2-19k)

(8) ECCS lines (Figure 19E.2-19c)

(9) Drywell inerting/purge lines (Figure 19E.2-19i)

(10) Wetwell/drywell vacuum breaker lines (Figure 19E.2-19g)

The bypass paths for atmospheric control system crosstie lines (Figure 19E.2-19h) require inadvertent opening of two normally closed motor operated valves. Since the seismic analysis does not consider a fail-open mode for normally closed valves, these bypass paths are not included in the analysis.

In the bypass analysis of Subsection 19E.2.3.3, several potential bypass pathways were excluded from detailed analysis on the basis of various reasons. The reasons are discussed in Subsection 19E.2.3.3.2 and Table 19E.2-1. These reasons were reviewed to determine whether they remain valid in regard to seismic events. All but one of the reasons are based on configuration details that would not be affected by an earthquake. RHR wetwell and drywell spray lines were excluded on the basis that the pipes are designed for higher internal pressures than will be seen in actual operation and would thus have a very low probability of breaking. In this case, the seismic event could increase the probability of a break in these lines. However, these pipes have very high seismic capacity with very low probability of breaking due to a seismic event.

An event tree was constructed for each of the above events. These event trees are shown on Figures 19I-16 through 19I-25. All event trees start with the earthquake as the initiating event followed by a core-damaging accident. If there is no core damage there is no large release. The HCLPF and random failure probability are shown for each branch point, and the sequence HCLPFs using convolution and min-max methods are also shown on the figures.

Figure 19I-16 is for suppression pool bypass via main steam lines. Following the earthquake and accident, the question is asked whether or not there is a break in a main steam line outside containment. If there is a break, the question is asked whether or not at least one MSIV in each steam line closes to isolate the break. For the case where there is no break, there could still be a bypass release to the main condenser if a turbine bypass valve is open—unless the MSIVs are closed to isolate the break.

Figure 19I-17 is an event tree for bypass via feedwater or standby liquid control lines. These lines inject into the RPV and are protected from reverse flow by redundant check valves. These check valves provide isolation of upstream breaks provided that one of the valves closes in the line with the break.

Figure 19I-18 is for bypass via reactor instrument, CUW instrument, LDS instrument, LDS sample or containment atmosphere monitoring lines. These lines are also protected by check valves, a single valve in each line.

Figure 19I-19 is for bypass via either the RCIC steam supply line or the CUW suction line. Both of these lines are protected by motor operated isolation valves which require power. Since offsite power is lost due to the earthquake, emergency power is required.

Figure 19I-20 is for bypass via the post accident sampling lines. These lines are also isolated by motor operated valves.

Figure 19I-21 is for bypass via the drywell sump drain line. This line is protected by a motor operated isolation valve and a check valve.

Figure 19I-22 is for bypass via the SRV discharge lines. If there is a break in an SRV discharge line during a core-damaging accident, and that SRV is open, a bypass pathway will exist. In this analysis, it is assumed that the SRV will be open during the accident.

Figure 19I-23 is for bypass via any of the ECCS lines. The lines of concern are the HPCF and LPFL warm-up and discharge lines. These lines are protected by motor operated isolation valves and check valves.

Figure 19I-24 is for bypass via drywell inerting/purge lines. These lines are protected by air operated valves.

Figure 19I-25 is for bypass via wetwell/drywell vacuum breaker lines. It requires an inadvertent opening of a vacuum breaker (check valve) to initiate a bypass during a severe accident.

19I.8 References

19I-1 R.P. Kennedy, et al., “Assessment of Seismic Margin Calculation Methods”, NUREG/CR-5270, Lawrence Livermore National Laboratory, March 1989.

Table 19I-1 ABWR Systems and Components/Structures Fragilities

System/Component	MED_CP (A_M)*	LOG_STD (β_C)*	HCLPF† (in g)
1. Plant Ess. Structures (SI)			
- Reactor Building			
- Containment			
- RPV Pedestal			
- Control Building			
- Reactor Pressure Vessel Support			
2. Support Systems (PW)			
a. AC Power (ACP)			
- Diesel Generator			
- Transformer (480 V AC)			
- Motor Control Center			
- Cable Tray			
- Circuit Breaker			
- Inverter			
b. Service Water (SW)			
- Pump (Motor Driven)			
- Heat Exchanger			
- Valve (Motor Operated)			
- Check Valve			
- Room Air Cond. Unit			
- Piping			
- SW Pump House			
- AC Ducting			
c. DC Power (DCP)			
- Batteries			
- Cable Tray			
3. High-Press Core Flooder (UH)			
- Pump (Motor Driven)			
- Injection Valve (Motor Op)			
- HPCF Piping			
- Check Valve			
- Switchover Valve (MO)			

Table 19I-1 ABWR Systems and Components/Structures Fragilities (Continued)

System/Component	MED_CP (A_M)*	LOG_STD (β_C)*	HCLPF† (in g)
4. Reactor Core Is. Cooling (UR)			
- Pump (Turbine Driven)			
- Steam Sup. Valve (MO)			
- Discharge Valve (MO)			
- Min Flow Valve (MO)			
- Check Valve			
- RCIC Piping			
- Switchover Valve (MO)			
5. Low-Press Core Flooder (V1)			
- Pump (Motor Driven)			
- Check Valve			
- Injection Valve (MO)			
- Discharge Valve (MO)			
- LPCF Piping			
6. RHR Heat Exchanger (HX)			
- Heat Exchanger			
7. Reactivity Control Sys. (C)			
- Fuel Assemblies			
- CRD Guide Tube			
- CRD Housing			
- Shroud Support			
- Hydraulic Control Unit			
8. SRVs Close (PC, PC1)			
- Safety Relief Valve			
9. Depressurization (X)			
- Safety Relief Valve			
10. Level & Press. Control (LPL)			
- Safety Relief Valve			
11. Inhibit ADS (PA)			
- Safety Relief Valve			

Table 19I-1 ABWR Systems and Components/Structures Fragilities (Continued)

System/Component	MED_CP (A_M)*	LOG_STD (β_C)*	HCLPF† (in g)
12. Standby Liq. Cont. Sys. (C4)			
- SLC Tank			
- SLC Pump			
- Valve (Motor Operated)			
- SLC Piping			
13. Firewater System (FW)			
- FW Tank‡			
- Pump (Diesel Driven)			
- Injection Valve (Manual)			
- FW Piping			
- Valve (Manual)			

* Fragilites and HCLPF values are not part of DCD. (Refer to SSAR).

† $\mathrm{HCLPF} = A_m \times \exp(-2.326x\beta_c)$

‡ Firewater tank may be designed and built to a lower capacity if provision is made for a pumper truck housed in such a manner that it will survive a SSE and a hose that will reach an alternate water supply.

Table 19I-2 Seismic Margins for ABWR Accident Sequences (Convolution Method) (Not part of DCD Refer to SSAR).

Table 19I-3 Seismic Margins for ABWR Accident Classes (Convolution Method)

	With Random Failure			Without Random Failure		
Accident Class	HCLPF* (in g)	MED_CAP* (A_m)	LOG_STD* (β_c)	HCLPF* (in g)	MED_CAP* (A_m)	LOG_STD* (β_c)
IA						
IB2						
IC						
ID						
IE						
IV						
IA-P, IE-P						

* Fragilities and HCLPF values are not part of DCD (Refer to SSAR).

Table 19I-4 HCLPF Derivation for the ABWR Accident Sequences (MIN-MAX Method) is not part of DCD (Refer to SSAR)

Figures 19I-1 through 19I-25 are not part of DCD (Refer to SSAR).

19J Not Used

19K PRA-Based Reliability and Maintenance

19K.1 Introduction

In this appendix, the results of the PRA are reviewed to determine the appropriate reliability and maintenance actions that should be considered throughout the life of an ABWR plant so that the PRA remains an adequate basis for quantifying plant safety. These actions comprise a part of the plant's reliability assurance program (RAP).

Paragraph 8.8, "Maintenance and Surveillance", of the ABWR Licensing Review Bases (Reference 19K-1), reads in part, "GE is to provide in the SSAR the reliability and maintenance criteria that a future applicant must satisfy to ensure that the safety of the as-built facility will continue to be accurately described by the certified design." This appendix provides the PRA based reliability and maintenance actions which should be considered for incorporation into the future applicant's (i.e., the applicant referencing the ABWR design) operating and maintenance procedures required by Standard Review Plan (SRP) Subsection 13.5.2. As indicated in Table 1.8-19, SRP 13.5.2 is an interface requirement to be provided by the utility applicant referencing the ABWR design.

19K.2 General Approach

To determine the appropriate reliability and maintenance-related activities that should be considered to assure that plant safety is maintained as operation proceeds, results of PRA and other analyses were reviewed. The objective of the review was to determine the relative importance of prevention and mitigation features of the ABWR in satisfying the key PRA goals related to core damage frequency (CDF) and frequency of offsite release. Also considered were the initiating events that had significant impact on CDF. From this review (Subsection 19K.3), the most important plant features were identified.

The PRA was further reviewed (Subsections 19K.4 through 19K.10) for other important features, the failure of which was not addressed directly in Subsection 19K.3, to supplement the above list. Finally (Subsection 19K.11), the individual features identified in Subsections 19K.3 through 19K.10 were reviewed to determine appropriate maintenance and surveillance actions.

19K.3 Determination of "Important Structures, Systems and Components" for Level 1 Analysis

To determine which plant structures, systems and components (SSCs) are the most important with respect to CDF, the Level 1 analysis results were analyzed. The SSCs were listed in order of Fussell-Vesely (FV) importance, or the percent of cutsets that contribute to the CDF, as calculated by the CAFTA code. A second criterion for selecting SSCs was to consider those SSCs with high "risk achievement worth", or the increase in CDF if that SSC always fails. The 21 SSCs of greatest importance, in that they had modest FV importance are shown in Table 19K-1. Five additional SSCs with modest values of risk achievement worth were considered. Not shown in Table 19K-1 are several human error contributions. Significant human errors are addressed in Subsection 19D.7.

The 26 SSCs in Table 19K-1 were further evaluated to eliminate those with a combination of low values for both FV importance and risk achievement worth. The five SSCs meeting this criterion are so indicated. However, one of those five is retained because of its designation as a “critical task” in the human factors evaluation of Subsection 18E.2. The other four are not considered further in this subsection.

The remaining 22 designated SSCs of Table 19K-1 should be included with important SSCs being considered for periodic testing and/or preventive maintenance (PM) as part of the Reliability Assurance Program (RAP) of the plant owner/operator. The reliability and maintenance actions suggested for the listed SSCs are identified in Subsection 19K.11.

A second table, Table 19K-2, was prepared to show those SSCs with small to moderate values of risk achievement worth. These SSCs all have very low Fussell-Vesely importance, indicating a low probability of failure. However, if they fail, the impact on CDF is not negligible. Most of these SSCs have the same risk achievement worth because their failure would result in failure of the RCIC system to perform its function.

Initiating events that are significant contributors to CDF in the Level 1 analysis are listed in Table 19K-3. There are five such events which are shown. The three most significant events, accounting for more than one-half of the CDF, are all station blackout events. The next two events, contributing small fractions of CDF are isolation/loss of feedwater and manual reactor shutdown. All other initiating events contribute small amounts to CDF.

The components within the control of the COL applicant that are of most significance to limiting the frequency of station blackout are the diesel generators and the combustion turbine. The COL applicant should assure that maintenance and test activities for these components are appropriate to assure high reliability.

Systems that are most important to limiting the frequency of isolation/loss of feedwater are the Feedwater and Feedwater Control (FWC) Systems. The FWC System is triply redundant, having digital logic with self-checking. The automatic checking of the FWC System assures that its reliability remains high throughout operation. The COL applicant should assure that maintenance and test activities for risk-significant components in the FW System, the FW pumps and motors, are appropriate to assure high reliability.

Unplanned manual reactor shutdowns occur with a relatively short time for preparation, in contrast with a planned shutdown. To assure that the unplanned shutdowns will not cause undue risk to the plant, the training procedures should include adequate training, including simulator exercises, for such events so the operating crews can respond to plant conditions during such shutdowns on short notice.

The RAP activities for important SSCs identified by consideration of initiating events are included in Table 19K-4.

The relative importance of some ABWR features is not established by the Level 1 analysis described above because some important SSCs are not treated in the Level 1 calculation. To identify other important SSCs, the Level 2, seismic, fire, flood and shutdown analyses results were carefully reviewed by knowledgeable engineers who identified additional SSCs for the RAP. The important SSCs identified in these other studies are given in Subsections 19K.4 through 19K.10, and RAP activities are in Subsection 19K.11.

19K.4 Determination of "Important Structures, Systems and Components" for Level 2 Analysis

The Level 2 analysis evaluates the offsite release of fission products following core damage. Those analyses related to the consequences of core damage were reviewed, including source term sensitivity studies, deterministic analysis of plant performance, and containment event trees. Those systems which would be important with regard to mitigating a core damage event were considered as potential risk-significant SSCs. The following features were identified:

(1) The Automatic Depressurization System (ADS)

The ADS depressurizes the RPV so that the low pressure systems can inject water. Even if no water injection is available, the depressurization via one safety/relief valve (SRV) eliminates the potential for direct containment heating in event of RPV failure. The SRVs are important SSCs for the ADS since they are the components that function to release steam to reduce RPV pressure.

(2) The AC-independent Water Addition (ACIWA) System

The ACIWA System has two major benefits. First, it can inject water into the RPV to prevent core damage or facilitate in-vessel recovery. Second, it helps protect the containment by flooding the lower drywell (diverse from LDF) to cool corium in event of core melt and vessel failure. The ACIWA System can also be used to reduce high drywell temperature when operated in the drywell spray mode.

Also, for sequences with loss of containment heat removal, the ACIWA System adds thermal mass to the containment, significantly delaying the time of rupture disk opening. The important SSCs for the ACIWA System are the valves, the diesel-driven pump, and the onsite fire truck as they provide for the addition of water to the core and/or drywell.

(3) The Lower Drywell Flooder (LDF)

The LDF System was selected because it is important in providing cooling for corium released from the reactor vessel and in scrubbing fission products released from the corium in the event all the automatic and manual systems fail to inject water. The LDF fusible plug valves are important SSCs for the LDF System since they provide for flooding of the lower drywell.

(4) The Containment Overpressure Protection System (COPS)

The COPS is important since it prevents containment failure and assures a fission product release path through the suppression pool. This serves to limit the potential offsite dose after a core damage event. Sequences which result in slow pressurization will lead to a COPS operation, as opposed to the drywell failure. Since the suppression pool scrubs fission products before they enter the wetwell air space, this results in a much lower source term than does the case of a drywell head failure.

The COPS will also reduce the potential for a Class II sequence to lead to core damage. The predominant mechanism for core damage in Class II sequences is failure of containment or reactor building structures causing damage to long term heat removal equipment. Operation of the COPS directs the gas flow to the stack, preventing damage to the equipment. The COPS SSCs identified by the analysis are the rupture disks, which prevent containment failure and limit offsite doses after core damage, the isolation valves, and the flow lines.

(5) The RHR System

The RHR System is a primary source of decay heat removal. Decay heat removal is necessary to prevent fission product release from the containment in the unlikely event of a severe accident. Also, the drywell spray function of the RHR is an important feature in limiting the consequences of the Level 2 analysis. The technical specifications and valve and pump inservice testing (Table 3.9-8) requirements for the RHR System were reviewed and it was concluded that except for a maintenance requirement on the RHR Non-safety Related Valve, no additional reliability and maintenance actions are needed in the RAP for the RHR System.

The RAP activities for important SSCs identified by this Level 2 analysis are given in Table 19K-4.

19K.5 Determination of “Important Structures, Systems and Components” for Seismic Analysis

The seismic analysis considers the potential for core damage from plant damage resulting from a seismic event. The results of the seismic analysis identified key features by consideration of those SSCs important to reactor shutdown or to decay heat removal which could potentially be damaged by seismic action.

The seismic margins analysis calculated high confidence, low probability of failure (HCLPF) accelerations for important accident sequences and classes of accidents. The analysis showed that all SSCs in the analysis have HCLPF significantly greater than that of the safe shutdown earthquake (SSE). Because an important failure mode for beyond design bases earthquakes is the failure of the RHR heat exchanger in such a manner as to drain the suppression pool, the RHR heat exchanger was assigned a reasonably high HCLPF in the ABWR PRA-based seismic margins analysis.

The two methods that were used to identify important SSCs from the standpoint of seismic analysis are the following:

(1) Identification of the SSCs whose failure would provide the shortest path to core melt in terms of the number of failures required, and comparison of the seismic capacities of those SSCs.

(2) Identification of the most sensitive SSCs in terms of their effect on accident sequence and accident class HCLPFs resulting from variation of component seismic capacities.

The primary containment and the Reactor Building are the Category I structures in the design certification scope with the lowest values of HCLPF, but since both have high values of HCLPF no special RAP activities are deemed necessary for these structures. Other SSCs identified by the seismic analysis as being important are as follows:

- The diesel generators, 480 VAC transformers, motor control centers and circuit breakers of the emergency AC Power System
- The batteries and cable trays of the DC Power System
- The heat exchangers of the Residual Heat Removal System
- The pumps, pump house and air conditioners of the Service Water System
- The SLC tank, valves, and piping and the motor driven pumps of the Standby Liquid Control System

- The valves, piping, and diesel-driven pump of the Fire Water System
- The discharge lines of the SRVs of the Nuclear Boiler System

The RAP activities for important SSCs identified by this seismic analysis are given in Table 19K-4.

19K.6 Determination of “Important Structures, Systems and Components” for Fire Analysis

The fire analysis considers the potential for core damage from plant damage resulting from a fire. The important SSCs identified by this analysis are the room fire barriers, which prevent the fire from spreading to other rooms, the Smoke Removal System, which maintains pressure differentials to exhaust smoke rather than allow it to reach other areas, and the remote shutdown panel and control which are needed following a fire in the control room or HVAC failure in the control room.

The RAP activities for important SSCs identified by this fire analysis are given in Table 19K-4.

19K.7 Determination of “Important Structures, Systems and Components” for Flood Analysis

The flood analysis considers the potential for core damage from plant damage resulting from a flood. The important SSCs identified by this analysis are the ECCS room, RCW rooms and control and reactor building external water tight doors, which prevent water from flowing into rooms other than the one with the leak; isolation valves on the Reactor Service Water System and anti-siphon capability, which limit the amount of water spilled into the control building; circuit breakers that will trip RSW pumps, which also limits the amount of water spilled into the control building; isolation valves in the Circulating Water System (CWS); circuit breakers that will trip CWS pumps; level switches in the turbine building condenser pit and control building RCW rooms; sump pump operation; overfill lines in reactor building sumps on floor BIF; and room drain lines.

The RAP activities for important SSCs identified by this flood analysis are given in Table 19K-4.

19K.8 Determination of “Important Structures, Systems and Components” for Shutdown Analysis

The shutdown analysis considers the potential for core damage during shutdown. Potential core damage during shutdown arises when the RHR System is lost. The important SSCs identified by this analysis are the ADS System, the RHR System for shutdown cooling and in the low pressure flooder (LPFL) mode, the High Pressure Core Flooder (HPCF) System, the AC-independent Water Addition (ACIWA) System, and the Control Rod Drive (CRD) System. Also important are the support systems, AC power and DC power. The important components are SRVs of the ADS System, valves and pumps of the RHR System and of the HPCF, ACIWA and CRD Systems.

The RAP activities for important SSCs identified by this shutdown analysis are given in Table 19K-4.

19K.9 Identification of Important Systems with Redundant Trains

Several plant systems have multiple trains of which only one is required to operate to perform the system safety function, the other trains providing redundancy. Because of this redundancy, components of the systems may not show up in a listing of high importance components. However, it is possible that operation or maintenance activities related to these systems could introduce some common cause failures which could affect all similar trains of a given system and, thereby, render all trains of such systems incapable of performing their safety functions. Engineering judgment was used to identify the multiple train systems having important safety functions that should be checked in addition to any identified component tests or maintenance. The systems selected are the RHR System in the shutdown cooling and the low pressure flooder (LPFL) mode, the High Pressure Core Flooder (HPCF) System, the Reactor Water Cleanup (CUW) System, the Reactor Service Water (RSW) System, and the AC Electrical System.

A single train of each of these systems should be designated for RAP by the COL applicant and the train should be given a walkdown inspection every refueling outage. The inspection should verify that system equipment is being operated and maintained properly so that there is no reason to suspect that other trains of the same system have problems that would preclude the system from performing its safety functions. The RAP activities for trains of systems identified by this analysis are given in Table 19K-4.

19K.10 Identification of Important Capabilities Outside the Control Room

Most safety-related actions by plant operators are conducted from inside the control room. However, in some sequences it is necessary for the operators to take appropriate action from stations outside the control room. Engineering judgment was used to identify activities that the operators should be capable of performing outside the control room, during internal flood, during reactor shutdown, or when the control room is inaccessible, such as in event of a fire.

The identified activities outside the control room are:

(1) Execution of the emergency operation procedures for operating the remote shutdown panels

(2) Manual operation of the RCIC System from outside the control room

(3) Closing water tight doors that are open (if there is flooding in the intact ECCS division) before opening doors to attempt corrective action

(4) Manual lineup of the combustion turbine generator and emergency diesel generators to non-safety-related buses

(5) Manual alignment of the AC-independent Water Addition System

(6) Manual bypass of the regenerative heat exchanger in the Reactor Water Cleanup System

(7) Connection of the diesel fire truck to the AC-independent Water Addition System after a seismic event

The RAP activities identified by these considerations are given in Table 19K-4.

19K.11 Reliability and Maintenance Actions

The individual SSCs identified as being “important” in Subsections 19K.3 through 19K.10 were reviewed to determine the appropriate reliability and maintenance actions. These actions are defined in this subsection.

The important SSCs are tabulated in Table 19K-4, showing the failure mode or cause, the recommended maintenance, the test or maintenance intervals and the basis for intervals, and the unavailability or failure rate. Where several components in one system are identified, such as for the RCIC, the ACIWA, and COPS, only the system unavailability is given. If the owner/operator cannot demonstrate each component meeting its unavailability assumption, the PRA assumptions will still be valid if the system unavailability assumption is met.

19K.11.1 Component Inspections and Maintenance

The two component types with the highest FV importances in the Level 1 analysis are the combustion turbine generator and the emergency diesel generators. Maintenance activities to assure high reliability of these components are discussed in 19K.11.10 and 19K.11.11.

The system of greatest FV importance is the RCIC System, which has been assigned a small value of unavailability for test and maintenance. The amount of time the RCIC System is unavailable because of test and maintenance should be monitored to assure that it remains within the specified assumption annually. Sensitivity studies of increased SSC unavailabilities showed that an increase in RCIC unavailability would cause the greatest increase in estimated core damage frequency of any SSC. The RCIC System was also found to be the most sensitive system to increased outage time assumptions. The highest contributor to uncertainties in the CDF as well as the CDF estimate was RCIC test and maintenance.

Multiplexers which provide multiple signals to several systems are identified by the Level 1 analysis as high importance components. Safety system multiplexers have a built-in self test that checks circuits frequently. In addition, one of four multiplexers can be bypassed and tested during plant operation without loss of system function. Such tests provide a complete simulation of the multiplexer signals, more than included in the self-test. During plant outages more detailed multiplexer tests are possible, including a complete system test and identification of signal errors. These tests will include verification that the remote multiplexing units function properly. Multiplexer tests that are suggested as part of the RAP are given in Table 19K-4.

The turbine of the RCIC System is an important component, as identified in Table 19K-1. Periodic startup and operation of the RCIC turbine is one way to monitor this turbine, and less frequent turbine inspection and refurbishment are also recommended. The RCIC pump is tested at the same time by measurement of speed, flow rate, differential pressure, and vibration. The turbine lube oil pump operation and many of the RCIC valves are also tested when the turbine testing is done. These RAP activities are included in Table 19K-4.

Trip logic units (TLUs) for the Reactor Protection System (RPS) represent another high importance component. Functional tests of these TLUs are performed at frequent intervals by the online, self-test feature of ABWR solid-state logic. Additional offline, semi-automatic, end-to-end (sensor input to trip actuator) testing of TLUs, which exercises the safety system logic and control logic processes, is important because it allows the detection of failures not sensed by the online system. The TLU tests that are suggested as part of the RAP are given in Table 19K-4.

Station batteries receive periodic checks in accordance with plant technical specifications. These checks will be adequate to assure that the batteries will have the reliability assumed in safety analyses.

For the normally closed, fail closed (NCFC) injection valves, the steam supply valves and the bypass valves of the RCIC System, which normally are not required to operate during plant operation, a quarterly full stroke test is judged to be appropriate for the RAP. Such tests are in compliance with ASME Code requirements for valves in nuclear plants. Detailed disassembly, inspection and refurbishment of valves would be done less frequently. The normally open, fail open (NOFO) bypass valves should be considered for similar tests. Suggested RAP activities and frequencies, and the basis for each suggested activity, are shown in Table 19K-4 for identified failure modes.

The HPCF maintenance valve is normally locked open, and its failure mode is being left closed following maintenance. To prevent this human error from occurring, administrative controls should require independent verification of the valve position following maintenance, positive control of the key to the valve lock, and control room verification of the valve position prior to startup. The RAP activities are in Table 19K-4.

The RCIC isolation signal logic should have a logic functional test every three months to assure it is functioning properly as shown in Table 19K-4.

Reliability of offsite power sources cannot be completely controlled by the plant. However, to assure that plant equipment does not contribute to power losses, inspection of switchyard equipment should be performed with a frequency of at least once every six months in accordance with site administrative procedures. Such inspections should include confirmation of secure structural mounting of equipment, physical condition of insulators and other supporting apparatus, and visual inspection of transformers and other oil filled equipment for oil leaks. Infrared thermography should be used to detect hot spots on electrical equipment and connections. All supports and supporting structures should be examined for structural integrity. In addition, suggested RAP activities given in Table 19K-4 for protective relay testing and for control power source components are recommended.

Common-cause miscalibration of RHR flow meters, and of Level 8 sensors, and common-cause failure (CCF) of digital trip modules (DTMs), and of Level 2 sensors, will have acceptable probabilities if adequate administrative controls are exercised. Calibration procedures for RHR flow meters and for Level 8 sensors should include notes about the safety importance of these instruments. Historical trend analysis should be performed for Level 2 sensors at each calibration. The procedure for testing DTMs should include a warning about their importance to safety. Suggested RAP activities are given in Table 19K-4.

The CCF of safety relief valves (SRVs) can be kept to an acceptably low probability if the SRVs receive the appropriate inservice inspection, if identified problems receive root cause analysis and correction, and if the configuration and qualified life of the valves at the site (or elsewhere) is maintained correctly, including consideration for aging and wear of parts. The SRV control panel can also be tested, separate from valve operation, to assure that it works properly. An inservice check to detect for valve leakage that can lead to setpoint drift is the temperature alarm on the tail pipe. The inservice inspection of SRVs is included in Table 19K-4 for RAP.

Isolation check valves of the NBS are leak tested at refueling outages, and that test demonstrates that the valves move from open to closed. Subsequent plant operation of the feedwater system opens the valves, giving assurance that they have ability to open. The NBS manual isolation valve has a stroke test at each refueling outage to assure that it can function. Testable check valves of the RCIC System can also be checked at each refueling to assure that they would function properly if conditions required a change in position. These valve tests are included in Table 19K-4.

19K.11.2 RCIC System Testing

The Level 1 analysis identified the Reactor Core Isolation Cooling (RCIC) System as one whose failures contribute substantially to CDF. Failure of RCIC to start or failure to continue operation after start are failure modes that are identified as significant. To provide assurance that the RCIC operation will be reliable, it is suggested that the system be started and operated long enough to demonstrate stable operation at least once every three months. The flow rate of RCIC should be measured to verify that it meets design requirements for injection into the RPV. Quarterly tests are with flow to the suppression pool. The RCIC System test will accomplish many of the RCIC turbine, pump and valve tests and will demonstrate that the Division 1 distribution panel is functioning. Components of RCIC that have been identified as significant, including many valves and instruments, are included in Table 19K-4 with identified failure modes and suggested RAP activities.

19K.11.3 Depressurization

The ADS technical specifications were reviewed, and it was concluded that no additional reliability and maintenance actions are needed. Testing of ADS System SRVs is included in Table 19K-4 with the other RAP activities.

19K.11.4 Lower Drywell Flooder (LDF)

In order to assure a dry cavity at the time of vessel failure, it is important that there be negligible probability of premature or spurious actuation of the passive flooder valves at temperatures less than 533 K (500°F) or under differential pressures associated with reactor blowdown and pool hydrodynamic loads.

Activities suggested for RAP are given in Table 19K-4 and discussed below.

(1) The ten fusible plug valve flanges and outlets should be inspected every refueling outage to assure there is no leakage.

(2) Two of ten fusible plug valves should be removed, inspected and their temperature setpoints tested every two refueling outages. (See testing and inspection requirements, Subsection 9.5.12.4.)

19K.11.5 AC-Independent Water Addition (Firewater) System

Inspection and testing of this system should be included in RAP. However, because of the importance of manual alignment, lining up the firewater should be specifically included in the training programs to assure that the system benefits are obtained. Specific procedures are required to be developed by the COL applicant to align the ACIWA System for vessel injection or drywell spray. See Subsection 19.9.7.

The strategy discussed below is recommended to test key components to assure that pumps and valves are operable and that there is no significant flow blockage in the flow paths from the Fire Water System to the reactor pressure vessel and to the drywell spray. Component testing is included in Table 19K-4.

(1) Onsite fire truck (pumper) maintenance should be conducted in accordance with the utility's normal fire protection maintenance procedures. A site service test of fire truck performance should be performed annually or after any major repairs in conformance with Chapter 11 of NFPA 1901, "Standard on Automotive Fire Apparatus". These tests should demonstrate that the pumper/engine combination is capable of meeting the performance requirements of the original certification or acceptance tests. Fire truck reliability for supporting the water injection function is assumed to be high. A satisfactory service test should consist of pumping water to the ground or back to the suction source as follows:

   (a) Twenty minutes of pumping 100% rated capacity, preferably at draft, at 1.034 MPa net pump pressure

   (b) Ten minutes of pumping 70% rated capacity at 1.379 MPa net pump pressure

   (c) Ten minutes of pumping 50% rated capacity at 1.724 MPa net pump pressure

   Engine speed should be recorded for each condition. A "spurt" test need not be conducted, but if care is taken to ensure that the pump does not cavitate, running the pumper with wide open throttle at 1.138 MPa net pump pressure may give a good indication of engine condition.

(2) As a part of the normal testing required by the utility's fire protection procedures, the following tests should be considered:

   (a) Once every two refueling outages or every four years (which ever is most convenient) the fire truck should be used to pressurize the fire protection system and test the flow capacity. Suction should be from both fire protection tanks and the ultimate heat sink water supply.

   (b) Once every two refueling outages or every four years the flow capacity of both the AC-driven and the direct diesel-driven fire pumps should be tested. This flow test can be alternated with the fire truck flow test (2a above). The diesel-driven fire water pump is assumed to have high reliability for supporting the water injection function.

(3) Once every two years the RHR non-safety-related valve (E11-F103C of Figure 5.4-10, Sheet 7) which must operate to provide flow to the vessel, or to the drywell spray or wetwell spray, should be manually opened and closed. Safety-related valves E11-F101C and E11-F102C are exercised every three months as part of the valve inservice testing program.

(4) Once every four years the AC-independent Water Addition (ACIWA) System flow and flow monitoring instrumentation from the fire protection system (FPS) to the RHR main loop should be tested. This can be accomplished during a reactor shutdown by initially isolating and closing off the branch lines of the RHR main loop C (however, the heat exchanger throttle valve E11-F004C remains open) and stopping both pumps, C001C and C002C. After ACIWA valves E11-F101C and E11-F102C are opened to apply the FPS pressure to the RHR main loop, the shutoff head pressure should be verified. With the RHR main loop closed off, no flow should occur. Then for a short time period, the flushing drain to the radwaste using valves E11-F029C and E11-F030C, Figure 5.4-10, Sheet 6, can be opened. The resulting flow can be measured with flow meter E11-FE012B, Figure 5.4-10, Sheet. 4.

Throttling valve E11-F030C can be used to turn the flow on and off and limit the flow to the desired rate and duration. The flow duration should be minimized to reduce the load to radwaste. The test should be repeated first with valve E11-F101C closed, then with the fire truck hose connection and valves E11-F101C and E11-F103C opened, Figure 5.4-10, Sheet 7.

(5) Once every five years all fire protection and RHR piping which forms the AC-Independent Water Addition System should be tested to ensure that it is structurally intact and properly supported.

(6) Seismic-related inspections listed in Subsection 19K.11.7 should be done.

19K.11.6 Containment Overpressure Protection System (COPS)

The COPS is identified in Subsection 19K.4 as important to limiting fission product release. Suggested system component testing as part of RAP is identified in Table 19K-4. Also, system flow testing and special operator training should be considered for inclusion in the RAP.

(1) Air-operated valves (AOVs) in series with rupture disks should be maintained in the same manner as containment isolation valves. It is suggested that, during preoperational testing and during each R/M outage, each valve be exercised and proper open and closed local and control room indications be checked. Any position other than full open should alarm in the control room.

After valves are returned to the open position, indication should be verified locally and in the main control room. These tests are included in Table 19K-4.

(2) Rupture disks should be maintained as required by the ASME code. The rupture disk manufacturer should perform the necessary tests to certify that the rupture disks will open at a pressure within 5% of the rated value. Every five years, the disks should be tested and replaced. These tests are included in Table 19K-4.

(3) A flow test should be conducted every five years to assure that there are no obstructions in the pressure relief path.

(4) Special training on operator actions following rupture disk opening should be included in the plant training program.

19K.11.7 Seismic-Related Inspections

The seismic capability of the following equipment is identified (Subsection 19K.5) as risk-significant: emergency diesel generators, 480 VAC transformers, motor control centers and circuit breakers of the AC Power System; batteries and cable trays of the DC Power System; the SLCS tank, valves and piping and motor driven pumps of the Standby Liquid Control System; the Service Water System pumps, pump house and air conditioner; the heat exchanger of the RHR System; the valves, piping and diesel-driven pump of the Fire Water System, and the discharge lines of the SRVs. For this equipment, the seismic related inspections detailed in Subsection 19H.5 should be conducted once every 10 years or after any earthquake equal to or greater than that corresponding to the cumulative absolute velocity (CAV) shutdown threshold.

19K.11.8 Plant Structures

No maintenance activities other than those already associated with the inservice surveillance of the seismic instruments defined in Subsection 3.7.4.5 are needed for seismic events. The seismic instrumentation program (Subsection 3.7.4) is designed to provide information on the input ground motion and resultant responses of representative Category I structures and equipment in the event an earthquake occurs sufficient to activate the seismic instrumentation. If the earthquake exceeds that corresponding to the CAV shutdown threshold, the plant is shut down, manually if necessary, and a detailed post-earthquake evaluation is undertaken. When it is determined that plant structures and equipment were not damaged, the plant can be safely re-started on the basis of seismic considerations.

19K.11.9 Hydraulic Control Units and Control Rod Drives

The technical specifications associated with the hydraulic control units and control rod drives were reviewed. It was concluded that no additional reliability and maintenance actions are needed beyond those in technical specifications.

19K.11.10 Emergency Diesel Generators

Maintenance for the emergency diesel generators is expected to be performed in accordance with site procedures and the manufacturer's recommendations. Surveillance testing is required in accordance with Regulatory Guide 1.9, "Design, Qualification and Testing of Diesel Generators", and with the surveillance requirements described in the Technical Specifications (Subsection 16.11.1) beginning with SR 3.8.1.4. Seismic-related inspections noted in Subsection 19K.11.7 should be done.

Maintaining emergency diesel generator reliability is a basic part of the station blackout rule (10CFR50.63). A reliability assurance program is required which maintains a target reliability. In view of the existing requirements noted above, it is judged that additional reliability and maintenance activities are not needed.

19K.11.11 Combustion Turbine Generator

Maintenance for the combustion turbine generator (CTG) is expected to be performed in accordance with site procedures and the manufacturer's recommendations. Suggested surveillance testing includes quarterly operation at rated speed and rated load until temperatures reach steady state values, approximately one hour. Also quarterly there should be a check of oil levels and assurance that there are no oil or fuel leaks. Quarterly the oil should be sampled and analyzed for acceptable quality. At each refueling/maintenance outage CTG fuel oil and lube oil should be inspected for deterioration; and replaced as necessary. Also, the fuel, lube oil and air filters should be replaced. There should be a thorough inspection of the entire assembly to assure that the inlet and outlet plenums are not blocked or deteriorating. Also, a complete visual inspection of the power unit should be made to assure that support bolts are secured and that there are no cracks and no blown gasket or engine hot spots. These tests and preventive maintenance activities are included in Table 19K-4.

19K.11.12 Fire Protection

The room fire barriers, the Smoke Removal System, and the remote shutdown panel and control were determined to be relatively important (Subsection 19K.6). Fire barriers, including penetrations, should be inspected periodically to assure that they retain their integrity with respect to confining a fire. The Smoke Removal System should be operated annually to demonstrate that it will be able to maintain a negative pressure in a room with a fire so that probability of propagation of fire and/or smoke to other rooms is low.

Smoke Removal System testing will be performed on each smoke removal zone in the reactor building, the service building and the radwaste building. Smoke Removal System testing will be patterned after damper alignment intended for smoke removal operation of the system. This consists of reducing normal exhaust from adjoining zones, to increase their pressure, and bypassing exhaust filters or small exhaust fans in the zone being tested to increase its exhaust flow rate. This will establish a pressure differential between zones to reduce the possibility that smoke will get into zones not directly affected by a fire.

Personnel entry to an area experiencing a fire is gained from an adjacent fire area which, by design, is at a positive pressure with respect to the area containing the fire. The pressure differential is sufficient to provide adequate velocity through the open door to push the combustion products back into the zone of the fire. The flow through the open door into the area of the fire and out of the area through the fire's exhaust duct system is enhanced by the positive pressure of the non-fire area. The HVAC Systems with recirculated air are manually switched over to a once-through system during a fire or test, so there is no direct mixing of smoke from one room to another.

The differential pressure between zones will be greater if all doors are closed, but each zone is relatively large, so one or two open doors between zones will not have a significant impact on the tests or on Smoke Removal System operation during a fire. Personnel should be advised that it is permissible to open doors during a test (or during a fire), but that doors should normally be closed at those times. This will allow personnel access to all related areas, and will not unduly restrict fire fighting personnel in event of a fire.

The remote shutdown panel should be tested periodically to show that it can perform its functions that will lead to safe shutdown. These RAP activities related to fire protection are included in Table 19K-4.

19K.11.13 Flood Protection

The important SSCs for flood protection are the water tight doors on external entrances to the control and reactor buildings and in ECCS and RCW rooms, the RSW and CWS isolation valves, anti-siphon capability, the circuit breakers that trip RSW and CWS pumps and water level sensors in the turbine building condenser pit and control building RCW rooms; sump pump operation; overfill lines on reactor building sumps on floor BIF; and room drain lines (Subsection 19K.7). Periodically room water barriers should be inspected to assure that they will prevent the spread of flooding, room drain lines should be checked to ensure no blockage exists, RSW isolation valves (MOVs) should be stroke tested (normally accomplished by switching from one pump to the standby pump in a given loop), CWS isolation valves should be stroke tested, the ability of RSW and CWS pump circuit breakers to trip upon receipt of a trip signal should be demonstrated, as well as RSW System anti-siphon capability. These RAP activities are included in Table 19K-4.

19K.11.14 Shutdown Protection

The shutdown analysis (Subsection 19K.8) identified as important components the SRVs of the ADS System and valves and pumps of the RHR system (including the LPFL mode) and of the HPCF, ACIWA and CRD Systems. RAP activities for SRVs are covered in Subsection 19K.11.3, and those for ACIWA components are covered in Subsection 19K.11.5. Testing of valves and pumps of the HPCF and RHR Systems and for the LPFL function of the RHR are covered by the technical specifications and valve and pump inservice testing (Table 3.9-8) for these systems. These testing requirements were reviewed and it was concluded that no additional reliability and maintenance actions are needed. This RHR testing also provides adequate assurance that the suppression pool temperature will be maintained below its high temperature limit (Table 19K-2).

The CRD System is normally operating, but system flow can be increased by opening some partially closed valves and / or by operating the second pump in addition to the operating one. The RAP activity, in Table 19K-4, is to review the CRD operating procedures and verify that they include steps to increase flow when necessary in a manner consistent with GE’s Service Information Letter, “Increase CRD System Flow to RPV After Shutdown for Emergency”, SIL 200, Rev. 1, Supplement 1.

During plant shutdown the normal cooling for the reactor will be by one division of the RHR System, in the shutdown cooling mode. This RHR division is powered by its divisional AC power with instrumentation power from the divisional DC power. A second RHR division of safety system with its supporting AC and DC power will be in standby, ready to operate at any time. (Electrical equipment from other systems is expected to be operating on the power systems that are in standby for the RHR function.) The third division of safety system is completely available for maintenance.

During shutdown, the failure of the operating RHR loop is one initiating event with an assumed low probability. Testing of key RHR components, consistent with Table 3.9-8 for in-service testing, is identified with RAP activities in Table 19K-4. Operators should monitor RHR loop failure rate and take corrective action if the failure rate exceeds the assumed probability during operation.

Testing and maintenance activities will be possible on AC and DC Power Systems in the third division which is in maintenance. Inspections related to reliability of offsite AC Power Systems are discussed in Subsection 19K.11.1, as are periodic checks on station batteries. Testing of emergency diesel generators and the combustion turbine generator are covered in Subsections 19K.11.10 and 19K.11.11, respectively. Since the two operating power systems are continuously monitored, it is not necessary to identify additional special tests or maintenance as part of the RAP for the AC and DC Power Systems.

19K.11.15 Prevention of Intersystem LOCA

The Reactor Water Cleanup (CUW) System provides a negligible benefit in the ABWR PRA by removing decay heat at high pressure. It would only be used in this mode if the containment cooling mode of the RHR system was disabled. During all operating modes, an unisolated CUW break could cause serious consequences, therefore these CUW isolation valves must be capable of automatically isolating against a differential pressure equal to the operating pressure of the reactor coolant system in the event of a LOCA in the CUW. If the automatic isolation valves fail to close, the operator can close the remote manual shutoff valve from the control room to terminate the LOCA. The RAP activities to assure reliability of these isolation valves are listed in Table 19K-4.

19K.11.16 Determination of "Important Structures, Systems and Components" for Suppression Pool Bypass Analysis

The suppression pool is an important containment feature for severe accident progression and fission product removal, since releases from the reactor vessel are either directly routed to the pool (e.g., transients with actuation of ADS) or pass through the pool via the drywell-wetwell connecting vents.

If an event leads to pressurization of the wetwell to the extent that the containment rupture disks open, the vacuum breakers would open to equalize pressures in the wetwell and drywell. The breakers would then close, thereby isolating the drywell from the wetwell. Failure of a DW-WW vacuum breaker to close following the assumed event would provide a significant bypass from the drywell into the wetwell airspace. If the rupture disk is open and one of the vacuum breakers has not closed there would be a direct pathway from the drywell to the wetwell and to the environment.

The following are critical to assuring a low risk from wetwell / drywell vacuum breaker bypass:

(1) A low probability of vacuum breaker leakage

(2) A low probability that the vacuum breakers fail to close

(3) A high availability of drywell or wetwell sprays (and ACIWA as a backup) to condense steam which bypasses the suppression pool.

Recommendations for testing of DW-WW vacuum breakers and ACIWA System RAP activities are included in Table 19K-4.

19K.12 References

19K-1 "GE ABWR Licensing Review Bases", August 1987.

Table 19K-1 ABWR SSCs of Greatest Importance for CDF, Level 1 Analysis

SSC	Fussell-Vesely Importance %*	Risk Achievement Worth*
Combustion Turbine Generator		
Emergency Diesel Generator		
RCIC System (Unavailable, Test or Maintenance)		
Multiplex Transmission Network (CCF)		
RCIC Turbine		
RCIC Pump		
Trip Logic Units		
Remote Multiplexing Units		
RCIC Turbine Lubrication System		
HPCF B (Unavailable, Test or Maintenance)†		
Station Batteries (CCF)		
Single Offsite Power Line†		
RCIC Min Flow Bypass Valve E51-F011 (NOFO)‡		
RCIC Min Flow Bypass Valve E51-F011 (NCFC)‡		
RCIC Injection Valve E51-F004 (NCFC)‡		
RCIC Steam Supply Valve E51-F037 (NCFC)‡		
HPCF Maintenance Valve E22-F005Bƒ		
RCIC Isolation Signal Logic		
Both Offsite Power Sources		
HPCF Pump†		
SRVs†		
RHR Flow Transmitters (CCF Miscalibration)		
SRV (CCF)		
Level 2 Sensors (CCF)		
Level 8 Sensors (CCF Miscalibration)		
Digital Trip Modules (CCF)		

* Not part of DCD (Refer to SSAR).

† SSCs with low FV importance and low risk achievement worth. Not considered further for RAP on the basis of Level 1 analysis.

‡ Valves that are closed during normal operation, and fail to open when required during a transient, are designated NCFC. Technically, they are "fail as is" conditions, which is closed. The minimum flow bypass valve is closed during normal operation, but during transients requiring RCIC operation, the bypass valve opens. Failure of this valve to open at that demand is shown as NCFC. Later in the transient this bypass valve, which is normally open at this time, should close on demand. If it fails to close, the shorthand description NOFO is used.

ƒ SSC with low FV importance and low risk achievement worth, but retained because of human factor importance.

Table 19K-2 ABWR SSCs With Moderate Risk Achievement Worth For CDF, Level 1 Analysis*

SSC	Fussell-Vesely Importance %†	Risk Achievement Worth†
RCIC Turbine Exhaust Isolation Valve F039 Limit Switch Fails		
RCIC Steam Supply Bypass Valve F045 Limit Switch Fails		
Div 1 Transmission Ntwk Failure (EMS)		
1st ESF RMU Div 1 Fails		
2nd ESF RMU Div 1 Fails		
RCIC Flow Sensor E51-FT007-2 Fails		
RCIC Isolation Valve F036 Fails (NOFC)		
RCIC Isolation Valve F035 Fails (NOFC)		
RCIC Isolation Valve F039 Fails (NOFC)		
RCIC Check Valve E51-F003 Fails to Open		
RCIC Check Valve F038 Fails to Open		
RCIC Outboard Check Valve F005 Fails to Open		
NBS Isolation Check Valve B21-F003B (FW Isolation) Fails Closed		
NBS Isolation Check Valve B21-F004B (FW Isolation) Fails Closed		
NBS Manual Valve B21-F005B (FW Isolation) Fails Closed (NOFC)		
RCIC Pres Sensor PIS-Z605 Miscalibrated		
RCIC Flow Sensor FT-007-2 Miscalibrated		
RCIC Pressure Sensor E51-PIS-Z605 Fails		
Failure of Division 1 Distribution Panel		
SP Temp High (Loss of Pump Head)		
SLU/EMS Link for Div 1 SLU 1 Fails (RCIC Fails)		
SLU/EMS Link for Div 1 SLU 2 Fails (RCIC Fails)		

* EMS = Essential Multiplexing System
ESF = Engineered Safety Feature
RMU = Remote Multiplex Unit
SLU = Safety System Logic Unit
† Not part of DCD (Refer to SSAR).

Table 19K-3 ABWR Initiating Event Contribution to CDF, Level 1 Analysis

Initiating Event	Events Per Year*	Total CDF *	Percent CDF Contribution*
Station Blackout for Less Than Two Hours			
Station Blackout for Two to Eight Hours			
Station Blackout for More Than Eight Hours			
Isolation/Loss of Feedwater			
Unplanned Manual Reactor Shutdown			

* Not part of DCD (Refer to SSAR).

Table 19K-4 Failure Modes and RAP Activities

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
RCIC System	System failure	See following items	See below	See below	*
RCIC System	Unavailable due to test or maintenance	Monitor unavailable time, compare with assumed 2%	Annually	Level 1 analysis	†
Multiplexers	Common cause failure of all MUX to give proper signals	System functional test	3 months	Experience	*
		Complete system test, error check	2 years	Experience	
One ESF RMU for Div 1 or one SLU/EMS Link for SLU Div 1	Failure of remote multiplex unit or link between RMU and safety system logic unit	System functional test	3 months	Experience	*
		Complete system test, error check	2 years	Experience	
RCIC Turbine & Pump (System Test)	Mechanical failure to operate	Turbine startup and operation; measure pump vibration velocity & displacement, flow, speed, diff. pressure.	3 months	Experience‡	†
		Turbine inspection, refurbishment	5 years	Experience‡	
RPS Trip Logic Units	Failure to trip upon demand	System functional test	3 months	Experience	*
		Complete system test, error check	R/M outage	Experience	
RCIC Turbine Lube System	Lube oil pump failure	Lube oil pump operation and oil pressure check	3 months	Experience	†
RCIC Check Valve F038	Failure to open	Open and close during system test	2 years	Table 3.9-8	†
RCIC Check Valves F003 & F005	Failure to open	Open and close test	R/M outage	Experience‡	†
RCIC Isolation Signal Logic	Failure to provide isolation signal when conditions warrant	Logic functional test	3 months	Experience	†

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
RCIC Min Flow Bypass Valve (NOFO or NCFC)	Failure to operate because of mechanical problems	Stroke test	3 months	Experience‡; ASME Code ISI	†
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary.	10 years	Low failure rate; ASME Code ISI.	
	Failure to operate because of electrical problems	Electrical circuit test	3 months	Experience‡	
RCIC Injection Valve and Turbine Steam Supply Valve	Failure to open because of mechanical problems	Stroke test	3 months	Experience‡; ASME Code ISI	†
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary.	10 years	Low failure rate; ASME Code ISI.	
	Failure to open because of electrical problems	Electrical circuit test	3 months	Experience‡	
RCIC Isolation Valves (NOFC)	Spurious failure because of mechanical problems	Stroke test	3 months	Experience‡; ASME Code ISI	†
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary.	10 years	Low failure rate; ASME Code ISI.	
	Spurious failure because of electrical problems	Electrical circuit test	3 months	Experience‡	
Limit Switches on RCIC Turbine Exhaust Isolation Valve and Steam Supply Bypass Valve	Failure of switch to change position when valve movement occurs	Observation of limit switch actuation during valve stroke test	3 months	Experience‡	†
RCIC Flow Sensor FT-007-2	Sensor fails	Calibration of sensor	R/M outage	Experience	*
	Miscalibration	Review calibration procedures for note about potential safety considerations	R/M outage	Judgment	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
RCIC Pressure Sensor PIS-Z605	Sensor fails	Calibration of sensor	R/M outage	Experience	†
	Miscalibration	Review calibration procedures for note about potential safety considerations	R/M outage	Judgment	
NBS Isolation Check Valves 003B & 004B	Fails to open	Leak rate test and subsequent operation of valves	R/M outage	Experience	*
NBS Manual valve F005B (NOFC)	Normally open valve fails closed	Stroke test	R/M outage	Experience	*
HPCF Maintenance Valve	Failure to open valve after maintenance	Independent verification of valve position following maintenance; position verification before startup	After maintenance, before startup	Judgment	*
Switch Yard Equipment	Failure results in loss of offsite power	Inspect switch yard equipment for signs of incipient failure, such as insecure structures, degraded insulators, leaking oil. Use thermography to detect hot spots on transformers, insulators, circuit breakers & connectors. Repair as necessary. See also the following items.	3 years	Experience	*
Switchyard Protective Relay	Relay failure to open or close on demand	Calibration, maintenance and test	1 year	Industry practice	ƒ
Auxiliary Relay Panels	Failure to provide power to loads	Routine cleaning and inspection	2 years	Industry practice	ƒ
Radio Batteries for microwave and fiber optic equipment	Battery failure	Routine test and maintenance	2 years	Industry practice	ƒ
Battery Chargers	Failure to provide charging current to batteries	Routine test and maintenance	18 months	Industry practice	ƒ

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Feedwater Pumps	Failure during operation	Walkdown/external visual observation: -Oil level -Leaks -Land vibration	1 week	Experience	
		Motor winding temperature, bearing temperature	1 week	Experience	
		Seal leakage, temperature, pressure	1 month	Experience	
		Oil sample/analyze	3 months	Experience	
		Performance data: -Discharge pressure -Inlet pressure -Flow rate -Peak vibration velocity -Motor current	3 months	Experience	
RHR Flow Meters	Common mode miscalibration	Review calibration procedures for note about potential safety considerations	Annual	Judgment	*
Level 2 Sensors	Common mode failure	Analyze Level 2 calibration data for trends of drifting or other CCF indications	R/M Outage	Judgment	*
Level 8 Sensors	Common mode miscalibration	Review calibration procedures for note about potential safety considerations	Annual	Judgment	*
Digital Trip Modules	Common cause failure to trip	Review trip unit test procedure to assure note about potential safety considerations	Annual	Judgment	*

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Wetwell/Drywell Vacuum Breakers	Fail to close or leakage after close	Cycle through full open to full close. Check for leakage	R/M Outage	Experience	*
ADS System SRVs	Failure of several SRVs to open on demand or failure to remain open	Inspect and replace degradable parts and test for correct operation	5 years (max)	Environmental qualification	*
		Remove valve, test for setpoint pressure, adjust setpoint as necessary, test for seat leakage, repair. Stagger testing of valves, 50% at one outage	3 years	Experience, ANSI/ASME OM-1	
		Control Panel Test	3 months	Experience	
LDF Fusible Plug Valves	Failure to open at temperature	Two of ten plugs replaced; tested to verify temperature setpoint	2 R/M outages	Judgment	*
	Leakage	Inspect for leakage	R/M outage	Judgment	
ACIWA System	System unavailable	See following items	See below	See below	*
ACIWA Flow Instrumentation	Failure to accurately monitor flow	Measure zero flow and full system flow	4 years	Judgment	**
ACIWA Manual Valves (in RHR System)	Stuck closed	Stroke test	3 months	Experience[‡]; ASME Code ISI	**

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Firewater System Pumps on Fire Truck	Failure of pumps to provide required flow at pressure	20 min pump at 100% rated flow, 1.13 MPa (150 psi)	1 year	Judgment	**
		10 min pump at 70% rated flow, 1.48 MPa (200 psi)	1 year	Judgment	
		10 min pump at 50% rated flow, 1.82 MPa (250 psi)	1 year	Judgment	
	Failure of system to deliver required flow	Test system flow with fire truck pumps, water from tanks & from UHS	4 years	Judgment	
		Test system flow with AC-driven and diesel-driven pumps, water from tanks & from UHS	4 years	Judgment	
ACIWA Diesel Pump	Failure to pump on demand	Pump start test	3 months	Experience	**
		Pump flow test	4 years	Experience	
RHR Non-Safety-Related Valve	Failure to open on demand	Manually open and close valve	2 years	Experience	**
Piping of AC-Independent Water Addition System	Piping failure that precludes successful operation	Piping visual inspection under operating pressure to assure no leaks	5 years	Judgment	**
		Piping support visual inspection to assure structural adequacy	5 years	Judgment	
COPS System	System failure	See following items	See below	See below	*
COPS AOVs	Inadvertently left closed following maintenance	Stroke test; position indication check; verification of local and control room indication following test	R/M outage	Experience	††
COPS Rupture Disks	Failure to open on demand	Disk replacement	5 years	ASME Code	††
		Verification of actuation within ±5% of rated pressure	5 years	ASME Code	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
COPS Flow Lines	Flow blockage	Flow test to assure no blockage in line	5 years	Judgment	††
Fire Barriers Between Rooms	Failure to retain integrity	Inspection of fire barriers, including seals and penetrations	1 year & after major maintenance	Judgment	N/A
Smoke Removal System	Failure to maintain low room pressure	Operate system to assure that it functions as designed	1 year	Judgment	N/A
Remote Shutdown Panel	Failure to provide control for reactor shutdown	Demonstrate ability to shut down reactor and remove decay heat by operation at remote shutdown panel	R/M outage	Judgment	N/A
RCIC System	Failure to start or operate RCIC from remote location	Start and operate RCIC from stations outside the main control room	10 years	Judgment	N/A
Control and Reactor Building and ECCS Room Watertight Doors	Failure to retain integrity	Inspection of watertight doors, including penetrations	1 year & after major maintenance	Judgment	*
RSW and CWS Isolation Valves	Failure to close on demand	Stroke test	1 month	Experience	*
RSW and CWS Pump Circuit Breakers	Failure to trip pump on demand	Breaker trip test to assure trip on demand	6 months	Judgment	*
CRD System Flow Increase	Failure to increase CRD flow in shutdown	Review CRD operating procedures to assure that steps to provide increased flow are consistent with SIL 200	2 years	Judgment	*
DC Div 1 Distribution Panel	Panel failure	Panel function is demonstrated by system test	3 months	Experience	*

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
RSW Anti-siphon Capability	Isolation valves don’t close after pumps trip	Open RSW motor breakers with isolation valves open and monitor system flow rate	6 months	Judgment	*
Room Sump Level Switches	Failure to detect water in sump	Observation of proper operation upon actuation	Annual	Judgment	*
Div 1 EMS Transmission Network	Network failure	System functional test	3 months	Experience	*
		Complete system test, error check	2 years	Experience	
Sump Pumps	Failure to pump water out of sump	Start sump pump and observe operation	Annual	Judgment	*
Overfill Line	Line clogged	inspect lines for debris	5 years	Judgment	*
	Water seal dry	Observe level in seal	Weekly	Judgment	N/A
Room Drain Lines	Line clogged	Inspect lines for debris	5 years	Judgment	N/A
Combustion Turbine Generator (CTG)	Failure to start and run	Start and operate CTG at rated speed and load for 1 hour	3 months	Experience	*
		Check oil levels, check for leaks	3 months	Experience	
		Sample, analyze oil. Replace as necessary	3 months	Experience	
		Inspect lube oil and fuel oil for deterioration. Replace oil filters as necessary; inspect inlet and outlet plenums and entire assembly	R/M outage	Experience	

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
Structures of Emergency AC Power EDGs, 480 VAC Transformers, MCCs & circuit breakers; DC Batteries and Cable Trays; RHR Heat Exchangers; SLC Tank, Valves, Piping & Pumps; Valves, Piping & Pump of ACIWA; SWS pumps, pump house and air conditioner; & SRV Discharge Piping of the NBS	Structural failure of supports during seismic event	Seismic walkdown to assure structural integrity	10 years	Judgment	N/A
		Visual inspection, support structures & devices.	10 years	Judgment	N/A
		Post-earthquake evaluation	After OBE or larger quake	Judgment	N/A
Single Train of RHR System (Shutdown Cooling & LPFL Modes)	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	*
Single Train of HPCF System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	*
Single Train of CUW System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	N/A
Single Train of RSW System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	N/A
Single Train of AC Electrical System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgment	N/A
Emergency Diesel Generator	Failure to start and run	Start up to full load	1 month	Tech Spec	*

Table 19K-4 Failure Modes and RAP Activities (Continued)

Component	Failure Mode/Cause	Recommended Maintenance	Test or Maintenance Interval	Basis	Unavailability, Failure Rate
CUW Isolation Valves (NO, FAI)	Failure to operate because of mechanical problems	Stroke test	3 months	Experience‡¶; ASME Code ISI	*
		Visual and penetrant inspection of stem, ultrasonic inspection of stem; replace if necessary	10 years	Low failure rate; ASME Code ISI	
	Failure to operate because of electrical problems	Electrical circuit test	3 months	Experience‡	
CUW Remote Manual Shutoff Valve (NO, FAI)	Failure to operate because of mechanical problems.	Stroke test	Refueling outage	Judgement (non-safety-related)	*
	Failure to operate because of electrical problems.	Electrical circuit test	Refueling outage	Judgement (non-safety-related)	
Operating RHR Shutdown Cooling Loop	Failure to Operate because of mechanical or electrical problems	See following items	See below	See below	*
RHR Pumps	Failure to provide adequate flow at desired pressure	Discharge pressure test Inlet pressure test Flow test Vibration test	3 months	Table 3.9-8	‡‡
RHR Injection Valves, F005	Failure to operate	Stroke test	Cold shutdown	Table 3.9-8	‡‡
RHR Isolation Valves, F010, F011	Failure to operate	Stroke test	Cold shutdown	Table 3.9-8	‡‡
RHR Admission Valves, F012	Failure to operate	Stroke test	3 months	Table 3.9-8	‡‡

* Not part of DCD (refer to SSAR).

† RCIC component failure rates are included within the system unavailability.

‡ These types of valves and turbines have been used in operating BWRs, so there is much experience to guide owners/operators in care of the equipment.

ƒ Switchyard component failure rates are included within the switchyard equipment failure rate.

** ACIWA component failure rates are included within the system unavailability.

†† COPS component failure rates are included within the system unavailability. (Failure of the rupture disks to actuate upon demand before structural failure of the containment dominates failure of COPS.)

‡‡ RHR component failure rates are included within the system unavailability.

19L ABWR Shutdown Risk Evaluation

19L.1 Purpose

The purpose of this study is to review the potential risk associated with ABWR operation while the plant is shut down. Events that have a potential to lead to accidents when the ABWR plant is shut down for maintenance or refueling are identified and reviewed against ABWR plant features which prevent and mitigate these accidents.

Additional information on ABWR shutdown risk is contained in Appendix 19Q.

19L.2 Conclusions

It is concluded that the ABWR plant is adequately protected against accidents during shutdown conditions. It is judged that the probability of core damage during shutdown periods is negligible and therefore it is concluded that no modifications to the ABWR plant design are required. It is also concluded that a detailed probabilistic risk assessment (PRA) for the ABWR shutdown conditions is not required.

19L.3 Introduction

General Electric completed a PRA for the ABWR plant as part of Tier 2. The internal event PRA (Section 19.3) provided an extensive analysis of transients and accidents that initiate during power operation. The seismic PRA (Section 19.4) also consisted of events that initiate during power operation. In both PRAs, it was judged that the risks during shutdown conditions would be low with respect to those during power operations for several reasons:

(1) Most of the transients that disturb power operations do not apply to the shut down plant

(2) Low system pressure reduces the already small frequency of loss of coolant events due to pipe break

(3) Low decay heat means long time periods are available to restore cooling capability should residual heat removal system cooling be interrupted

However, the NRC has requested (Reference 19L-1) that GE review the risks associated with shutdown in more detail to support the conclusion that such risks are low.

Shutdown risks have not been studied in detail in the past. In the Reactor Safety Study (Reference 19L-2) the shutdown risks were estimated to be negligible. EPRI conducted a somewhat detailed review of the shutdown risks for the Zion plant, a pressurized water reactor (PWR) (Reference 19L-3), and concluded that the mean core damage frequency (CDF) for the shutdown conditions is about a factor of four lower than the corresponding value for power operation. In a subsequent study (Reference 19L-4) for the Seabrook plant, another PWR, the shutdown risk was calculated to be about a factor of 5 lower than that for power operation. A recent French study (Reference 19L-5) has concluded that shutdown risks for the Paluel PWR plant constitute about 60% of the total plant risk. Recently, the NRC has launched studies to estimate the risk associated with shutdown conditions for two plants: Surrey (PWR plant, analyzed by Brookhaven) and Grand Gulf (BWR plant, analyzed by Sandia National Laboratories). The results of these studies are expected in 1993.

Appendix 19Q contains additional information on ABWR shutdown risk including a risk assessment of the loss of an operating RHR system during shutdown. This risk assessment evaluates the conditional core damage probability given a loss of one RHR train. Minimum sets of systems are identified that, if administratively controlled to not be in maintenance, will ensure an acceptably low conditional core damage probability. Other items discussed in 19Q are:

- ABWR features to minimize shutdown risk
- Procedures for completion of outage plans
- Use of freeze seals
- Evaluation of potential vulnerabilities due to new ABWR features
- How ABWR features could mitigate past events at operating BWRs

19L.4 Scope of the Study

19L.4.1 Mode of Reactor Operation

The various modes of ABWR reactor operation as noted in the plant technical specifications are shown in Table 19L-1.

The ABWR PRA (Section 19.3) covers periods of power operation (Mode 1) and start up periods (Mode 2), whereas this shutdown study covers periods of cold shutdown (Mode 4) and refueling (Mode 5). Periods of hot shutdown (Mode 3) are not included in either study. Hot shutdown periods are expected to be relatively small compared to those previously analyzed (Modes 1 and 2) and considered in this study (Modes 4 & 5). Also, hot shutdown can be seen as an extension of the shutdown process started during Mode 1 and the incremental increase in risk during this mode of operation is judged to be small since the safety systems available for achieving hot shutdown continue to be available during hot shutdown. Loss of RHR in Mode 3 is discussed in Subsection 19Q.7.

The types of events during Modes 4 and 5 considered in this study are as follows:

- Reactivity Excursion Events

- Reactor Pressure Vessel Draining Events
- Loss of Cooling Events
- Loss of Decay Heat Removal Events

19L.4.2 Noncore-Related Events

Events which occur inside the containment that are not related to the fuel in the reactor core, but have a potential to release radioactivity to the environment were not included in the PRA, but are addressed in this study. Events outside the containment, such as the rupture of the liquid radwaste tank, are not reviewed in this study since they are judged to be negligible contributors to ABWR Plant risk.

19L.4.3 Summary of Types of Events Considered

The types of events considered in this shutdown risk study are summarized as follows:

(1) Reactivity Excursion Events (Subsection 19L.5)

(2) Reactor Pressure Vessel Draining Events (Subsection 19L.6)

(3) Loss of Core Cooling (Subsection 19L.7)

(4) Loss of Decay Heat Removal Events (Subsection 19L.8)

(5) Non-Core-Related Events (Subsection 19L.9)

19L.5 Reactivity Excursion Events

Reactivity events which have a potential to occur during power operations are examined for their likelihood to occur during shutdown conditions. In addition, events which have a potential to occur only during shutdown conditions are also reviewed.

19L.5.1 Control Rod Drop Accident

The ABWR fine motion control rod drive (FMCRD) is equipped with several new and unique features to prevent a control rod drop accident compared with locking piston control rod drives (LPCRD) used in the boiling water reactors (BWR) currently in operation. Three modes of failure that could lead to a control rod drop accident have been identified and a summary of the event causes and preventive and mitigative features included in the FMCRD design is provided in Table 19L-2.

Subsection 15.4.9 provides a detailed review of the control rod drop accident during power operation and describes the ABWR features that prevent and mitigate the accident. The following discussion extends the review to shutdown conditions (operating Modes 4 & 5).

For the rod drop accident to occur during power operation, the control rod must stick initially and then physically separate from the drive on a control rod withdrawal command. Later the same control rod becomes unstuck and drops freely resulting in a rod drop accident. For the rod drop accident to occur during operating Modes 4 & 5, in addition to the above failures, the reactor must also be critical. Since the reactor is subcritical in these modes, even with the above sequence of failures, it is impossible for a rod drop accident to occur. The only time when the above sequence of events could potentially result in a rod drop accident is when it occurs in conjunction with the withdrawal of an adjacent control rod for reasons such as testing. As will be shown by the following consideration and analysis, the probability of a control rod accident during operating Modes 4 & 5 is negligible.

The ABWR features that help prevent control rod accidents are as follows:

(1) Each FMCRD is equipped with dual Class 1E separation detection devices that will detect the separation of the control rod from the CRD if the control rod and hollow piston stick and separate from the ballnut of the CRD. The separation switches can also detect if the blade separates from the hollow piston, even with the hollow piston still resting on the ballnut. The separation detection device is in operation at all times. When the separation has been detected, the interlocks will prevent further rod withdrawal (i.e., will initiate a rod block). Also, an alarm signal will be initiated in the control room to warn the operator.

(2) The hollow piston part of the FMCRD is equipped with a latch mechanism. If the hollow piston is separated from the ballnut and the rest of the drive due to a stuck rod, the latch will limit any subsequent rod drop to a distance of 20.32 cm (8 inches). (More detailed descriptions of the FMCRD system are presented in Subsection 4.6.1.)

(3) There is a unique, highly reliable bayonet-type coupling between the control rod blade and the control rod drive. The coupling spud at the top end of the hollow piston engages and locks into a mating socket at the base of the control rod. The coupling requires a 45-degree rotation for engaging or disengaging. Once locked, the drive and rod form an integral unit that must be manually unlocked by specific procedures before the components can be separated. This feature practically assures that the rod and the drive are never accidentally separated, and offers protection against the rod drop failure Mode 2 (Table 19L-2).

(4) Procedural coupling checks are enforced to assure proper coupling.

(5) Interlocks have been provided to assure that inadvertent criticality does not occur because a control rod is withdrawn adjacent to another control rod.

The Class 1E separation detection device and the control rod withdrawal interlock help prevent each of the three control rod drop failure modes listed in Table 19L-2. The other features that help prevent specific failure modes are discussed below.

Control rod drop failure requires the following events/failures:

For Failure Mode 1:

(1) Operator withdraws two control rods for testing,

(2) A third adjacent control rod sticks and unsticks at specific times,

(3) Class 1E separation detection of the third control rod or rod block fails,

(4) Operator tries to withdraw the third control rod and the interlock fails,

(5) Operator ignores alarm and continues withdrawal of the third rod, and

(6) Hollow piston latch of the third rod fails.

For Failure Mode 2:

(1) Operator withdraws two control rods for testing,

(2) A third adjacent control rod sticks and unsticks at specific times,

(3) Positive bayonet coupling of the third control rod experiences structural failure,

(4) Class 1E separation detection of third control rod or rod block fails,

(5) Operator tries to withdraw the third control rod and the interlock fails, and

(6) Operator ignores alarm and continues withdrawal of the third rod.

For Failure Mode 3:

(1) Operator withdraws two control rods for testing;

(2) Operator installs a third adjacent control rod drive without coupling, and fails to detect the error during procedural coupling checks;

(3) The third control rod sticks and unsticks at specific times;

(4) Class 1E separation detection of the third rod or rod block fails; and

(5) Operator ignores alarm and continues withdrawal of the third rod.

It is clear from the above discussion that multiple hardware failures and human errors have to occur to cause a rod drop accident. Even without a detailed analysis, it can be seen that the rod drop accident frequency is negligible. It is therefore concluded that the rod drop accident is unlikely to occur during Modes 4 & 5 and is therefore not a safety concern for the ABWR.

19L.5.2 Control Rod Ejection Accident

The control rod ejection accident during the ABWR power operation starts with a major break in the FMCRD housing weld between the housing and the RPV, or a major break in the drive mounting bolts or a drive spool piece. The accident can also be started with a break in the drive insert line. Following the break, the reactor pressure exerted on the CRD coupling pushes down the hollow piston and the ballnut with a large force. The shaft screw and the motor are forced to unwind, resulting in the rod being ejected. For the control rod ejection accident to occur during operating Modes 4 & 5, in addition to the above failures, the reactor must also be critical. Since the reactor is subcritical in these modes, even with the above sequences of failures, it is impossible for a control rod ejection accident to occur. Similarly, the low pressures associated with these operating modes makes the break in the FMCRD housing or drive insert line extremely unlikely. The only time when the above sequence of events (i.e. those that cause control rod ejection accident during power operation) could potentially result in a control rod ejection accident during operating Modes 4 & 5 is when it occurs in conjunction with a reactor hydro-test and withdrawal of an adjacent control rod withdrawn for reasons such as scram time testing. A summary of the causes of the rod ejection accident and the ABWR preventive and mitigative features is provided in Table 19L-3. As will be shown by following consideration and analysis, the probability of control rod ejection accident during operating Modes 4 & 5 is negligible.

The ABWR features that prevent and mitigate control rod ejection accidents are:

(1) A break in the FMCRD housing (or weld between housing and vessel or drive mounting bolts or drive spool piece) is mitigated by integral internal blowout supports (“shootout restraints”) (Subsection 4.6.1.2.2.9) which physically prevent the control rod from being ejected.

(2) A break in the drive insert line is mitigated by the following:

(a) Ball check valve in the CRD insert port.

(b) Electromechanical brake

The FMCRD design incorporates an electromechanical brake keyed to the motor shaft. The brake is normally engaged by a passive spring force. It is disengaged when the spring load is overcome by the energized magnetic force. The braking torque between the motor shaft and the

CRD spool piece is sufficient to prevent control rod ejection in the event of a failure in the pressure retaining parts of the drive mechanism. The brake is designed so that its failure will not prevent the control rod from rapid insertion (scram). Additional details on the electromechanical brake are provided in Subsection 4.6.1.

(c) Holding torque provided by the permanent magnet in the step motor prevents rod from being ejected during operating Modes 4 & 5 when the reactor is not under pressure.

Control rod ejection can occur only under the following conditions:

(1) Failure of FMCRD housing, etc., coupled with failure of integral internal blowout support of one FMCRD when an adjacent drive has been withdrawn for testing and reactor is undergoing hydro-test (i.e. reactor is at pressure).

(2) Break in any one of the FMCRD insert pipes coupled with the failure of the corresponding ball check valve in the insert port and failure of the corresponding FMCRD electromechanical brake when an adjacent drive has been withdrawn for testing and reactor is undergoing hydro-test.

During operating Modes 4 & 5, the time duration that the reactor is at pressure due to hydro-test is very small. Also, because of multiple independent failures required, the probability of a control rod ejection accident through above sequences is judged to be negligibly low. It is therefore concluded that the control rod ejection accident is unlikely to occur during operating Modes 4 & 5 and is therefore not a safety concern for the ABWR.

19L.5.3 Refueling Error

Refueling errors resulting in the loading of fuel bundles in two adjacent uncontrolled cells could result in a reactivity accident. Uncontrolled cells are fuel cells in which control blades have been withdrawn. An accident can result from inserting a fuel bundle into a fueled region of the core which has withdrawn control blades.

Preventive and mitigative features in the ABWR plant are summarized in Table 19L-4 and discussed below:

(1) In the ABWR plant there is very little incentive for unloading the entire core. Generally, utilities resort to unloading the whole core when there is a need to maintain a large number of control rod drives during a refueling outage. In the case of ABWR, very few FMCRD need to be removed for maintenance and therefore there is very little incentive for unloading the whole core.

(2) During refueling, only one rod can be withdrawn. This is because Technical Specifications require that the gang/single selector switch in the Rod Control and Information System (RC and IS) be placed in the single position during refueling. Any attempt to withdraw a second rod results in a rod block initiated by the RC and IS.

(3) With mode switch in the REFUEL position, if any one control blade has been removed, then the refueling interlocks prevent hoisting another fuel assembly over the vessel (Subsection 9.1.4.2.7.1).

Therefore, for this accident to take place, the following events must occur.

(1) Utility decides to unload the whole core or perform control blade shuffling in parallel with refueling.

(2) One control blade is removed and its CRD is valved out of service.

(3) The rod block fails and the operators remove the adjacent control blade, and its CRD is valved out of service.

(4) Operator starts loading the fuel bundles. All fuel cells adjacent to withdrawn blades have been loaded except for the last fuel bundle.

(5) The last bundle is lowered into the empty uncontrolled fuel cell.

(6) The control room operator fails to observe SRNM multiplication.

(7) The reactor goes critical and high flux initiates a scram signal but valved out drives cannot scram.

As a consequence of this accident, local fuel failures can be expected. GE has studied this problem and has issued a service information letter (SIL-372) (Reference 19L-7), to assist the operating plants. In that study, GE has found that for the BWR plants following GE’s guidelines, the probability of this accident is negligible. The probability of this accident is also expected to be negligible for the ABWR plants.

19L.5.4 Rod Withdrawal Error

During shutdown, there is a potential for the reactor to become critical if two adjacent control rods are withdrawn inadvertently. The ABWR features that prevent and mitigate this event are as follows:

(1) During refueling, Technical Specifications only allow one rod to be withdrawn at a time. Any attempt to withdraw a second rod results in a rod block by the rod withdrawal interlock.

(2) If the rod block fails and the rod is withdrawn, the reactor will scram on a high flux signal. The scram system is in operation at all times during shutdown.

Therefore, for this event to take place, the following events must occur:

(1) Operator withdraws one control rod for testing.

(2) Operator decides to test a second control rod without inserting the first control rod (i.e., operator does not follow procedures).

(3) The second control rod is adjacent to the first control rod which was withdrawn for testing.

(4) The interlock designed to prevent the withdrawal of the second rod fails.

(5) Rod fails to scram as designed.

The refueling interlock and the scram systems are highly reliable. The combined probability of operator error and failure of the above systems resulting in a rod withdrawal error is judged to be negligible.

19L.5.5 Fuel Loading Error

During refueling, there is a potential for the reactor to become critical if a fuel loading error is followed by withdrawal of a potentially high worth control rod. The ABWR features that prevent and mitigate this event are as follows:

(1) Operators follow specific core loading procedures.

(2) During core loading, interlocks prevent withdrawal of more than one control rod.

(3) Following the full core loading, an as-loaded core verification process is completed.

(4) If the reactor does become critical on a control rod withdrawal, it will be followed by a scram immediately, since the neutron monitoring system is in operation during refueling.

Therefore, for this event to take place, the following events must take place:

(1) Operators fail to follow fuel loading procedures and commit specific loading errors.

(2) Core verification fails to reveal the fuel loading error.

(3) Operator withdraws a control rod for testing.

(4) Reactor fails to scram.

It should be noted that not all fuel loading errors can initiate this accident. For fuel loading error to be a concern, the high worth fuel bundles must be loaded at the wrong location. The combined probability of this error plus the others listed above is judged to be negligible. Therefore, it is concluded that a fuel loading error during refueling is not a concern for the ABWR plant.

19L.5.6 Conclusion

It is concluded that, during operation Modes 4 and 5, reactivity excursion events have a negligible probability of occurrence and are therefore not a safety concern for the ABWR plant.

19L.6 Reactor Pressure Vessel Draining Events

There is a potential for draining the reactor vessel during operating Modes 4 and 5, either as a result of hardware failures or operator errors or a combination of both. There is a potential for draining the vessel during maintenance activities such as the CRD or reactor internal pump removal and replacement. There is also a potential for draining the vessel when systems feeding to and bleeding from the RPV are in continuous operation. The control room operator routinely monitors the water level and takes corrective actions such as isolating the appropriate valve when the water level drops for unexplained reasons. Certain other corrective actions initiate automatically. A discussion of these drain paths and the preventive and mitigative features of the ABWR design are discussed below.

19L.6.1 FMCRD Replacement

FMCRD replacement can take place only during operating Mode 5. The replacement is done in two steps. First the control blade is withdrawn until the blade back-seats on the guide tube to provide a metal to metal contact. This provides the seal for preventing the reactor water from draining.Then the CRD spool piece is removed at which time the spindle adaptor seats on the splined spindle adaptor back seat to prevent any leakage of water from the RPV. The drive can then be removed and replaced. This arrangement of preventing vessel draining through back-seating of the control blade is the same as the one used in the operating BWR plants. There is still a potential for the operator to remove the blade inadvertently. The probability of this error is minimized through administrative controls. Occasionally a small amount of water leakage is experienced due to imperfect sealing of the control blade. However, based on hundreds of reactor-years of operating experience, it is judged that the probability of draining the vessel during FMCRD replacement is negligible.

19L.6.2 Reactor Internal Pump

There is a potential for draining the RPV while the reactor internal pumps (RIP) are undergoing maintenance or replacement. Two such maintenance activities, replacement of the RIP motor and replacement of the RIP impeller are discussed below and summarized in Table 19L-5.

19L.6.2.1 RIP Motor Replacement

This activity is carried out only during operating Mode 5. After the bolts are loosened at the bottom, the whole pump moves down by about 6 mm until the impeller backseats to prevent leakage of reactor water when the motor cover is removed. A secondary seal is then provided inflated with the help of a portable pump. At this point, the RIP motor can be removed and replaced.

19L.6.2.2 RIP Impeller Replacement

Impeller replacement can be carried out only after the RIP motor is removed as described above. Following the removal of the motor, a temporary cover plate is bolted at the bottom. The impeller is then removed from the top. The seal is provided by the bolted cover plate at the bottom. After the impeller is removed, a cap is installed on the RPV bottom head at the impeller shaft opening to provide additional protection against draining the RPV.

19L.6.2.3 Potential for Draining

Nuclear plants with RIPs have been in operation for over 10 years. Over 500 RIPs and motors have been removed and reinstalled in the European BWR plants without any problem. This has demonstrated that the replacement activities can be carried out without draining the vessel. For draining to occur, as a minimum, the impeller backseat and the inflatable seal have to fail when the motor is being replaced. Administrative procedures assure that impeller removal does not start until the RIP motor is removed and the temporary motor cover plate is bolted. In the most likely failure scenario, it is possible that the sealing between the impeller shaft backseat and the sealing provided by the inflatable seal may not be perfect. However, such failures are detectable, and result only in a small leakage [6.3×10^{-5} m^3/s (less than one gallon per minute)]. Under these conditions, the operator can always bolt the temporary bottom plate if needed. During impeller replacement, for drainage to occur, the impeller shaft nozzle cap must fail (or be dislodged), finally the bottom plate must also fail. During maintenance on the inflatable seals, a plug is placed over the impeller diffuser inside the RPV to prevent draining. Subsection 19Q.4.2 contains additional information on RIP maintenance.

19L.6.3 Control Rod Drive Hydraulic System

During operating Modes 4 & 5, the control rod drive hydraulic system (CRDHS) continues operating with one pump running to provide purge water to the FMCRDs.

With one pump in operation, the head of the pumping water can easily overcome the pressurized head of the RPV; hence, there is no possibility of draining the RPV. In the event that neither pump is in operation, there is a potential for draining the RPV through the CRDHS as discussed below, summarized in Table 19L-6, and shown in Figure 19L-1. As will be shown by the following considerations and analysis, the probability of draining the RPV through the CRD hydraulic system is negligible.

19L.6.3.1 Path 1

When neither pump is in operation, the scram valves will open due to low hydraulic control unit (HCU) charging header pressure, and will stay open if

(1) The reactor protection system (RPS) scram logic is not reset, or

(2) There is no instrument air available to the scram valve, or

(3) The scram pilot solenoid valves are disconnected from the RPS scram circuits.

This, combined with the failures of the CRD ball check valve and check valve (F115), and the mechanical failure of the HCU maintenance isolation valves (F101, F140) and HCU drain valve (F113) to isolate when closed by the operator or the operator error to leave them open, will lead to drainage of the RPV into the CRD hydraulic system.

Multiple failures are necessary for path 1 to occur. Should they occur in one HCU, only 2 CRD’s will be affected. In addition, the size of the piping connection between the RPV and CRDHS, being only 32A allows for a discharge rate which will provide enough time to remedy the situation. Therefore, the probability of draining the RPV through this path is judged to be negligible.

19L.6.3.2 Path 2

In the event where neither pump is in operation and the scram valves fail to open, there is still another potential path for draining the RPV through the CRDHS. Similar to the failures that resulted in path 1, (Subsection 19L.6.3.1) the CRD ball check valve must fail, and the HCU maintenance isolation valves (F101 and F140) must be open by operator error or mechanical failure. In addition, the scram valve must fail to open, the test port valve (F141) must be open by operator error or by mechanical failure, and testing equipment (or lack of) must fail. A drainage of the RPV through this path would lead to contamination of the plant environment.

Again, multiple failures are necessary for path 2 to occur; and, should a failure occur, only 2 CRDs will be affected and the slow discharge rate will provide time to correct the situation. Therefore, the probability of draining the RPV through this path is judged to be negligible.

19L.6.3.3 Path 3

Path 3 is similar to path 2 with the exception that the test port valve (F141) remains closed, the check valve (F138) must fail and HCU isolation valve (F104) must fail open or be left open by the operator. Such an event could cause drainage of the RPV water into the CRDH System. As with all other paths in this system, multiple combinations are needed for an event to occur and the drainage rate will be slow. Therefore, the probability of draining the RPV through this path is judged to be negligible.

19L.6.3.4 Conclusion

In conclusion, because of the multiple failures required in each HCU, it is judged that the probability of draining the vessel through the CRDHS during shutdown is negligibly low. Also, because of the small drain line size, adequate time is available to remedy the situation should vessel drain start. It is therefore concluded that during operating Modes 4 & 5, draining of RPV through failures in CRDHS is not a safety concern for the ABWR plant.

19L.6.4 Reactor Water Cleanup System

During the operating Modes 4 & 5, the reactor water cleanup (CUW) system is used in conjunction with the fuel pool cooling and cleanup system (FPC) to provide continuous cleaning of the reactor water. During these modes, both pumps operate to provide 100% capacity. Reactor water flows from the RPV via both the RPV bottom head line and a shared nozzle with the RHR suction line. There is a potential for draining the RPV through the CUW System during shutdown mode as discussed below, summarized in Table 19L-7 and shown in Figure 19L-2. As will be shown by the following considerations and analysis, the probability of draining the RPV through the CUW system is negligible.

19L.6.4.1 Path 1

During Modes 4 and 5, one potential path for RPV drainage occurs when valves F500 and F501 are open (failed open or inadvertently opened by operator). Reactor water will drain to the low conductivity waste (LCW) sump in the drywell through a 50A diameter vessel nozzle. This path is unlikely to occur because valves F500 and F501 are in series, F500 is locked closed, and both valves are under administrative control. However, should this drain path be established, when the LCW drywell floor sump water level reaches high level, a persistent alarm is annunciated in the main control room to alert the operator for proper action. Also, drainage will be slow because of the small (50A diameter) size of the vessel drain nozzle, thereby allowing adequate time to correct the situation. Because of the above features, the probability of draining the RPV by this path is judged to be negligible.

19L.6.4.2 Paths 2 & 3

Paths 2 and 3 are dependent on the normally closed valves F055A and F055B. Both are used for chemical flushing and decontamination before maintenance. Should either of these two valves be left open during operating Modes 4 or 5 (either by equipment failure or by operator error), reactor water will drain into the reactor building. Floor drain sumps are provided in the reactor building to collect waste from the equipment drains. If the water level in the drain sumps reaches a high level, an alarm is annunciated in the main control room to alert the operator. Should paths 2 or 3 occur, the drain path, a 50A diameter pipe, will allow sufficient time to correct the situation. Should no corrective action be taken manually, on reaching reactor water level 2, valves F002 & F003 will be isolated automatically, terminating the event. Also, the operator monitors the reactor water level in the control room and takes mitigative actions. Because of all these preventive and mitigative features, the probability of draining the RPV by this path is judged to be negligible.

19L.6.4.3 Path 4

Valves F022, F024 and F025 are normally closed. During the plant startup mode, excess water generated by reactor water level swell is dumped in a controlled manner to the main condenser. Flow control valve F022 regulates the blowdown flow. Should all three be inadvertently left open or fail open at the same time during operating Modes 4 or 5, RPV water will drain to the suppression pool. There are a number of preventive and mitigative features in the ABWR design. The valves are redundant and the valve status (open, closed) is indicated in the control room for all three valves. In the unlikely event that reactor water is drained through this path, high flow will be detected by flow element FT-017 and signals will be sent to the leak detection system to isolate the CUW system. Furthermore, if this drain path is established, it will terminate on reactor level 2 isolation of valves F002 & F003. Also, the operator monitors the reactor water level in the control room and takes mitigative actions. Because of all these preventive and mitigative features, the probability of draining the RPV by this path is judged to be negligible.

19L.6.4.4 Path 5

Path 5 is dependent on valves F022 and F023. Both are normally closed during operating Modes 4 and 5. During startup, excess water generated by reactor water level swell is dumped in a controlled manner to the LCW collector tank. Flow control valve F022 modulates the blowdown flow. If both valves are left open (by operator error or equipment failure), RPV water will drain to the LCW collector tank. There are a number of preventive and mitigative features in the ABWR design. The valves are redundant and the valve status (open, closed) is indicated in the control room for all three valves. In the unlikely event that RPV water drains through these valves, high flow will be detected by flow element FT-017 which will send a signal to the leak detection system to isolate the CUW System. If established, the drain path will terminate on reactor level 2 isolation as before. Also, the operator monitors the reactor water level in the control room and takes mitigative actions. Because of all these preventive and mitigative features, the probability of draining the RPV by this path is judged to be negligible.

19L.6.4.5 Path 6

Valve F056, which is used for chemical washing and decontamination before maintenance, is normally closed during operating Modes 4 & 5. If it fails open or is inadvertently left open by the operator when the CUW pump is in operation, reactor water will drain into the reactor building. Similar to path 2, floor drain sumps are provided in the reactor building to collect waste from the equipment drains and high water levels in these sumps will activate an alarm to alert the operator. Since this is a small diameter pipe, the slow drainage rate will allow sufficient time to correct the situation before level 2 is reached at which point the path will terminate on reactor level 2 isolation signal. Also, the operator monitors the reactor water level in the control room and takes mitigative actions. Because of all these preventive and mitigative features, the probability of draining the RPV by this path is judged to be negligible.

19L.6.4.6 Path 7

Path 7 is similar to path 4. If valves F022 and F025 are inadvertently left open or fail open at the same time, RPV water will drain to the main condenser. The two valves in series and the valve status indicator help lower the possibility of this path occurring. In the unlikely event this path were to occur, flow element FT-017 will detect the high flow and signal the leak detection system to isolate the CUW system. If unmitigated, the drain path will be terminated on reactor water level 2. As in the case for path 4, the probability of draining the RPV through this path is judged to be negligible.

19L.6.4.7 Conclusion

Because of the multiple failures or operator errors required for each of the above paths to occur, and the leak detection instrumentation in the drywell and reactor building that will alert the operator, it is judged that the probability of draining the RPV during shutdown mode through the reactor water cleanup system is negligibly low. Furthermore, as a mitigative measure, at reactor water level 2, CUW system valves F002 and F003 isolate the reactor from the CUW system. In practically all cases, even if all the above features should fail, the RPV drain will stop automatically when the RPV outlet nozzle is uncovered. At that point, there is still 1.6 meters of water over the top of the active core. It is therefore concluded that draining of the RPV through CUW system failures is not a safety concern for the ABWR plant.

19L.6.5 Residual Heat Removal System

The ABWR residual heat removal (RHR) system is a closed system consisting of three independent pump loops (A, B, and C—where B and C are similar) which inject water into the vessel and/or remove heat from the reactor core or containment. Loop A differs from B and C in that its return line goes to the RPV through the feedwater line whereas loop B & C return lines go directly to the RPV. In addition, loop A does not have connections to the drywell or wetwell sprays or a return to the fuel pool cooling system. However, for purposes of this analysis, the differences are minor and the three loops can be considered identical. The RHR system has many modes of operation, each mode making use of common RHR system components. These components are actuated by the operator; hence, the operation is subject to operator error which could potentially lead to drainage of the RPV. Potential paths for draining the RPV through the RHR system during operating Modes 4 and 5 are discussed below, summarized in Table 19L-8, and depicted in Figure 19L-3. Of the various modes of RHR operation it was judged that the potential for RPV draining was the greatest during the shutdown cooling mode. Therefore, the potential RPV draining paths start with the RHR in the shutdown cooling mode of operation. As will be shown by the following consideration and analysis, the probability of draining the RPV through the RHR system is negligibly low. Even if all the preventive and mitigative features fail, RPV draining will stop when the RHR shutdown cooling nozzle is uncovered at which point there is still 1.6 meters of water over the top of the active fuel.

19L.6.5.1 Path 1 (Loop B and C only)

During the shutdown cooling mode of operation, pump C001 is in operation and valves F010, F011, F012, F004, F005 and F007 are normally open. One potential path will occur if valve F026 is open (by mechanical failure or operator error). This will lead to drainage of RPV water to HCW (high conductivity water). The preventive and mitigative features are as follows: valves F010 and F011 will isolate the reactor from the RHR system at reactor water level 3; the operator monitors reactor waterlevel in the control room and correctly responds to control room indicators and alarms; and the drain path is only a 40A diameter line allowing sufficient time for corrective action. Because of all these preventive and mitigative measures, the probability of draining the RPV by this path is judged to be negligible.

19L.6.5.2 Path 2

With the pump running during the shutdown cooling mode of operation, path 2 will be established if the liquid waste flush valves (F029 and F030) are open by mechanical failure or operator error. Through this route, RPV water will drain to radwaste via a 150A diameter pipe. To prevent this from occurring, valves F029 and F030 are required to be closed during shutdown cooling mode, and if open, their open status will be indicated in the control room. Also at reactor water level 3, valves F010 and F011 will isolate the system. Finally, the operator monitors the reactor water level in the control room and takes corrective actions. Because of all these preventive and mitigative measures, the probability of draining the RPV by this path is judged to be negligible.

19L.6.5.3 Path 3

During the shutdown cooling mode of operation, if the suppression pool return valve (F008) is open (by mechanical failure or operator error), potential draining path 3 will be established. This path will drain reactor water to the suppression pool. The preventive and mitigative features are as follows: an interlock prevents opening of valve F008 if F012 is open and vice versa and indicators in the control room will show the status of F008 and the reactor water level which will prompt the operator to correctly respond to these control room indicators and alarms. In addition, valves F010 and F011 will isolate the RHR system at reactor level 3. Because of all these preventive and mitigative measures, the probability of draining the RPV by this path is judged to be negligible.

19L.6.5.4 Path 4

The fuel pool isolation valves (F014 and F015) are normally closed during shutdown cooling mode. Potential path 4 is established when the fuel pool isolation valves are open (by mechanical failure or operator error). By this path, reactor water will drain into the fuel pool through a 300A diameter pipe. The preventive and mitigative features are as follows: valve F014 is equipped with a key lock; and valves F010 and F011 will isolate the system at reactor water level 3. Also the operator should correctly respond when alerted by control room alarms and indicators. Because of all these preventive and mitigative measures the probability of draining the RPV by this path is judged to be negligible.

19L.6.5.5 Path 5

Potential draining path 5 will occur if the drywell spray isolation valves (F017, F018) are opened inadvertently or fail to close during the shutdown cooling mode of operation. If this path is established, RPV water will be sprayed in the drywell through a 250A diameter pipe. The preventive and mitigative features are as follows: during shutdown cooling, with the drywell pressure low, valves F017 and F018 cannot be opened at the same time because they are interlocked such that both can be opened simultaneously only if the drywell pressure is high. The status of valves F017 and F018 is indicated in the control room. Futhermore, the isolation valves F010 and F011 will isolate on reactor level 3 and the operator monitoring the water level in the control room will take corrective actions to further mitigate this drain path. Because of these preventive and mitigative measures, the probability for draining the RPV by this path is judged to be negligible.

19L.6.5.6 Path 6

During shutdown cooling mode operation, the wetwell spray isolation valve, F019 is normally closed. If F019 is open (by operator error or mechanical failure), RPV water will be sprayed in the wetwell through a 100A diameter pipe. This event is unlikely to occur since it requires F019 to be open, the operator to incorrectly respond to control room alarms and indicators, and the failure of valves F010 and F011 to isolate the reactor from the RHR system at level 3. Because of these preventive and mitigative measures, the probability of draining the RPV by this path is negligibly low.

19L.6.5.7 Path 7

During shutdown cooling mode operation, opening of normally locked closed valves F016 (by mechanical failure or operator error) establishes drain path 7 between the RPV and the fuel pool. However, since the fuel pool is at a higher elevation than the RPV, water cannot drain from the RPV to the fuel pool when the RHR pumps are not operating, and therefore this path is not a concern for the ABWR plant.

19L.6.5.8 Path 8

Potential path 8 will occur during shutdown cooling mode of operation if the normally closed valve F001 is open (inadvertently or by mechanical failure). Path 8 will drain RPV water to the suppression pool through an 450A diameter pipe. The preventive and mitigative features in the design are as follows: both F010 and F011 are interlocked to be opened only when the RPV is depressurized, F012 is interlocked such that it cannot be opened unless F001 is closed, and similarly, valve F001 cannot be opened unless valve F012 is closed. If the RPV drain path is established, draining will stop on reactor level 3 isolation of valves F010 & F011. Also, the operator monitors the reactor level in the control room and takes corrective actions. Because of all these preventive and mitigative measures, the probability of draining the RPV through this path is judged to be negligible.

19L.6.5.9 Path 9

Path 9 has the potential to drain reactor water to the suppression pool. The minimum flow valve, F021, will automatically open when pump C001 is running and the flow through the main loop (downstream of F004 and F013) is below the low flow setpoint. The valve will automatically close when the low setpoint is reached indicating sufficient flow. Inadvertent opening of this valve will divert the flow to the suppression pool. The preventive and mitigative features in the design are as follows: valve F021 closes on receipt of normal flow signal in the main loop, the isolation valves F010 and F011 will isolate on reactor level 3 and the operator monitors the reactor water level in the control room and will take corrective actions to mitigate the event. Because of all these preventive and mitigative measures, the probability of draining the RPV by this path is negligibly low.

19L.6.5.10 Conclusion

Because of the multiple failures or operator errors required for each of the above paths to occur, and the numerous key locks, valve interlocks and control room indicators to prevent such paths, it is judged that the probability of draining the vessel during shutdown, through the RHR system is negligibly low. Furthermore, as a mitigative measure, in all cases, at reactor water level 3, valves F010 and F011 isolate the reactor from the RHR system. Even if all these safety features fail, the RPV draining will stop automatically when the RHR shutdown cooling nozzle is uncovered at which point there is still 1.7 meters of water over the top of the active fuel. It is therefore concluded that draining of RPV through failures in RHR System is not a safety concern for the ABWR plant.

19L.6.6 Summary of Reactor Pressure Vessel Draining Events

Based on a review of maintenance activities which have the potential to drain the RPV and based on a review of the operation of water systems which are connected to the RPV, it is concluded that during operating Modes 4 and 5, draining of the RPV is not a safety concern for the ABWR plant.

19L.7 Loss of Core Cooling

19L.7.1 Introduction

During operating Modes 4 & 5, with the RHR system in operation in the shutdown cooling mode, no steam is being produced in the reactor and therefore there is no need for making up reactor coolant inventory using core cooling systems. Thus loss of core cooling capability in itself is not a concern unless either the RHR system becomes unavailable causing loss of coolant inventory through evaporation or the RPV is drained. As discussed in Subsection 19L.6 the probability of draining the RPV is negligible. The remaining sequences where loss of core cooling becomes a potential concern are discussed below.

Subsection 19Q.7 contains additional information on the risk associated with loss of core cooling during shutdown.

19L.7.2 Success Criteria

Many systems continue to be available for cooling the core during operating Mode 4.

A list of core cooling systems that satisfy the core cooling success criteria are as follows:

- CRDHS, or
- HPCF B or C, or
- LPFL A, B or C, or

- 1 feedwater pump + 1 condensate pump + 1 condensate transfer pump, or
- AC-independent Water Addition System

Note that no credit is taken for the RCIC because of lack of steam in the reactor. If none of these systems are available initially, the reactor will heat up and be repressurized. If one of the high pressure make up systems is recovered, then immediate coolant makeup is possible. However, should one of the failed low pressure core cooling systems be recovered, the reactor will have to be depressurized prior to coolant injection.

The systems that satisfy the core cooling success criteria for operating Mode 5 are essentially same as those for operating Mode 4. One difference is that if none of these systems are available during operating Mode 5, the reactor will not be pressurized since the pressure vessel head has been removed. An additional difference is that if none of these sources of water is available, a flexible hose connected to the AC independent water addition system from any outside source of water can be used to cool the core since the decay heat rate diminishes substantially by the time operating Mode 5 is reached. It is thus concluded that loss of core cooling is more limiting for operating Mode 4 than for operating Mode 5. Therefore, the remainder of this review focuses on operating Mode 4.

19L.7.3 Review of Accident Sequence

The sequence of concern starts with a loss of RHR event. It is assumed that the low pressure core flooder LPFL (A, B & C) are unavailable and for core damage to occur the loss of RHR must be followed by failure of all remaining core cooling systems that meet the success criteria. Based on results of the internal event PRA, it is clear that the combined probability of failure of all systems is dominated by support system failures, especially offsite and onsite power failures. Table 19L-9 shows the dependency of the core cooling systems on power support systems. The ABWR plant technical specifications require that during operating Mode 4, at least one offsite AC power source and two diesel generators be available. In addition, the combustion turbine generator is expected to be available.

It is judged that the time window during which operating Mode 4 is most vulnerable to accidents is the first week of operation in that mode. Following that period, decay heat levels are low enough that there is a high probability of recovering a failed system. During the first week, the most dominant cut-set for core damage is expected to consist of the following basic events:

(a) Loss of offsite power during the first one week period of operating in Mode 4 with no recovery, and

(b) Failure of diesel generators, and

(c) Combustion turbine generator failure to start, and

(d) Failure of operator to initiate the AC-independent Water Addition System, and

(e) Failure of operator to recover any one of the failed systems.

The combined failure probability of all these systems is negligible, even when excluding operator failure to recover.

It is recognized that there are other cutsets that could contribute to core damage. Also, at certain times, some of the systems may be unavailable due to maintenance. (The plant technical specifications control the number of safety systems that can unavailable at any given time.) On the other hand, the above calculation takes no credit for power or equipment recovery, even though sufficient time is available. Therefore, it is judged that even after the above considerations are factored in, the combined failure probability would be negligible.

19L.7.4 Conclusion

It is concluded that loss of core cooling capability during operating Modes 4 & 5 is a negligible contributor to ABWR plant risk.

19L.8 Loss of Decay Heat Removal Events

19L.8.1 Introduction

In the ABWR internal event PRA, (Section 19.3) accident sequences were analyzed to a point where the reactor is in a condition of hot stable shutdown with the reactor mode switch in shutdown, the reactor subcritical, pressures and temperatures stabilized and within limits, containment and suppression pool cooling being maintained and vessel water level controlled. The heat removal systems were evaluated for the first 20 hours of operation. Therefore, the shutdown risk evaluation for operating Mode 4 begins at 20 hours after shutdown. Twenty hours of shutdown cooling results in a reactor coolant temperature of 294.85 K (51.7°C) or less. It takes about 2 to 3 days to reach operating Mode 5. Therefore, evaluation for operating Mode 5 starts at about 48 hours after reactor shutdown.

Subsection 19Q.7 contains additional information on the risk associated with loss of decay heat removal during shutdown conditions.

19L.8.2 Accident Initiators

The core cooling and heat removal systems are either available or in operation at the onset of operating Modes 4 and 5. (Scenarios involving failure of these systems prior to shutdown are analyzed in the internal event PRA.) This means, prior to operating Mode 4, at least 20 hours of core and containment cooling has been successfully in operation. At this point, accidents involving loss of the intended RHR heat removal function can be initiated only as follows:

- Internal failures in the RHR System, or
- Failures in the RHR support systems such as offsite and onsite power, or service water, or
- Improper operation of the RHR system (flow diversion by operator).

19L.8.3 Success Criteria

The ABWR plant features many redundant means of removing decay heat. In the internal event PRA, depending upon the sequence, credit has been taken for the following:

- Main condenser (normal heat removal path)
- RHR (3 redundant loops)
- Reactor water cleanup heat exchanger

An overpressure relief rupture disk (containment vent) has been added to the ABWR design and this can also be used to remove the containment heat under certain conditions.

The success criteria for operating Mode 4 are given in Table 19L-10. It should be noted that even though the reactor is at low pressure, main condenser and CUW heat exchangers can still be used to remove decay heat following failure of the RHR system. The overpressure relief rupture disk comes into play when the containment is pressurized following the loss of all heat removal systems.

During operating Mode 5 both the RPV and drywell heads are open and the containment is thus “vented” already. Complete failure of heat removal functions would result in initially heating the pool of water and eventually, in the worst case, boiling the water. For all practical purposes this is similar to removing the containment heat through the overpressure relief rupture disk (vent) following which the suppression pool begins to boil. In both cases water makeup to the respective pools is necessary. In other words, operating the reactor in Mode 5 can be seen as operating with a vented containment, and if heat removal functions are lost during this mode, the only action needed is to make up the water inventory lost by evaporation.

There is sufficient time available to provide the makeup water and therefore loss of RHR during operating Mode 5 is not judged to be a safety concern. Therefore, the rest of this review focuses on operating Mode 4.

19L.8.4 Review of Accident Sequence

At the start of the event, the core cooling as well as the heat removal functions are in operation. Initially, the heat is removed by the main condenser and after the reactor pressure is reduced, if the reactor is not isolated, the RHR system is engaged in the shutdown cooling mode. If the reactor is isolated, core cooling is provided by the high pressure system and the heat rejected to the suppression pool through the SRVs is removed by the RHR system in the suppression pool cooling mode. At about 20 hours into the event, with the reactor temperature at approximately 294.85 K (51.7°C), the RHR system fails as a result of internal failures, or support system failures. Loss of RHR function is the initiator. Success criteria are listed in Table 19L-10.

The probability that all these systems will fail due to unrelated problems is judged to be negligible. It is more likely that these systems will fail as a result of failures in the support systems. Table 19L-11 shows the power related support systems for the systems listed in the success criteria. The ABWR plant technical specifications require that during operating Mode 4, at least one offsite AC power source and two diesel generators be available. In addition, the combustion turbine generator is expected to be available. The most likely accident initiator is the loss of offsite power. If power is not recovered in time (say 24 hours), and the diesel generators and the combustion turbine generator fail to start, then the only heat removal system available is the overpressure relief rupture disk.

The combined failure probability of this event sequence is negligible. It is recognized that this analysis does not include all the failure paths and does not account for equipment that are unavailable due to maintenance. On the other hand, it should also be noted that failure of heat removal function does not automatically lead to core damage as has been assumed above. Only a fraction of these sequences lead to core damage as would be evident if detailed containment event trees were developed. On balance, it is concluded that the probability of core damage, resulting from a loss of containment heat removal function during operating Mode 4 is negligible. It has also been identified that no problems are anticipated during operating Mode 5 as long as the water evaporated by boiling is periodically made up. Thus, in summary, it is concluded that loss of containment cooling function during operating Modes 4 and 5 pose a negligible threat to the ABWR plant safety.

19L.9 Noncore-Related Accidents

19L.9.1 Introduction

Many noncore-related accidents can be postulated during operating Modes 4 & 5. However, it is judged that the consequences of any accident that does not involve fuel bundles is negligible. Thus for instance, drainage of the radwaste tank is not considered in this study. Accidents considered here are the fuel bundle drop accident, spent fuel cask drop accident, loss of fuel pool cooling, and drainage of fuel pool.

19L.9.2 Fuel Drop Accident

The fuel handling accident can only be assumed to occur as a sequence of failures in the fuel assembly lifting mechanisms which will result in the dropping of a fuel bundle and the subsequent release of radioactive materials from damaged rods. A detailed probabilistic analysis of such an accident was not performed based upon the following considerations.

(1) The probability for the failure of mechanisms involved in fuel handling either through mechanical failure or human action is assumed very small based upon the small number of cases seen to date throughout the nuclear industry in the handling of literally thousands of fuel bundles. Therefore, initially the probability of a fuel bundle being dropped is very small.

(2) Given the occurrence of a fuel bundle being dropped, the radiological consequences depend upon the distance of fall, the angle of impact, and the surface onto which the bundle would fall. For the exposure time during which any fuel bundle is being moved from point A to point B, the potential consequences are a function of probability involving distance of fall, type of bundle being dropped (exposed or fresh), and surface onto which the bundle can fall (steel, concrete, other bundles and their exposure history).

(3) Based upon the reasoning in paragraph 2 above and upon current operating experience, the more probable fuel drop events result in damage to no to a few rods (less than 10). Considering the factor of radioactive decay prior to handling exposed fuel, the use of safety systems (the failure of which would reduce the potential accident probability), volatility and migrability of the fission products through water pools and potential in plant transport analysis, maximal whole body and thyroid doses less than a few tenths of a milliRem at extremely low probabilities could be expected at the site boundary.

(4) Given releases for larger events at lower probabilities and the factors above, doses up to one millirem at even lower probabilities are estimated.

It is therefore concluded that this accident is a negligible contributor to ABWR plant risk.

19L.9.3 Spent Fuel Cask Drop Accident

The spent fuel cask drop accident is discounted as a credible accident based upon the following logic.

(1) The probability of dropping a spent fuel cask during handling is extremely low due to the mechanical interlocks and safety systems used. During handling the cask is moved via a type 1 crane with redundant rigging with both procedural and mechanical interlocks to prevent movement of the cask over areas such as the spent fuel pool.

(2) During handling from the cask loading pit to the cleaning pit the cask lid is in place and the height of lift is limited such that a fall would not result in any significant damage to the cask and no damage to the cask contents. The cask is sealed in the cleaning pit and given that a drop occurs over the hatch in transient to the loading dock, the maximum fall would not be expected to result in sufficient impact to damage the cask based upon cask design requirements from DOE.

(3) Even assuming potential releases from a cask, the minimum time for fuel movement is generally one year after removal from the core which results in the decay of all volatile isotopes except Kr-85. Owing to Kr-85's low gamma energy, such a release would result in doses which are less than 0.1E–06 Sv or accidents of probability on the order of or less than 1.0E–06 at the site boundary.

It is therefore concluded that this accident is a negligible contributor to ABWR plant risk.

19L.9.4 Loss of Fuel Pool Cooling

In the ABWR plant, the fuel pool cooling and cleanup FPC system is backed up by the RHR system (2 of the 3 loops). Therefore, fuel pool cooling function is lost only if both the FPC and two loops of RHR system become unavailable. Even if these systems become unavailable, adequate time will be available for repairs to be made to restore the failed systems before fuel damage occurs. Providing makeup water alone will mitigate the accident and many sources of water exist including fire or potable water. The combined probability of loss of FPC and RHR and failure to repair the failed system or provide makeup water is judged to be negligible and therefore it is concluded that this event is a negligible contributor to ABWR plant risk.

19L.9.5 Drainage of Fuel Pool

FPC system is designed with no piping penetrations or drain paths which can drain the fuel pool. Further, there are no potential paths for siphoning water from the pool. Thus it is impossible to drain the pool inadvertently. For fuel pool drainage to occur, the pool liners must fail causing leakage of water from the pool. A postulated means of damaging the liners is dropping of a heavy load, such as the fuel transfer cask in the fuel pool. In WASH-1400 (Reference 19L-2) the probability of draining the pool by this postulated accident was estimated to be negligible. The WASH-1400 analysis is judged to be applicable for ABWR also, and it is therefore concluded that this event is a negligible contributor to ABWR plant risk.

19L.10 References

19L-1 Dino C. Scaletti, "Summary of Meetings with GE on ABWR—item 4", USNRC, June 8, 1990.

19L-2 "Reactor Safety Study—An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants", WASH-1400 (NUREG-75/014), USNRC, October 1975.

19L-3 "Zion Nuclear Power Plant Residual Heat Removal PRA", NSAC/84, NSAC/EPRI, July 1985.

19L-4 "Seabrook Station Probabilistic Safety Study—Shutdown (Modes 4, 5 and 6)", New Hampshire Yankee, May 1988.

19L-5 A. Villemeur, et al. (Electricite de France), "Living Probabilistic Safety Assessment of a French 1300 MWe PWR Nuclear Power Plant Unit: Methodology, Results and Teachings", Published at TUV-Workshop on Living PSA Application, Hamburg, FRG, May 7-8, 1990.

19L-6 "Advanced Light Water Reactor Utility Requirements Document, Volume II, Chapter 1, Appendix A: PRA Key Assumptions and Ground Rules", Draft, EPRI, August 1988.

19L-7 "Recommended Technical Specifications for Fuel Loading", service information letter No. 372, General Electric, June 1982.

Table 19L-1 ABWR Modes of Operation

Mode*	Title	Reactor Mode Switch Position	Average Reactor Coolant Temperature, K (°C)
1	Power Operation	Run	Any temperature
2	Startup	Startup/Hot Standby	Any temperature
3	Hot Shutdown	Shutdown	>366.45 K (> 93.3°C)
4	Cold Shutdown	Shutdown	≤366.45 K (≤ 93.3°C)
5	Refueling	Shutdown or Refuel	≤366.45 K (≤93.3°C)†

* In Modes 1 through 4, fuel is in the reactor vessel with the reactor vessel head closure bolts fully tensioned. In Mode 5, fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

† Technical specification states “any temperature”, but in this mode the temperature will be below boiling point.

Table 19L-2 Control Rod Drop Accident

Cause/Event Hardware	Cause/Event Operator	Preventive and Mitigative Features
Failure Mode 1		
1. —	Two control rods withdrawn for test.	—
2. A third adjacent rod sticks (still coupled to hollow piston)	—	—
3. —	The third control rod is withdrawn	Interlock prevents withdrawal of the third control rod
4. Separation of ballnut and hollow piston in the third control rod	Operator misses alarm & continues withdrawal of the third control rod	Class 1E separation detection
5. The third control rod unsticks and drops	—	Rod block + hollow piston latch
Failure Mode 2		
1. —	Two control rods withdrawn for test.	—
2. A third adjacent control rod sticks	—	—
3. —	The third control rod is withdrawn	Interlock prevents withdrawal of third control rod
4. Rod to hollow piston separation occurs in the third control rod	Operator misses alarm & continues withdrawal of the third control rod	1. Positive bayonet coupling 2. Class 1E separation detection
5. The third control rod unsticks and drops	—	Rod block
Failure Mode 3		
1. —	Two control rods withdrawn for test	—
2. —	A third adjacent control rod is installed without coupling	—
3. —	Error in the third control rod not detected during coupling check	
4. —	The third control rod withdrawn	Interlock prevents withdrawal of third control rod

Table 19L-2 Control Rod Drop Accident (Continued)

Cause/Event: Hardware	Cause/Event: Operator	Preventive and Mitigative Features
5. The third rod sticks	—	—
6. Rod to hollow piston separation in the third control rod	Operator misses alarm & continues withdrawal of the third control rod	Class 1E separation detection
7. The third control rod unsticks and drops	—	Rod block

Table 19L-3 Control Rod Ejection Accident

Cause/Event: Hardware	Cause/Event: Operator	Preventive and Mitigative Features
Failure Mode 1		
1. Reactor under hydro test	—	This occurs during a small fraction of time during shutdown
2. —	Two control rods withdrawn for testing	—
3. Break in the adjacent FMCRD housing or weld between housing and vessel or CRD mounting bolts or CRD spool piece	None	Integral internal blowout support (“shootout restraints”)
Failure Mode 2		
1. Reactor under hydro test	—	This occurs during a small fraction of time during shutdown
2. —	Two control rods withdrawn for testing	—
3. Break of insert pipe in the adjacent CRD	None	1. Ball check valve in insert port 2. FMCRD electro-mechanical brake

Table 19L-4 Refueling Error

Cause/Event		Preventive and Mitigative Features
Hardware Failure	Operator Error/Action	
1. —	Utility plans to offload all fuel bundles or perform multiple control blade shuffles	No incentive for unloading all fuel bundles because very few FMCRDs need to be maintained during refueling
2. —	One CRD removed	
3. —	Adjacent CRD removed	Interlock prevents withdrawal of second CRD
4. —	Operator starts loading the fuel bundles, the last bundle is lowered into the empty uncontrolled fuel cell	Automatic refueling machine interlocked to prevent hoisting a fuel assembly over the vessel

Table 19L-5 Potential for Draining RPV During RIP Maintenance

Cause/Event		Preventive and Mitigative Features
Activity	Cause	
Replacement of RIP motor	Potential leakage path from RPV to outside due to pressure difference	1. Impeller backseats to prevent leak 2. Inflatable seal provides backup seal
Replacement of RIP impeller	Same as above	1 & 2. Same as above since initially the motor is removed 3. Temporary motor cover plate is bolted 4. Impeller removal results in the loss of the pump shaft backseat seal, but impeller diffuser cap is inserted in the impeller cavity to provide additional protection
Maintenance on inflatable seal	Same as above	1. Plug over RIP nozzle inside RPV prevents draining

Table 19L-6 Potential for Draining RPV Through Control Rod Drive Hydraulic System at Shutdown

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
1A	Both CRD pumps + CRD ball check valve + HCU maintenance isolation/drain valves (F101, F140, F113) + Check valve (F115)		Drain RPV water into CRDHS	Pump required to run continuously Multiple failures necessary Potential draining pipes are only 16 A each allowing sufficient time for mitigation
1B	CRD ball check valve + Check valve (F115)	Both pumps off + HCU maintenance isolation/drain valves (F101, F140, F113) left open	See 1A	See 1A
2A	Both CRD Pumps + CRD Ball Check Valve + HCU maintenance isolation valves (F101, F140) + Scram valve closed + Test port valve open (F141) + Test equipment		RPV water leaks into HCU environment	See 1A
2B	CRD ball check valve + Scram valves closed	Both pumps off + HCU maintenance isolation valves (F101, F140) open + Test port valve (F141) open + No test fixture in test port	See 2A	See 1A

Table 19L-6 Potential for Draining RPV Through Control Rod Drive Hydraulic System at Shutdown (Continued)

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
3A	Both CRD pumps + CRD ball check valve + HCU maintenance isolation/drain valves (F101, F140, F104) + Scram valve closed + check valve (F138)		See 1A	See 1A
3B	CRD ball check valve + Scram valve closed + Check valve F138	Both pumps off + HCU maintenance isolation valves (F101, F140, F104) left open	See 1A	See 1A

Table 19L-7 Potential for Draining RPV Through Reactor Water Cleanup System

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
1A	Valve F500 fails open + Valve F501 fails open		Reactor water drains to low conductivity waste sump	Drain line is only 50 A diameter Leak detection alarm in main control room Valves are redundant Operator monitors reactor water level in the control room and takes corrective action
1B		Valve F500 open + Valve F501 open	See 1A	Valve F500 under key lock + administrative control + valves are redundant See 1A
2 & 3 A	Valve F055A fails open or Valve F055B fails open		RPV water drainage into reactor building	Drain Line is only 50 A diameter Leak detection alarm in main control room Path terminates on reactor Level 3 isolation signal Operator monitors reactor water level in the control room and takes corrective action
2 & 3 B		Valve F055A open or Valve F055B open	See 2A	See 2A

Table 19L-7 Potential for Draining RPV Through Reactor Water Cleanup System (Continued)

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
4A	Valve F022 fails open + Valve F024 fails open + Valve F025 fails open		RPV water drainage to suppression pool	Redundant (3) valves CUW isolation on high flow Control room indicator of valve status Path terminates on reactor level 3 isolation signal Operator monitors reactor water level in the control room and takes corrective action
4B		Valve F022 open + Valve F024 open + Valve F025 open	See 4A	See 4A
5A	Valve F022 fails open + Valve F023 fails open		RPV water drainage to LCW collector tank	Redundant (2) valves CUW isolation on high flow Path terminates on reactor Level 3 isolation signal Operator monitors reactor water level in the control room and takes corrective action
5B		Valve F022 open + Valve F023 open	See 5A	See 5A

Table 19L-7 Potential for Draining RPV Through Reactor Water Cleanup System (Continued)

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
6A	Valve F056 fails open		See 2A	See 2a
6B		Valve F056 open	See 2A	See 2A
7A	Valve F022 fails open + Valve F025 fails open		See 4A	See 4A
7B		Valve F022 open + Valve F025 open	See 4A	See 4A

Table 19L-8 Potential for Draining RPV Through Residual Heat Removal System

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
1A [A] =	[Pump C001 running + Valve F011 open + Valve F010 open + Valve F012 open + Valve F026 fails open]		Drain RPV water to HCW	1. F010 & F011 isolation on reactor level 3 2. Operator monitors level in control room and takes corrective action 3. Drain line is only 50 A diameter allowing sufficient time for corrective action
1B		[A] + Valve F026 inadvertently opened	See 1A	See 1A
2A	[A] + Valve F029 fails open + Valve F030 fails open		Drain RPV water to radwaste 150 A diam. pipe	1. Requires multiple valve failures/openings 2. Indicators in control room will show F029 and F030 open 3. F010 & F011 will isolate on reactor level 3 4. Operator monitors level in control room and takes corrective actions
2B		[A] + Valve F029 inadvertently open + Valve F030 inadvertently open	See 2A	See 2A

Table 19L-8 Potential for Draining RPV Through Residual Heat Removal System (Continued)

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
3A	[A] + Valve F008 fails open		Drain RPV water to suppression pool	1. Valve interlock between F008 + F012 2. Indicators in control room will show F008 open 3. F010 & F011 will isolate on reactor level 3 4. Operator monitors level in control roon and takes corrective action
3B		[A] + Valve F088 inadvertently opened	See 3A	See 3A
4A	[A] + Valve F014 fails open + Valve F015 fails open		Drain RPV water to fuel fool via 300 A diameter pipe	1. Requires multiple valve failures/openings 2. F014 is key locked 3. Indicators in control room show F014 and F015 open 4. F010 & F011 will isolate on reactor level 3 5. Operator monitors level in control room and takes corrective actions
4B		[A] + Valve F014 inadvertently opened + Valve F015 inadvertently opened	See 4A	See 4A

Table 19L-8 Potential for Draining RPV Through Residual Heat Removal System (Continued)

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
5A	[A] + Valve F017 fails open + Valve F018 fails open		Drain RPV water to drywell via spray through 250 A line	1. Requires multiple valve failures/openings 2. F017 and F018 interlocked such that both can be opened simultaneously only if the drywell pressure is high 3. Indicators will show F017 and F018 open 4. F010 & F011 will isolate on reactor level 3 5. Operator monitors level in control room and takes corrective actions
5B		[A] + Valve F017 inadvertently opened + Valve F018 inadvertently opened	See 5A	See 5A
6A	[A] + Valve F019 fails open		Drain RPV water to wetwell spray via 100 A diameter pipe	1. Requires valve failure/opening 2. Indicators in control room will show F017 open 3. F010 & F011 will isolate on reactor level 3 4. Operator monitors level in control room and takes corrective actions

Table 19L-8 Potential for Draining RPV Through Residual Heat Removal System (Continued)

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
6B		[A] + Valve F019 inadvertently opened	See 6A	See 6A
7A	[A] + Valve F016 fails open		Drain RPV water to fuel pool via 300 A diam. pipe	1. F016 is a locked closed manual valve 2. F010 & F011 will isolate on reactor level 3 3. Operator monitors level in control room and takes corrective actions
7B		[A] + Valve F016 inadvertently opened	See 7A	See 7A
8A	[A] + Valve F001 fails open		Drain RPV water to suppression pool via 450 A diameter pipe	1. Valve interlock between valves F001 and F012 2. F010 & F011 will isolate in reactor level 3 3. Operator monitors level in control room and takes correction actions
8B		[A] + Valve F001 inadvertently opened	See 8A	See 8A

Table 19L-8 Potential for Draining RPV Through Residual Heat Removal System (Continued)

Path	Equipment Failure	Operator Error	Results	Preventive/Mitigative Measures
9A	Minimum flow valve F021 opens during low flow during shutdown cooling mode		Reactor water is diverted to suppression pool	1. F021 closes on nominal flow signal in the shutdown cooling mode 2. F010 and F011 will isolate the reactor level 3 3. Operator monitors level in control room and takes corrective actions
9B		Operator inadvertently opens minimum flow valve F021 during shutdown pool cooling mode	See 9A	See 9A

Table 19L-9 Dependency of Core Cooling Systems on Electrical Power

System	Power Systems: Offsite Power	Combustion Turbine	DG1	DG2	DG3	Div 1 DC	Div 2 DC	Div 3 DC	Diesel Driven Fire Water Pump
RCIC						XX			
HPCF (B)	OR	OR		OR			XX		
HPCF (C)	OR	OR			OR			XX	
FW (A)	OR	OR	OR*	OR*	OR*				
FW (B)	OR	OR	OR*	OR*	OR*				
FW (C)	OR	OR	OR*	OR*	OR*				
CRD (A)	OR	OR	OR*	OR*	OR*				
CRD (B)	OR	OR	OR*	OR*	OR*				
LPFL (A)	OR*	OR	OR			XX			
LPFL (B)	OR*	OR		OR			XX		
LPFL (C)	OR*	OR			OR			XX	
Firewater†	OR*	OR	OR*	OR*	OR*				OR
Condensate									
(A)	OR	OR	OR*	OR*	OR*				
(B)	OR	OR	OR*	OR*	OR*				
(C)	OR	OR	OR*	OR*	OR*				
(D)	OR	OR	OR*	OR*	OR*				

* Assumes manual feedback capability for combustion turbine distribution system

† AC-independent water addition system

Notes:

DG1 - Diesel generator 1

FW - Feedwater

LPFL - Low pressure core flooder

OR - Redundant supply to other ORs

XX - Loss of this power supply means loss of system

Table 19L-10 Success Criteria for Long-Term Heat Removal for Operating Mode 4

Function	Success Criteria
Containment heat removal during operating Mode 4	RHR-A or B or C* or Normal heat removal using main condenser† or Reactor water cleanup‡ or Overpressure relief rupture disc*f*

* RHR can be operated in either the suppression pool cooling or the shutdown cooling mode. Shutdown cooling requires the reactor to be at low pressure.

† Reactor will have to be pressurized and MSIVs opened for establishing this path.

‡ Reactor may have to be pressurized to use the CUW system efficiently to remove decay heat or reactor water could be drained to the main condenser hotwell through the CUW system and reactor water makeup obtained from HPCF, feedwater, CRD hyrdraulic system, or the AC independent water addition system.

f Reactor will have to be pressurized and heat transferred to the suppression pool through safety/relief valves. Long-term suppression pool makeup will be required to compensate for water lost through evaporation and reactor water makeup must be obtained from any of the methods indicated in Note ‡ above.

Table 19L-11 Dependency of Heat Removal Systems on Electrical Power

System	Offsite Power	Combustion Turbine	DG1	DG2	DG3	Div 1 DC	Div 2 DC	Div 3 DC
RHR (A)	OR	OR	OR			XX		
RHR (B)	OR	OR		OR			XX	
RHR (C)	OR	OR			OR			XX
CUW (A or B)	OR	OR	OR*	OR*				
Overpressure Relief†								
Main Condenser	XX							

* Assumes feedback capability for combustion turbine distribution system.

† Does not need power source for operation. Also, the function provided by the overpressure relief can be provided by operator opening one of the containment doors.

Notes:

DG1 - Diesel generators

OR - Redundant supply to other ORs

XX - Loss of this power supply means loss of system

Table 19L-12 ABWR Seismic PRA: Highest Class I Accident Frequency Sequences

Seq. No.	Structural Integrity	Offsite Power	Onsite Power or Service Water	SRV	Scram	ADS Inhibit	Stuck Open Relief Valve	Flow Control	RCIC	HPCF	ADS	LPFL	Fire Water	RHR	Frequency*
	Failure Events														
1		X	X		X				X						
2		X	X										X		
3	X														
4		X	X						X				X		
5		X	X		X								X		
6		X	X								X				
7		X	X		X				X						
8		X							X	X		X			
9		X		X	X										
10		X	X						X		X				
11	X													X	
12		X	X							X		X	X	X	
13		X			X	X	X			X		X			
14		X	X						X				X	X	
15		X	X		X			X						X	

* Not part of DCD (Refer to SSAR).

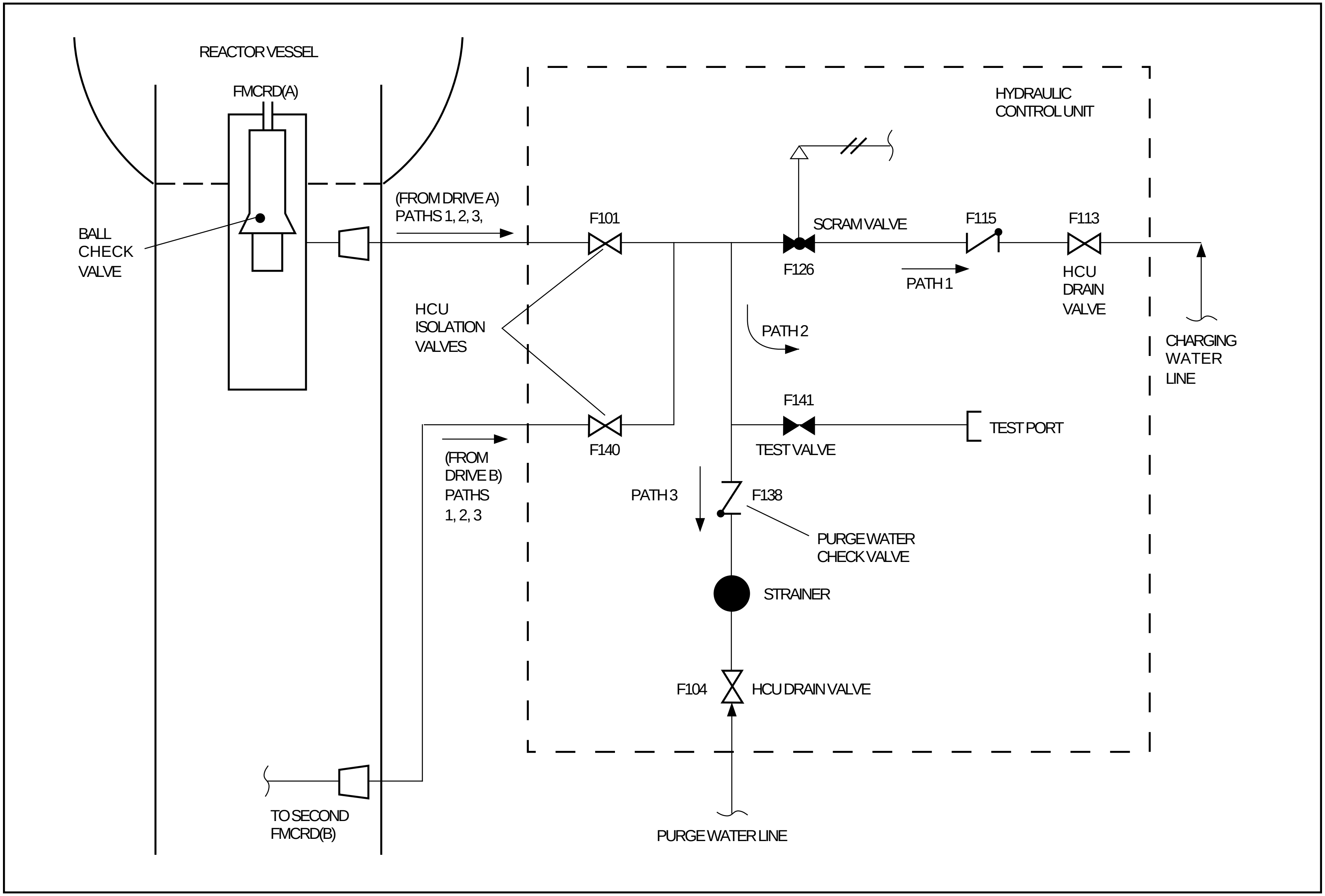


Figure 19L-1 Potential Paths for Draining RPV Through Control Rod Drive Hydraulic System

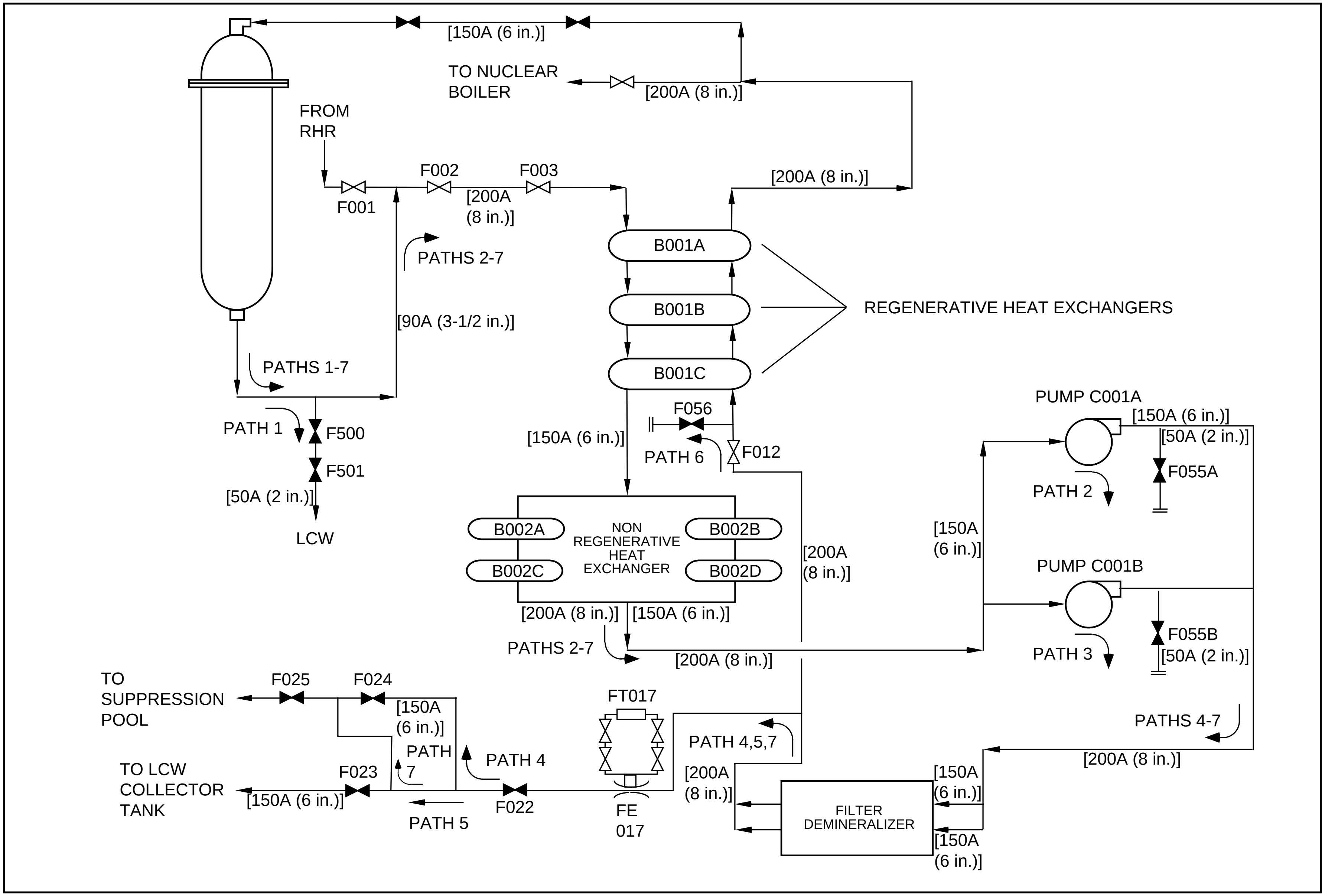


Figure 19L-2 Potential Path for Draining RPV Through Reactor Water Cleanup System

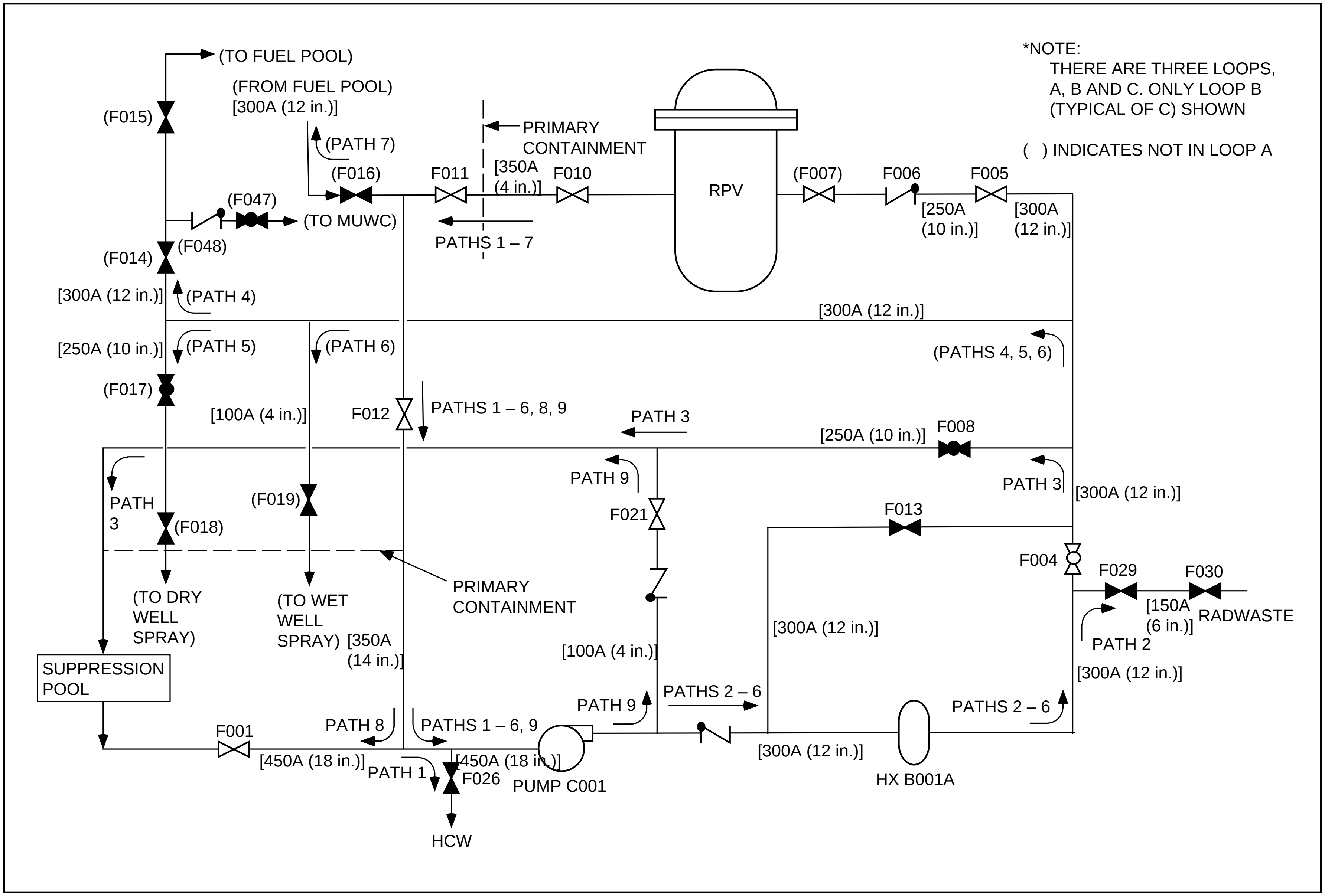


Figure 19L-3 Potential Path for Draining RPV Through Residual Heat Removal System (Pump On)

19M Fire Protection Probabilistic Risk Assessment

19M.1 Introduction

As part of the Advanced Boiling Water Reactor (ABWR) design certification process, the USNRC requested that General Electric expand upon earlier considerations of the subject of fire risk. Through discussions with the NRC it was mutually agreed that a fire screening analysis approach was appropriate. It was further agreed that the Fire Vulnerability Evaluation (FIVE) Methodology Plant Screening Guide (Reference 19M-1) being developed by the Electric Power Research Institute (EPRI) provided an appropriate vehicle for performing this analysis.

The FIVE methodology provides procedures for identifying fire compartments for evaluation purposes, defining fire ignition frequencies, and performing quantitative screening analyses of fire risk. The criterion for screening acceptability is that the risk of core damage from any postulated fire be less than an acceptably small criterion. Any fire scenarios not meeting this criterion require more detailed consideration.

Five bounding fire scenarios and corresponding ignition frequencies were developed on the basis of the FIVE methodology. The first three of these consider the impact of fires which incapacitate each of the three divisions of emergency power, and thus the ECCS equipment which is dependent on each for successful performance. The fourth scenario considers the impact of a fire in the control room with the assumption that the only ECCS functions available are those that can be controlled and operated from the remote shutdown panel, and the RCIC, which can be manually operated outside of the control room. The fifth and final scenario examines the consequences of a fire in the turbine building based upon the assumption that resulting loss of off-site power bounds the possible outcomes of this initiator.

19M.2 Basis of the Analysis

This analysis is prepared with Figures 5.1 and 6.0 and related text sections from the FIVE Methodology Draft Report as the basis. In performing this analysis the ABWR was broken into three major groupings as follows:

(1) A safety-related building grouping consisting of the reactor building except primary containment, control building except the control room complex, and the intake structure. This grouping contains all of the equipment required for safe shutdown except that within primary containment and the control room complex. The buildings are subdivided by three hour rated fire barriers into fire areas corresponding to the safety divisions. Each division is considered as a unit, although each division encompasses several fire areas in three buildings. For these groupings, it is conservatively assumed that a fire at any location in a divisional fire area results in the immediate loss of function of the

division. This precludes having to calculate the rate of spread and possible magnitude of a fire within a fire area. The requirement that the fire containment system be capable of confining any fire within the fire area of origin is documented in Subsection 9.5.1.

(2) Control room complex—The control room complex contains safety-related equipment from all four divisions in a single fire area and therefore must be uniquely analyzed. The redundant system to the control room is the remote shutdown panel. For the purposes of this analysis, remote manual operation of the RCIC system is also included as a method of mitigation.

(3) Turbine building—As documented in Subsection 9A.5.5.1, fire induced failure of the small amount of safety-related sensors located in this building cannot prevent safe shutdown of the plant. The turbine building is included in the analysis because a turbine building fire could result in a plant shutdown concurrent with a loss of off site power.

19M.3 Summary of Results

All three major groupings were determined to be “Significant Fire Areas” by the screening procedures outlined in FIVE Methodology Figure 5.1, which is included as Figure 19M-1. They were then screened out by the Step 2 path of the procedures outlined in FIVE Figure 6.0, which is included as Figure 19M-2, on the basis that:

(1) The product of the Fire Ignition Frequency and the probability that the redundant or alternate systems would not be available was less than the acceptance criteria.

(2) The redundant or alternate systems for which credit was taken are in fire areas other than the one experiencing the fire and therefore the fire cannot affect the redundant or alternate systems. Fire areas are separated by three hour fire rated barriers.

A summary of the results of the analysis is given in Table 19M-1. Note that the core damage frequencies per year are less than the acceptance criteria for all cases. Originally, the remote shutdown panel included controls for just three safety-relief valves. For this configuration, the core damage frequency for a fire in the control room was greater than the acceptance criteria, even though credit was taken for local manual operation of the RCIC (See Subsection 19M.7 for COL license information). A control switch for a fourth SRV was added to the remote shutdown panel. This dropped the probability of core damage for a control room fire to less than the acceptance criteria. This is considered a very conservative estimate because of the conservative assumptions that a fire in one area disables all potentially affected equipment. Taking credit for the distance between fire sources and targets would reduce the core damage probability to a fraction of the calculated low probability. For this reason, it was judged not appropriate to add these results to other core damage frequencies estimated elsewhere in Chapter 19.

The analyses required to calculate the fire ignition frequencies and combine them with the PRA models are included in Subsections 19M.4 through 19M.6.

19M.4 Phase I Scenario and Phase II Fire Frequency Analysis

This analysis is prepared with Figures 5.1 and 6.0 from the FIVE Methodology Draft Report as the basis. Subsections of the analysis bear the titles of the applicable FIVE Methodology sections or the applicable blocks from the FIVE Methodology figures, flagged with "(FIVE)" for identification. The subsections detail the method of compliance or source of information requested by the block.

19M.4.1 Phase I Qualitative Analysis (FIVE)

19M.4.1.1 Identify Plant Fire Areas (FIVE, Use Table 1 Matrix)

For the purpose of this analysis the ABWR has been broken into three major groupings as follows:

(1) A safety-related building grouping consisting of the reactor building except primary containment, control building except the control room complex, and the intake structure. This grouping contains all of the equipment required for safe shutdown except that within primary containment and the control room complex. The buildings are subdivided by rated fire barriers into fire areas corresponding to the safety divisions. Each division is considered as a unit, although each division encompasses several fire areas in three buildings. For these groupings, it is assumed that a fire at any location in a divisional fire area results in the immediate loss of function of the division.

The assumption of the immediate loss of function for the division is a conservative adaptation of the FIVE methodology and makes it unnecessary to calculate the rate of fire growth and spread within a fire area, provided the results of the analysis confirm that the probability of the redundant/alternate systems being unavailable is less than the acceptance criteria.

(2) Turbine building—As documented in Subsection 9A.5.5.1, fire induced failure of the small amount of safety-related sensors located in this building cannot prevent safe shutdown of the plant. The turbine building is included in the analysis because a turbine building fire could result in a loss of off site power and/or a plant shutdown.

(3) Control room complex—The control room complex contains safety-related equipment from all four divisions in a single fire area and therefore must be uniquely analyzed. The redundant system to the control room is the remote shutdown panel. For the purposes of this analysis, remote manual operation of the RCIC system is also included as a method of mitigation.

Primary containment was determined to not be a significant fire area because:

(1) It is inerted during plant operation.

(2) A fire in containment cannot prevent safe shut- down of the plant. (Subsection 9.5.1.0.2.)

(3) The containment spray system could serve as a fire suppression system if the need did arise.

(4) The FIVE Analysis excluded the containment.

The equipment in primary containment was included in the probabilistic failure models as required to support safe shutdown, however.

Fire in non safety-related buildings other than those listed above was not considered as the buildings are separated from the equipment required for safe shutdown by three hour rated fire barriers. Therefore, fire in the non safety-related buildings cannot prevent safe shutdown.

A table of fire areas for the safety related buildings (reactor and control) is provided as Table 9A.6-1. The fire areas listed on the table are as shown on the fire area separation drawings, Figures 9A-1 through 9A-18. Except as described above for the control complex and primary containment, fire areas are assigned to specific safety divisions.

As stated above, screening of the fire areas is on a grouped basis. A fire in any one of the grouped fire areas is assumed to result in the immediate loss of function for the equipment of the division of the grouped area, and screening is on that basis. For example, all of the Division 1 fire areas are screened as a group because it is possible for a fire to occur at any location in the Division 1 fire area and the fire is assumed to result in immediate failure of Division 1 for purposes of evaluating the effects on safe shutdown.

The relative location of the fire areas has been specified such that there should be no reason for the detail designer to route piping or cable trays of non-conforming divisions though divisional fire areas. It is an interface requirement (Subsection 9.5.13.12) that the utility confirm that the routing of piping and cable trays during the detailed design phase conforms with the fire area divisional assignment documented in the fire hazard analysis.

19M.4.1.2 List Safe Shutdown Systems (FIVE, Use Table 1 Matrix)

In this conservative analysis, credit is taken for only safety-related systems for accomplishing safe shutdown. Not all safety-related systems are safe shutdown systems. A differentiation is not made, however, as a fire involving a safety-related system not required for safe shutdown could result in the loss of a divisional power supply common to the safety-related system required for safe shutdown. Rather than analyzing all possible interactions between systems within a division, the worst case is assumed. Therefore, damage to any safety-related system is considered to have the possibility of affecting a system required for safe shutdown during a fire situation. Safety-related equipment is identified by a 1, 2, 3, or 4 in the "Electrical Division" column of Table 9A.6-1. The system identification is included in the master parts list number shown on the table.

The acceptability of possible spurious operation (e.g. motors randomly starting or stopping, valves randomly opening and closing, etc.) was also addressed and found acceptable as a result of the analysis for uncontrolled acts by an inside saboteur. The results of the study is provided in Appendix 19C.

19M.4.1.3 Identify Safe Shutdown Systems in Each Fire Area (FIVE, Use Table 1 Matrix)

The devices are sorted by system for each room on Table 9A.6-2. The safety division and room number are also shown on the table. This allows cross comparison of the table and the fire separation drawings to determine what systems are in each fire area. A column listing the fire area for each device will be added to the table so that cross reference to the fire protection drawings is not required to determine the fire area for each piece of equipment.

19M.4.1.4 Shutdown Equipment In Fire Area (FIVE)

If there is safety-related equipment in any fire area, it is assumed that either the equipment is required for safe shutdown or its loss by fire could affect equipment required for safe shutdown. The answer for this block is assumed to be "yes" if there is safety-related equipment in the area.

19M.4.1.5 Fire Causes Demand For Safe Shutdown Equipment (FIVE)

It is assumed that accomplishment of safe shutdown must be possible with a fire at any location in the plant. This block is always "yes".

19M.4.1.6 Significant Fire Areas (FIVE)

The above screens confirm that the reactor building except primary containment, the control building except the control room complex, the intake structure, the turbine building and the control room complex should be termed to be "significant fire areas".

They must be subjected to the screening depicted by Figure 6.0 of the FIVE Methodology.

19M.4.2 Phase II Quantitative Analysis (FIVE)

19M.4.2.1 Identify Fire Compartments For Evaluation Purposes (FIVE)

The fire areas established in the ABWR design meet the requirements for fire areas as defined in section 2.2 of the FIVE Methodology Draft Report. The ABWR fire protection design is on the basis of separation being on a fire area basis. Conservatively, credit is not taken for the lesser separation allowed by the fire compartment definition of section 2.4 of the FIVE Methodology Draft Document. Most ABWR fire areas encompass more than one room. The separation between rooms within a fire area is similar to that required for fire compartments in the FIVE methodology. Separation within a fire area by room does tend to limit the consequences of a fire within a fire area, but no credit is taken for this in the ABWR analysis.

To summarize, screening of the fire areas is by a grouping of the intake structure, reactor building and control building except primary containment and the control room complex; the turbine building; and the control room complex. The fire areas of each safety division external to the control room complex and primary containment are considered as a group. A fire in any one of these grouped areas is assumed to result in immediate loss of function for the equipment in the grouped area and screening is on that basis.

19M.4.2.2 Evaluate Fire Vulnerability Frequency (FI) For Fire Compartment (FIVE)

This evaluation is done by the FIVE methodology on the basis of grouped fire areas and not fire compartments.

19M.4.2.3 Determine Fire Ignition Frequency (FI) (FIVE) [Figure 6.3.1.2]

Figure 6.3.1.2 of the FIVE report is used and the results are entered in the appropriate locations on Table 1. See Subsection 19M.5.2 for the calculations for the fire ignition frequency. There are no significant fire areas with a fire ignition frequency less than 1E-6 per year.

19M.4.2.4 Choose (FIVE)

The choice is always for Step 2. The ability of the plant to accommodate the complete burnout of any fire area without recovery is a design requirement and is always assumed. Step 3, separation of redundant safety-related systems by less than a rated fire barrier, is not available for a new plant design.

19M.4.2.5 Probability For Redundant/Alternate System Unavailable (FIVE)

The Level 1 PRA models are used to determine the probability of failure of the redundant systems external to the grouped fire area. The PRA models were combined with the fire ignition frequency for the calculation. The results of the analysis are entered in the appropriate locations on Table 19M-1.

All significant fire areas have a fire induced core damage frequencies of less than the acceptance criteria.

19M.4.2.6 Can Fire Affect Redundant/Alternate Path (FIVE)

It is concluded that a fire cannot affect the redundant/alternate paths in other divisional fire areas since only physical separation by rated fire barriers (as described in Subsection 9.5.1 and confirmed in the fire hazard analysis, Appendix 9A) is relied upon. All fire areas screen out.

19M.5 Calculation of the Fire Ignition Frequency

19M.5.1 General Comments On Completion Of FIVE Table 3

- Fire Compartment Boundaries: (FIVE)

  All boundaries are three hour rated structures such as walls, ceilings, floors and doors. All penetrations are closed by penetrations with a fire rating equal to the rating of the structure penetrated.

- Inside Fire Area: (FIVE)

  No general comments.

Fire Ignition Frequency

- Step 1.1 (FIVE)
  - Selected Fire Location (FIVE, Table 1.1)

    No general comments.

- Step 1.2 (FIVE)
  - Location Weighting Factor (WF_L) (FIVE, Table 1.2)

    The location weighting factors for the ABWR are summarized on Table 19M-2. The first two columns of the table were copied from Table 1.1 of the FIVE methodology. The third and fourth columns apply specifically to the ABWR. The rationale for determining each factor is stated on the table.

Division 4 is not included in the table because a very small fraction of the plant fire areas are Division 4 areas. Division 4 only provides additional instrumentation and control logic in support of the other three divisions which contain all of the safety-related depressurization, core cooling and containment heat removal capacity. Loss of Division 4 can therefore only have an impact which is a fraction of the loss of any one of the other 3 safety-related divisions.

Some equipment which was in the reactor building in operating plants is located in the lower portion of the control building for the ABWR. Separation from the control room complex by three hour rated fire barriers is provided for this equipment and it is considered in conjunction with the reactor building for purposes of the analysis. A fire in a single divisional area in either building is assumed to cause loss of that division without recovery. Considering the two locations as one provides an ABWR basis similar to the basis for the fire frequency data for the FIVE methodology.

- Step 1.3 (FIVE)

  – Ignition Source Weighting Factor (FIVE)

Potential Fixed Ignition Sources—It appears that the list of potential ignition sources in Table 1.2 of the FIVE methodology report represents all types of significant ignition sources that have resulted in fires in the existing plants and are therefore all inclusive. Potential ignition sources which are in existing plants but which have not ignited a fire may experience a fire in the future. Since they have not yet served as an ignition source, the frequency would be less than any data given on the FIVE tables. Any new potential fire sources unique to the ABWR have been considered. The new potential sources are the fine motion control rod drive (FMCRD) power supplies and the reactor internal pump (RIP) adjustable speed drives. These two new sources are included in the reactor building analysis.

In the FIVE methodology, the ignition source weighting factors are fractional numbers calculated on the basis of the number of ignition sources in the fire compartment (fire area) being considered divided by the total number of similar ignition sources for the plant. For example, if there are 30 electrical cabinets in the reactor building and 10 of them are in a fire compartment (fire area) being analyzed, the ignition source weighting factor would be 10/30 or 0.33 for the fire compartment. Counted quantities are used to calculate fractions.

The specific rationale for derivation of the weighting factor for each potential ignition source is given below for each Ignition Source Data Sheet (ISDS), Table 3 (FIVE).

19M.5.2 Completed Ignition Source Data Sheets and Notes

The fire compartment fire frequency was determined for the applicable building areas by completing Ignition Source Data Sheets, Tables 19M-3 through 19M-10 and their associated notes, for the applicable areas. A summary of the results follows:

Building Area	Fire Compartment Fire Frequency[1]
Reactor/Control Building/Intake Structure	
Division 1 Fire Areas	
Division 2 Fire Areas	
Division 3 Fire Areas	
Turbine Building	
Control Room Complex	

1 Not a part of DCD (Refer to SSAR).

These fire compartment fire frequency values were used as input to the probability risk assessment models.

19M.5.3 Completed Ignition Source Data Sheets and Notes

Completed ignition source data sheets and the associated explanatory notes for each sheet are included as Tables 19M-3 through 19M-10.

19M.6 Calculation of Core Damage Frequencies

19M.6.1 Methodology

The calculations were based upon original ABWR functional fault trees for the reactor water injection and heat removal functions, which included a gas turbine generator as a diverse source of emergency power. The fault trees and input data are described in detail in Chapter 19. Fault tree analyses were performed using the CAFTA computer program.

Functional fault trees were developed to reflect the reduced injection and heat removal capabilities defined by each of the five bounding fire scenarios. Estimates of expected core damage frequency were developed for each scenario by applying results of these functional fault tree analyses to accident sequence event tree structures developed for the ABWR internal events PRA, and described in Chapter 19. The isolation / loss of feedwater event tree, Figure 19D.4-2, was selected for evaluation as a conservative representation of the sequence of events for fires which lead to divisional power loss and for control room fires.

The consequences of a turbine building fire were determined to be bounded by a loss of off-site power event, and therefore the loss of off-site power event trees, Figures 19D.4-3 through 19D.4-9, were used as the basis for its assessment.

Conservative estimates of fire initiating event frequencies and assumed consequences were developed in a preceding task using the EPRI FIVE methodology. The initiating event frequencies obtained and used in this analysis are as follows:

Initiating Event and Assumed Consequences	Annual Frequency[1]
Fire disabling electrical Division 1	
Fire disabling electrical Division 2	
Fire disabling electrical Division 3	
Control room fire limiting ECCS control to remote shutdown panel	
Turbine building fire resulting in loss of off-site power	

1 Not a part of DCD (Refer to SSAR).

19M.6.2 Results

Calculated core damage frequencies for each of the initiating events are summarized in Table 19M-11. Breakdowns of core damage frequency by accident class for each initiator are provided in Tables 19M-12 and 19M-13. It should be noted that the core damage frequencies for Class II events are reduced very significantly prior to summation in the “TOT. CDF” columns and rows of the latter two tables. This accounts for recovery actions for Level 1 PRA Class II events identified in the Level 2 containment event tree analyses in Subsection 19D.5.

Event trees used for the ABWR fire risk screening analysis for divisional and control room fires are illustrated in Figures 19M-3 through 19M-6. Those for the turbine building fire event are given in Figures 19M-7 through 19M-13. As indicated in Tables 19M-12 and 19M-13, only event sequences categorized as accident Class I are found to contribute significantly to core damage frequency. It can also be seen from the tables that all postulated fire events pass the screening criteria.

The event tree figures and summary tables show the main contributor to core damage frequency for each initiator leading to a divisional power loss to be that sequence in which both high and low pressure injection systems fail, following successful scram and SRV performance and loss of feedwater. In the case of the control room fire event, assumed inability to recover feedwater or inject condensate at low pressure increases the values of both high and low pressure Class I sequences. The probability of failure to manually depressurize also has greater impact in this latter event, since only four SRVs can be controlled from the remote control location in the modified scenario, and opening of three is required for success. For the divisional fire sequences, failure to depressurize is essentially determined by human error. Turbine building fire core damage frequency is dominated by station blackout event sequences.

The core damage frequency initially calculated for control room fires (initiating event CR) was greater than that predicted for a divisional electrical fire, and did not pass the FIVE Methodology screen. This was due to the provision of capability at the remote shutdown location to control a single loop for high pressure injection (HPCF) as well as only three safety relief valves for depressurization. With respect to the latter, successful operation of all three valves would be necessary to prevent core damage in the event of a need to depressurize. Therefore, a more detailed analysis was required for this initiator, as well as consideration of possible system control capability modifications to the remote shutdown control system.

Potential courses of action to reduce control room fire risk which were identified and evaluated included the following:

- Providing control capability for a fourth SRV at the remote shutdown control panel, and
- Taking credit for operating the RCIC system from outside the control room if determined to be practical, i. e., from the motor control center and locally at the RCIC.

Examination of the latter possibility led to the conclusion that successful operation of the RCIC system from outside the control room would be practical, and it is an interface requirement that the applicant provide an emergency operating procedure for manual operation of the RCIC.

Results of these evaluations are documented in Table 19M-14. It can be seen that neither of the above actions by itself satisfies the screening criterion. In combination, however, the criterion is met, and with incorporation of the above two actions no further analyses are required to demonstrate acceptably low fire risk for the ABWR.

19M.7 COL License Information

19M.7.1 Manual Control of RCIC

Subsection 19M.3 requires local manual control of RCIC as one means of mitigation in case of a control room fire. It is a requirement that a procedure for local operation of the RCIC be provided by the COL applicant.

19M.8 References

19M-1 "Fire Vulnerability Evaluation Methodology, FIVE, Plant Screening Guide", Electric Power Research Institute, Preliminary Draft.

Table 19M-1 Fire Risk Screening Analysis Summary

Initiators and Conditions	Fire Ignition Frequency[1]	Core Damage Frequency Per Year[1]
Safety-Related Buildings		
Division 1 Fire		
Division 2 Fire		
Division 3 Fire		
Control Room Fire With Remote Control of 4 SRVs and RCIC		
Turbine Building Fire		

1 Not a part of DCD (Refer to SSAR).

Table 19M-2 Weighting Factors for Adjusting Generic Location Fire Frequencies for Application to Plant-Specific Locations (References FIVE Table1.1)

Plant Location (Table 1.1 Of Five)	Weighting Factors[1] (Wfl) (Table 1.1 Of Five)	Weighting Factor (Wfl) ABWR Analysis	WFL Value
Auxiliary Building (PWR)	The number of units per site and divide by the number of buildings.	Not Applicable.	N/A
Reactor Building (BWR)[2]	The number of units per site and divide by the number of buildings.	One unit divided by three divisionally grouped fire areas. In effect, the reactor building is divided into three separate buildings by the three hour rated fire barriers for the divisional fire areas.	0.33
Diesel Generator Room	The number of diesels and divide by the number of rooms per site.	Three diesels per site divided by three rooms per site.	1
Switchgear Room	The number of units per site and divide by the number of rooms per site.	One reactor per site divided by nine switchgear rooms (2-TB, 3-RB and 4-CB) per site.	0.11
Battery Room	The number of units per site and divide by the number of rooms per site.	One reactor per site divided by five battery rooms (Divisions 1, 2, 3 and 4 and non divisional in CB) per site.	0.20
Control Room	The number of units per site and divide by the number of rooms per site.	One reactor per site divided by one control room complex per site.	1
Cable Spreading Room	The number of units per site and divide by the number of rooms per site.	Not applicable, due to the multiplexed systems there are no cable spreading rooms in the ABWR. This is a significant difference between the plants characterized in FIVE and the ABWR.	N/A
Intake Structure	The number of units per site and divide by the number of rooms per site.	One unit divided by three single division fire areas. This is equivalent to three separate intake structures per site.	0.33
Turbine Building	The number of units per site and divide by the number of rooms per site.	One reactor per site divided by one turbine building per site.	1

Table 19M-2 Weighting Factors for Adjusting Generic Location Fire Frequencies for Application to Plant-Specific Locations (References FIVE Table1.1) (Continued)

Plant Location (Table 1.1 Of Five)	Weighting Factors[1] (Wfl) (Table 1.1 Of Five)	Weighting Factor (Wfl) ABWR Analysis	WFL Value
Radwaste Area	The number of units per site and divide by the number of radwaste areas.	One reactor divided by one radwaste building per site. (Since the radwaste building is a grouping of fire areas separate from any area containing safety-related equipment, a fire in the radwaste building cannot affect safe shut- down of the plant.)	1
Transformer Yard	The number of units per site and divide by the number of switch-yards.	One reactor divided by one switchyard.	1
Plant-Wide Components (cables, transformers, elevator motors, hydrogen recombiner/ analyzer).	The number of units per site.	One reactor per site.	1

Notes:

1. The analyst must identify the number of like locations when determining the number of building, e.g., a 480 volt load center is “like” a switchgear room.
2 Reactor building does not include containment.

Table 19M-3 Fire Compartment-Division 1 Ignition Source Data Sheet (ISDS) (Taken from Draft FIVE Methodology)

Not part of DCD (Refer to SSAR)

Table 19M-3 Fire Compartment-Division 1 Ignition Source Data Sheet (ISDS) (Taken from Draft FIVE Methodology) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-4 Fire Compartment - Division 2 Ignition Source Data Sheet (ISDS) (Taken from Draft FIVE Methodology)

Not part of DCD (Refer to SSAR)

Table 19M-4 Fire Compartment - Division 2 Ignition Source Data Sheet (ISDS) (Taken from Draft FIVE Methodology) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-5 Fire Compartment - Division 3 Ignition Source Data Sheet (ISDS) (Taken from Draft FIVE Methodology)

Not part of DCD (Refer to SSAR)

Table 19M-5 Fire Compartment - Division 3 Ignition Source Data Sheet (ISDS) (Taken from Draft FIVE Methodology) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-6 Reactor and Control Building Fire Areas Explanatory Notes (for Tables 19M-3,4 & 5) (FIVE, Table 3)

Not part of DCD (Refer to SSAR)

Table 19M-6 Reactor and Control Building Fire Areas Explanatory Notes (for Tables 19M-3,4 & 5) (FIVE, Table 3) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-6 Reactor and Control Building Fire Areas Explanatory Notes (for Tables 19M-3,4 & 5) (FIVE, Table 3) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-6 Reactor and Control Building Fire Areas Explanatory Notes (for Tables 19M-3,4 & 5) (FIVE, Table 3) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-7 Fire Compartment - Turbine Building Ignition Source Data Sheet (ISDS) (Taken from Draft FIVE Methodology)

Not part of DCD (Refer to SSAR)

***Legend for Table 19M-7**

Not part of DCD (Refer to SSAR)

Table 19M-8 Turbine Building Explanatory Notes (For Table 19M-7)(FIVE, Table 3)

Not part of DCD (Refer to SSAR)

Table 19M-8 Turbine Building Explanatory Notes (For Table 19M-7)(FIVE, Table 3) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-8 Turbine Building Explanatory Notes (For Table 19M-7)(FIVE, Table 3) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-9 Fire Compartment - Control Room Complex Ignition Source Data Sheet(ISDS)*
(Taken from Draft FIVE Methodology)

Not part of DCD (Refer to SSAR)

Table 19M-10 Control Room Complex Explanatory Notes (For Table 19M-9) (FIVE, Table 3)

Not part of DCD (Refer to SSAR)

Table 19M-10 Control Room Complex Explanatory Notes (For Table 19M-9) (FIVE, Table 3) (Continued)

Not part of DCD (Refer to SSAR)

Table 19M-11 ABWR Fire Screening Analysis Summary

Initiators and Conditions	Core Damage Frequency (per year)[1]
Safety-Related Buildings	
Division 1 Fire	
Division 2 Fire	
Division 3 Fire	
Control Room Fire With Remote Control of 4 SRVs and RCIC	
Turbine Building Fire	

1 Frequencies are not part of DCD (Refer to SSAR).

Table 19M-12 Divisional and Control Room Fire Risk W/Remote Control of RCIC & 4SRVs for CR Fires

	Accident Class											
INIT.Event	**1A[1]**	**1B-1**	**1B-2**	**1B-3**	**1C**	**1D[1]**	**II[1]**	**IIIA**	**IIID**	**IV**	**TOT.**	**CDF[1]**
D1D		–	–	–	–			–	–	–	–	
D2D		–	–	–	–			–	–	–	–	
D3D		–	–	–	–			–	–	–	–	
CR		–	–	–	–			–	–	–	–	

1 Not Part of DCD (Refer to SSAR).

Table 19M-13 Summary of ABWR Risk Screening Analyses for Turbine Building Fire

	Accident Class											
INIT Event	**1A**	**1B-1**	**1B-2**	**1B-3**	**1C**	**1D**	**II**	**IIIA**	**IIID**	**IV**	**TOT. CDF**	**Percent**
TE2[1]		–	–	–	–		–	–	–	–		
TE8[1]		–	–	–	–			–	–	–		
TE0[1]		–	–	–	–			–	–	–		
BE2[1]		–	–	–	–		–	–	–	–		
BE8[1]	–					–	–	–	–	–		
BE0[1]	–	–			–	–	–	–	–	–		
TOT. CDF[1]												
PERCENT[1]											–	

1 Not Part of DCD (Refer to SSAR).

Table 19M-14 ABWR Control Room Fire Risk Screening Analysis Summary

Conditions of the Control Room Fire Analysis	Core Damage Frequency (per year)[1]
Remote control of 3 SRVs	
Remote control of 4 SRVs	
Remote control of 3 SRVs and RCIC	
Remote control of 4 SRVs and RCIC	

1 Not Part of DCD (Refer to SSAR).

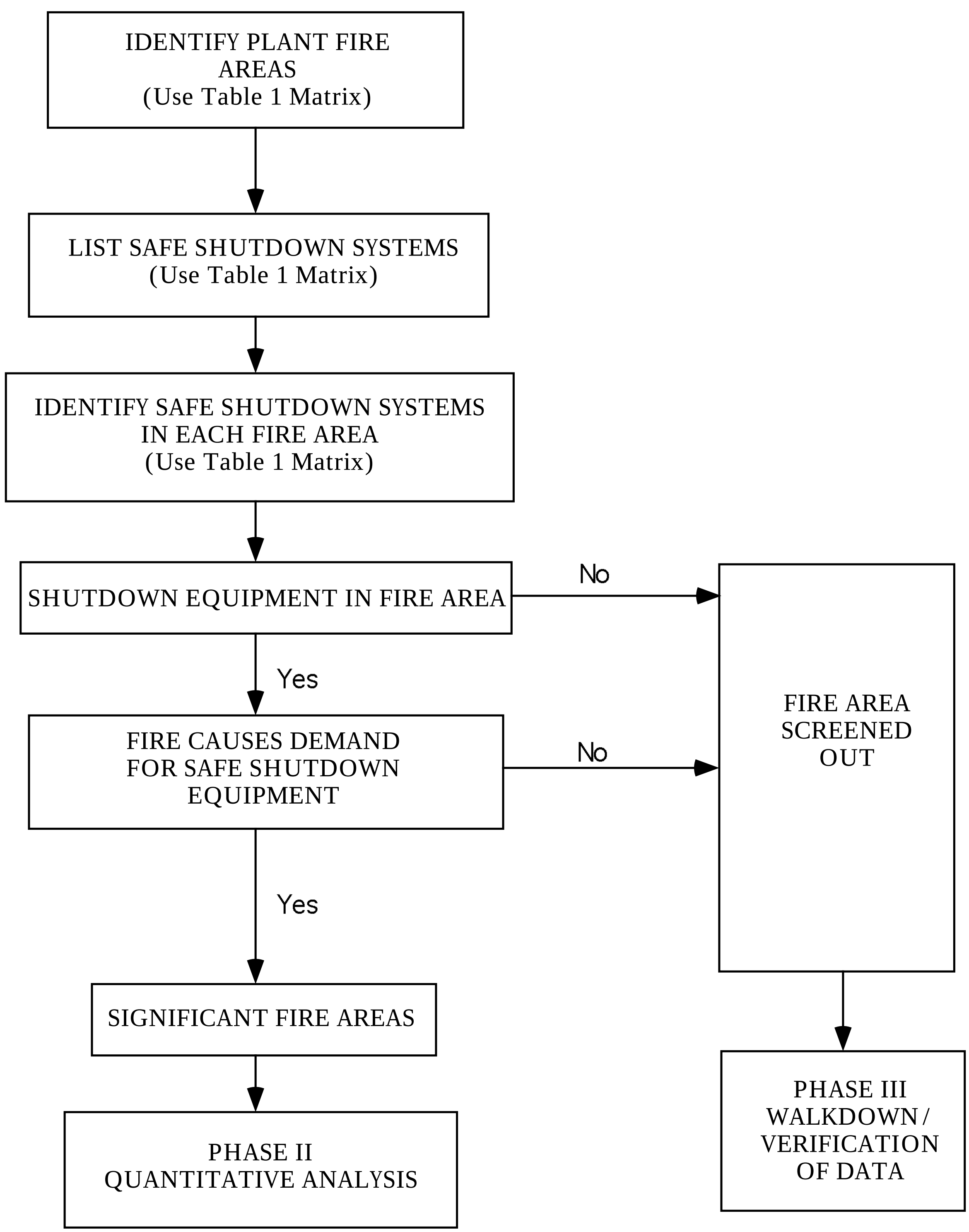


Figure 19M-1 Phase I Qualitative Analysis Flow Chart

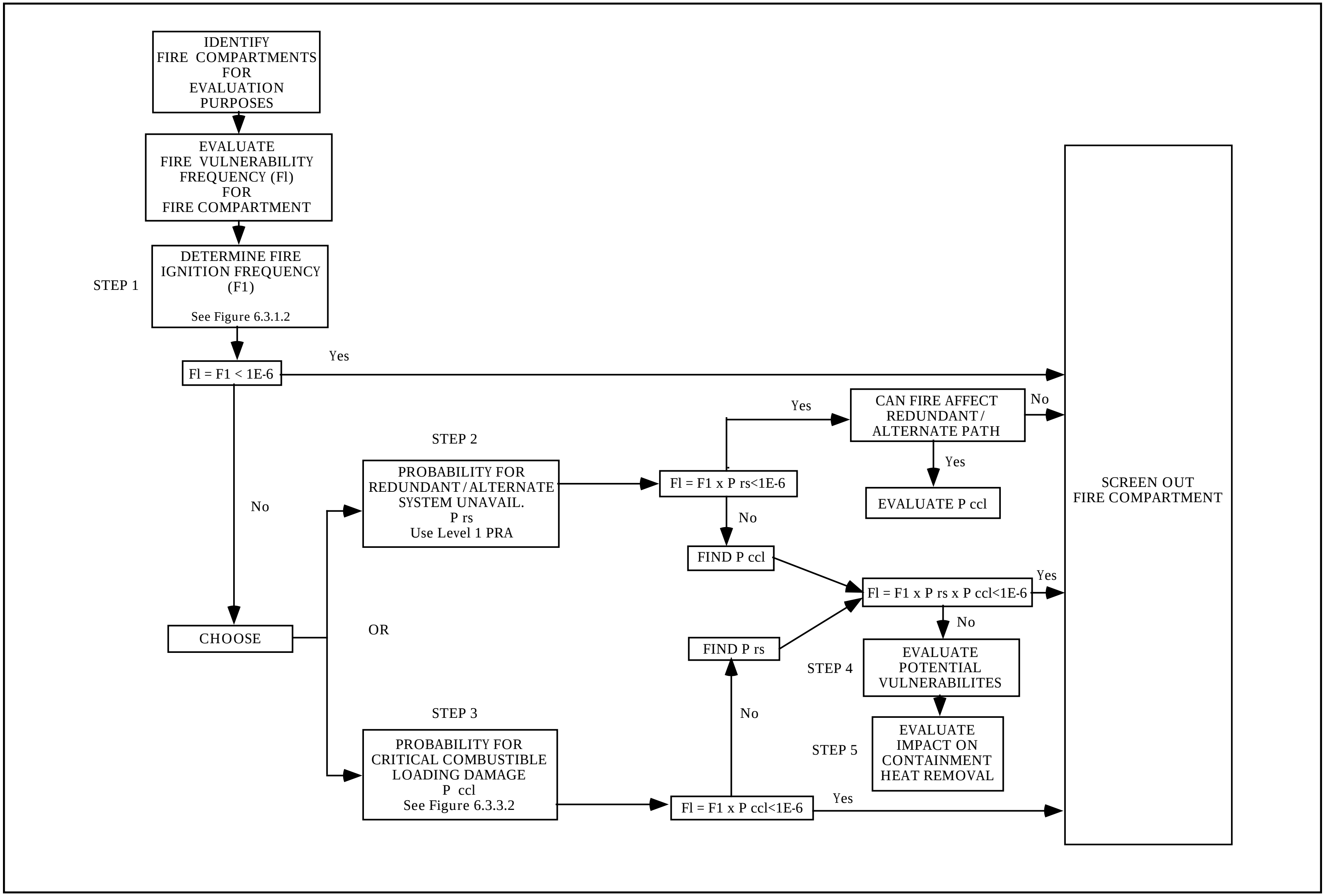


Figure 19M-2 Phase II Qualitative Analysis Flow Chart

Figure 19M-3 Division 1 Electrical Fire

Not part of DCD (Refer to SSAR)

Figure 19M-4 Division 2 Electrical Fire

Not part of DCD (Refer to SSAR)

Figure 19M-5 Division 3 Electrical Fire

Not part of DCD (Refer to SSAR)

Figure 19M-6 Control Room Fire

Not part of DCD (Refer to SSAR)

Figure 19M-7 Turbine Building Fire (Loss of Offsite Power and Station Blackout Event Tree)

Not part of DCD (Refer to SSAR)

Figure 19M-8 Loss of Offsite Power Event Tree (Recovery time: 30 min< t < 2 h)

Not part of DCD (Refer to SSAR)

Figure 19M-9 Loss of Offsite Power Event Tree (Recovery time: 2 hrs< t < 8 h)

Not part of DCD (Refer to SSAR)

Figure 19M-10 Loss of Offsite Power Event Tree (Recovery time: t > 8 h)

Not part of DCD (Refer to SSAR)

Figure 19M-11 Station Blackout Event Tree (Recovery time: 30 min < t < 2 h)

Not part of DCD (Refer to SSAR)

Figure 19M-12 Station Blackout Event Tree (Recovery time: 2 hrs< t <8 h)

Not part of DCD (Refer to SSAR)

Figure 19M-13 Station Blackout Event Tree (Recovery time: t > 8 h)

Not part of DCD (Refer to SSAR)

19N Analysis of Common-Cause Failure of Multiplex Equipment

19N.1 Introduction

The effect of common-cause failures of the ABWR multiplexing equipment (EMUX) on each safety function is included in the PRA analysis of each of the transient and LOCA initiating events (Appendix 19D). The fault tree designators for EMUX CCF are CCFMUX, CCFTLU, and ILCCFH. The probability values used in the PRA analysis are based on random probabilities of failure and common-cause beta-factor. The effect on total core damage frequency (CDF), as evaluated, is found to be significant.

Because of the importance of the multiplexing equipment to ABWR instrumentation and control, a supplemental study of EMUX CCF has been performed to further investigate the effects of the use of common instruments, multiplexers, and transmission networks for reactivity control (scram), ECCS (core cooling and decay heat removal), and LDIS (isolation).

The safety system logic and control (SSLC) has four independent divisions of instrumentation having separate sensors, actuators and multiplexing equipment. The only restriction regarding assignment of sensors and actuators to remote multiplexing units within a division is that reactor pressure vessel (RPV) wide-range and narrow-range water-level sensors are always assigned to different RMUs.

The primary effect considered in this analysis is that due to common-cause failure of automatic initiation of the ECCS and RPS functions. The study also examines the effects of EMUX common-cause failure on containment isolation.

19N.2 Results and Conclusions

The effects of EMUX CCF on total core damage frequency are found to be significant for transient and LOCA initiating events as analyzed in the PRA (Subsections 19N.5.1 - 19N.5.3). Additional “special” initiating events have been analyzed and found to not be affected by EMUX CCF (Subsection 19N.5.4) Common-cause failure of the multiplexing equipment during normal plant operation at power has also been examined as a potential accident initiator, and found to be a negligible CDF contributor (Subsection 19N.5.5).

The PRA analysis contains several conservatisms in regard to the evaluation of the effect of EMUX CCFs on CDF.

(1) As a simplification, the CCF probabilities were derived using the beta-factor method. Use of the “multiple-Greek” method of analysis, as described in Reference 19N-1, would provide smaller CCF probabilities where more than two failures are involved.

(2) The mean time between failures (MTBFs) used in the analysis to represent the component reliabilities treated all failures as functional failures; whereas a substantial fraction of the failures would be minor and would not fail the function.

(3) Multiple equipment failures generally do not occur simultaneously. Usually there will be a noticeable time period between the first and any subsequent failures, thus, providing advance information on a potentially developing problem. If the first failure is detected and its cause determined before subsequent failures occur, loss of system functions can be avoided and corrective action can be taken.

The potential causes of common failure of multiple divisions of EMUX have been identified as the following:

- Earthquake
- Loss of DC Power
- Loss of Cooling
- Sensor Miscalibration
- RMU Miscalibration
- Set Point Drift
- Maintenance / Test Error
- Manufacturing Error
- Electromagnetic Interference
- Fire
- Software Fault

These eleven potential common causes have been examined (Subsection 19N.4) and only three of them appear to be credible:

(1) RMU miscalibration,

(2) maintenance / test error, and

(3) software fault.

All three of these potential causes could exist across division boundaries in spite of physical separation and electrical independence. Because of the existence of these three potential causes of common-cause EMUX failure, several precautions are being taken regarding defense against them:

(1) To eliminate the RMU miscalibration as a credible source of EMUX common-cause failure, administrative procedures will be established to perform cross-channel checking of RMU outputs at the main control room SSLC instrumentation, as a final checkpoint of RMU calibration work.

(2) To eliminate maintenance / test error as a credible source of EMUX common-cause failure, a thorough post-maintenance test (Subsection 7.1.2.1.6 (4), (5), (6), Protection System Inservice Testability) will be conducted using the surveillance test controller (STC) that is provided in each instrumentation division as part of the EMUX and SSLC designs. The STC contains pre-programmed test sequences for each sensor type and each safety-related system supported by EMUX and SSLC. The tests cannot be changed by the maintenance technician; the technician only selects which system is to be simulated. The STC then injects appropriate simulated sensor signals (traceable to and automatically checked against known standards) into the RMUs of the EMUX. Failure of the calibration standards is alarmed. Testing is dynamic; i.e., the STC injects ramp-type analog signals over the full range (including abnormal upscale and downscale) of the simulated transmitters and also injects pulse, contact closure or frequency-modulated signals as required by the system under test. In this way, the full transmission capability of EMUX and the functional control and interlock logic in SSLC are tested. Test results are monitored either at the EMUX outputs in the control room or local area, or at the SSLC outputs, depending upon where test or maintenance was performed. The STC logs the test results, which can also be sent to the process computer or printed out. The STCs are normally off, have continuous self-test, and are operated one at a time, so they are not subject to CCFs of their own. Since the logged test results can verified independently by control room personnel, a single technician can safely maintain multiple divisions of EMUX.

The test features described above check the electronic circuitry from the signal conditioning and A / D converter inputs through the digital processing electronics. Transmitter calibration and other sensor calibration activities will require two technicians for the four safety divisions. Each will calibrate his division to the inputs of the RMUs and then check the other’s work. This will then be repeated for the remaining two divisions.

(3) To prevent any unidentified EMUX faults / failure modes (e.g., an undetected software fault) from propagating to other EMUX divisions, so that such unidentified faults are effectively eliminated as a credible source of EMUX common-cause failure:

(a) Chapter 16, “Plant Operating Technical Specifications” will incorporate requirements on the “Limiting Conditions of Operation” and “Required Action” that must be followed in the event of a failure of a single division of EMUX and in the event of a failure of multiple divisions of EMUX.

(b) The plant operating procedures will include the appropriate detailed procedures necessary to assure that the ABWR plant operations are maintained within compliance with the governing “Plant Operating Technical Specifications” during the periods of divisional EMUX failure. These will also include the appropriate symptom-based procedures to assure that adequate core cooling is maintained in the hypothetical event of an entire EMUX system failure.

19N.3 Basis for the Analysis

The design features of the EMUX that are of most importance to and form the basis for this analysis are the following:

(1) There is complete separation of RMUs, DTMs, SLUs, TLUs, sensors and ECCS actuators, etc., between the four safety divisions of control and instrumentation.

(2) Within a given division, the only restriction regarding assignments of sensors and actuators to RMUs is that wide-range and narrow-range reactor water level sensors cannot be input to and processed by the same RMU.

(3) There is separation of DTM and TLU modules within a division along the lines of “deenergize to operate” and “energize to operate” functions, i.e., RPS, and MSIV signals are processed by different DTM and TLU modules than the DTM and SLU modules used for ECCS control and PCV isolation (PCV isolation is also deenergize-to-operate).

(4) The RMUs are connected by a separate multiplexing system (EMUX), in each division, which is a redundant or reconfigurable control data network of high reliability (MTBF=100,000 hours).

(5) All data communications to and from other divisions of control and instrumentation, and all data communications to nondivisional systems are electrically isolated.

(6) Comparison of a sensed input to a setpoint for generating a trip is done by a DTM. Coincident 2/4 trip logic processing for generating a divisional output trip is done by a TLU or SLU.

(7) Loss of data communications in any division to the RPS (and deenergize-to-operate isolation functions) will result in a trip (and isolation, respectively) in the failed division due to the fail-safe design.

(8) Manual scram is implemented by hard wire to the scram pilot valve solenoids and does not depend on the correct operation of the DTM or TLU.

(9) A bypass of the RPS output logic unit is a manual division out-of-service bypass, which allows repair of the DTM or TLU of that division without a half scram condition or half MSIV isolation condition. Only one division can be bypassed at a time.

(10) To reduce the probability of spurious initiation of ECCS, two SLUs are used in parallel within a division, with 2/2 voting at the final channel output to initiate equipment actuation. If one ECCS SLU is in a failed condition, it is automatically bypassed, the control room is alerted, and the remaining SLU operates with 1/1 logic until the failed SLU is restored.

(11) RMUs and EMUXs are self-tested every 15 minutes and repaired/replaced in an average time of 4 hours.

(12) Control room indications, annunications, and alarms associated with EMUX-transmitted control signals are dependent on correct operation of EMUXs.

(13) Vital plant parameters are hard-wired to the remote shutdown panel independent of EMUX.

In addition to the design features listed above, the following assumptions and ground rules also supply the basis for this analysis:

(1) Common-cause failure of all RMUs or all EMUX networks cannot be ruled-out as impossible or incredible. The reason for this is that several potential common causes can be postulated. (Subsection 19N.2.)

(2) The probability of common-cause failure of interdivisional RMUs or EMUXs is extremely low. The reasons for this are the common-cause defenses built into the design—physical separation, electrical separation, asynchronous operation, optical isolation, natural convection cooling ability, and the self-testing feature—in addition to the special defenses discussed in Subsection 19N.2.

(3) RMUs may be postulated to have common-cause failures of the energize-to-trip mode or the deenergize-to-trip mode, but not of both modes simultaneously.

(4) EMUX transmission may be postulated to have common-cause failures of the energize-to-trip mode only. Failure of the deenergize-to-trip mode is considered to not be possible.

(5) Simultaneous failure of all RMUs or EMUXs in the energize-to-trip mode would result in an automatic scram and MSIV and PCV isolation valve closure, and loss of automatic ECCS initiation capability. Some ECCS could be initiated manually from the remote shutdown panel.

(6) In addition to complete failure of energize-to-trip or deenergize-to-trip functions, the RMUs may have common-cause calibration errors.

19N.4 Potential Causes of and Defenses Against EMUX CCF

Because of the high degree of independence between divisions in the ABWR design, the probability of simultaneous failures in multiple divisions is very low. If there were no identifiable common failure cause, the random probability of failure of n divisions would be the nth power of the probability of a single division. In the presence of potential common failure causes, the probability of multiple failures may increase. The identified potential common failure causes are listed in Subsection 19N.2. A discussion of the nature and credibility of each of these potential common failure causes and the defenses against them follows in Subsections 19N.1 through 19N.4.12.

19N.4.1 Earthquake

The multiplex equipment consists of solid-state electro-optical modules, which are vibration and shock resistant by nature. In addition, the equipment is designed and tested to very high acceleration levels (7-10g). Earthquakes of magnitudes above 2g have never been experienced, are not expected to occur, and if they did occur would have much more serious consequences than loss of EMUX equipment. Even allowing for magnification above ground level, earthquake does not appear to be a credible cause of concern.

19N.4.2 Loss of D.C. Power

Common-cause loss of DC power has been examined intensively in an EPRI analysis (Reference 19N-1). Most of the identified potential common causes were found to either result in gradual degradation and/or be self-announcing. The consequences of actual loss of all DC power would be far more serious than the loss of EMUX equipment since most control instrumentation in the plant's safety equipment depends on DC power. (Loss of DC power is evaluated as part of the station blackout analysis of

Appendix 19D.) Loss of DC power does not constitute a significant cause of common-cause EMUX failure.

19N.4.3 Loss of Cooling

It is a design requirement that the ABWR EMUX equipment must be capable of continuous operation at 323.15 K (50°C), and must be capable of continuous operation in its installed condition without fans. This is not a problem for present-day low-power solid-state electronic equipment, and the maximum anticipated ambient temperature is 313.15 K (40°C). Loss of cooling is not a credible common cause.

19N.4.4 Sensor Miscalibration

Sensor miscalibration does not represent a common-cause failure of EMUX equipment per se, but is identified here because of the fact that there is a reduction in the number of sensors in the ABWR multiplexed instrumentation configuration relative to earlier designs, and the sensors are shared between safety functions.

A reduction in the number of sensors does not necessarily degrade reliability or availability. In fact, simpler systems are usually more reliable than more complex systems. When additional components are used redundantly in a system to improve reliability, a point may be reached where the system reliability is dominated by common-cause failure, and additional redundancies add little, if any, improvement in system reliability.

Sharing of sensors raises the possibility of common-cause sensor miscalibration error between safety functions. For the limiting-risk case, where low RPV water level is the sole sensed initiation condition, reactor trip and ECCS initiation have different sets and types of sensors. ECCS is initiated by two sets of wide-range water level sensors and reactor trip is initiated by a separate set of narrow-range sensors. With proper maintenance procedures and special precautions, the possibility of common-cause miscalibration resulting in loss of automatic initiation of both safety functions is very remote.

In summary, a reduction in sensors from earlier designs has little effect on core damage frequency or risk due to the separation of functions, diversity of sensor types, different modes of operation, and use of multiple trip units for different trip set points. Sensor miscalibration is not a credible cause of common-cause failure in the ABWR multiplexed instrumentation.

19N.4.5 RMU Miscalibration

Only the analog-to-digital converters of the RMUs require calibration. The calibration is automatic and computer-controlled. Calibration is accomplished by comparison to voltage, resistance and time references that are verified against external laboratory standards. The EMUX transmission equipment is self-calibrating. The technician only initiates calibration by pushing a button. In addition, the self-test feature of the equipment detects certain types of calibration faults.

The above factors minimize the likelihood of miscalibration, but do not eliminate miscalibration as a possible (credible) common cause. Administrative controls will be used during cross-channel checking to assure that miscalibration is not propagated by transmission of bad signals from one division to another.

19N.4.6 Setpoint Drift

Setpoints are digital and programmed into non-volatile memory locations; therefore, there is no setpoint drift. Setpoint drift is not a credible cause. (Setpoints could be incorrectly set initially, as discussed in Subsection 19N.4.7.)

19N.4.7 Maintenance/Test Error

The EMUX equipment has a built-in provision to prevent bypassing multiple divisions simultaneously. This feature would not prevent common maintenance or test errors that were done consecutively and were latent by nature, such as set points being erroneously set. Periodic surveillance, as required by the technical specifications, includes verification of setpoints. The self-test feature of the equipment will also identify some types of maintenances/test errors.

Although the features discussed above will minimize the likelihood of common-cause maintenance/test errors, they do not eliminate maintenance/test errors as a credible common cause. Administrative controls will be used to further reduce the likelihood of most of these types of errors by not allowing the same technician to work on multiple divisions. (See the discussion in Subsection 19N.2.)

19N.4.8 Manufacturing Error

Solid-state electronic manufacture is a largely automated process subjected to multiple tests at successive levels of assembly (component, circuit, board and instrument level). Safety-related equipment is further qualified by extensive burn-in to uncover premature failures. The equipment is also subjected to very thorough check-out and test during installation. It is difficult to conceive of a type of manufacturing error that could escape all inspections and tests and cause concurrent failure in multiple channels at a later time. Manufacturing error does not appear to be a credible cause.

19N.4.9 Electromagnetic Interference (EMI)

EMI is a potential cause of failure of solid-state electronic equipment. EMI can enter a circuit through any of several paths—power supplies, adjacent equipment, adjacent cabling, or input signals. In the case of the EMUX equipment, none of these paths would affect multiple divisions since the divisions are widely separated physically and are electrically independent. In addition, the nature of electro-optics reduces the susceptibility to EMI. Fiber-optic transmission lines are not subject to EMI and will not propagate transients between lines. EMI is not a credible common cause.

19N.4.10 Fire

The four divisions of remote EMUX equipment are located in separate rooms of the reactor building and are separated by barriers. The fiber optic transmission cables have fire-resistant protective covering. A localized fire would affect only one division. A more wide-spread fire might affect two divisions, but a fire large enough to affect three or four divisions would have more far-reaching effects than the loss of EMUX transmission. Because of the physical separation, common-cause failure of remote EMUX equipment due to fire does not appear to be a credible concern.

A fire in the main control room could affect multiple divisions to the same extent that it would affect habitability of the room and other control functions. In such eventuality, the remote shutdown panel would be used for control.

19N.4.11 Software

The EMUX equipment is programmed to perform the multiplex function, self-test, and calibration. The software that provides the programming is subject to extensive “debugging” procedures and strict quality control and test requirements (verification and validation). Nevertheless, it is not impossible that an undetected “bug” could remain. If such were the case, it would most likely affect all divisions. It would not necessarily cause all divisions to fail simultaneously. Common-cause software fault is a credible, although unlikely, possibility. To provide additional defense against software CCF, technical specification requirements and administrative procedures will be established, as discussed in Subsection 19N.2, to assure taking of appropriate action in the event of failure of individual multiplex divisions.

19N.4.12 Summary

Of the eleven potential common causes examined, only three appear to be credible:

(1) RMU miscalibration

(2) Maintenance / test error

(3) Software fault

All of these potential causes could exist across division boundaries in spite of physical separation and electrical independence. In all cases, administrative controls will be applied to minimize the probability of common-cause failure.

The failure that would result in a significant contribution to core damage frequency would be complete failure during plant operation of three or four divisions of EMUX that transmit signals from wide-range water level sensors. This condition could result in failure to automatically initiate ECCS. Since failure of EMUX equipment is annunciated, the operator would be aware of the need for manual initiation of ECCS. Appropriate instrumentation and control is available at the remote shutdown panel, if needed.

19N.5 Discussion of the Effect on Core Damage Frequency

The three primary safety functions that are necessary to prevent core damage are reactivity control, core cooling, and decay heat removal. The effects of EMUX CCF are included in the quantification of core damage frequency in the internal events analysis of Appendix 19D. Additional discussion is given herein to provide further information and insight into the nature of EMUX CCF contribution to core damage frequency. The isolation function does not contribute directly to core damage frequency and is evaluated separately in Subsection 19N.6.

The most demanding condition requiring safety action is the condition of decreasing water level in the reactor pressure vessel (RPV) during power operation. This condition requires immediate reactivity control (scram) to slow the rate of inventory loss, increased water injection into the vessel to maintain or increase the water level (ECCS), and eventually a means of removing decay heat from the containment (main condenser or RHR). The limiting condition regarding automatic initiation and control of the three safety functions is a situation where the only sensed abnormal condition is the decreasing water level. This could occur with a feedwater trip or malfunction, a turbine trip, or closure of the main steam isolation valves (MSIVs). These three plant responses could result from a large variety of causes, including generator trip, loss of offsite power, loss of condenser vacuum, load rejection, recirculation pump trip, and others. For purposes of this analysis, all of these events resulting in decreasing water level are grouped and designated as “plant transients”.

19N.5.1 General Plant Transient Events

In the ABWR, automatic response of the safety functions to a plant transient producing decreasing water level is initiated by signals transmitted through the EMUX. Initiation of ECCS and closure of some isolation valves is by the presence of an energizing signal. Initiation of RPS (scram) and MSIV and PCV closure is by a deenergizing signal or absence / loss of energization.

There are four independent divisions of sensors and EMUX equipment. Simultaneous loss of transmission capability on any two of the four divisions would result in a scram on loss of energization. Loss of transmission capability on any three divisions simultaneously would result in loss of automatic initiation of ECCS and loss of low-pressure permissive signals for reactor shutdown cooling. When a single division is lost, the control room is alerted and that division is bypassed by the operator. Bypassing of a division results in that division becoming inoperative; ie, that division cannot contribute to scram, isolation, or ECCS initiation. Technical specification requirements govern actions to be taken under those conditions.

Because of the high degree of independence between divisions in the ABWR design, the probability of simultaneous failures in multiple divisions is very low. If there were no common failure cause, the random probability of failure of n divisions would be the nth power of the probability of failure of a single division. In the presence of potential common failure causes, the probability of multiple failures could increase. Potential multiple failure causes are listed in Subsection 19N.2. Defenses against these common-cause failures are discussed in Subsections 19N.2 and 19N.4. These defenses provide a high degree of independence between instrumentation channels and divisions in the EMUX control data network.

The relationship of the safety function initiation and the EMUX is depicted in a simplified event tree, shown on Figure 19N-3. This event tree is for a plant transient initiating event and loss of transmission capability from three or four divisions of EMUX transmission of wide-range RPV water level signals. Loss of transmission of narrow-range water level sensor RMUs due to common-cause failure would not affect the results since scram would be automatically initiated by loss of energization. The purpose of this event tree is to provide a means for examining the effect of common-cause failures of safety function initiating signals. Random failures of instrumentation and failures of mechanical execution of the safety function are evaluated in Appendix 19D.

The first safety response to a plant transient is a reactor trip and scram. Because of the deenergize-to-trip feature, a scram would be initiated, even with a common-cause failure of all EMUX transmission. (A loss of transmission through the EMUX would result in a plant scram at any time, even without a plant transient. That event is evaluated in a later subsection—Subsection 19N.5.5.) Common-cause failure of transmission would also result in closure of the MSIVs.

Given a successful scram, the next essential safety function is to maintain water level in the reactor pressure vessel. The limiting case for common-cause failure of the EMUX is common-cause failure of three or four of the individual remote multiplexing units processing wide-range RPV water level signals. Since ABWR has motor-driven feedwater pumps, closure of the MSIVs would not cause loss of feedwater unless the feedwater pumps tripped because of the transient. If the feedwater pumps did not trip, RPV water-level could be maintained as long as there was water in the condenser hotwell. In ABWR, the condenser hotwell inventory is automatically replenished from the condensate storage tank. If the feedwater pumps were tripped, they could be started manually from the control room, since the feedwater control system is independent of the EMUX. If necessary, sufficient ECCS pumps could be started manually from the remote shutdown panel to provide water to the RPV. Automatic initiation of ECCS would not occur because of the common-cause failure of EMUX to transmit wide-range RPV water level signals.

In the event that the motor-driven feedwater pumps were tripped and could not be restarted, the operator would need to manually start ECCS pumps in a relatively short time (approximately 30 minutes). The operator can extend the time available by starting the second CRD pump as instructed by the emergency operating procedures (EOPs). This extension of available time is not included in the internal events analysis of Appendix 19D.

To manually start some ECCS pumps, the operator may have to use the remote shutdown panel, since manual start signals from the control room are normally transmitted through the EMUX and may not be operable. The operator would have correct indication of RPV water level in the control room since water level is hard wired in addition to being transmitted through the EMUX. He also would be aware of the reactor scram. If control is not possible from the control room, the EOPs will tell the operator to proceed to or send someone to the remote shutdown panel where true indications and means of control are supplied through independent channels. In this simplified bounding analysis, failure of the operator to manually start ECCS pumps would result in uncovering of the reactor core and eventual core damage.

In the event that the operator successfully recovered feedwater or started ECCS pumps, the RPV water level would be maintained above the top of the fuel and no direct core damage would ensue. Eventually (within 20–24 hours—or longer if the main condenser were available) decay heat removal would be required to prevent excessive heatup of the suppression pool and containment. Initiation of decay heat removal would be accomplished by the operator through manual start and valve lineup of RHR in the suppression pool cooling mode. Later in the shutdown procedure, the operator would realign RHR in the shutdown cooling mode. In this analysis, proper action by the operator to provide pump initiation and valve lineup is all that is considered. Mechanical failure of pumps, valves, or other equipment is evaluated in Appendix 19D.

In this simplified analysis, if the operator fails to initiate decay heat removal, it is assumed that the containment will eventually fail and ECCS equipment will also fail due to harsh environmental conditions. This is a conservative simplification, since the ABWR has a containment overpressure protection system.

The effect of common-cause EMUX failure on CDF is included in the quantification of the event trees in Appendix 19D for transient-initiated and LOCA events. The random unavailability of the RMUs and TLUs is derived from an expected mean time between failures (MTBF) and a mean time to detect and repair a failure (MTTR). The random unavailability of the EMS is derived from an expected MTBF and an MTTR. The MTBF values are estimated, based on information from the supplier. The MTTR value is based on the use of a self-test feature which detects a failure within 15 minutes on the average, and the existence of spare replacement units on hand at the plant. The self-test feature detects most of the failures. The remaining failures are detected by surveillance testing conducted quarterly.

The beta-factor model used to estimate the common-cause failure probability is based on the premise that the common-cause failure probability is a function of the random unavailability of the individual units, as well as the existence of potential common causes. The beta-factor is simply the ratio of the common-cause failure probability to the total failure probability. Stated another way, the beta-factor represents the proportion of total failures that are multiple failures due to a common cause.

If there were sufficient experience data for multiple failures of solid-state multiplexing equipment, the experience data would be used directly and there would be no need for use of the beta-factor model. However, there is a dearth of multiple-failure data pertaining to solid-state multiplexer equipment, particularly equipment with a self-test feature. The alternative is to evaluate or estimate the relative susceptibility of the EMUX to multi-divisional failures through use of the beta-factor.

A recent report by the Electric Power Research Institute (EPRI) (Reference 19N-1) discusses the beta-factor model and lists representative values for beta. The values listed generally range from 0.1 down to about 0.01, but there is no value given specifically for solid-state multiplexing equipment. Considering the defenses in the ABWR design, particularly the self-test feature, a lower value for beta is justified. The self-test feature of the EMUX equipment provides detection of failures in 15 minutes, and on-hand spare modules provides restoration of operability within 4 hours. This feature limits the available time for propagation of multiple failures to 4.25 hours, and essentially eliminates several of the more likely causes of multiple failures.

A data summary of Licensing Event Reports (Reference 19N-2) pertaining to common-cause failure of instrumentation equipment derives beta-factors for several types of instrumentation equipment. Although there is no summary specifically for solid-state multiplexer units, there is a summary for “signal conditioning equipment.” Direct applicability and use of these data for the Appendix 19D analysis is not warranted, and the data are not used directly. All of the data have very large bands of uncertainty. The derived median values for beta provide some indication that the beta-factor used in the Appendix 19D analysis may be conservative.

The ABWR PRA indicates that the total core damage frequency for the ABWR design will be very low. The PRA analysis also indicates that potential EMUX CCFs during plant transient events are significant contributors to the low total CDF. EMUX CCFs appear in many of the top cutsets. An importance analysis indicates that all three EMUX CCFs have relatively high “risk achievement worth”, i.e., increases in the CCF probabilities would result in significant increases in total CDF. The defenses against EMUX CCFs in the plant design (Subsection 19N.4) and the administrative procedures prescribed in Subsection 19N.2 should prevent increases in EMUX CCF probabilities above the values used in the PRA analysis. Conservatisms in this part of the PRA tend to somewhat overestimate the importance of EMUX CCFs.

19N.5.2 Loss of Feedwater Event

The previous analysis considered the effect of loss of transmission capability of the EMUX, that is, an instance where the EMUX failed to transmit an energization signal. The reverse failure mode would be failure to lose the energization signal for RPS due to common-cause failure of the narrow-range water level sensor RMUs to properly sense a Level 3 condition. For many plant transients, automatic scram would occur due to increased neutron flux or other direct-input signals to the RPS logic. For purposes of this analysis, an initiating event is used that would require response of the narrow-range RMUs that sense a Level 3 water-level condition. A feedwater pump trip can be used to represent such an event.

The probability of common-cause failure in this mode is much lower than for the loss-of-transmission mode since most of the identifiable common causes would not cause a failure in this mode. The EMUX failure in this mode could result in failures of automatic scram. There is a very high probability that the operator would provide manual scram based on independent indications of the feedwater pump trip. Since the MSIVs would not close, the power conversion system would remain in operation. Based on past operating experience, there is a high probability that the operator would recover feedwater in addition to initiating manual scram. If feedwater were not recovered before low water level (Level 2) was reached, ECCS would be initiated automatically by means of transmission through the wide-range water-level sensor RMUs.

Initiation of decay heat removal would not be affected by the EMUX failure in the deenergize-to-trip mode.

Failure of the deenergize-to-trip mode of the narrow-range water level sensors does not contribute to core damage frequency for the ABWR.

19N.5.3 Loss of Coolant Accidents

Because of the low frequency of occurrence, LOCA events are very small contributors to ABWR core damage frequency. The probability of a coincidental common-cause EMUX failure together with a LOCA is an extremely low probability event. The possibility of a common-cause EMUX failure occurring as a result of a LOCA, where the LOCA would provide the common cause, is highly unlikely because of the locations and physical separation of the EMUX divisions.

19N.5.4 Other Initiating Events

Other initiating events that have been considered on past PRAs include the following:

(1) Loss of offsite power

(2) Loss of DC power

(3) Inadvertent open relief valve

(4) Loss of service water

(5) Loss of instrument air

19N.5.4.1 Loss of Offsite Power

Loss of all offsite power would have no direct effect on EMUX operability since EMUX equipment operates completely on divisional DC power. A loss of offsite power would cause a small increase in the conditional probability of loss of DC power since DC power is supplied by batteries or an AC converter-charger. The probability of loss of DC power is very low as discussed below in Subsection 19N.5.4.2.

19N.5.4.2 Loss of DC Power

Each division of the EMUX is powered by a division of DC power. Loss of all divisions of DC power would result in loss of EMUX transmission capability. The annual probability of loss of DC power on one essential bus is extremely small. The complete loss of DC power to all four divisions of essential power is considered to be essentially zero since the four divisions are independent, loss of DC power on any one division is alarmed, and the station batteries are routinely tested. Very few credible causes of common-cause failure of multiple DC buses have been identified (Reference 19N-1).

19N.5.4.3 Inadvertent Open Relief Valve

An inadvertent open relief valve (IORV) as an initiating event is treated in this analysis as just another plant transient. Although the plant response is somewhat different for an IORV, there is no peculiar impact on EMUX operation or response, and common-cause failure of EMUX would have the same effect on plant response as it would in any other plant transient event.

19N.5.4.4 Loss of Service Water

Loss of essential service water has been hypothesized and studied as an initiating event since loss of service water could disable some ECCS equipment. Service water is not used directly by any EMUX equipment and is not used for room cooling. The effects of loss of service water on essential safety equipment is evaluated in the system fault trees of Appendix 19D.

19N.5.4.5 Loss of Instrument Air

Instrument air is not used by EMUX equipment. As with essential service water, loss of instrument air would not affect EMUX equipment or this analysis.

19N.5.5 CCF of EMUX During Normal Plant Operation

Results of the above analyses indicate that common-cause failure of EMUX equipment in response to a demand from a plant transient or other off-normal event is a very small contributor to core damage frequency. This subsection examines the effect of a common-cause EMUX failure at a random time during normal plant operation (EMUX failure as an initiating event).

The limiting failure in this case would be common-cause failure of the three or four divisions of remote multiplexing units transmitting the signals from the narrow-range and wide-range water level sensors. If only the narrow-range transmission channels failed, the plant would scram on loss of energization, and ECCS would be initiated automatically through the wide-range RMUs. If only the wide-range water level sensor RMUs failed, the plant would not scram from that failure alone and there would be no demand on ECCS unless a plant transient occurred. Thus, both wide-range and narrow-range RMUs must fail in multiple divisions to cause a condition of concern and a potential accident initiator. In that event, the plant would scram and ECCS would not be automatically initiated.

Using the beta-factor method of CCF evaluation, the expected frequency of common-cause failure of all RMUs in three or four divisions would be equal to the product of the expected frequency of random failure of a single RMU and a beta-factor. In this case, the beta-factor should be lower than for the transient-initiated event since twice as many RMUs must fail; however, the assignment of a specific value to beta in this case is extremely uncertain.

Because of the great degree of uncertainty in any quantitative analysis that could be performed at this level, it appears preferable (and sufficient) to make a qualitative judgement. Since two or three EMUX divisions must fail in two distinct modes involving separate equipment, and they must fail in a nearly simultaneous manner, i.e., in a sufficiently short interval to not allow mitigating action to be taken, the expected frequency of occurrence must be extremely low.

Even if the initiating event should occur, there are still means of providing water injection to the core in time to prevent core damage, and to provide decay heat removal The contribution to CDF for this initiating event is certainly extremely small.

Further defenses against this event are discussed at the end of Subsection 19N.7.

19N.6 Discussion of the Effect on Isolation Capability

Failure of the Leak Detection and Isolation System (LDIS) does not have a direct effect on core damage frequency. The primary purpose of the LDIS function is to isolate the reactor and associated primary equipment and certain fission products in the event of a loss-of-coolant accident. A simplified event tree for a LOCA with common-cause loss of transmission capability of all RMUs is shown on Figure 19N-4. For this condition, MSIVs and PCV isolation valves would close on loss-of-signal.

The largest expected initiation frequency for a LOCA is for a small LOCA and is very small. The conditional probability of common-cause unavailability of RMUs is extremely small. There is no identifiable mechanism by which the LOCA could increase the probability of common-cause RMU failure.

With the MSIVs closed and the reactor shut down, the operator would have sufficient indications that an event had occurred and that the water level indication was erratic. Following operating procedures, the operator would send someone to the remote shutdown panel to monitor plant conditions and initiate necessary safety functions. There is a very high probability that isolation valves would then be closed manually within a reasonable time. The location of the remote shutdown panel is such that it cannot be made inaccessible by a LOCA. With any reasonable judgmental value assigned to failure of the operator to provide manual isolation, the total expected frequency of failing to isolate in response to a LOCA is negligible.

One additional isolation failure event should be considered—the effect of failing to isolate in a severe accident situation with a severely damaged core. In accident sequences resulting in core damage because the operator failed to maintain water inventory to the reactor (given an EMUX CCF), it is possible that he would also fail to close isolation valves.

The consequences of failure to isolate in the presence of a damaged core are not necessarily severe. If the MSIVs are closed, as they would be in this event, then the primary effect of failing to isolate is contamination of piping and equipment exiting the reactor vessel. Since all return lines to the RPV have check valves, check valve failures would have to occur to contaminate return lines and upstream equipment.

Contamination of the lines and equipment exiting the RPV would be mostly by steam or other aerosols rather than liquid, since the RPV water level would be very low. To provide a pathway into the reactor building or a release to the environment, there would also have to be a break or leak in the piping or piping components, since all systems are closed. In a severe accident scenario, there are larger and more likely potential bypass paths than isolation failure, and the consequences of failing to isolate would be very mild in comparison.

19N.7 Summary

This analysis has focused on the use of common multiplexing equipment in the EMUX. Because it is possible to identify feasible causes of multiple failures, the possibility of common-cause failure of identical multiplexing units has been studied. In view of the number and types of defenses built in to the EMUX design, the probability of common-cause failure should be very low. Because of the lack of multiple-failure experience data on equipment of this type, it has been necessary to predict the common-cause failure probability by use of an analytical model. The model used is a simple model—the beta-factor model—that hypothesizes that common-cause failure probability is proportional to the random failure probability of a single unit. The proportionality factor is beta. The hypothesis may not be true in all cases, and there is a great deal of uncertainty in assigning a value to beta.

Beta represents the fraction of total failures that would involve multiple identical units. The expected value of beta is dependent on the nature of the possible causes, how and how fast failures would propagate between units, and what defenses exist to the causes. There is no established method for quantifying these factors. In the absence of good and sufficient data, assignment of a value to beta is a matter of judgement. Values that have been used for beta range from 0.1 down to 0.001 and lower. Values of beta between 0.1 and 0.01 are common for mechanical equipment. Values below 0.01 are more common for instrumentation. The value used in the analysis of Appendix 19D may be conservative, considering the defenses in the ABWR EMUX design.

Using a conservative value for EMUX beta, the results of the Appendix 19D analysis show that use of the ABWR multiplexed shared-sensor configuration results in very little contribution to core damage frequency in response to demands from plant transients or off-normal events. This is because of the high availability on demand of the limiting equipment, the RMU. The high availability of the RMU is due to the self-test capability and the resulting short mean time to detect and recover from a failure. This same self-test feature is the best protection against common-cause failures, since multiple failures must all occur within an average time interval of 4.25 hours. This study tends to confirm the conclusions of the Appendix 19D analysis in regard to the effect on CDF of EMUX CCF in response to transient and LOCA initiated events.

Also of potential concern is common-cause failure of EMUX as an initiating event. The EMUX must be available at all times when the plant is operating because of the “fail-safe” (deenergize-to-trip) design for scram and MSIV closure. A simultaneous common-cause failure of two EMUX divisions at any time during plant operation would result in a plant trip, even though all plant parameters were normal. In a sense, this is a “false alarm” that results in a scram, which is a potential accident initiating event. If the third and/or fourth division of EMUX equipment also failed simultaneously, there could be a loss of automatic initiation of ECCS.

The expected frequency of occurrence of common-cause EMUX failure during normal operation is a function of the EMUX reliability, including D.C. power reliability. Fast recovery time due to the EMUX self-test feature does not help if two divisions fail simultaneously, since a plant trip is immediate. (The self-test feature is a major defense if the CCFs do not occur simultaneously.) The probability and expected frequency of occurrence of such an event is extremely low. Administrative controls will be imposed to minimize the probability of progressive common-cause failures. With the present design, the frequency of occurrence can be further reduced only by increasing the reliability of the remote multiplexing unit.

One type of administrative action that will effectively eliminate several common causes including software faults is establishment of required action to be taken in the event of functional failure of a single EMUX channel during plant operation. The action to be taken in the event of functional failure of an EMUX channel during plant operation is to re-establish operability and determine the cause of the failure as soon as possible. During the period of repair/replacement and diagnosis, the remaining channels are monitored closely. In the event of a second channel failing before the first channel is restored, the safest available action is immediately taken as prescribed by technical specifications and/or emergency operating procedures.

The sensitivity of core damage frequency to EMUX MTBF and beta can be seen from the event tree of Figure 19N-3. The RMU CCF probability or frequency is a direct function of both of these reliability elements. In turn, the core damage frequency is directly proportional to the RMU CCF probability and the initiating event frequency. If the RMU MTBF was twice as high, the core damage frequency would be reduced by half. In like manner, uncertainty in the initiating frequency propagates directly into uncertainty in CDF.

19N.8 References

19N-1 "Procedures for Treating Common Cause Failures in Safety and Reliability Studies", EPRI NP-5613, February 1988.

19N-2 Meachum, T.R., and Atwood, C.L., "Common-Cause Fault Rates for Instrumentation and Control Assemblies", EG and G Idaho, Inc., May 1983.

Figure 19N-1 Not Used

Figure 19N-2 Not Used

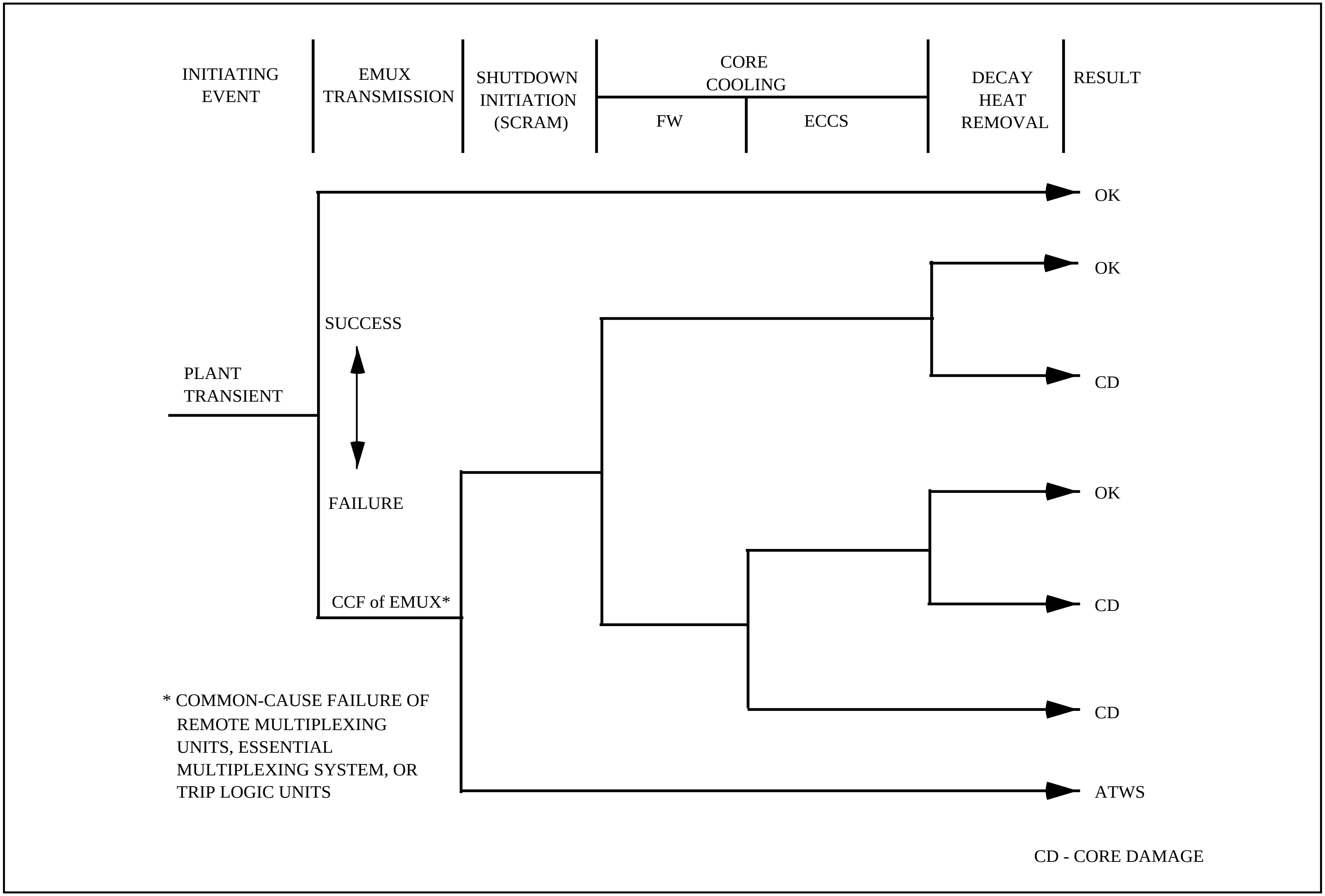


Figure 19N-3 Event Tree for Analysis of Common-Cause Failure of EMUX

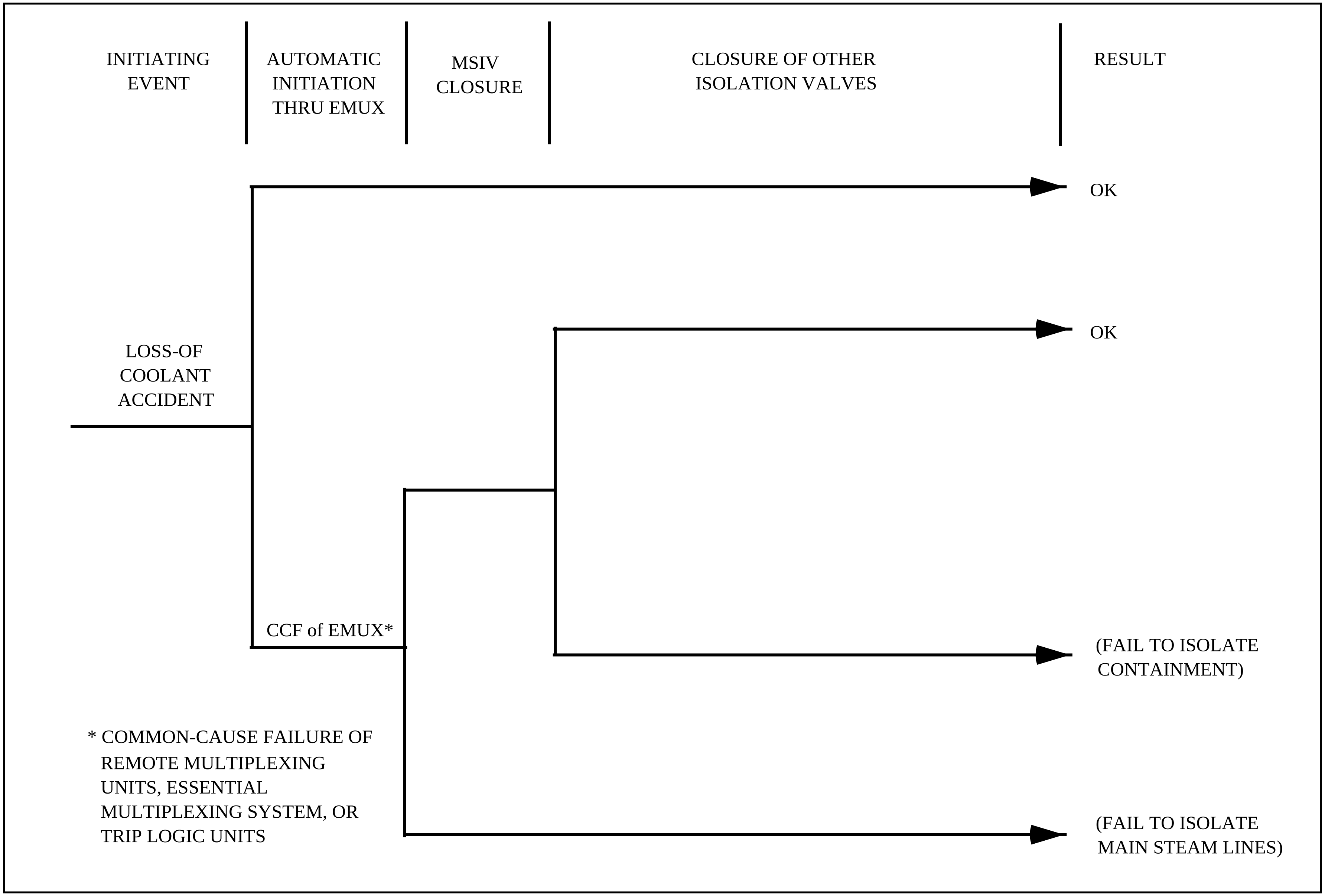


Figure 19N-4 Event Tree for Failure to Isolate Due to EMUX CCF

190 Not Used

19P Evaluation of Potential Modifications to the ABWR Design

This Section is not part of the DCD (Refer to Attachment A of the "Technical Support Document for the ABWR*"). Attachment A of the Technical Support Document for the ABWR is not incorporated by reference in the DCD.

* Revision 1 - December 1994. See letter from J.F. Quirk, GE, to R.W. Borchardt, NRC, dated December 21, 1994.

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Figures 19QA-19A through 19QA-20ck are not part of the DCD (Refer to SSAR Section 19QA)

19Q ABWR Shutdown Risk Assessment

19Q.1 Introduction

Due to events at operating plants in the past several years such as the loss of offsite power at Vogtle on March 20, 1990 and the loss of decay heat removal (DHR) at Diablo Canyon on April 10, 1987, the shutdown risk associated with nuclear power plants has become more of a concern to the industry. On January 17, 1992 the NRC issued Draft NUREG-1449, “NRC Staff Evaluation of Shutdown and Low Power Operation.” In NUREG-1449 the NRC staff identified some safety issues that may result in new regulatory requirements.

As part of the certification process for the advanced boiling water reactor (ABWR), an evaluation of the shutdown risk associated with the ABWR was completed. This Appendix discusses the design and procedural features of the ABWR that contribute to the conclusion that the ABWR shutdown risks are negligible.

19Q.2 Evaluation Scope

The ABWR shutdown risk evaluation covers the important aspects of NUREG-1449 as well as specific items requested by the NRC.

The evaluation encompasses plant operation in Modes 3 (hot shutdown), 4 (cold shutdown), and 5 (refueling). The ABWR full power PRA covered operation in Modes 1 (power operation) and 2 (startup / hot standby). This evaluation addresses conditions for which there is fuel in the reactor pressure vessel (RPV). It includes all aspects of the Nuclear Steam Supply System (NSSS), the containment, and all systems that support operation of the NSSS and containment. It does not address events involving fuel handling outside the primary containment or fuel storage in the spent fuel pool.

The evaluation was broken down into several topics covering design, procedures, and ABWR features that have the potential to prevent / mitigate past operating events that are considered precursors to loss of decay heat removal capability and fuel damage. The design issues included: decay heat removal, inventory control, containment integrity, electrical power, reactivity control, and instrumentation. Guidelines for generation of ABWR procedures are covered in a separate section, as well as the risk implications of using freeze seals during ABWR maintenance.

In NUREG-1449 it was pointed out that due to the increased level of maintenance activity while shutdown, the potential for fires and flooding in operating nuclear plants is considered higher during shutdown. These topics are covered separately to highlight the ABWR features designed to minimize the shutdown risks from fires and flooding.

In order to evaluate the ABWR features that are capable of preventing or mitigating safety significant events that have occurred at operating plants in the past, a study was completed of specific past events that resulted either in a loss of offsite power or a challenge to DHR. Loss of power events as described in NUREG-1410, “Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operation at Vogtle Unit 1 on March 20, 1990,” were evaluated and ABWR features which could have prevented/mitigated the event were described. A total of 74 loss of power events were evaluated. In a like manner, events described in NSAC-88, “Residual Heat Removal Experience Review and Safety Analysis–Boiling Water Reactors,” were reviewed along with certain loss of DHR events from INPO Significant Evaluation Reports (SERs) and Significant Operating Experience Reports (SOERs) and NRC Information Notices. Over 100 precursor events to loss of DHR were reviewed.

To ensure that new features (i.e., different than current operating BWRs) of the ABWR do not introduce any additional vulnerabilities to operation of the plant, a failure Modes and Effects Analysis (FMEA) was completed on these new features. The FMEA focused on the potential safety impact of identified failure modes and why these do not contribute to increased risk of ABWR shutdown operation.

Lastly, a detailed reliability study was completed of the ABWR DHR function. Probabilistic Risk Assessment (PRA) models including Fault and Event Trees were completed for all DHR and makeup systems. Based on PRA results, minimum sets of systems were identified that, if available, would result in acceptable shutdown risk.

Based on this shutdown risk evaluation, input has been provided to other parts of Tier 2. Systems and components important to safety were identified for inclusion in the reliability assurance program. COL action items such as a need for shutdown procedures and important operator actions were specified. Plant features important to risk reduction were identified and made part of Tier 1.

19Q.3 Summary of Results

The ABWR design has been evaluated for risks associated with shutdown conditions (i.e., Modes 3, 4, and 5). The evaluation included the following shutdown risk categories discussed in NUREG-1449:

- Decay heat removal
- Inventory control
- Containment integrity
- Loss of electrical power
- Reactivity control

The evaluation also included shutdown risk reduction features of the ABWR design due to instrumentation, flooding and fire protection, use of freeze seals, and procedure guidelines. ABWR features that are not part of current domestic BWR designs were evaluated to determine if any new shutdown risk vulnerabilities would be introduced. Finally, minimum sets of plant systems that if available would meet a goal of an acceptably low conditional core melt probability were identified.

The results of this shutdown risk evaluation demonstrate that the ABWR incorporates design features which make the plant risk during shutdown negligible. This conclusion is based on the following principal ABWR features which are capable of mitigating shutdown risks:

Shutdown Risk Concern	Principal ABWR Feature
Decay Heat Removal	Three physically and electrically independent RHR and support systems
Inventory Control	Multiple makeup systems and sources
Loss of Electrical Power	Two offsite and four onsite power sources
Reactivity Control	RPS, ARI, and standby liquid control systems and interlocks to prevent accidental reactivity excursions

The ABWR is adequately protected from internal flooding by floor drains, sump pumps, watertight doors, water level alarms, automatic isolation of flow sources, equipment mounted 20.32 cm (8 inches) off the floor, and the ability to fully contain potential flood sources (where appropriate).

Adequate protection from fire is provided by means of fire barriers and physical separation of the three independent safety divisions. Use of fire detectors, alarms, automatic fire suppression, manual fire water system, and a trained crew of fire fighters keep the risk related to fire at a negligible level.

To assure the flood and fire related risks are kept low during shutdown, the shutdown procedures that the COL applicant is required to develop have been identified.

Based on an FMEA of the new features incorporated into the ABWR that are different from operating domestic BWR plants, it is concluded that none of the new features will introduce additional shutdown vulnerabilities.

Instrumentation was identified that is available during shutdown to adequately monitor the status of the plant and operation of systems which will result in low levels of shutdown risk.

Guidance was presented on how freeze seals could be used during maintenance on unisolatable valves to minimize the risk associated with loss of the freeze seal.

Recommendations on outage planning procedures were presented to ensure that activities scheduled during outages take into account plant status and potentially high risk periods or configurations during shutdown. It was pointed out that the single most important element of reducing shutdown risk is proper outage scheduling of maintenance on systems and support systems capable of removing decay heat or supplying inventory makeup.

An analysis of 70 loss of power and over 100 loss of DHR precursor events at operating BWRs confirmed that the ABWR design features would prevent or mitigate the most safety significant of these events.

The PRA model for analyzing the loss of DHR accident initiation identified about 12 systems that can be used to prevent core damage. The resultant core damage frequency was negligible but the focus of the study was to identify minimum combinations of systems that, if available, would result in a conditional core melt of an acceptably low probability given a loss of RHR event. It was found that generally about four of the 12 systems are sufficient to meet the goal.

In all cases, the minimum type and number of systems required by technical specifications (e.g., RHR) plus systems normally operating during shutdown (e.g., CRD and fire water) are sufficient to maintain adequate shutdown safety margins.

Many such combinations are possible, but certain specific combinations of minimum sets of systems have been identified to provide guidance to the COL applicant. Additional minimum sets of systems can be identified by the COL applicant, if needed, by using the PRA model. These combinations of systems identified will allow COL owners much flexibility in preparing outage plans to ensure that shutdown safety margins are adequate at all times.

19Q.4 Features to Minimize Shutdown Risk

As part of the process for certifying the ABWR design, the NRC requested that General Electric provide a specific discussion of ABWR features that minimize shutdown risk.

The list of ABWR shutdown risk features is presented in Table 19Q-1. The features are grouped by risk categories as discussed in NUREG-1449, “NRC Staff Evaluation of Shutdown and Low Power Operation.” Fire protection was not discussed in NUREG-1449 but was added to the list based on discussions with the NRC. The risk categories are:

- Decay Heat Removal
- Inventory Control

- Containment Integrity
- Electrical Power
- Flooding Control
- Reactivity Control
- Fire Protection

NUREG-1449 also discussed reactor coolant system pressurization but this was not included in the list because it is mainly a PWR issue. BWR shutdown pressure control concerns are ultimately inventory (i.e., LOCA) concerns and are addressed under Inventory Control.

The ABWR has been designed with the minimization of risk being a high priority. PRA methods have been very influential in the design of the ABWR. The ABWR features described in Table 19Q-1 along with appropriate Technical Specifications and utility operating and maintenance procedures (which contain insights gained from risk based evaluations) all result in the conclusion that during shutdown conditions the ABWR is adequately protected against accidents and the estimated core damage frequency is negligible.

The following subsections describe the shutdown risk concern, past experience at operating BWRs for each risk concern, and the ABWR features that contribute towards minimizing shutdown risk for each concern.

19Q.4.1 Decay Heat Removal

Shutdown Risk

Loss of decay heat removal (DHR) while shutdown can lead to fuel uncovery and damage. It can be initiated by loss of the operating RHR System or by loss of an intermediate or ultimate heat sink. If loss of DHR occurs shortly after shutdown, bulk boiling of reactor coolant and fuel uncovery can happen quickly (i.e., less than one half hour for bulk boiling and approximately five hours to core uncovery if no protective action is taken).

Past Experience

There has never been a loss of DHR in a BWR which resulted in actual core uncovery but several precursors to such an event have occurred in the past. Subsection 19Q.11 discusses many of these precursor events and describes ABWR features that could have prevented or mitigated each event.

For BWRs, the most common precursor events involved temporary loss of RHR due to various reasons including inability to open Shutdown Cooling (SDC) valves inside containment and isolation of SDC due to low water level in the RPV or loss of power to the Reactor Protection System (RPS). In all of these cases, redundant loops of RHR or alternate DHR methods were available.

ABWR Features

The ABWR contains many features to minimize the loss of DHR. The ABWR contains three divisions of RHR and associated support systems that are electrically and physically separated. This is the first line of defense in maintaining DHR. One RHR loop could be in maintenance and if a single failure were to occur to the operating loop, the third loop could be placed into service. It is also possible, if conditions warrant, to run RHR loops in parallel. In this case, failure of one loop would not result in even a temporary loss of DHR.

In the unlikely event that all RHR loops were unavailable, several alternate methods of DHR from the RPV could be used. Steam from the RPV could be directed to the main condenser (if available). Makeup to the RPV could be supplied by many sources as discussed in Subsection 19Q.4.2. Other potential heat sinks include the suppression pool (via the safety relief valves), or under certain conditions the Reactor Water Cleanup System, or the Fuel Pool Cooling and Cleanup System (if the reactor water level is raised to the refueling level). As a final method, if the RPV head was removed, bulk boiling of reactor coolant in the RPV with adequate makeup would prevent fuel damage.

From the above it can be seen that there are multiple methods to maintain DHR in the ABWR such that the shutdown risk associated with loss of DHR is negligible.

19Q.4.2 Inventory Control

Shutdown Risk

Loss of inventory control can lead to uncovering the fuel and damage by overheating. Reduction of reactor coolant inventory is more likely when the plant is shut down because additional paths for diversion of coolant (e.g., RHR System) are operable. In addition, there are shutdown activities such as test and maintenance that require seldom used valve line-ups and plant configurations which increase the probability of operator errors associated with inventory control.

Past Experience

As discussed in Subsection 19Q.11, events at operating plants have resulted in reduction of reactor coolant inventory. For BWRs this typically involved diversion of reactor coolant from the RPV to the suppression pool due to improper valve line-ups (e.g., opening suppression pool suction valve before SDC suction was fully closed) or valve leakage (e.g., RHR pump mini-recirculation valve). Other inventory losses were due to leaking RHR heat exchanger tubes, placing a partially drained RHR loop online following maintenance, and buckling of an RHR heat exchanger due to marine growth.

In all cases, the loss of inventory was either recovered due to operator action or automatically stopped by isolation of SDC on low RPV level.

ABWR Features

The ABWR contains several design features to minimize the potential for inventory loss. Indication of RPV level is displayed to the operator in the control room continuously during all modes of plant operation (including refueling). To ensure that an adequate level is maintained in the RPV, multiple sources of makeup exist including:

- Suppression pool
- Condensate storage tank
- Main condenser hotwell
- AC-independent Water Addition System

To minimize the potential for pipe breaks, RHR system valves are interlocked with reactor system pressure to ensure that low pressure RHR piping is not exposed to full system pressure. In the event that the interlocks fail or are bypassed, the RHR piping is capable of withstanding full reactor pressure without rupture.

During shutdown there are many maintenance tasks and evolutions that could lead to potential draining of the RPV. These include: CRD and Reactor Internal Pump (RIP) removal and replacement, and failures or operator errors associated with operation of the Reactor Water Cleanup System and the RHR System. These potential drainage paths are discussed below.

CRD Replacement

CRD replacement for the ABWR will use the same procedure followed for current operating BWRs. The CRD is withdrawn to the point where the CRD blade back seats onto the CRD guide tube. This provides a metal to metal seal that prevents RPV drainage when the CRD is removed. The many years of BWR experience with CRD removal gives a high degree of assurance that the risk from this operation will be negligible for the ABWR.

See Subsections 4.6.1.2.1 and 4.6.2.3.4 for additional information on CRD replacement and maintenance.

RIP Motor and Impeller Replacement

Nuclear plants with RIPs have been in operation for over 15 years. Over 500 RIPs and motors have been successfully removed and reinstalled in European BWR plants. This has demonstrated that replacement activities can be carried out without draining the vessel.

Replacement of RIP motor and impeller involves the following steps. The RIP lower bolts are loosened and the pump allowed to move downward approximately 6.25 mm (1/4-inch) to the point where the impeller becomes backseated. An integral inflatable seal is then actuated as a backup sealing device to assure no RPV leakage occurs. The RIP motor can then be removed. Following motor removal, a temporary cover plate is bolted to the bottom. The impeller is then removed from the top. The bolted cover plate prevents leakage of coolant from the RPV. After the impeller is removed, a plug is installed on the RPV bottom head at the impeller nozzle to provide additional protection against draining the RPV.

During maintenance activities on the RIP, there are two periods when the potential for leakage is greatest: when removing the motor and when completing maintenance on the secondary seals. In both these cases, the temporary bottom cover plate is removed. During motor removal, the primary and secondary seals prevent leakage but they could fail. In this case, only small leakage could occur because of the tight clearances between the RIP housing and the impeller shaft. If the seals were to leak, the bottom cover could be bolted in place to prevent further leakage. Maintenance on the secondary seals requires removal of the motor, impeller and shaft, and the temporary bottom cover. A temporary plug is installed in the RIP diffuser before removing the bottom cover plate. This temporary RIP diffuser plug is designed so that it can not be removed unless the RIP motor housing bottom cover is in place. Due to the multiple operator errors required to cause a major leak during RIP maintenance, the risk from RIP maintenance is considered negligible.

See Subsection 5.4.1.5 for additional information on RIP motor and impeller maintenance.

Control Rod Drive Hydraulic System

During operating Modes 4 and 5, the Control Rod Drive Hydraulic System (CRDHS) continues operating with one pump running to provide purge water to the FMCRDs. With one pump in operation, the head of the pumping water can easily overcome the head of water in the RPV; hence, draining the RPV is unlikely. In the event that neither pump is in operation, there are several potential paths for draining the RPV through the CRDHS.

With neither CRD pump operating, the scram valves will open due to low Hydraulic Control Unit (HCU) charging header pressure. The scram valves may remain open due to operator error in not resetting the RPS logic or other system failures such as loss of instrument air to the scram valve. This combined with multiple mechanical failures to check valves and operator errors in CRD hydraulic system valve lineups could result in RPV drainage through the CRD hydraulic system. Multiple failures are required for RPV leakage to occur and even if a leak were to develop, only two CRDs would be affected and the leak would be small since it would occur in a 32A (1-1/4-inch) line. Therefore, the probability of draining the RPV through the CRD hydraulic system is considered negligible.

Reactor Water Cleanup System (CUW)

During shutdown, the CUW provides continuous cleaning of the reactor coolant. Water is removed from the RPV through the RHR shutdown cooling suction nozzle and a line attached to the RPV bottom head and after passing through a series of heat exchangers and a filter demineralizer is returned to the RPV either via an attachment to the upper head or through the feedwater lines and spargers.

Potential drainage paths exist due to several maintenance flush and drain valves and CUW discharge paths to the low conductivity water (LCW) sump and the main condenser. The latter two paths are used during reactor startup to control excess reactor water due to heat up and thermal expansion.

For any of the potential flow paths described above to result in RPV drainage, multiple failures of equipment and operator errors must occur. In addition, if the RPV were to start draining all but one of the potential flow paths (LCW sump) would be automatically isolated on low RPV level. The flow path to the LCW sump is controlled by two valves in series one of which is locked closed and both are under administrative control. If drainage were to occur, LCW sump well level alarms would annunciate in the control room. Also, the line is only 50A (2 inches) in diameter and so the flow rate would be slow enough to allow ample operator time to mitigate the leak.

Because of the multiple failures and operator errors that must occur to cause RPV drainage through the CUW and the automatic RPV isolation logic to stop most potential flow paths, the risk of RPV drainage though the CUW is considered negligible.

Residual Heat Removal System

The ABWR Residual Heat Removal (RHR) System is a closed system consisting of three independent pump loops (A, B, and C—where B and C are similar) which inject water into the vessel and/or remove heat from the reactor core or containment. Loop A differs from B and C in that its return line goes to the reactor pressure vessel (RPV) through the feedwater line whereas loop B and C return lines go directly to the RPV. In addition, loop A does not have connections to the drywell or wetwell sprays or a return to the fuel pool cooling system. However, for purposes of this analysis, the differences are minor and the three loops can be considered identical. The RHR System has many modes of operation, each mode making use of common RHR System components. Protective interlocks are provided to prevent the most likely interactions of mode combinations.

The operator has five mode selection switches available that will automatically perform the required valve alignment for the mode selected. This feature reduces the chance of operator error by only requiring one action, selection of the mode switch, to realign several valves. Only one mode at a time can be operational, thus precluding potential undesirable multiple mode interactions. The five modes are:

(1) RHR initiation

(2) RHR suppression pool cooling

(3) RHR shutdown cooling

(4) RHR standby

(5) RHR drywell spray

There are two basic ways that the ABWR RPV water level can potentially be decreased through the RHR System during shutdown cooling. The first way is through operator error in opening manual isolation valves that are used for RHR System maintenance. These paths are to the High Conductivity Water sump and the Liquid Radwaste Flush System. These valves are normally closed during the shutdown cooling mode of plant operation. These are 50A (2-inch) and 150A (6-inch) lines respectively. Inadvertent opening of these valves would result in a relatively slow RPV level decrease which would be alarmed to the operator in the control room such that there would be adequate time to respond. If the operator failed to notice the decreased RPV level, an alarm would annunciate in the control room and the RPV isolation valves would automatically close on low RPV level. The fuel would remain covered with water and no fuel damage would occur.

The second way that RPV level could decrease would be for one of the motor operated valves (MOVs) in the RHR System to open inadvertently or by operator error. Most of the MOVs in the RHR System are interlocked to prevent inadvertent diversion of RPV water (e.g., the shutdown cooling (SDC) suction line is interlocked so that the suppression pool suction and return valves and wetwell spray valve must be closed before the SDC valve can be opened, the shutdown cooling suction valve must be fully closed before the suppression pool suction or return valve can be opened, the two series dry well spray valves cannot be opened at the same time unless the drywell pressure is high). Thus loss of RPV level through these paths is not likely. Loss of RPV level through the wetwell spray valve requires a mechanical failure or an operator error to open the valve when not required. The only other potential path is via the RHR pump mini-flow valve. This valve is designed to open to allow water flow back to the suppression pool if the RHR pump is running at shutoff head. This is a pump protection feature. The valve opens and closes automatically depending on measured RHR flow.

Whether the potential flow path is caused by mechanical failure or operator error, two features exist to mitigate the loss of RPV level. On a low RPV level signal, both RPV isolation valves close to stop all flow out of the RPV. The RPV low level setpoint is 3.81

meters above the top of the fuel. Even if the low RPV level isolation feature were to fail (after a previous valve mechanical failure or operator error), flow out of the RPV would automatically stop when the RHR shutdown cooling nozzle is uncovered. At this point, 1.7 meters of water would still be above the top of the active fuel. Therefore, the draining of the RPV via the RHR System to the point of uncovering the fuel and causing fuel damage is not considered credible for the ABWR.

Another potential for loss of inventory control is through the use of freeze seals on piping attached to the RPV. Subsection 19Q.8 discusses how freeze seals will be used on the ABWR and why the risks associated with freeze seals will be small.

In summary, the ABWR contains many redundant and diverse features such that, along with the use of experience proven administrative controls, loss of inventory control is not a significant safety concern.

19Q.4.3 Containment Integrity

Shutdown Risk

A breach of containment integrity is not by itself an issue of high safety significance but, in conjunction with other initiating events, could increase the severity of the initiating event. A breach of containment integrity followed by breach of another radiological barrier or boiling of the reactor coolant could lead to a direct release to the atmosphere. Attachment 19QB discusses potential offsite releases following boiling in the RPV with the head removed and shows that releases would be a small fraction of normal operating limits. In addition, the PRA results in Subsection 19Q.7 indicate that the risk of RPV boiling is low.

During refueling of the BWR, the primary containment is open and cannot be readily closed since the drywell head is removed. Nonetheless, loss of containment integrity has not been an issue for BWRs in the past.

The probability of core melt during shutdown is low, but if a core melt were to occur when the primary containment was open, the suppression pool may be bypassed resulting in high offsite doses.

ABWR Features

During shutdown with the drywell head removed, the ABWR has the secondary containment which can be automatically isolated on high radiation from a radiological boundary breach or fuel handling accident.

The Standby Gas Treatment System (SGTS) filters air from the secondary containment to reduce potential contamination to the atmosphere.

The ABWR secondary containment and use of the SGTS results in a negligible risk concern for loss of containment integrity.

19Q.4.4 Electrical Power

Shutdown Risk

A loss of all offsite power challenges the onsite sources to power safety-related equipment to maintain safe shutdown. Loss of individual buses (AC or DC) affects divisional train capability and results in loss of redundancy to complete required safety functions.

Past Experience

As discussed in Subsection 19Q.11, loss of power events have occurred at many nuclear power plants. There have been several cases of a total loss of offsite power which, in some instances, led to loss of shutdown cooling and increases in coolant temperatures of as much as 333 K (140°F).

The majority of total loss of offsite power events were due either to severe weather or operator errors. Several losses of onsite power events were due to objects falling on transformers while operators were performing maintenance activities in the switchyard. In other cases, switching errors resulted in temporary loss of power to vital buses or offsite power.

ABWR Features

The ABWR electrical system has the following features to prevent or mitigate potential loss of power events:

- Three physically and electrically independent Class 1E emergency diesel generators
- Two independent sources of offsite power
- Three unit auxiliary transformers powering three Class 1E and non-1E power buses
- Combustion turbine generator (CTG) that can be used to power any of the Class 1E or non-1E buses

The ABWR electrical power system contains redundancy and diversity of electric power sources. This allows sources to be in maintenance during shutdown and still have adequate power sources to meet potential equipment failures. Even in the case of a loss of offsite power, the CTG has the ability to start a feedwater or other pump for DHR or inventory makeup if required. This means that the ABWR can use alternate sources of DHR with only onsite power sources.

In the event that one phase of the main transformer were to fail, an installed spare is available to return the preferred source of offsite power to service without the need to procure and deliver a new transformer.

As discussed more fully in Subsection 19Q.11, the ABWR electrical power distribution system has features that are capable of mitigating potential loss of power events that have occurred at operating plants in the past. The design features described above in conjunction with appropriate Technical Specifications and other administrative controls result in an electrical distribution system that is able to maintain an adequate level of redundancy and capacity even with equipment out for maintenance or testing. This ensures that safety margins can be maintained at all times during shutdown and normal plant operation.

19Q.4.5 Reactivity Control

Shutdown Risk

Reactivity control during shutdown may be a concern because local criticality can be achieved through movement of control rods or errors in fuel handling that may not be adequately detected by installed neutron detectors. Also at lower temperatures, the inherent negative reactivity feedback available at normal operating temperature and pressure is less able to mitigate potential power excursions.

While overall core shutdown margins are adequate to protect the fuel as long as procedures are followed, inadvertent withdrawal of two adjacent CRDs or fuel handling errors can lead to fuel damage.

Past Experience

A few isolated cases of BWR shutdown reactivity control concerns have been identified in the past and were attributed to operator errors (e.g., withdrawing the wrong control rod).

Reactivity excursion events could occur due to any one of the following:

- Control Rod Drop
- Control Rod Ejection
- Refueling Error
- Rod Withdrawal Error
- Fuel Loading Error

Control Rod Drop

While shutdown, the only time a control rod drop could occur is during control rod testing. If two control rods associated with one Hydraulic Control Unit (HCU) are fully withdrawn, a rod block signal prevents withdrawal of a third control rod. If the rod block signal were to fail and the operator were to incorrectly select an adjacent control rod for withdrawal, a latch mechanism exists such that if the rod were to become stuck and decouple from its drive it could only drop a maximum of 20.32 cm(8 inches). In addition, a Class 1E separation detection system would sense a separated control rod drive and initiate a rod block signal.

Due to the combination of events required to cause a control rod drop including operator error coincident with multiple mechanical failures, the ABWR rod drop accident risk is considered negligible.

Control Rod Ejection

For a control rod ejection accident to occur while shutdown, RPV pressure would have to be increased (e.g., during a hydrostatic test). The series of events that would have to occur are:

(1) During RPV hydrostatic testing one or two control rods associated with an HCU are withdrawn for testing and,

(2) A break occurs:

(a) In the CRD housing of an adjacent rod which also results in failure of the internal control rod anti-ejection supports ("shootout restraints")

or

(b) In the CRD insert pipes coupled with failure of both its ball check valve and electro-mechanical brake.

Due to the short amount of time that the RPV undergoes hydrostatic testing and the multiple failures required for a control rod ejection to occur, the risk from this event is considered negligible.

Refueling Error

During refueling, inserting a fuel bundle into a fueled region of the core which has a withdrawn control rod blade could result in a reactivity accident.

The ABWR features that prevent or mitigate refueling errors are:

(1) An interlock with the mode switch in the REFUEL position which prevents hoisting another fuel assembly over the vessel if a control blade has been removed.

(2) While shutdown, only two control rods can be withdrawn at a time. Any attempt to withdraw a third control rod would result in a rod block signal being initiated by the rod control and instrumentation system. During refueling, technical specifications allow only one rod to be withdrawn at a time.

(3) The operator would be alerted to a refueling error by the source range neutron monitoring system.

Due to the combination of operator errors, interlock failures, and core configuration required for this event to occur, refueling accident risks are considered negligible.

Rod Withdrawal Error

If two adjacent control rods are withdrawn at the same time, the reactor may become critical. To prevent this, the ABWR has an interlock which prevents adjacent control rods being withdrawn at the same time. Two control rods associated with an HCU can be withdrawn at the same time but these rods are separated by at least two cells. An interlock prevents withdrawal of a third rod. If the interlock fails and the rod is withdrawn, the rods would scram on a high flux signal.

The coincident failures of the rod withdrawal interlock and Reactor Protection System in conjunction with operator error, which are required to cause a rod withdrawal error are considered improbable and the risk negligible.

Fuel Loading Error

This event is similar to a refueling error. In this case the refueling procedure is not followed and a higher than design core reactivity configuration is formed. If not identified by the core verification process, subsequent control rod testing may result in inadvertent criticality and power excursion. A high flux scram would terminate the excursion.

The risk from a fuel loading error is considered negligible because of the combination of events required for the accident to occur.

Summary of Reactivity Control

The ABWR refueling interlocks, control rod design, Reactor Protection System operability during shutdown, and strict administrative controls all combine to support the conclusion that shutdown Reactivity Control is a negligible risk concern for the ABWR design.

19Q.4.6 Summary of Shutdown Risk Category Analysis

The ABWR design was evaluated against shutdown risk categories from NUREG-1449. The analysis took into account past experience at operating BWRs. The conclusion from this analysis is that the ABWR design contains multiple features to minimize potential risk during shutdown for the major shutdown risk categories.

19Q.5 Instrumentation

The ABWR instrumentation system contains many features that help reduce shutdown risk. These features are contained in the basic design of the instrument systems and in the type and number of parameters monitored.

During shutdown, the main concern from a risk perspective is removal of decay heat from the fuel in the RPV. The large volume of water in the spent fuel pool and low probability of draining makes the risk associated with fuel pool operation relatively low. The smaller reactor pressure vessel (RPV) volume and relatively high decay heat load of the fuel increases the cooling requirements and decreases the available time to recover from loss of decay heat removal (DHR). Thus, to minimize shutdown risk, the instrumentation system must monitor RPV level and water temperature, status of makeup sources and heat sinks, and display these to the plant operators in a reliable and easy to understand manner.

Design Features

The ABWR utilizes redundant channels of safety-related instruments for initiating safety actions and monitoring plant status. This is accomplished by a four division correlated and separated protection logic complex called the safety system logic and control (SSLC). The SSLC receives signals from the redundant channels of instrumentation, displays information to the operator, and makes decisions on safety actions.

The safety system setpoints are determined by analysis and experience, factoring in instrument errors, drift, repeatability, safety margins, and the need to minimize spurious actuations. The system provides continuous automatic online testing of the logic and offline semi-automatic end-to-end (sensor input to trip actuator) testing. This combination meets all current regulatory requirements.

Specific instrumentation features important to shutdown operations include:

- Automatic initiation of ECCS to ensure adequate RPV makeup.
- Four channels of instrumentation to allow for bypass during maintenance and testing while still retaining redundancy. (The two-out-of-four logic reverts to two-out-of-three during maintenance bypass).
- Continuous monitoring for detection of fires or flooding in safety-related and other areas.
- Operability of the Reactor Protection System (RPS) during shutdown to mitigate potential reactivity excursions.
- Interlocked refueling bridge operation to prevent reactivity excursion.

- Automatic isolation of shutdown cooling (SDC) on low level in the reactor pressure vessel (RPV) to ensure against fuel uncovery.
- Interlocked residual heat removal (RHR) valves (SDC and suppression pool) to reduce the potential for diversion of coolant from the RPV to the suppression pool.
- Ability to control shutdown plant status from the remote shutdown panel in the event that the control room becomes uninhabitable.
- Ability to monitor radiation levels throughout the plant to detect breaches in radiological barriers.

Parameters Monitored

The key shutdown parameters monitored by the ABWR instrumentation system include:

- RPV level, water temperature, and pressure
- Neutron flux
- Drywell and wetwell pressure and temperature
- Suppression pool temperature and level
- Reactor, turbine and control building flooding level
- RHR flow rate, temperature, pump motor trip, and loop logic power failure
- CUW outlet temperature
- Fire detection in various buildings
- Electric power distribution system parameters (e.g., power, voltage, current, frequency)
- Operation of fire water system

19Q.6 Flooding and Fire Protection

The ABWR has been designed to minimize the risks associated with fires and flooding through the basic layout of the plant and the choice of systems to enhance the plants tolerance to fires and flooding.

Plant Layout

The plant layout is such that points of possible common cause failure between safety-related and non-safety-related systems have been minimized. As an example, the control room is situated between the reactor building and the turbine building. Thus safety-related equipment and controls that are used to shutdown and maintain long term cold shutdown of the plant cannot be impacted by failures of non-safety-related systems in the turbine building. Likewise, non-safety-related systems / equipment in the turbine building that could be used to reach and maintain cold shutdown (e.g., condensate, main condenser) are not affected by failures of safety-related equipment, therefore, interactions between reactor and turbine building systems are minimized.

Normal and alternate preferred power is supplied through the turbine building to the reactor building for safety-related loads. These non-safety-related power sources are backed-up by safety-related diesel generators located in the reactor building. The diesel generators are thus not affected by events in the turbine building.

The buildings are laid out internally so that fire areas of the same division are grouped together in block form as much as possible. This grouping is coordinated from building to building so that the divisional fire areas lineup adjacent to each other at the interface between the reactor and control building. An arrangement of this fashion naturally groups piping, HVAC ducts, and cable trays together in divisional arrangements and does not require routing of services of one division across space allotted to another division.

A major difference between the ABWR and current reactor designs is that due to the multiplexing of plant systems, there is no need for a cable spreading room. This removes a significant source of potential fires that could lead to core damage both during normal plant operation and shutdown conditions.

Systems

The ABWR has three independent safety-related divisions, any one of which is capable of maintaining the reactor in a safe cold shutdown condition. With this arrangement, a single division may be out for maintenance and a single random failure could occur which disabled another division, but the third division could be available to ensure continued DHR. In addition, there are non-safety-related systems such as condensate that can be used to maintain cold shutdown.

In general, systems are located and grouped together by safety division so that; with the exceptions of the primary containment, the control room, and the remote shutdown room (when operating from the remote shutdown panels); there is only one division of safe shutdown equipment in a fire area. Complete burnout of any fire area without recovery will not prevent continued decay heat removal (DHR), therefore, complete burnout of a fire area is acceptable from a public risk perspective.

The separation exception in the primary containment is made because it is not practical to divide the primary containment into three fire areas. The design is deemed acceptable because:

(1) Sprinkler coverage is provided by the containment spray system.

(2) Only check valves and ADS / SRV valves (if the RPV head is on) are required to operate within containment to provide DHR. A fire could not prevent the operation of a check valve nor would it prevent a safety valve from being lifted on its spring by pressure. The high pressure pumps are capable of providing water to the core up to the set point of the SRVs. Thus, a fire could not prevent injection of water to and relief of steam from the reactor vessel.

(3) In addition, maximum separation is maintained between the divisional equipment within primary containment.

All divisions are present in the control room and this cannot be avoided. The remote shutdown panel provides redundant control of the DHR and ECCS functions from outside of the control room. The controls on the remote shutdown panel are hard wired to the field devices and power supplies. The signals between the remote shutdown panel and the control room are multiplexed over fiber optic cables so that there are no power supply interactions between the control room and the remote shutdown panel.

There are some areas where there is equipment from more than one safety division in a fire area. Each of these cases is examined on an individual basis to determine that the encroachment is required and that failure in the worst conceivable fashion is acceptable. These are documented in Subsection 9A.5.5 under “Special Cases—Fire Separation for Divisional Electrical Systems.”

Divisions I and II 125 VDC and 120 VAC power supplies, reactor building cooling water pumps and heat exchangers, emergency chillers and emergency HVAC Systems are located in the control building. Since these systems are required for DHR if the function of the control room is lost, they are separated from the control room complex and its HVAC System by rated fire barriers. A fire resulting in the loss of function of the control room will not affect the operation of the remote shutdown or remote shutdown support systems.

When the plant is shutdown and, if due to normal maintenance or other work, fire barriers must be breached between two safety divisions, the third division must be operable and its barriers checked to ensure they are intact.

Fire Containment

The fire containment system is a combination of structures and barriers that work together to confine the direct effects of a fire to the fire area in which the fire originates. The fire containment system is comprised of the following elements:

(1) Concrete fire barrier floors, ceilings and walls which must be at least 15.24 cm (6 inches) thick if made from carbonate and silicious aggregates. Other aggregates and thicknesses are acceptable if the type of construction has been tested and bears a UL (or equal) label for a three hour rating.

(2) Fire doors, which are required to have a UL (or equal) label certifying that they have been tested for a three hour rating per ASTM E119, including a hose stream test.

(3) Electrical penetrations which are required to have been type tested to ASTM E119, including a hose stream test.

(4) Piping penetrations which are required to have been type tested to ASTM E119, including a hose stream test.

(5) Fire dampers for any HVAC duct penetrating a fire barrier and which must have a rating of three hours. The only fire dampers separating divisions are in the HVAC duct for secondary containment (six total). The plant arrangement minimizes fire dampers.

(6) Fire rated columns and support beams, which are required to be of reinforced concrete construction or, if of steel construction, enclosed or coated to provide a three hour rating.

(7) Backup of the fire barrier penetration seals by the HVAC Systems operating in the smoke removal mode. This backup feature is accomplished in the reactor and control buildings by maintaining a positive static pressure for the redundant divisional fire areas with respect to the fire area with the fire. Leakage is into the fire impacted area under sufficient static pressure to confine smoke and heat to the fire area experiencing the fire, even if there is a major mechanical failure of the penetration seal.

Other aspects of the ABWR design that minimize the risk due to fires while shutdown are:

- HVAC Systems dedicated to the divisional areas which they serve.
- A smoke control system to remove smoke and heat from the affected area, to control the pressure in a room due to a fire, assure that any fire barrier leakage is into the fire area experiencing the fire, and supply a clean air path for fire suppression

personnel. The HVAC System has been designed for the dual purposes of HVAC and smoke control.

- Fire alarm systems.
- Fire suppression system to automatically initiate, where appropriate, and extinguish fires.
- Manual fire suppression equipment such as hand held CO_2 or chemical fire extinguishers, and water hoses.
- Administrative controls to ensure that at least one safety division is available with intact barriers at all times.

Fires During Maintenance

When the plant is shutdown, maintenance activities may require breaching the fire barriers for one or more activities. The recommended outage philosophy regarding fire barrier integrity is that through administrative controls, one division of safety equipment will be available (i.e., not in maintenance) and its physical barriers will be intact. This division will be in standby and one other division will be operating to remove decay heat and complete other required functions (e.g., fuel pool cooling, CRD purging, reactor water cleanup). The third division could then be fully in maintenance. In this configuration, a fire in any one division would not result in loss of decay heat removal capability. If the fire were to occur in the intact division, the fire barriers would restrict the fire to that division only and the operating division could continue to remove decay heat. For fires in either of the other two divisions, even if the barriers between the two divisions were breached, the intact division would be available to remove decay heat. See Subsection 19.9.24 for COL license information requirements.

As discussed more fully in 19Q.7, the COL applicant must identify a minimum set of systems that will not be in maintenance such that the conditional probability of core damage due to certain initiating events is maintained acceptably low. The minimum set selected should take into account fires in various locations of the plant. If the above outage philosophy is followed, the risk from fires during shutdown conditions will be low.

Flooding

Many of the features that are designed to mitigate fires also serve to protect the plant from damage due to flooding. Physical separation of safety divisions not only prevents propagation of fires but also restricts or prevents flooding of safety-related equipment. The fire barriers will also prevent potential water from entering a divisional area due to flooding from non-divisional sources or contain water in the fire area for divisional water sources.

Other aspects of the ABWR design that minimize the risk from flooding are the practice of not routing unlimited sources of water (e.g., service water) through ECCS room areas and ensuring that other large water sources (e.g., suppression pool) can be contained without damaging equipment in more than one safety division if a flood were to occur.

A review has been completed of all ABWR internal flood sources and the results show that during shutdown conditions at least one safety division would be unaffected by water damage for any postulated flood. Features, beside separation, that contribute to this low level of risk are: Adequately sized room floor drains, water level alarms and automatic isolation of flood sources for potentially affected rooms, mounting motors and other electrical equipment at least 20.32 cm above floor level, and using watertight doors. As was discussed under fire protection, administrative controls will be implemented to assure that at least one safety division with intact barriers is available at all times during plant shutdown. The seals on the doors seat with water pressure from floods outside the room but only small leakage past the seals is expected from flooding in the ECCS room. Therefore, during shutdown if maintenance tasks require breaching the barriers of two divisions, flooding in the intact division will not cause damage to equipment in all three divisions. Additional details on the ABWR flood mitigation capability is contained in Appendix 19R.

Summary of Fire and Flood Features

The ABWR has been designed to minimize the risk due to fires or flooding during shutdown conditions by plant configuration and system design. Divisional separation, both physically and electrically, as well as fire/flooding mitigation systems exist to reduce plant risks from these potential accidents. Along with these design features, administrative controls are implemented to ensure that at least one safety division is not in maintenance and its physical barriers are intact.

19Q.7 Decay Heat Removal Reliability Study

19Q.7.1 Introduction

As part of the ABWR shutdown risk evaluation, a reliability assessment of the decay heat removal (DHR) capability was completed. Decay heat removal reliability has received increasing attention due to events such as those at Vogtle and Diablo Canyon where decay heat removal systems were made inoperable due to loss of electric power and other causes.

Attachment 19QC summarizes approximately 200 events at operating plants which were either loss of decay heat removal events or precursors to such events. The relatively large number of events underscores the potential for loss of decay heat removal events and the potential for associated core damage.

19Q.7.2 Purpose

The purpose of this study is to determine the minimum number of systems that might be available during shutdown to ensure that the risks associated with loss of decay heat removal events are acceptable. That is, given a loss of the operating RHR System for any reason, the subsequent conditional probability of core damage remains acceptably low using only those systems that are potentially available (i.e., system not in maintenance but which could experience random failures).

The results of this study provide guidance regarding various combinations of systems, that if kept available during a plant outage, will ensure that the risk associated with loss of DHR Systems will be acceptable. A utility may choose to keep more systems available but as long as a minimum set is made available, shutdown risk will be considered acceptable. This minimum set of systems will give a utility flexibility in scheduling maintenance activities for DHR Systems. The minimum sets described in this study are representative of acceptable combinations. There may be additional sets of equipment that were not included in this study which would also result in acceptable risk levels.

The minimum sets of systems take into account plant conditions (i.e., modes) and the fuel decay heat generation rate as a function of time. Both safety and non-safety (e.g., power conversion) systems are included in the minimum sets.

19Q.7.3 Summary

Using probabilistic risk assessment (PRA) techniques, an acceptable level of shutdown risk was demonstrated for various minimum sets of equipment and systems that were assumed to be available (i.e., not in maintenance).

These minimum sets were determined for an initiating event involving loss of an operating RHR System. The three primary causes of a loss of the RHR System were identified to be the following:

(1) Mechanical or electrical failures in the operating RHR System

(2) Loss of the operating service water pump associated with the operating RHR System

(3) Loss of offsite power

Loss of operating service water pump and offsite power were evaluated separately as the cause for loss of the operating RHR because of their impact on other DHR Systems. Each potential cause for loss of the RHR System was considered an initiating event.

Success criteria were determined for each initiating event, taking into account decay heat load and plant operating mode. Minimum complements of systems that will prevent core damage given the initiating event and the time dependent core decay heat generation rate were then identified.

Event trees were generated based on the assumed initiating event and applicable success criteria. System failure probabilities were determined with the help of fault tree analysis.

The results from the study are summarized in Tables 19Q-3, 19Q-4, and 19Q-5. The tables show that significant flexibility exists for completion of system maintenance during outages while still maintaining adequate safety margins. These minimum sets of systems can be used by utilities for initial outage planning and for evaluating changes to outage schedules to ensure adequate safety margins are maintained at all times during the outage. The risk goal can, in general, be met by just those systems required to be operable (and therefore available) by the ABWR Technical Specifications plus normally operating systems (e.g., CRD, fire water, CUW, FPC).

19Q.7.4 Methodology

The methodology used in this study was the same utilized in full power PRAs (i.e., event trees and fault trees). The plant is assumed to be shutdown with decay heat being removed by the RHR System in the shutdown cooling (SDC) mode. Loss of the operating RHR System is then assumed. The loss could occur due to mechanical or electrical component failures of the RHR System, loss of service (i.e., cooling) water pump in the same division as the operating RHR System, or loss of offsite electrical power. The three types of failures are assumed to be initiating events.

For each initiating event, the success criteria were determined. The success criteria are the minimum complement of systems that are capable of preventing core damage. As the decay heat load is dependent on the time following shutdown, the minimum systems required to remove the decay heat will also be time dependent. Therefore, the success criteria have been determined as a function of time. Subsection 19Q.7.6 discusses the success criteria in more detail.

With the help of the success criteria, event trees for each initiating event were developed for each period. Subsection 19Q.7.7 discusses the event trees.

The branch points on the event trees model the probability of success and failure for each system included in the success criteria. The failure probability for each system was evaluated by a fault tree analysis. The fault trees model potential system failures due to mechanical failure of components, loss of electric power to pumps or valves, or operator errors associated with manual actions (e.g., valve line ups or remote control of pumps and valves). Unavailability due to maintenance was modeled as follows. For a system included in a minimum set, the maintenance unavailability was taken to be 0 (i.e., the system is assumed to not be in maintenance). For a system not in a minimum set, the maintenance unavailability was taken to be 1. In other words, the system was assumed to be completely unavailable. This is a very conservative assumption because it is unlikely that all systems allowed to be in maintenance would all be in maintenance at the same time. In addition, some systems in maintenance might be returned to service in time. The fault trees used in this study are contained in Attachment 19QA. A mission time of 24 hours was used for this study. The loss of RHR event is assumed to terminate successfully if the mitigating systems start and run for a period of 24 hours. It is assumed that provisions for long term maintenance of decay heat removal will be made within 24 hours. This assumption is consistent with other full power PRAs.

A number of deterministic analyses were performed and documented in Attachment 19QB. These include the estimation of time available for operator action and human reliability analysis to estimate the probability of operator error under various conditions.

The event trees were quantified with an initiating event frequency of 1.0. Thus the core damage probability that is obtained by this evaluation yields the conditional probability of core damage given a loss of decay heat removal event. The event trees were quantified assuming various complements of systems to be available. The various minimum complements of systems that met the goal were selected for inclusion in Tables 19Q-3 through 19Q-5.

Maintenance of the suppression pool was not modeled in this study. If the suppression pool level must be lowered for any reason, several options exist, such as: off loading all fuel in the RPV to the spent fuel pool or making systems available which do not rely on the suppression pool as a source of water (e.g., condensate, fire water, HPCF). From a risk perspective, the suppression pool should only be drained during periods when it is not relied upon for a source of water or heat sink in performance of an ECCS function. If the above recommendations are followed, the suppression pool unavailability will have a negligible impact on core damage frequency during shutdown.

19Q.7.5 Core Damage Probability Goal and RPV Boiling

The conditional core damage probability goal was selected for this study for the following reasons. The initiating event frequency for loss of an RHR System is not included in this probability goal, but can be conservatively assumed. In the analysis, it is conservatively assumed that all systems not explicitly required to be kept out of maintenance are totally unavailable (i.e., all in maintenance).

The ABWR meets the NRC goal of an overall core damage frequency of 1.0E–04 and a large release goal of 1.0E–06 per reactor-year. In reality, loss of RHR events occur less than assumed and more importantly, not all systems allowed to be in maintenance will all be in maintenance at the same time. Typically, the results show that more than six to ten systems are allowed to be under maintenance and there is a very low probability that all the systems will be simultaneously under maintenance. An analysis using more realistic maintenance unavailability assumptions results in lower core damage frequency estimates. The simplifying assumption of 0 or 1 for maintenance unavailability allows for the calculation of core damage probabilities without having to model maintenance unavailability for each system. This avoids discussion of overlapping maintenance periods for systems during outages. These conservative assumptions allow for a straightforward determination of minimum system availabilities that also meet the NRC risk goals.

In Mode 5 with the RPV head removed, it is assumed that successful DHR can be achieved by allowing water in the RPV to boil and making up lost water by various water sources. Boiling under these conditions is an effective means of DHR but it is not desirable because the resultant pressure buildup in secondary containment could cause loss of containment integrity (i.e., steam release to the atmosphere). Calculations presented in Attachment 19QB show that the boiling release rates, assuming no core damage, are well below allowable limits for normal plant operations.

Equipment in the reactor building would be exposed to the steam environment including: CRD, CUW, RHR, HPCF, and FPC. No other areas in the plant would be exposed to the steam environment that operators would need to enter to assure continued decay heat removal capability.

The RHR and HPCF Systems are qualified for a harsh environment and their operation would not be affected by the steam. The impact of the steam on operation of CUW or FPC is a moot point because either the systems had previously failed or the decay heat load exceeded their capacity or boiling would not have occurred.

The CRD System is not qualified for a steam environment but due to its hardy construction it would be expected to operate for some period of time. Depending on the decay heat load, the time to boiling could vary between 4 - 26 hours. After 4.5 days, the time to boiling is approximately 15 hours and after 14 days is 26 hours. Therefore, for most of the outage, the CRD System could be relied upon for makeup for a significant period of time following loss of normal decay heat removal before being damaged by the steam environment.

There would also be non-safety-related equipment in other buildings that would not experience the steam environment which could be relied upon for makeup (e.g., condensate). The fire water system can also be used for makeup at low pressure.

The fire water system ties into the RHR System through a connection on the outside of the reactor building. Three RHR valves inside the reactor building must be manually opened to inject fire water into the RPV. Adequate time would be available to open these valves following loss of RHR before boiling occurred so that the operator would not be affected by the steam environment. All other operator actions to mitigate loss of RHR can be performed outside the reactor building.

19Q.7.6 Success Criteria

In order to prevent core damage given an initiating event, sufficient systems must be available to ensure that the core decay heat is removed and the fuel remains covered by water. No fuel damage will occur as long as the fuel remains covered by water. There are three ways to achieve success:

(1) Remove decay heat directly from the coolant in the RPV

(2) Remove decay heat indirectly by condensing the steam produced, and provide makeup water to the RPV

(3) Allow the coolant to boil in the RPV and provide makeup water to the RPV to keep the core covered

These three ways to achieve success are discussed in detail below:

(1) Direct Decay Heat Removal from RPV

Recovery of the failed RHR System, use of one of the other two RHR Systems (SDC) or the Reactor Water Cleanup (CUW) System (under certain plant conditions) is sufficient for success. The CUW System capacity is temperature dependent and requires both pumps and nonregenerative heat exchangers (the regenerative heat exchangers must be bypassed). In Mode 5, the Fuel Pool Cooling and Cleanup (FPC) System can be used after the reactor cavity is flooded. FPC alone after 10 days is sufficient to remove all the decay heat. Both FPC pumps and heat exchangers and the supporting systems are required. CUW can remove the entire decay heat 8 days after shutdown.

(2) Decay Heat Removal and RPV Water Makeup

Under certain plant conditions the main condenser, if available, can be used to remove decay heat by condensing steam. The MSIVs must be opened and a condensate return path to the RPV is required. If the condenser is unavailable, steam can be released through the SRVs into the suppression pool and RPV makeup can be supplied by several sources. The availability of the SRVs is not explicitly modeled. At least one SRV is expected to be operable in the safety mode (i.e., spring pressure) even if power is not available.

High pressure makeup can be accomplished by the HPCF, CRD, or feedwater and condensate systems. Low pressure makeup is available from the condensate, LPFL or AC-independent Water Addition Systems. Low pressure makeup may require depressurization of the RPV by actuation of ADS or individual SRVs.

(3) In Mode 5 with the RPV head removed, boiling of water in the RPV with adequate makeup from low or high pressure sources is considered success for the purposes of this study.

Mitigation of loss of offsite power requires recovery of offsite power or use of the emergency diesel generators or combustion turbine generator. The AC-independent Water Addition System can be used for make up in the event of a loss of all AC power.

The success criteria and event trees do not explicitly model maintenance of RPV water level for availability of the Reactor Water Cleanup System or RHR. RPV level is assumed to be maintained by automatic activation of ECCS (i.e., LPFL or HPCF) in Modes 3 - 4 and Mode 5 (reactor cavity unflooded). In Mode 5 with the reactor cavity flooded, RPV level control is assured since the water level will be 7.01 m (23 feet) above the RPV flange.

Table 19Q-2 summarizes the loss of RHR success criteria.

19Q.7.7 Accident Progression and Event Trees

Loss of RHR may initiate from a failure in the operating RHR System, loss of operating Service Water pump, or loss of offsite power. The accident progression for each of the above initiators is discussed below.

19Q.7.7.1 Loss of RHR Due to Failure in the Operating RHR System

Following reactor shutdown, the plant is cooled down by rejecting steam to the main condenser and making up water loss in the RPV by the feedwater system. The RHR System in the SDC mode can be initiated at about 1.034 MPa which corresponds to approximately 456 K (360°F). The RHR System is then used to cool down to either Mode 4 [less than 367 K (200°F)] or Mode 5 (refueling). Loss of the operating RHR loop is assumed to occur sometime after it has been initiated.

Loss of RHR in Mode 3 or 4

Figures 19Q-1 and 19Q-2 are the event trees for loss of RHR in Mode 3 or 4, respectively. The following discussion applies to both event trees. Following loss of the operating RHR loop (event tree node RHR), the operator has to recognize the event and start following the correct procedure (OP). The sequence of events following the successful outcome at this node is described first. The operator can identify the failed system and request the maintenance crew to restore it to operation. An analysis showed that for the decay heat load at this time, water in the RPV would begin to boil in 1.3 hours. Using a typical mean time to repair for the RHR System, and 1.3 hours as the time for recovery, the system recovery probability was determined (REC).

If the failed RHR System cannot be recovered, the operator could initiate one of the other two RHR Systems, if available, in the shutdown cooling mode (R). If all RHR Systems fail, the RPV would pressurize and the main condenser could be made available (V2) by opening the MSIVs, drawing a vacuum in the condenser, and operating the feedwater and condensate pumps for makeup.

If the main condenser fails or is unavailable, the operator can use the CUW System to remove the decay heat (W2) if the RPV temperature is above 386 K (234°F).

If all DHR means are unavailable, the only path to success is to keep the core covered by either high pressure or low pressure sources. The high pressure sources are feedwater and condensate (Q), HPCF (UH), or a CRD pump (C). The HPCF initiates automatically whereas the other two systems require operator action. If all these fail, the operator must depressurize the RPV by actuation of individual SRVs or ADS will initiate automatically (X) on low water level in the RPV. Successful depressurization would make the LPFL (VI), condensate (CDS), or AC-independent Water Addition (FW) Systems available.

Failure to depressurize the reactor or failure of FW leads to core damage.

If at node OP, the operator fails to follow the correct procedure, the reactor coolant temperature and pressure in the RPV will rise, the SRVs will open and discharge steam to the suppression pool and eventually the HPCF will initiate (UH) on low RPV water level. If HPCF fails, ADS will actuate on low water level (X). Failure to depressurize will lead to core damage. Following successful reactor depressurization, LPFL will inject on low water level(VI). Failure to inject with LPFL leads to core damage.

Loss of RHR in Mode 5

Figure 19Q-3 shows the event tree for loss of RHR in Mode 5 less than 3 days after shutdown. This sequence is the same as the previous one except that since the RPV head is removed, the main condenser and feedwater pumps are unavailable and ADS is not required as the RPV cannot become pressurized. Also, at this low temperature, CUW by itself is not capable of removing all the decay heat generated within three days of shutdown.

Figure 19Q-4 shows the event tree for loss of RHR in Mode 5 for 3 - 8 days after shutdown. Figure 19Q-5 shows the event tree for loss of RHR in Mode 5 for the period 8 - 10 days and Figure 19Q-6 shows the event tree for greater than 10 days. The differences in these event trees are that for the period 8 - 10 days CUW alone is success (W2) and beyond 10 days FPC alone (FPC) is success.

19Q.7.7.2 Loss of RHR Due to Loss of Service Water

Figures 19Q-7 through 19Q-16 show the event trees for loss of the Division A or C operating service water pump. The scenarios are basically the same as for a loss of RHR except that loss of the operating service water pump may impact other DHR or makeup systems in addition to the operating RHR pump. Loss of both service water pumps in Division A or B (operating and standby pumps) also results in loss of CUW and FPC. For Division B, the HPCF(B) is also lost (Division A contains RCIC which is not available during shutdown). Likewise, loss of both service water pumps in Division C causes loss of HPCF(C) in addition to the Division C RHR pump. Loss of Division B service water is identical to loss of Division A service water, therefore no event trees were developed specifically for Division B.

19Q.7.7.3 Loss of RHR Due to Loss of Offsite Power

Figures 19Q-17, 19Q-18 and 19Q-19 show the event trees for loss of offsite power in Modes 3, 4, and 5, respectively. The success criteria are the same but longer time is available for recovery in Mode 5. Following a loss of offsite power, it is possible to recover power in time to prevent core damage. If power is not recovered, the available DG will start automatically, and if the DG fails, CTG can be manually initiated. Following loss of all AC power, the AC-independent Water Addition System can be used for make up if the RPV can be depressurized by opening SRVs.

19Q.7.8 System Fault Trees

The unavailability of a system to perform its safety function on demand given a loss of RHR was evaluated by fault tree analysis. Eleven fault trees were used in this analysis. The fault trees are contained in Attachment 19QA.

Five of the fault trees: HPCF, RHR (SDC), RHR (LPFL), Reactor Building Service Water, and ADS, were taken from the full power PRA with modifications (e.g., maintenance unavailability and operator actions) to reflect shutdown conditions. The other six fault trees were developed specifically for the shutdown PRA and include:

- Reactor Water Cleanup,
- Fuel Pool Cooling,
- Main Condenser,
- CRD,
- Condensate, and
- Feedwater.

The fault trees model system unavailability due to mechanical failures, loss of power, and operator errors. As previously mentioned, maintenance unavailability is either assumed to be 1 or 0 (i.e., system is in or out of maintenance).

The unavailability of the AC-independent Water Addition System was estimated based on the assumed operator error in manually initiating the system.

19Q.7.9 Results and Conclusions

19Q.7.9.1 Introduction

The event trees described in the previous subsection were evaluated and the core damage probability calculated with certain systems assumed unavailable due to maintenance. In general, the minimum set of equipment assumed to be available was initially taken as that required by the Technical Specifications for the given operating mode. Combinations of systems were made available until a set resulted in a conditional probability of less than the selected threshold. Each of these sequences that met the acceptance criteria is considered a minimum set for assuring acceptable shutdown risk.

Minimum sets were obtained for each of the three loss of RHR initiators:

- Loss of Operating RHR System
- Loss of Operating RSW Pump
- Loss of Offsite Power

Tables 19Q-3 through 19Q-5 list certain minimum sets of systems that meet the acceptance criteria for loss of the operating RHR System initiator for the three major configurations during shutdown. The configurations are: Modes 3 or 4, Mode 5 prior to flooding the reactor cavity, and Mode 5 after the reactor cavity has been flooded. The effect of changes in decay heat, as a function of time, will be discussed for each of the three plant conditions.

With about 12 systems available, and about four needed to meet the goal, many minimum sets can be identified. In order to simplify the selection of minimum set systems, the following maintenance philosophy was assumed: all of division C in maintenance; division B, ADS, and combustion turbine generator (CTG) are available. Although the CTG is not covered by Technical Specifications, its availability is assumed to be controlled by Administrative Procedures. Other maintenance philosophies can be adopted and the model used to identify appropriate minimum systems. Additional details of the plant configuration based on selected maintenance philosophy is as follows. The plant is being cooled through use of RHR “A” and its support systems (i.e., service water “A”, RCW “A”, electric power division “A”). Other division “A” systems, including EDG “A” may be in maintenance unless specifically included as a support system in one of the minimum sets. All division “B” systems are assumed to be not in maintenance, although they may become unavailable due to random failures or operator errors. All division “C” systems are assumed to be in maintenance. For the above assumed configuration, one of the isolation valves for RHRB is powered by division “C” (due to single failure concerns with containment isolation). If RHRB is required, division “C” power can be made available momentarily. This configuration was selected because it is one that meets minimum technical specification requirements (i.e., 2 ECCS and 2 RHR Systems available). Other configurations could have been selected but this one is typical and the resulting minimum sets identified will demonstrate the low risk associated with loss of decay heat removal for the ABWR and the flexibility afforded utilities for outage maintenance scheduling while still maintaining low risk levels. If one of the assumed power sources becomes unavailable, the utility should make another power source (e.g., a second EDG) available to ensure the safety criterion will be met. Normal surveillance testing should be used to assure the availability of these systems.

19Q.7.9.2 Loss of RHR Initiator

The minimum set for loss of RHR is discussed first. Table 19Q-3 lists some minimum sets of systems that if available during Mode 3 or 4 meet the core damage criterion. As can be seen, if the 2 ECCS Systems are assumed to be RHR, then only a CRD pump plus AC-independent Water Addition or CUW plus AC-independent Water Addition need be made available. This is not restrictive since one pump from CRD and firewater are usually available for other reasons (e.g., CRD to purge the FMCRDs and AC independent water addition for fire protection) and CUW is usually operable during this period. The table shows five different minimum sets. This is indicative of the flexibility for performing ABWR shutdown maintenance while still maintaining risk margins.

Table 19Q-4 lists some minimum sets of systems for Mode 5 during 2 - 3 days after shutdown. In this configuration the RPV head bolts have been detensioned and the head is off but the reactor cavity has not been flooded. For this Mode 5 configuration, fewer systems are available than during Mode 3 or 4 or after flooding the reactor cavity but enough systems are available to ensure adequate risk margins. Also, this is a relatively short duration of the outage. The main condenser is not available since the RPV cannot be pressurized. Fuel pool cooling cannot be used because the RPV and fuel pool have not been connected together and CUW capacity is not sufficient to remove all the decay heat due to the low RPV temperature and high decay heat load. The table shows three minimum sets of systems which meet the risk criteria. As was noted for Modes 3 and 4, the CRD pump which is normally available in addition to fire water and RHRB meet the core damage criterion. Another minimum set might be RHRB (SPC) condensate, and fire water.

Table 19Q-5 lists nine minimum sets for Mode 5 following 3 days after shutdown when the reactor cavity is flooded. RHR plus condensate and fire water meet the criterion. After 8 days, CUW and firewater along with either CRD or condensate meet the criterion, and after 10 days, FPC, CRD, and condensate could be a minimum set. As time following shutdown increases, more systems become able to remove decay heat and greater time is available for operator actions prior to boiling or core damage.

As Tables 19Q-3 through 19Q-5 illustrate, many combinations of systems can be made available to ensure adequate shutdown risk while still allowing for maintenance to be performed on systems. As previously mentioned, these minimum sets are only a few of the possible combinations that will ensure adequate shutdown risk margins. Other minimum sets can be identified for different assumed plant conditions. An important point that is illustrated by the minimum sets identified in this study is that under all shutdown plant conditions, minimum technical specification requirements plus systems that are normally operating or available during shutdown (e.g., CUW, FPC, CRD, and fire water) are enough to ensure adequate shutdown risk margins.

19Q.7.9.3 Loss of Service Water (SW) Initiator

If loss of the operating SW pump is assumed to be the initiating event, all the minimum sets in Tables 19Q-3 through 19Q-5 would be applicable. This is because there is only one RHR pump per division, while one standby pump supports the operating pump in each division of service water. Therefore, RHR (A) is not lost due to the failure of the operating RSW pump. With RHR (A) available, the minimum sets previously identified without RHR (A) must still be able to meet the acceptance criteria. Loss of the operating RHR pump is the limiting condition for this analysis.

19Q.7.9.4 Loss of Offsite Power Initiator

For a loss of offsite power initiator, calculations have shown that the probability of failing both the EDG and the CTG along with not recovering offsite power within 80 minutes is extremely small. For core damage to occur, loss of AC independent water addition or ADS must also occur. This scenario is less than the acceptance criteria and thus meets the criterion. Since the above maintenance philosophy assumes only EDGB and the CTG are available, no additions to the minimum sets already identified are required for the loss of offsite power initiator.

19Q.7.9.5 Adequacy of Technical Specifications

From the above results, the following can be stated regarding adequacy of the ABWR Technical Specifications. In Mode 5, the onset of boiling is most dependent on water level (or total inventory), and thus the most vulnerable condition is at low water level prior to flooding up of the reactor cavity. In this condition, not only is the time to boiling relatively insensitive to decay heat level, but RHR in shutdown cooling (SDC) is the only source of decay heat removal. This is the basis for the Technical Specifications requiring that two loops of RHR SDC be available in this condition; one normally operating and one in standby. Results of the analysis show that given the loss of the operating RHR pump, with one RHR loop in standby there is a small probability of the onset of boiling. This is acceptable given the short time duration the plant is expected to be in this unique condition and the benign consequences that are calculated to result so long as core damage is avoided. Clearly, a utility could further reduce the likelihood of boiling by assuring that the third division of RHR SDC provided in the ABWR design is available during these conditions. Thus, during the early stages of the transition from Mode 4 to Mode 5 (prior to flood-up), the availability of the third loop of RHR SDC further reduces shutdown risk. However, given the other compensatory measures available to delay the onset of boiling and prevent core damage (e.g., condensate, AC independent water addition, CRD), the two loops of RHR SDC required by ABWR Technical Specifications are more than adequate during these plant conditions.

19Q.7.9.6 Contribution of Human Errors to CDF

The ABWR design is relatively insensitive to human error contributions to CDF during shutdown for the following reasons. Although several potential human errors have been identified (e.g., failure to recognize the loss of operating RHR loop and failure to manually actuate systems such as condensate, fire water, Reactor Water Cleanup, and Fuel Pool Cooling), multiple systems and paths for decay heat removal and makeup are available during shutdown to mitigate these errors. Also, automatic actuation of makeup from LPCF and HPCF and multiple alarms to alert the operator to potential unsafe conditions during shutdown (e.g., high RHR temperature, sump pump alarms, fire detection, low RPV level, high area radiation, and high neutron flux) all contribute to the conclusion that the ABWR is tolerant of human errors.

The methodology used in this study does not allow for a quantitative estimate of the impact of human errors to CDF, but based on the above discussion it is considered to be low.

19Q.8 Use of Freeze Seals in ABWR

Freeze seals are used for repairing and replacing such components as valves, pipe fittings, pipe stops, and pipe connections when it is impossible to isolate the area of repair any other way. Freeze seals have successfully been used in pipes as large as 700A (28 inches) in diameter.

The ABWR design has eliminated a significant amount of piping associated with the Reactor Coolant System (RCS)(e.g., no recirculation loops). This by itself will reduce the necessity for freeze seals in ABWRs over other plant designs.

In addition to reduced RCS piping, the ABWR design has most piping connected to the reactor pressure vessel (RPV) enter at a level significantly higher [152.4 cm(5 feet)] than the top of active fuel. Inadvertent draining from these lines will automatically stop without exposing the fuel. The only piping connection below the top of active fuel (Reactor Water Cleanup System) is small in size [<50A (< 2 inches)]. If a freeze seal were required on this line and it were to fail, several sources of makeup are available to refill the RPV to prevent core uncovery.

Whenever freeze seals or other temporary boundaries are used in the ABWR, administrative procedures will be necessary to ensure integrity of the temporary boundary. Also, mitigative measures will be identified in advance and appropriate backup systems made available to ensure no loss of coolant inventory occurs.

An option that a utility could choose is to off-load all the fuel in the RPV to the spent fuel pool when repair or maintenance of an unisolatable valve must be completed.

The selected method for working on unisolatable valves must take into account adequate safety margins, personnel experience with freeze seals, availability of backup systems, and the potential impact on other outage activities.

See Subsection 19.9.23 for COL license information requirements.

19Q.9 Shutdown Vulnerability Resulting from New Features

The ABWR has incorporated many new design features that do not exist in current operating domestic BWRs. These features have been added based on past operating experience, advances in technology since earlier designs were finalized, and the results of detailed probabilistic risk assessments (PRAs).

In order to evaluate the potential shutdown risk associated with these new features, a Failure Modes and Effects Analysis (FMEA) was completed for each new feature. The feature is identified followed by potential failure mode(s). The possible method for detecting each failure mode is then presented followed by the potential impact on safe shutdown and any preventive or mitigating feature that may exist. Finally, the overall shutdown vulnerability evaluation is described.

The FMEA is contained in Table 19Q-6. As the results presented in Table 19Q-6 show, there are no identified vulnerabilities resulting from implementation of new design features in the ABWR that affect shutdown risk.

19Q.10 Procedures

The ABWR has been designed to minimize risk associated with plant operations both at normal power and shutdown conditions. As previously mentioned, PRA techniques have been employed to identify potential accident scenarios and, where appropriate, design modifications have been included to reduce estimated risks. In addition to the physical plant design and configuration, the ABWR will incorporate operating procedures that are based on rigorous engineering evaluations including safety analyses. These procedures will be prepared consistent with NUMARC Guidelines presented in NUMARC 91-06, “Guidelines to Enhance Safety During Shutdown.”

Each utility must generate plant specific operating procedures based on individual site characteristics and training program requirements. A procedures guideline will be completed for the ABWR to address shutdown conditions. The guideline will provide insight into two general areas:

(1) Effective outage planning and control

(2) Maintenance of key shutdown safety functions:

   (a) Decay heat removal capability,

   (b) Inventory control,

   (c) Electrical power availability,

   (d) Reactivity control, and

   (e) Containment integrity (primary and secondary).

Outage Planning and Control

Although design features help, shutdown risk can best be minimized through appropriate outage planning and control procedures. Planning is important because of the large number and diversity of tasks that must be completed during the outage. Safety and support systems must be taken out of service for maintenance. This reduces redundancy of safety systems. If alternate means are not utilized to backup the lost safety system, a reduction in safety margin may occur. The ABWR contains multiple normal and alternate systems to complete all required shutdown safety functions. Availability of normal and alternate systems must be made known to all personnel involved in planning and execution of the outage. This is an ever-changing situation during outages and proper planning and tracking of activities is required to ensure safety margins are maintained.

The plant specific procedures for outage planning and control should ensure that the appropriate focus is maintained on the following activities:

(1) Documentation of outage philosophy including organizations responsible for outage scheduling. This should address not just the initial outage plan but all safety significant changes to the schedule.

(2) Ensuring that all activities, particularly higher risk evolutions, receive adequate resources. The plan should consider scope growth and unanticipated changes.

(3) Ensuring that the "defense in depth" concept that is central to power operation be maintained during shutdown to ensure that safety margins are not reduced. Safety systems must be taken out of service for maintenance but alternate or backup systems can be made available if proper planning is completed.

(4) Ensuring that all personnel involved in outage planning and execution receive adequate training. This should include operator simulator training to the extent practicable. Other plant personnel, including temporary personnel, should receive training commensurate with the outage tasks they will be performing.

(5) After completion of outage planning, but prior to final approval, a review of the schedule should be completed by an independent safety review team. The main objective of this review is to assure that the defense in depth principal will not be violated at any time during the outage.

See Subsection 19.9.25 for COL license information requirements.

Shutdown Safety Issues

Procedures for outage planning and control address general aspects of risk reduction during shutdown. Specific shutdown procedures are required to maintain key safety functions during shutdown (See Subsection 19.9.25 for COL license information requirements). The following guidelines should be used for each key shutdown safety function.

(1) Decay Heat Removal Capability

The normal method of Decay Heat Removal (DHR) is through use of the Residual Heat Removal System (RHR) in the shutdown cooling mode. As discussed in Subsections 19Q.7 and 19Q.11, there have been many events at operating plants that have resulted in partial or total loss of DHR. A recovery strategy should be established to address loss of normal RHR. This should include identification of alternate DHR Systems as well as personnel responsible for execution of the recovery plan. In addition to recovery plans, outage planning should emphasize availability of DHR by postponing maintenance on RHR Systems to later in the outage when decay heat loads have been reduced or to when the core has been off-loaded to the spent fuel pool. In the case of core off-load, procedures should be prepared to ensure maintenance of spent fuel pool cooling.

(2) Inventory Control

If DHR were to be lost, the time to reactor coolant boiling and core uncovery will be determined by the initial coolant inventory and makeup capability. Procedures should be prepared to ensure that adequate coolant inventory is maintained at all times during shutdown. Also, plant activities or configurations where a single failure can result in loss of inventory should be identified and compensatory measures established. Specific activities for the ABWR that should be reviewed for the potential of inventory deduction are: Use of freeze seals (see Subsection 19Q.8 for a more complete discussion); removal of control rods, control rod drives, and reactor internal pumps; RHR valve actuations or leakage leading to diversion of RPV coolant to the suppression pool (e.g., RHR pump mini-flow valve failure / leakage, switching shutdown cooling from one division to another); and inadvertent actuation of safety relief valves.

(3) Electrical Power Availability

As discussed in Subsections 19Q.4.4 and 19Q.11, loss of electrical power during shutdown has resulted in loss of DHR in the past. The ABWR has two sources of offsite (preferred) and four sources of onsite electrical power. Procedures should be utilized to ensure that defense in depth for electrical power sources is maintained. Maintenance of power sources should reflect the current plant conditions. Availability of normal and alternate power sources should be ensured especially during periods of higher risk evolutions (e.g., unbolting the RPV head prior to flooding the reactor cavity). Many of the loss of power events discussed in Subsection 19Q.11 were caused by operator errors (e.g., switching errors, inadequate maintenance / testing procedures) and grounding of transformers in switchyards due to movement of equipment by cranes and trucks. All maintenance and switchyard activities should be reviewed to identify single failures or procedural errors that could result in loss of power to vital buses during shutdown. Procedures should be developed for implementation of alternate sources of power including applicable breakers and bus locations, required tools, and sequence of steps to be performed.

(4) Reactivity Control

Shutdown reactivity control for the ABWR is maintained by core design analysis and interlocks that restrict fuel and control rod drive movements. Procedures are required to ensure that the core is loaded per design requirements and that unauthorized fuel movement does not occur simultaneous with CRD mechanism maintenance. If the refueling sequence must be altered, new shutdown margin analyses should be performed. All fuel movements should be verified by knowledgeable trained personnel.

(5) Containment Integrity

The ABWR primary containment will not be available during most of the refueling outage but procedures should be developed to ensure its availability during Mode 3 and during Mode 4 (if appropriate). During all modes, procedures should be available to ensure that secondary containment can be maintained functional as required, especially during higher risk evolutions.

Procedure Reviews

An important part of procedures implementation is a review of the adequacy of all operating procedures. All shutdown operating procedures should be reviewed periodically to ensure that the defense in depth concept is being maintained given the actual events occurring at each site. This review should include not only procedure adequacy but dissemination of the outage philosophy to all personnel involved in scheduling and executing the outage plan and training of personnel including temporary personnel. This review should be documented and retained as a permanent plant record.

19Q.11 Summary of Review of Significant Shutdown Events: Electrical Power and Decay Heat Removal

As part of the certification process for the ABWR design, the NRC has requested that General Electric complete a review of significant shutdown events in operating plants and discuss ABWR features which could prevent or mitigate such events.

To complete this evaluation, a review was made of operating events involving loss of offsite power (LOOP) and loss of Decay Heat Removal (DHR). These two areas appear to have the greatest potential for causing core damage during shutdown based on past experience. The sources utilized for information on past shutdown events were:

- “Residual Heat Removal Experience Review and Safety Analysis”, NSAC-88, March 1986
- “Loss of Vital AC Power and the Residual Heat Removal System during Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990”, NUREG-1410, June 1990
- “NRC Staff Evaluation of Shutdown and Low Power Operation”, NUREG-1449, March 1992
- Selected INPO SEO Reports and NRC Information Notices

The results of this evaluation are contained in Attachment 19QC, Tables 19QC-1 and 19QC-2 for LOOP and loss of DHR respectively. The following is a discussion of the results for each event type.

LOOP

NUREG-1410 contains a discussion of 70 LOOP events at operating plants both PWR and BWR. Although the response to LOOP events will differ for PWRs and BWRs, the initiating events are similar in that offsite and onsite power configurations are similar for both reactor types. The events evaluated in NUREG-1410 occurred between 1965 and 1989. Two additional recent LOOP events were added to this list and are included in Table 19QC-1.

The LOOP events can be grouped into the following categories:

- Loss of all offsite power sources due to various reasons including weather, operator errors or grid upset
- Loss of one or more offsite sources with at least one offsite source remaining
- Isolation of offsite power due to onsite electrical faults
- Degraded offsite or onsite power sources resulting from errors in maintenance activities

As discussed in Table 19QC-1, the ABWR electrical distribution system has several features which would prevent or mitigate every precursor event evaluated in this study. Prevent or mitigate in this case means that at least one Class 1E power supply would be available to energize equipment to maintain plant cold shutdown.

The main features of the electrical system are:

- Two independent sources of offsite power
- Three physically and electrically independent Class 1E emergency diesel generators
- Three unit auxiliary transformers powering three Class 1E and three non-1E power buses
- Combustion Turbine Generator (CTG) that can be used to power any of the Class 1E or non-1E power buses

The above features of the ABWR electrical distribution system, along with appropriate Technical Specifications and other administrative controls, assures that adequate power sources would be available to mitigate potential electrical events such as those described in Table 19QC-1.

Loss of DHR

NSAC-88 contains a discussion of 90 loss or degradation of DHR events during the seven year period 1977 through the end of 1983. The source for these events were Licensee

Event Reports (LERs). Other events described in INPO SEO reports and NRC information notices were also reviewed and included in the study.

Summary of DHR Events

The results of this evaluation are contained in Table 19QC-2. Not all of the events discussed in NSAC-88 are contained in Table 19QC-2. Those events that were due to random failures of single components and did not result in loss of DHR or other significant plant effects were not evaluated further. If the single failure resulted in loss of coolant, overpressurization, flooding, or loss of Shutdown Cooling (SDC) function, the event was included and the applicable ABWR feature to prevent or mitigate the event was discussed.

For the purposes of this study, prevention or mitigation means that, given the DHR challenge event, the ABWR design would either not be susceptible to the postulated failure or it has design features that could be relied upon to ensure that the fuel in the RPV remained covered with water at all times.

Of the events described in Table 19QC-2, some were single failures of RHR System components that resulted in either delayed achievement of shutdown cooling (SDC), reduction in reactor pressure vessel (RPV) water level, or a temporary loss of SDC. In all of these events, the fuel remained covered with water and alternate means of DHR remained available (e.g., Reactor Water Cleanup System, main condenser, and ECCS Systems). In the cases of delayed or temporary loss of SDC, RPV water temperature increases ranged from 261 K - 333 K (10° - 140°F). In all cases, SDC was restored and alternate means of DHR were not used although available. Operator errors associated with improper valve lineups or incorrect maintenance were identified. In these cases, delays in implementing SDC or temporary loss of SDC occurred while the error was corrected. In a few cases, marine growth caused failure of one or more RHR heat exchangers which resulted in temporary loss of SDC while other RHR loops or alternate cooling paths were implemented. In one case, a freeze seal failure in the RHRSW caused 56.8 m^3 (15,000 gallons) of water to damage ECCS power supplies resulting in temporary isolation of SDC.

None of the events described above and in Table 19QC-2 resulted in fuel being uncovered. The flexibility of the RHR System and the several alternate means of DHR that were available served to mitigate the component failures or operator errors.

Summary

Significant shutdown events in operating plants have been reviewed to determine ABWR features which could prevent or mitigate the events. Loss of offsite power and loss or degradation events from published nuclear industry reports were the database for this review. The results of this review demonstrate that ABWR design includes many features that prevent or mitigate unacceptable consequences of typical past events.

The main features of the ABWR that will prevent or mitigate shutdown events are:

- Three divisions of ECCS and support systems that are physically and electrically independent
- Two independent offsite power sources
- Four onsite power sources (three emergency diesel generators and one combustion turbine generator)
- Plant configuration and structural integrity to minimize common mode failures due to fire and floods
- Appropriate Technical Specifications and other administrative controls to ensure availability of systems during periods of potentially high risk operations
- Several alternate means of DHR if normal systems were to fail or be out of service for maintenance
- Instrumentation availability during shutdown to monitor plant safety status and initiate safety systems when needed

19Q.12 Results and Interface Requirements

19Q.12.1 Insights Gained from the Analysis

Completion of the ABWR shutdown risk analysis has resulted in the following insights:

(1) The most important element in control of shutdown risk is adequate planning of maintenance on systems and support systems that can be used to remove decay heat or supply inventory makeup to the RPV.

(2) The ABWR design has incorporated a significant number of new design features relative to operating BWRs. Past events that have led to loss of decay heat removal capability or loss of offsite power can, in general, be mitigated by ABWR design features.

(3) The ABWR design has a very low risk associated with loss of decay heat removal. Adequate shutdown safety margins exist if only systems required by Technical Specifications and those that are already in operation (e.g., CRD, FPC, fire water) are relied upon. Minimum combinations of systems have been identified that, if available, will ensure adequate shutdown safety margins. Combinations other than those identified in this study may exist which also result in adequate shutdown risk margins. By taking advantage of these available decay heat removal and makeup systems, utilities can exercise much flexibility in outage maintenance scheduling while ensuring that adequate safety margins are maintained at all times during shutdown conditions.

(4) The above safety margins were calculated using very conservative estimates for human error probabilities. For all events analyzed during shutdown, sufficient time is available to prevent core damage that no extraordinary operator actions are required. ABWR safety is designed into the plant.

(5) Fire and floods during shutdown can be mitigated by ensuring, through administrative procedures that at least one safety division is not in maintenance and its physical boundaries remain intact. If it is decided to breach the boundaries of two safety divisions to complete maintenance tasks, an evaluation must be completed to ensure that a minimum set of systems capable of meeting the shutdown safety criterion will remain available if a fire or flood were to occur. This applies to flooding/fire in the intact division as well as the breached divisions.

(6) The minimum technical specification requirements plus systems normally operating during shutdown (e.g., CRD, fire water, CUW) are adequate to ensure that safety margins can be maintained during shutdown due to a loss of an operating RHR train. Also, no technical specification changes are required to mitigate fires or floods during shutdown. Administrative controls are recommended on maintenance activities during shutdown to ensure the availability of systems to mitigate loss of RHR, fires, and floods.

19Q.12.2 Important Design Features

The ABWR features identified as important contributors to the low level of risk associated with shutdown are discussed in Subsection 19.8.6.

19Q.12.3 Operator Actions

The following operator actions have been identified that are important to minimization of shutdown risk and have been included as COL action items:

- Ability to recognize failure of an operating RHR System.
- Rapid implementation of standby RHR Systems following the loss of the operating RHR System.
- Use of alternate means of decay heat removal using non-safety grade equipment such as CUW, FPC, or main condenser.
- Use of alternate means of inventory makeup using non-safety grade equipment such as AC independent water addition, CRD pump, feedwater, or condensate.
- How to utilize boiling for decay heat removal in Mode 5 with the RPV head removed including available makeup sources.

- Implementation of fire/flood watches during periods of degraded safety division physical integrity.
- Fire fighting during shutdown.
- Use of remote shutdown panel during shutdown.
- Instrumentation must be made available during shutdown to support the following functions:
  - Isolation of RPV
  - ADS
  - HPCF
  - LPFL
  - RPV water level, pressure, and temperature
  - RHR System alarms
  - EDG
  - Refueling interlocks
  - Flood detection and associated valve isolation and pump trips
- Procedures should be prepared to address the following tasks during shutdown:
  - Fire fighting with part of the fire protection system in maintenance
  - Outage planning to minimize risk using guidance from NUMARC 91-06
  - Use of freeze seals
  - Replacement of RIPs and CRD blades
  - Loss of offsite power
  - Increasing CRD pump flow when using it for inventory control
  - Maintenance of suppression pool as it relates to maintaining safety margins for decay heat removal
  - Ensure that one safety division is always available with intact fire/flood barriers.

19Q.12.4 Reliability Goals (Input to RAP)

The following assumed system unavailabilities were determined to be important in minimizing shutdown risk and are included in the ABWR Reliability Assurance Program:

System	Unavailability (Per Demand)
RHR (SDC)	†
RHR (LPFL)	†
HPCF	†
CRD	†
CTG	†
EDG	†
Offsite Power	†
ADS	†
DC Power	†

† Not a part of DCD (refer to SSAR)

19Q.12.5 Conclusions

The ABWR has been evaluated for risks associated with shutdown conditions and for all postulated events, the risk has been determined to be low. Multiple means of removing decay heat and supplying inventory makeup have been identified that along with appropriate Technical Specifications and outage procedures result in acceptably low shutdown risk levels for the ABWR.

Table 19Q-1 ABWR Features That Minimize Shutdown Risk

Category	Feature	Shutdown Risk Capability
Decay Heat Removal (DHR)	Residual Heat Removal (RHR) System	Three independent (100% capacity) divisions of RHR and support systems for normal DHR. Each RHR division has several DHR modes (e.g., SDC, SPC).
	Reactor Coolant Temperature Measurement	During shutdown, reactor coolant temperature is determined by measuring Reactor Water Cleanup (CUW) inlet water temperature.
	Shutdown Cooling Nozzle	The shutdown cooling mode of RHR uses suction piping that connects directly to a nozzle on the RPV instead of to an external piping system. This reduces the probability of losing RHR pump suction due to air entrapment or cavitation.
	Safety Relief Valves	Can be used as alternate means of decay heat removal by venting steam to the suppression pool. They are also actuated to depressurize the RPV to allow use of low pressure RHR or other low pressure systems.
	Suppression Pool	A potential heat sink and makeup source for decay heat removal. Pool temperature is monitored in the control room to indicate trends in pool temperature. This large heat sink allows sufficient time for appropriate operator actions.
	Reactor Water Cleanup System (CUW)	Can be used under certain conditions to remove decay heat. See Subsection 19Q.7 and Attachment 19QB for more details on this feature.
	RPV Boiling	When the RPV head is removed, boiling is an effective (although not preferred) heat transfer method as long as RPV water level can be maintained by available makeup sources.
	Condenser	The main condenser (if available) can be used for DHR.
	Remote Shutdown Panel (Two Divisions)	Cold Shutdown can be achieved and maintained from outside the control room if the control room is uninhabitable due to fire, toxic gas, or other reasons. The remote shutdown panel is powered by Class 1E power to ensure availability following a Loss Of Preferred Power (LOPP). Controls are hard wired and thus not dependent on multiplexing systems. A minimum set of monitored parameters and controls are included to ensure the ability to achieve and maintain cold shutdown.

Table 19Q-1 ABWR Features That Minimize Shutdown Risk (Continued)

Category	Feature	Shutdown Risk Capability
	Instrumentation	Adequate instrumentation is available to operators both inside and outside of the control room for monitoring shutdown conditions throughout the plant. Some of the safety significant parameters monitored during shutdown include: RPV water level, reactor coolant temperature, neutron flux, drywell pressure, RHR flow, reactor pressure, and suppression pool temperature and level. In addition to monitoring, signals are also available to actuate ECCS functions on low RPV water level, scram control rods on high flux, and close isolation valves on appropriate signals. Four divisions of instrumentation allow one division of monitoring sensors to be in maintenance without disabling the function, thus assuring availability of instrumentation during shutdown.
	Fuel Pool Cooling System	The Fuel Pool Cooling System (FPC) can be used for DHR under certain conditions during Mode 5 (refueling). See Subsection 19Q.7 and Attachment 19QB for more detail. The pool does not contain drains and includes antisiphon devices to prevent inadvertent drainage. The RHRS can be interconnected to the FPC to aid cooling of fuel in the pool if required.
Reactor Inventory	High Strength Low Pressure Piping	Low pressure piping connected to high pressure piping has been redesigned to a higher pressure rating up to the most remote closed valve and is therefore expected to withstand full reactor pressure on a rupture criteria basis. This minimizes the potential for loss of inventory.
	Interlocked RHR Valves (Mode Switch)	The RPV shutdown cooling suction valve must be fully closed before the suppression pool return or suction valves can be opened. Shutdown cooling suction valve cannot be opened until suppression pool suction and return valves are fully closed. This prevents inadvertent draining of the RPV to the suppression pool. Interlocks are part of mode switch design.
	RPV Isolation Valves	All large diameter [>50A (>2 inches)] isolation valves in the RHR and CUW Systems that connect to the RPV (except injection lines) automatically close on a low RPV water level signal. This reduces potential for the core being uncovered due to an inadvertent RPV drain down event.
	Makeup Control	If RPV level decreases, High Pressure Core Flooder (HPCF), Automatic Depressurization System (ADS), and Low Pressure Flooder (LPFL) Systems initiate automatically. If HPCF and LPFL Systems are in the test mode and a RPV low level signal is received, the systems automatically switch to the vessel injection mode.

Table 19Q-1 ABWR Features That Minimize Shutdown Risk (Continued)

Category	Feature	Shutdown Risk Capability
	Feedwater and Condensate Pumps	Three electric driven pumps that can be used during shutdown for makeup.
	High Pressure/Low Pressure Interlocks	Controls position of RHR valves to ensure that the RHR is not exposed to pressures in excess of its design pressure.
	Makeup Sources	Multiple sources of RPV makeup are potentially available while the plant is shutdown (e.g., main condenser hotwell, condensate storage tank, suppression pool, control rod drive system, AC-independent Water Addition System).
	No Recirculation Piping	Elimination of Recirculation piping external to RPV reduces probability of LOCA both during normal operations and while shutdown.
	RPV Level Indication	Permanently installed RPV water level indication for all modes of shutdown. Redundant sensors use a two-out-of-four logic configuration to ensure high reliability.
Containment Integrity	Containment	Reinforced concrete structure surrounds RPV to withstand LOCA loads and contain radioactive products from potential accidents during hot shutdown. Secondary containment permits isolation and monitoring all potential radioactive leakage from the primary containment.
	Standby Gas Treatment System	Removes and treats contaminated air from the secondary containment following potential accidents.
	Reactor Building Isolation Control	Automatically closes isolation dampers on detection of high radiation. These dampers are potential leakage paths for radioactive materials to the environs following breach of nuclear system barriers or a fuel handling accident.
Electrical Power	3 Diesel Generators	One diesel for each safety division. Independent, both electrically and physically, of each other to minimize common mode failure. Allows for diesel maintenance while still maintaining redundancy.
	Combustion Turbine Generator	Redundant and diverse means of supplying power to safety and non-safety buses in event of loss of offsite power and diesel generator failures.
	2 Sources of Offsite Power	Reduces risk of LOPP due to equipment failure or operator error.
	Electrical Cable Penetrations	Will prevent propagation of fire damage and water from postulated flooding sources.
	4 Divisions of DC Power	Electrically and physically independent. Includes batteries and chargers. Diverse means of electrical power for control circuits and emergency lighting.

Table 19Q-1 ABWR Features That Minimize Shutdown Risk (Continued)

Category	Feature	Shutdown Risk Capability
Flooding Control	Flood Monitoring and Control	Reactor, control, and turbine building flooding is monitored and alarmed in the control room. This alerts the operator to potential flooding during shutdown. Many flood sources (e.g., HVAC, EDG Fuel) are relatively small volume and are self limiting. Operation of the fire water system is alarmed in the control room to help the operator differentiate between a break in the fire water system and the need to extinguish a fire. Larger sources are mitigated by means of, equipment mounted at least 20.32 cm off the floor, floor drains, watertight doors, pump trips, valves closing, antisiphon capability, or operator actions except at the steam tunnel interface.
	Room Separation	The three divisions of ECCS are physically separated and self contained within flooding resistant walls, floors, and doors. ECCS wall penetrations located below the highest potential flood level in the reactor building first floor corridor will be sealed to prevent water entering the ECCS room from the corridor. No external potential flooding sources are routed through the ECCS rooms and potential flooding sources in other rooms will not overflow into the ECCS rooms and cause damage to ECCS electrical equipment. If ECCS flood barriers must be breached during shutdown, administrative controls ensure that at least one ECCS division is operable and all barriers in that division are maintained intact.
Reactivity Control	Refueling Interlocks	A system of interlocks that restricts movement of refueling equipment and control rods during refueling to prevent inadvertent criticality. When the mode switch is in the REFUEL position, a fuel assembly cannot be hoisted over the reactor vessel if a control rod is withdrawn. When the mode switch is in the REFUEL mode, only two control rods can be withdrawn at one time, but during fuel handling only one control rod can be withdrawn per Technical Specification requirements.
	Fuel Handling	Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain adequate shielding and cooling for spent fuel.
	CRD Supports and Brake	CRD supports limit the travel of a control rod in the event a control rod housing is ruptured. The brake limits the velocity at which a control rod can fall out of the core should a hydraulic scram line break. The internal blowout support prevents rod ejection due to failure of flange bolts or a spool piece. Both of these limit reactivity excursions and thus protect the fuel barrier.

Table 19Q-1 ABWR Features That Minimize Shutdown Risk (Continued)

Category	Feature	Shutdown Risk Capability
	Instrumentation	Reactor Protection System (RPS) high flux (set down) and manual scram functions are operable during shutdown.
Fire Protection	Divisional Separation	The three ECCS divisions are physically separated so that a fire initiated in one division will not propagate to another division. Procedures ensure that during shutdown, if fire barriers between divisions must be breached due to maintenance, at least one division will be available with barriers intact.
	Detection	Fire detection sensors that alarm in the control room are located throughout the plant and operate during shutdown. Actuation of the fire water system is alarmed in the control room. Also, during shutdown more personnel are located throughout the plant to identify, extinguish, and report potential fires.
	Suppression	Water and chemical fire suppression systems are located at appropriate plant locations.
	Water Supplies	Multiple water supplies and both electric and diesel powered fire pumps can deliver water to various locations in the plant during shutdown.
	Multiplexed systems	Eliminates the need for a cable spreading room which is a major fire concern in most plants.
	HVAC	Dual purpose HVAC/SMOKE Control System, divisionally separated, to control individual room pressure and assure clean air path for fire suppression personnel.

Table 19Q-2 Success Criteria for Prevention of Core Damage

System(s)	Comment
1 RHR (SDC) or	All times when available.
Main Condenser or	If available, open MSIVs and establish condensate return path to RPV.
CUW or	If temp 386 K (>234°F) or after 8 days (using 2 pumps and using 2 nonregenerative heat exchangers and with regenerative heat exchanger bypassed).
FPC or	Mode 5 only after 10 days. Both pumps and heat exchangers in each system required.
1 Feedwater + 1 Condensate or	High pressure injection.
1 HPCF or	High pressure injection.
1 CRD or	High pressure injection (After 1 day shutdown. Prior to one day two pumps required).
1 Condensate or	Low pressure injection (may need ADS).
1 LPFL or	Low pressure injection (may need ADS).
1 AC-Independent Water Addition System	Low pressure injection (may need ADS).

Table 19Q-3 Minimum Sets of Systems for Modes 3 and 4

	RHRB	Main Condenser	CUW	HPCFB	CRD	ADS	RHRB (CF)	Condensate	Fire Water
1)	*				*	*	*		*
2)	*		*			*	*		*
3)	*					*	*	*	*
4)	*	*				*	*		*
5)	*			*	*	*		*	

Table 19Q-4 Minimum Sets of Systems for Mode 5 (Unflooded)†

	RHRB	HPCFB	CRD	RHRB (CF)	Condensate	Fire Water
1)	*		*	*		*
2)	*				*	*
3)	*	*	*		*	

† 2 - 3 days after shutdown

Table 19Q-5 Minimum Sets of Systems for Mode 5 (Flooded)†

	RHRB	FPC	CUW	HPCFB	CRD	RHR (CF)	Condensate	Fire Water
1)	*						*	*
2)	*				*	*		
3)	*			*	*		*	
4)	*					*	*	*
5)‡			*		*			*
6)‡			*				*	*
7)*f*		*			*		*	
8)*f*		*					*	*
9)*f*		*			*			*

† 3 days after shutdown
‡ After 8 days
f After 10 days

Table 19Q-6 Shutdown Vulnerability Evaluation of new ABWR Features

Feature	Shutdown Failure Mode	How Detected	Potential Impact on Safe Shutdown	Preventive/Mitigative Feature	Vulnerability Evaluation
Reactor Internal Pumps (RIPS)	RPV leakage during maintenance	Visual identification of leakage	Inventory loss, fuel uncovery.	Multiple seals, administrative controls, diffuser plug cannot be removed unless RIP motor cover is in place	None, past experience with maintenance on RIPs indicates no concerns.
Combustion Turbine Generator (CTG)	Fails to start or pick up load.	No output voltage on demand or test.	Loss of electrical power redundancy.	Two independent offsite power sources and three Emergency Diesel Generators (EDGs).	None, adequate offsite and onsite power sources exist if CTG were to fail.
	Improper synchronization to existing power sources.	Loss of bus voltage when CTG output breaker closes on demand or test.	Loss of vital power	Two other divisions Capable of supplying vital power, auto synchronization circuit, administrative controls.	None, redundant power supplies and administrative controls/antisync circuit prevent any impact on safe shutdown.
Third EDG	Fail to start or pick up load.	No voltage on vital bus on demand or test.	Loss of power to one bus.	CTG capable of feeding any vital bus, two independent sources of offsite power.	None, increases number of onsite vital bus sources.

Table 19Q-6 Shutdown Vulnerability Evaluation of new ABWR Features (Continued)

Feature	Shutdown Failure Mode	How Detected	Potential Impact on Safe Shutdown	Preventive/Mitigative Feature	Vulnerability Evaluation
Third ECCS Division	Single failure results in loss of third ECCS division.	Safety function not completed (e.g., no ECCS flow given initiation signal) on demand or test.	Loss on one ECCS division.	Two other divisions capable of completing safety function.	None, increases number of ECCS divisions to complete safety functions, allows for ECCS maintenance without total loss of redundancy, separation reduces common mode failure susceptibility.
Micro Processor Based Safety Logic	Fails to initiate safety signal.	ECCS function not completed on demand or during test.	Loss of ECCS function.	High reliability with redundancy and self test feature.	None, increased reliability of ECCS logic.
Fine Motion Control Rod Drives (FMCRDs), Alternate Rod Insertion (ARI)	Fails to control CRD motion on demand.	CRD does not move when directed or spurious movement.	Reduced shutdown margin.	Only two CRDs can be withdrawn at a time, RPS active during shutdown (hi flux or manual trip).	None, adequate preventive mitigative features exist.
Two Independent Preferred Power Sources	Loss of offsite power.	No voltage on bus.	Loss of safety division power sources.	CTG and three EDGs.	None, increased number of onsite and offsite power sources.
Multiplex Control of System Sensor Interfaces	Loss of control power to ECCS.	ECCS functions not completed on demand or test.	Loss of ECCS function.	Self testing capability, high reliability with redundancy.	None, increased ECCS reliability and elimination of cable spreading room.

Table 19Q-6 Shutdown Vulnerability Evaluation of new ABWR Features (Continued)

Feature	Shutdown Failure Mode	How Detected	Potential Impact on Safe Shutdown	Preventive/Mitigative Feature	Vulnerability Evaluation
Closed Loop Reactor Building Cooling Water System (RCW)	Heat exchanger tube failure.	High temperature on RB equipment, water accumulation in RCW room alarms in control room.	Loss of safety equipment (e.g., RHR heat exchangers).	Redundant heat exchanger can supply necessary cooling.	None, closed loop RCW supplies cleaner water to safety equipment enhancing cooling capability (i.e., reduced fouling of heat transfer surfaces) as compared to direct cooling with service water.
	RCW Isolation Valve failed closed.	High Temperature on RB equipment.	See Heat Exchanger failure.	Three divisions of RCW.	See Heat Exchanger failure.
	RCW pump fails to supply water.	High temperature on RB equipment.	See Heat Exchanger Failure.	Redundant pump can supply necessary flow.	See Heat Exchanger Failure.
High Pressure Nitrogen Gas Supply to ADS and SRVs	Gas leak.	Loss of pressure in accumulators.	Loss of ADS/SRV capability to reduce RPV pressure and allow use of low pressure for Heat Removal (DHD) Systems.	Other high pressure DHR means exist (e.g., HPCF, feedwater/condensate RCIC). Can reduce RPV pressure through use of RCIC.	None, nitrogen supply instead of air reduces potential corrosion of valves and loss of system pressure due to compressor failures. More reliable than air systems.
	Bottle isolated due to valve closure (operator error).	Surveillance test.	See gas leak.	See gas leak.	See gas leak.
Enhanced Remote Shutdown Panel (e.g., 4 SRVs, HPCF)	Transfer and control switches fail to actuate fourth SRV and HPCF.	Safety equipment fails to actuate on demand or during test.	Loss of ability to control fourth SRV and HPCF from the remote shutdown panel.	Three SRV controls exist, local control of equipment is possible.	None, added features enhance shutdown safety.

Table 19Q-6 Shutdown Vulnerability Evaluation of new ABWR Features (Continued)

Feature	Shutdown Failure Mode	How Detected	Potential Impact on Safe Shutdown	Preventive/Mitigative Feature	Vulnerability Evaluation
Enhanced Suppression Pool Temperature Monitoring (64 T/Cs in four divisions instead of 16 in one division)	Fails to detect correct pool temperature.	During operation or test.	Loss of some redundancy in pool temperature monitoring.	None required.	None, enhancement to suppression pool monitoring function.
Suppression Pool Level Monitoring	Fails to detect correct pool level.	During operation or test.	Loss of pool water level monitoring capability.	Local indication.	None, does not perform a safety function.

Figures 19Q-1 through 19Q-19 are not part of the DCD (Refer to SSAR)

Figures 19QA-1a through 19QA-20ck are not part of DCD (Refer to SSAR Figures 19QA-1a through 19QA-20ck).

19QB DHR Reliability Study

19QB.1 Offsite Dose and Operator Recovery Calculations

This attachment covers five different calculations that were completed for various aspects of the ABWR Decay Heat Removal Reliability Study. The calculations are:

(1) Offsite doses following RPV boiling in Mode 5

(2) Time to reach RPV boiling for specific plant conditions and decay heat loads

(3) Time for RPV water level to reach top of active fuel (TAF)

(4) Human Reliability Analysis

(5) Decay heat removal capability of CUW and FPC

19QB.1.1 Offsite Doses

For the ABWR Decay Heat Removal Reliability Study, the success criteria for Mode 5 allows boiling of water in the RPV or spent fuel pool. The following calculation of offsite doses assuming boiling in the RPV and spent fuel pool substantiates why boiling is a viable success criteria in Mode 5.

The equation for calculating offsite doses is:

Dose = RR * DF * BR * DCF

Dose = Offsite dose for 24 hour period (Sv)

RR = Release rate for 24 hours

$$= \frac{\text{Decay Heat Load (joule/h)}}{\text{2039 joule/kg Water}} * 5.92\ (B_q/g\ \text{I} - 131) *$$

$$0.015\,(\text{I}-131\,\text{carryover}) * 24$$

DF = Dispersion factor = $1.2 \times 10^{-3}\ s/m^3$

BR = Breathing rate = $3.47 \times 10^{-4}\ m^3/s$

DCF = Thyroid dose concentration factor = $2.92 \times 10^{-7}\ Sv/B_q$

The values in the above equation such as I-131 carryover, I-131 concentration, and dispersion factor are conservative estimates based on ABWR Tier 2 analysis and regulatory guidance.

The decay heat loads at 3 and 14 days following shutdown are 17.29 x 10^6 w (5.9 x 10^7 Btu/h) and 9.378 x 10^6 w (3.2 x 10^7 Btu/h), respectively. Using the above equation, the doses for 24 hours at 3 and 14 days are 7.5 x 10^{-6} and 4.04 x 10^{-6} Sv, respectively. This is significantly below the FEMA limit of 0.05 Sv per 24 hours for normal plant operations. Thus boiling in Mode 5 will not exceed any offsite dose limits and is a viable success criteria.

19QB.2 Time to Reach Boiling

The time for an operator to recover a failed RHR system in the ABWR Decay Heat Reliability Study is conservatively based on the time to boiling in the RPV or the spent fuel pool. The following discussion addresses the calculation of time to boiling for the RPV and RPV plus spent fuel pool at various times after shutdown.

It is assumed that the initial temperature of the RPV or spent fuel pool is 333.15 K (140°F). This is typical for normal Mode 4 or 5 operation.

The equation for time to boiling is:

$$t = [\Delta T / \text{heat up rate } (°K/h)]$$

$$t = \frac{373.15°K - 333.15°K}{(\text{Decay Heat Rate} / \text{Mass of Water})}$$

Table 19QB-1 shows the results for time to boiling for the RPV alone at 2 and 3 days following shutdown and for the RPV plus spent fuel pool (i.e., reactor cavity flooded and fuel pool gates opened) at 3 and 14 days. As can be seen, the time for operator action varies from a little over an hour for the RPV alone to approximately one day for the RPV plus spent fuel pool 14 days after shutdown.

19QB.3 Time for RPV Water Level to Reach Top of Active Fuel

This subsection summarizes the calculations for the time to reach top of active fuel in Modes 3, 4, and 5. The results show it will take 6.4 hours in Mode 3, 13 hours in Mode 5, 15 hours in the early part of Mode 5 before flooding of the cavity, and more than a week after cavity flooding in Mode 5. Assuming that it takes 9.76 x 10^5 joules (925 BTU) to vaporize 0.4536 kg (1 lb) of water, the decay heat at a specific time is divided by 9.76 x 10^5 joules to find the rate of vaporization. Division of water mass by this vaporization rate results in the time for RPV water level to reach TAF. Table 19QB-2 shows the results.

19QB.4 Human Reliability Analysis (HRA)

19QB.4.1 Purpose

The purpose of this HRA is to calculate the human error probabilities (HEPs) for the decay heat removal reliability study.

19QB.4.2 Summary

Tables 19QB-3 and 19QB-4 show the HEPs which were calculated for various time frames and plant modes for two cases.

Case a Operator action required before water starts to boil.

Case b Operator action required to prevent core damage (CD).

However, it was decided that more conservative values should be used in the PRA. These values are also shown in these tables.

19QB.4.3 Methodology

The HEP calculations were performed conservatively using the procedure for normal human reliability analysis (HRA) in Table 8-1, Reference 19QB-1, with the following steps:

(a) The displays and alarms available to the operator were identified.

(b) The times to boiling and core damage were identified.

(c) The times for diagnosis and post-diagnosis actions were allocated.

(d) The HEPs for diagnosis and post-diagnosis actions were calculated using Figure 8-1 and Table 8-3 and 8-5 of Reference 19QB-1.

(e) Higher than calculated values were assigned conservatively for use in the PRA.

(f) It is assumed that at least two operators are in the control room at all times during shutdown.

19QB.4.3.1 Control Room and Alarms

Table 19QB-5 shows the relevant alarms which are available in the CR (Reference 19QB-2). Operator is alerted to the failure of the operating RHR by means of one of the RHR specific alarms. If none of these alarms work, he will be alerted to the RPV parameters alarm 2 (though RPV pressure and water level may not be available prior to boiling). With these multiple alarms, it is reasonable to assume that all operators will be promptly alerted to the RHR failure.

In Mode 5 with the reactor cavity flooded, the operator would be made aware of heating the fuel pool by many other indications. Personnel on the refueling floor will all sense the increased temperature and will see steam formation. If no personnel notice the fuel pool heatup, the operator would receive an alarm of low fuel pool level and initiation of fuel pool level make up.

19QB.4.3.2 Allocation of Times to Diagnosis and Post-Diagnosis Actions

The time available to the operators was allocated between time for diagnosis and time for post-diagnosis action. Table 19QB-6 shows the various times which were to calculate the HEPs. Column three gives the calculated times before boiling (case a), and core damage (case b), or the total time available for allocation. Columns 4 and 5 show the results of the allocation. Enough time is allocated to post-diagnostic actions, so that there is sufficient time for recovery of human errors, even if the required action must take place outside the control room.

19QB.4.4 Results and Conclusions

The results of this HRA study are documented in Tables 19QB-3 and 19QB-4. It is concluded that the operator has adequate instrumentation and alarms to diagnose the event. Adequate procedures and operator training will assure proper response to the loss of RHR event.

19QB.5 Decay Heat Removal Capability of CUW and FPC

The purpose of the following heat removal calculations is to determine heat removal capabilities of FPC and CUW after flooding the cavity as a function of time following shutdown. In Modes 3 and 4, FPC cannot be used but CUW is able to remove the decay heat because of increased capacity at higher temperatures. The results show that the CUW System alone is capable of removing the decay heat 8 days after shutdown because it keeps the RPV temperature below 373 K (212°F) within 24 h. The FPC System can be used 10 days after shutdown to keep the temperature below 339 K (150°F), which is the design limit for the FPC pumps. To perform these calculations, initial RPV temperatures of 333 K (140°F) and 325 K (125°F) were assumed for the CUW and the FPC, respectively an initial temperature of 333 K (140°F) and 325 K (125°F) was assumed for the CUW to account for the time that it takes to initiate the CUW System manually, because one FPC pump is working all the time, it takes a negligible amount of time to initiate the second pump.

19QB.6 References

19QB-1 Swain, A.D., “Accident Sequence Evaluation Program Human Reliability Analysis Procedure”, Sandia National Laboratories, NUREG /CR-4772, U.S. Nuclear Regulatory Commission, Washington, D.C., February 1987.

19QB-2 Interlock Block Diagram, IBD, 137C8326, Sh. 18, Rev. 2.

Table 19QB-1 Time to Boiling for the RPV and RPV Plus SFP

Mode	Days after Shutdown	Decay Heat (watts)	Decay Heat (Btu/h)	Mass of Water (kg)	Mass of Water (lbs)	Time to Reach Boiling (h)
4	2	2.0×10^7	6.8×10^7	5.0×10^5	1.1×10^6	1.2
5	3	1.7×10^7	5.9×10^7	5.0×10^5	1.1×10^6	1.3
5	3	1.7×10^7	5.9×10^7	5.4×10^6	1.2×10^7	15
5	14	9.4×10^6	3.2×10^7	5.4×10^6	1.2×10^7	27

Table 19QB-2 Time for RPV Water Level to Reach TAF

Mode	After Shutdown	Decay Heat (watts)	Decay Heat (Btu/h)	Mass of Water (kg)	Mass of Water (lbs)	Time to Reach TAF
3	4 hrs	4.19×10^7	1.43×10^8	4.4×10^5	9.8×10^5	6.4 h
4	2 days	2.0×10^7	6.8×10^7	4.4×10^5	9.8×10^5	13 h
5	3 days	1.7×10^7	5.9×10^7	4.4×10^5	9.8×10^5	15 h
5	3 days	1.7×10^7	5.9×10^7	5.4×10^6	1.2×10^7	7.8 days
5	14 days	9.4×10^6	3.2×10^7	5.4×10^6	1.2×10^7	14.5 days

Table 19QB-3 Probability of Failure to Diagnose

Case	Mode	Time After Shutdown	Prob. (Fail to Diagnose) Calculated	Prob. (Fail to Diagnose) Used in PRA
a	5	2 - 3 days	*	*
a	5	>3 days	*	*
b	3,4, and 5 (prior to flooding reactor cavity)	Any time After Shutdown	*	*
b	5 (after flooding reactor cavity)	Any Time	*	*

* Not a part of DCD (refer to SSAR).

Table 19QB-4 Probability of Failure to Start a Specified “Minimum-Set” System

Case	Mode	Prob. (Fail to Diagnose) Calculated	Prob. (Fail to Diagnose) Used in PRA
a	All	*	*
b	All	*	*

* Not a part of DCD (refer to SSAR).

Table 19QB-5 Control Room Alarms and Indications Aiding Diagnosis of “One RHR Lost“

RHR Specific	RPV Parameters
1. Pump discharge pressure high	1. Temperature
2. Pump motor over current	2. Pressure
3. RHR loop power failure	3. Water level
4. RHR loop logic failure	
5. RHR pump motor trip	
6. RCW outlet temperature high	

Table 19QB-6 Times Available (in Hours)

Case	Mode	Total Time To Event	Allocated Diagnosis Time (TD)	Time for Post-Diagnosis Actions (T_A)
a	5 (Days 2 to 3)	1.2	0.5	0.7
a	5 (after 3 days)	≥14	12	≥ 2
b	All	≥ 6.4	≥ 2.4	≥ 4

19QC Review of Significant Shutdown Events: Electrical Power and Decay Heat Removal

19QC.1 Review of Significant Shutdown Events

A review was made of operating events involving loss of offsite power (LOOP) and loss of Decay Heat Removal (DHR). These two areas appear to have the greatest potential for causing core damage during shutdown based on past experience. The sources utilized for information on past shutdown events were:

- "Residual Heat Removal Experience Review and Safety Analysis", NSAC-88, March 1986
- "Loss of Vital AC Power and the Residual Heat Removal System during Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990", NUREG-1410, June 1990
- "NRC Staff Evaluation of Shutdown and Low Power Operation", NUREG-1449, March 1992
- Selected INPO Reports (e.g., SOE and SOCR summaries) and NRC Information Notices.

The results of this evaluation are contained in Tables 19QC-1 and 19QC-2 for LOOP and loss of DHR, respectively.

19QC.1.1 Summary of DHR Events

Not all of the events discussed in NSAC-88 are contained in Table 19QC-2. Those events that were due to random failures of single components and did not result in loss of DHR or other significant plant effects were not evaluated further. If the single failure resulted in loss of coolant, over-pressurization, flooding, or loss of Shutdown Cooling (SDC) function, the event was included and the applicable ABWR feature to prevent or mitigate the event was discussed.

19QC.1.2 Summary

The results of this review demonstrate that ABWR design includes many features that will prevent or mitigate unacceptable consequences of typical past events.

The main features of the ABWR that will prevent or mitigate shutdown events are:

- Three divisions of ECCS and support systems that are physically and electrically independent
- Two independent offsite power sources

- Four onsite power sources (three emergency diesel generators and one combustion turbine generator)
- Plant configuration to minimize common mode failures due to fire and floods
- Appropriate Technical Specifications and administrative controls to ensure availability of systems during periods of potentially high risk operations
- Several alternate means of DHR if normal systems were to fail or be out of service for maintenance
- Instrumentation availability during shutdown to monitor plant safety status and initiate safety systems when needed

Table 19QC-1 Loss of Offsite Power Precursors

Event	Description	Applicable ABWR Features
Indian Point 1 and Yankee Rowe (11/9/65)	"Great Northeast Blackout"	ABWR has two independent offsite power sources. These are backed up by three physically and electrically separate trains of Class 1E AC power each containing an emergency diesel generator. These are further backed up by a permanent onsite Combustion Turbine Generator (CTG) which is capable of powering any one of the three trains if all three diesels were to fail. The CTG is also capable of supplying power to non-safety busses such that condensate pumps can also be used to provide reactor coolant make up.
Point Beach 1 (2/5/71)	Loss of all transmission lines, failure of three transformer differential relays, causing transformer lockout.	See discussion of Indian Point 2 and Yankee Rowe (11/9/65).
Ginna (3/4/71)	Plant siding fell into 34.5 kV line connecting sole startup transformer.	ABWR has two independent transformers powered by two independent offsite power supplies which reduces the probability of losing offsite power. In the event of losing offsite power, features described under Indian Point 2 and Yankee Rowe (11/9/65) can mitigate the event.
Palisades (9/2/71)	Transmission line fault, isolation breaker failure. Backup relay isolated 345 kV bus.	See discussion of Ginna (3/4/71).
San Onofre 1 (6/7/73)	138 kV auxiliary transformer out for maintenance. Ground fault operated differential relays, de-energizing other auxiliary transformers.	ABWR uses three auxiliary transformers. Each powers one of the three Class 1E and non-1E buses. In addition, a reserve auxiliary transformer is available to power all three Class 1E buses. CTG is also available which can power 1E and non-1E busses without using the auxiliary transformers.

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Oconee 1 (1/4/74)	230 kV switchyard isolated, 100 kV offsite source remained energized to supply power to the plant.	The ABWR also has two sources of offsite power.
Fort Calhoun (3/13/75)	Sole 161 kV backup offsite transmission line out for maintenance. 345 kV output breaker tripped (faulty protective relays), opening remaining connection to offsite power. Offsite power could have been supplied from 345 kV switchyard by opening generator disconnects.	See Indian Point 2 and Yankee Rowe (11/9/65) and Ginna (3/4/71).
Turkey Point 4 (5/16/77)	Loss of Offsite Power (LOOP)	See Indian Point 2 and Yankee Rowe (11/9/65).
Connecticut Yankee (6/26/76)	Protective relays operated when lines were re-energized after service, causing LOOP.	See Indian Point 2 and Yankee Rowe (11/9/65).
Indian Point 2 (7/13/77)	LOOP due to lightning strikes. Emergency Diesel Generators (EDGs) operated.	See Indian Point 2 and Yankee Rowe (11/9/65).
St. Lucie 1 (5/14/78)	Substation switching error.	ABWR has two offsite power sources so probability of one switching error resulting in loss of all offsite power is low. But if it were to occur, mitigation features exist as discussed in Indian Point 2 and Yankee Rowe (11/9/65).
Turkey Point 3 (4/4/79)	Loss of all 7 transmission lines due to weather.	See Indian Point 2 and Yankee Rowe (11/9/65).
Davis Besse (4/19/80)	One EDG out for maintenance. One 13.8 kV bus connected, other energized but not connected. Ground fault on 13.8 kV bus caused loss of non-nuclear instruments. Air was pulled into DHR pump, and pump was stopped by operator. Pump vented and restarted after 2-1/2 hours.	ABWR has two sources of non-1E power. A ground fault on one would not result in loss of all non-1E power. In addition, if all non-1E power were to be lost, no valves connected to the RHR System would automatically cycle and cause loss of NPSH to any RHR pump. Also, the ABWR has three independent (100%) RHR Systems such that loss of one would not result in loss of the ability to remove decay heat.

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
San Onofre 1 (4/22/80)	Maintenance error caused LOOP.	See St. Lucie 1 (5/14/78).
Prairie Island 1 (7/15/80)	Weather related LOOP.	See Indian Point 2 and Yankee Rowe (11/9/65).
San Onofre 1 (11/22/80)	Maintenance error caused LOOP.	See St. Lucie 1 (5/14/78).
Diablo Canyon 1 (10/16/82)	LOOP caused by brush fire.	See Indian Point 2 and Yankee Rowe (11/9/65).
Farley 2 (10/8/83)	Switchyard breaker failure during refueling.	See St. Lucie 1 (5/14/78).
Palisades (1/8/84)	Deliberate de-energization of offsite power to isolate faulty breaker. One EDG out for maintenance, other available but its service water pump was out for maintenance, and operators failed to recognize this before authorizing work on breaker. Available EDG overheated and was manually tripped.	See Indian Point 2 and Yankee Rowe (11/9/65) and Ginna (3/4/71). ABWR technical specifications require one offsite and one onsite power source be available at all times.
Sequoyah 1 (3/26/84)	Ground short on 500 kV switchyard breaker de-energized transformer. Startup transformer supplied power.	A similar event at an ABWR could be more easily mitigated due to the existence of the CTG and three EDGs.
Yankee Rowe (5/3/84)	One 115 kV line out for maintenance, other energized. Normal supply transformer energized. Temporary fault detection relay caused breakers from normal supply transformer to open.	See Sequoyah 1 (3/26/84).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Salem 1 (6/5/84)	One of three safety buses was out of service for maintenance and one of the batteries in the two remaining safety trains was out of service for replacement. Automatic transfer relay which should have energized this bus was removed and placed in Unit 2 and not replaced in Unit 1, loss of power to two buses resulted in two operable EDGs to start but loss of DC control to one of the trains prevented closing of the EDG output breaker. One EDG did energize one bus but EDG cooling water pump was powered by EDG which lost control power. EDGs ran for two hours without cooling water.	Each of the three ABWR safety trains have separate independent emergency power supplies and support systems so each diesel can supply power to its own cooling water pump. ABWR technical specifications require one offsite and one onsite power supply be available at all times.
Connecticut Yankee (8/24/84)	One 115 kV transmission line out for maintenance, one auxiliary transformer out for maintenance. Differential relay opened breakers to remaining auxiliary transformer.	See San Onofre 1 (6/7/73).
Point Beach 2 (10/22/84)	Breaker alignment errors during cross-tie between units caused LOOP.	ABWR design does not allow crossties between plants.
Indian Point 3 (11/16/84)	Object from roof fell onto startup transformer.	See San Onofre 1 (6/7/73) and Ginna (3/4/71).
Turkey Point 3 (4/29/85)	Startup and C transformer were both out of service. Offsite power supplied through main transformer. Relay failure resulted in loss of main transformer and LOOP. One EDG started and loaded its safety bus.	See Indian Point 2 and Yankee Rowe (11/9/65).
Turkey Point 3 (5/17/85)	Brush fire disabled station.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Waterford 3 (12/12/85)	Lightning caused loss of preferred offsite power source. Two EDGs started and loaded. Two sources of offsite power were available.	See San Onofre 1 (6/7/73).
Fort Calhoun (3/21/87)	One EDG and alternate offsite power source were out for maintenance, controls for other EDG bypassed to prevent auto-start. Maintenance error tripped offsite power; EDG had to be manually loaded.	See Indian Point 2 and Yankee Rowe (11/9/65) and Palisades (1/8/84).
Yankee Rowe (6/1/87)	Maintenance error caused loss of 2 of 3 safety buses.	See Sequoyah 1 (3/26/84).
McGuire 1 (9/16/87)	One offsite power source and 1 EDG out for maintenance. Test error caused loss of other offsite power source. Remaining EDG started and loaded. Offsite power restored after 25 minutes.	See Indian Point 2 and Yankee Rowe (11/9/65).
Crystal River (10/14/87)	One EDG out for maintenance. Safety buses cross-tied. Maintenance error caused loss of 1 safety bus. Cross-connect breaker then tripped and locked out. Dead bus transfer was required to close one cross-connect breaker. This required shutting the running EDG and resetting the under voltage lockout.	See Indian Point 2 and Yankee Rowe (11/9/65) and San Onofre 1 (6/7/73). Also, ABWR design does not allow safety buses to be cross-tied. Therefore, this event cannot occur in ABWR.
Crystal River (10/15/87)	One safety bus and its EDG out for maintenance. Maintenance error grounded offsite power supply. Remaining EDG started and loaded.	See Indian Point 2 and Yankee Rowe (11/9/65) and San Onofre 1 (6/7/73).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Wolf Creek (10/16/87)	One safety bus and 1 EDG out for maintenance. Error de-energized other bus. EDG output breaker opened and would not close due to anti-pump circuit preventing reclosure once it had been opened after EDG started on under-voltage. DHR lost for 17 minutes.	Anti-pump circuitry has been redesigned in the ABWR to allow closure following breaker trip when required.
Oconee 3 (9/11/88)	All offsite power going through 1 breaker. Maintenance error caused this breaker to open, and it could not be reclosed. No instruments to determine actual level and temperature of water in reactor core region (incore thermocouples not yet reconnected, and no power to RPV level transmitters).	See Indian Point 2 and Yankee Rowe (11/9/65). ABWR RPV water level instruments are powered by batteries and at least two divisions are required to be operable during shutdown to support ECCS automatic initiation functions.
Surry 1 and 2 (4/6/89)	Electrical fault and transformer lockout. This de-energized one safety bus in each unit. Unit 2 EDG started and loaded. Unit 1 EDG control in manual.	See Indian Point 2 and Yankee Rowe (11/9/65) and Point Beach 2 (10/22/84).
Diablo Canyon 1 (3/7/91)	Maintenance error caused power arc and LOOP. EDGs started and loaded.	See Indian Point 2 and Yankee Rowe (11/9/65).
Nine Mile Point (11/17/73)	One transmission line out for maintenance. Maintenance error caused loss of other line.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (4/15/74)	Lightning caused loss of all 345 kV lines. 23 kV line remain energized.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (5/26/74)	All 345 kV lines de-energized (cause unknown). 23 kV line remained energized.	See Indian Point 2 and Yankee Rowe (11/9/65).
Brunswick 2 (3/26/75)	One train of 230 kV buses for each unit out for maintenance. Relay error caused breakers on all five lines supplying remaining buses to open.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Quad Cities 2 (2/13/78)	Reduced voltage on grid caused under-voltage relays to trip breakers on both safety buses. System dispatcher increased grid voltage.	See Indian Point 2 and Yankee Rowe (11/9/65). ABWR has an alarm at 95% of rated voltage (degraded voltage). This gives operator 5 minutes to restore full voltage before offsite breakers would open.
FitzPatrick (3/27/79)	Maintenance error caused LOOP.	See Indian Point 2 and Yankee Rowe (11/9/65).
Browns Ferry 1 and 2 (3/1/80)	Ice storm caused loss of both offsite lines. Power supplied by Unit 3.	See Indian Point 2 and Yankee Rowe (11/9/65).
Monticello (4/27/81)	4.16 kV breaker was racked out under load. Breaker then shorted, causing loss of both safety buses.	See San Onofre 1 (6/7/73).
Quad Cities 1 (6/22/82)	Not really an event: Unit 1 supplied Unit 2 when Unit 2 scrammed.	See Browns Ferry 1 and 2 (3/1/80).
Pilgrim (10/12/82)	Storms failed 345 kV lines. 23 kV remained energized.	See Indian Point 2 and Yankee Rowe (11/9/65).
Brunswick 1 (4/26/83)	One offsite power source out for test. Maintenance error caused loss of second source resulting in LOOP.	See Indian Point 2 and Yankee Rowe (11/9/65).
Fort St. Vrain (5/17/83)	One EDG out for maintenance. 2nd EDG in parallel with offsite power. Storm caused LOOP, and 2nd EDG tripped on overcurrent due to faulty load sequencer and operating non-essential loads.	For the ABWR, the CTG could be used to power one of the safety buses if offsite power was not secure. In event of LOOP from any sources, features described under Indian Point 2 and Yankee Rowe (11/9/65) would mitigate the event.
Pilgrim (8/2/83)	Lightning caused loss of all 345 kV.	See Indian Point 2 and Yankee Rowe (11/9/65).
Oyster Creek (11/14/83)	Fire caused loss of power to 1 startup transformer. Switchyard de-energized to permit cleanup. Main generator disconnect links were removed, which allowed for use of unit transformer if necessary (wasn’t used).	See Ginna (3/4/71) and Sequoyah 1 (3/26/84).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Monticello (6/4/84)	One reserve transformer, 1 safety bus, 1 EDG out for maintenance. Procedure error caused loss of energized bus.	See Indian Point 2 and Yankee Rowe (11/9/65).
Quad Cities 2 (5/7/85)	Unit 2 dedicated EDG out for maintenance. Maintenance error caused LOOP to Unit 2. Unit 1 plus swing EDG powered Unit 2.	See Indian Point 2 and Yankee Rowe (11/9/65) and Browns Ferry 1 and 2 (3/1/80).
Millstone 1 (11/21/85)	Reserve station transformer out for maintenance. EDG out for maintenance. Maintenance error caused loss of 345 kV supply.	See Indian Point 2 and Yankee Rowe (11/9/65).
Peach Bottom 3 (4/13/86)	Explosion and fire in transformer caused loss of 1 startup transformer. Alternate startup transformer supplied power.	In the ABWR design, loss of the preferred offsite power source would result in all three emergency diesels starting and picking up respective 1E buses. Power could be manually transferred to the alternate preferred power source (reserve transformer) if desired depending on offsite power reliability.
Hope Creek (5/2/86)	Two of 4 EDGs out for maintenance. One of 3 offsite line out for maintenance. Inadvertent relay actuation caused LOOP to safety buses.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (11/19/86)	Storm failed all 345 kV. 23 kV remained energized.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (12/23/86)	One 345 kV out for maintenance. Flashover caused loss of other 345 kV. 23 kV still available.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Shoreham (3/18/87)	One of 3 EDGs out for maintenance, 1 safety bus out for maintenance. Current transformers shorted as a safety measure. This unbalanced relays serving both service transformers, but without actuating differential current relays. Three weeks later, condensate pump start caused differential relay trip, opening breakers from service transformer. Automatic fast transfer to reserve service transformer occurred, but unbalance caused it to trip. Two EDGs started and loaded.	See Peach Bottom 3 (4/13/86).
Pilgrim (3/31/87)	One 345 kV ring bus breaker out for maintenance. One 345 kV line lost due to storm. Other line isolated due to resultant breaker openings. 23 kV line still available.	See Indian Point 2 and Yankee Rowe (11/9/65).
Peach Bottom 2 & 3 (7/10/87)	Lightning caused loss of 1 of 2 offsite. This caused loss of 1 startup transformer. Other transformer remained in service.	See Peach Bottom 3 (4/13/86).
Vermont Yankee (8/17/87)	Both startup transformers and 1 of 2 345 kV main generator output breakers out for maintenance. Main generator disconnect links were removed. Unit auxiliary transformer energized by main transformer. Upset on grid caused other output breaker to open, causing LOOP. EDGs started, and backup source was still available.	See Indian Point 2 and Yankee Rowe (11/9/65).

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Pilgrim (11/12/87)	23 kV line out of service. Snow failed both 345 kV lines. Startup transformer de-energized due to arcing. EDGs started, and power was restored by removing main generator disconnect links and backfeeding to auxiliary transformer.	See Indian Point 2 and Yankee Rowe (11/9/65).
FitzPatrick (10/31/88)	One 115 kV line out for maintenance. High winds interrupted other 115 kV line. EDGs energized safety buses; efforts were directed at other systems, so shutdown cooling was unavailable for 95 minutes [RCS temperature increased 260 K (10°F)].	See Indian Point 2 and Yankee Rowe (11/9/65).
Nine Mile Point 2 (12/26/88)	One 115 kV line out for maintenance. Current transformer failure caused loss of other line. Out of service line was returned to service and EDGs also started and loaded.	See Indian Point 2 and Yankee Rowe (11/9/65).
Pilgrim (2/21/89)	345 kV lost due to cable failure. 23 kV line available, SBO EDG available. Disconnect links removed for backfeed.	See Indian Point 2 and Yankee Rowe (11/9/65) and Sequoyah 1 (3/26/84).
Browns Ferry 2 (3/9/89)	Bus fault on secondary side of station transformer. EDGs started.	See Ginna (3/4/71) and San Onofre 1 (6/7/73).
Millstone 1 (4/29/89)	Main generator disconnect links removed. Loads had been transferred to station service transformer. Design error in relay of load shed system caused opening of 4.16 kV breakers when reserve station transformer was de-energized. Normal station transformer remained energized.	ABWR undervoltage load shed system will not inadvertently trip 6900 volt loads. ABWR undervoltage relays sense power on bus independent of source.

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Browns Ferry 1 (5/5/89)	Ground faults opened breakers from 500 kV switchyard. Offsite power restored to safety buses from 161 kV switchyard through startup transformer.	See Peach Bottom 3 (4/13/86).
WNP-2 (5/14/89)	One safety bus out for maintenance. Two EDGs out for maintenance. Operator error caused LOOP to other safety buses. EDG started and loaded 1 safety bus.	In the ABWR, the two operable emergency buses could have been energized from either the combustion turbine generator or the alternate preferred offsite reserve transformer.
River Bend (6/13/89)	One of 4 preferred transformers out. Maintenance error tripped 1 preferred transformer, causing loss of power to 1 safety bus. EDG started and loaded. Maintenance error tripped main generator output breakers, causing LOOP to non-safety buses.	See WNP-2 (5/14/89).
Oyster Creek (3/9/91)	One EDG and 1 bus out for maintenance. Routine check revealed other EDG had faulty head gasket which would have caused failure if required. This left plant with only 1 source of power, the startup transformer.	ABWR has two offsite power sources, three diesel generators, and one combustion turbine generator.
Vermont Yankee (4/23/91)	LOOP due to improper maintenance in switchyard. While installing a new battery on non-1E 125 VDC bus, two vital DC buses were cross connected through a battery charger after defeating a mechanical interlock. When the battery charger breaker was opened to install the new battery, a voltage transient was sent through the entire DC control power system which caused both offsite power breakers to trip and lock open.	ABWR procedures do not allow independent vital buses to be cross connected. The multiple sources of onsite and offsite power reduces the need to attempt cross connecting buses. The ABWR has four physically separate and independent 125 VDC systems.

Table 19QC-1 Loss of Offsite Power Precursors (Continued)

Event	Description	Applicable ABWR Features
Diablo Canyon Unit 1 (3/7/91)	LOOP caused by boom of mobile crane shorting out 500 kV transformer. Standby startup transformer was out of service for maintenance. The three EDGs started and picked-up vital buses. Offsite power was restored in five hours.	ABWR has two independent preferred sources of offsite power.
Nine Mile Point (3/23/92)	LOOP while working on aux. boiler circuitry. Div. I diesel was out for maintenance. Div. II diesel started and loaded. Div. III (HPCS) started but tripped on over temperature due to lack of cooling water. All control room annunciators lost due to loss of A and B UPS.	ABWR offsite power supplies are physically and electrically separated so loss of both is not expected to occur due to common cause failure. Three independent electric divisions (including instrument UPSs) would reduce likelihood of simultaneous failure of all three divisions.

Table 19QC-2 Decay Heat Removal Precursors

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 3 79-002 January 8, 1979	Mode 4, Cold Shutdown. RHRS in operation on loop 'A'.	A slight reactor water level drop was detected and determined to be caused by leakage through the minimum flow recirculation valve for the 'A' RHR pump (MO-16A). Vessel level was maintained by use of the stay full pressurizing system. Attempts to eliminate the leakage by further closing the minimum flow valve resulted in its failure to the wide open position. This failure caused a loss of coolant to the suppression pool. The loss of vessel water level continued to the point of isolation of the shutdown cooling system on low water level, at which time the water level stabilized. The time required to raise the reactor water level, via the stay full system, clear the RHRS isolation and reestablish shutdown cooling with the 'C' RHRS pump, allowed the coolant to rise to about 366 K (200°F), causing a gaseous release via disassembled RCIC steam isolation valves.	Failure of the minimum flow recirculation valve associated with the 'A' RHRS pump.	ABWR component design and procurement will emphasize fabrication quality and proper maintenance to minimize individual component failures. However, if failure occurs, SDC would be temporarily lost but two other RHR trains would be available to re-establish DHR before any fuel damage occurred. In addition other heat removal systems (e.g., fuel pool cleanup and cooling (FPC), reactor water cleanup) are available for DHR depending on plant conditions. Other makeup sources (e.g., HPCF, feedwater, condensate, AC Independent water addition, CRDS) can be used if no DHR system is available and the reactor coolant begins to boil.
Hatch 1 August 13, 1979	Mode 5, Refueling. RHRS in operation.	The 'B' loop RHRS was placed in service in the shutdown cooling mode and vessel level was observed to be dropping. Valve E11- F004B was determined to be leaking to the suppression pool. A local leak rate test of the RHRS 'B' pump torus suction isolation valve showed the valve to be leaking in excess of specified criteria. Following corrective action, the valve was satisfactorily retested.	None reported.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Oyster Creek 81-038 August 27, 1981 August 28, 1981	Mode 5, Refueling. RHRS system in operation on loop 'C'. Reactor had been shutdown for 13 days.	This event consists actually of two separate events involving shutdown cooling heat exchanger tube leaks. On August 27, with reactor water temperature at 365 K (197°F), the 'C' shutdown cooling heat exchanger developed a tube leak resulting in reactor water leaking into the RBCCW system as indicated by the RBCCW process radiation monitor. About 2 minutes later, reactor water level began to decrease. The decrease occurred over approximately 10 minutes, with an estimated leak rate of 0.025 m^3/s (400 gpm). Reactor vessel water level was recovered by makeup supplied by the feedwater and condensate system. The 'C' loop was secured and temperature maintained below 373 K (212°F) by use of the 'A' shutdown cooling loop. (Continued on following page)	Circumferential through wall cracks in one tube of the 'A' heat exchanger and one tube of the 'C' heat exchanger, due to fatigue failure caused by flow induced vibration.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Oyster Creek 81-038 August 27, 1981 August 28, 1981 (continued)		On August 28, another RBCCW process monitor alarm was received and the RBCCW surge tank was reported to be overflowing. The 'A' shutdown cooling loop was isolated. The 'B' heat exchanger was out of service but was made serviceable in a few hours. Temperature was maintained by increasing flow to the CUW nonregenerative heat exchanger and increasing letdown to the main condenser. Water was pumped back to the reactor using a condensate pump. In addition to CUW and main condenser systems, the isolation condenser and ECCS systems were all available.		
LaSalle 1 82-039 June 9, 1982	Mode 3, Hot Shutdown. Plant cooldown in progress RHR loop 'A' being placed in service. (Prior to initial criticality.)	While placing RHR 'A' loop in service in the shutdown cooling mode, leakage was discovered at the 'A' RHR pump suction line. RHR loop 'A' was taken out of service for repairs. Alternate methods of decay heat removal were reactor recirc pumps and inboard main steam line drain with CUW.	Leaking flange on spool piece on 'A' RHR pump suction line, caused by thermal growth on heatup and cooldown.	ABWR has three independent RHR loops. Also, the main condenser and CUW are capable of removing decay heat in Mode 3.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 82-042 June 11, 1982	Mode 4, Cold Shutdown (Prior to initial criticality).	The unit was in cold shutdown following performance of reactor internals vibration testing. ‘B’ RHR system was operating in the shutdown cooling mode with all flow bypassing the ‘B’ RHR heat exchanger to maintain reactor temperature between 333 K (140°F) and 366 K (200°F). The ‘A’ RHR system was lined up for standby shutdown cooling. The ‘A’ and ‘B’ RHR suppression pool suction valves were out of service electrically for repair and the valves were manually closed. No backup means of decay heat removal was available due to the reactor building closed cooling water system being out of service. (No actual decay heat existed.) (Continued on following page)	Personnel did not recognize the potential vessel drain path that existed upon returning the system to a normal lineup from standby operation. The test procedure failed to recognize the current operating status of the RHR system in shutdown cooling. The level instruments tap off the downcomer region where shutdown cooling receives its suction. The Tech Specs were interpreted such that both shutdown cooling loops were required operable with one in operation, and that the idle pump could be out of service for only 2 hours. This was a conservative interpretation but it aggravated the event by imposing an arbitrary time restraint on the test.	ABWR procedures will clearly describe proper operational steps and the technical specifications will be based on minimizing plant risks during normal full power operation and shutdown conditions. The ABWR has three independent RHR systems and SDC is isolated on low RPV level.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 82-042 June 11, 1982 (Continued)		Testing of the 'A' RHR drywell spray outboard isolation valve was approved and performed in accordance with procedure. After the test was completed, the system was returned to standby operation. The restoration procedure directed the opening of the RHR 'A' heat exchanger bypass valve. When this valve was opened, water from the reactor vessel filled the previously drained RHR 'A' piping, draining about 11.36 m^3 (3,000 gallons) of water from the vessel. At 31.75 cm (12.5 inches) level, an automatic isolation of the shutdown cooling system occurred. The vessel level was restored, and the 'B' RHR loop was verified filled and vented, and shutdown cooling system suction isolation valves reopened. Reactor vessel level again decreased to about 25.4 cm (10 inches) and a second isolation occurred. It was determined that this second isolation resulted from the starting transient and resulting level drop in the downcomer region. Vessel level was again restored; and shutdown cooling unisolated, vented, and restarted; and the 'A' RHRs loop determined operable.		

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Grand Gulf N/A April 3, 1983	Mode 4, Cold Shutdown, after initial criticality. RHRS Loop ‘B’ in Shutdown Cooling.	Loop ‘A’ of the RHRS was lined up in the LPCI mode, and loop ‘B’ was lined up in the shutdown cooling mode for a surveillance test. After completion of the test, the operator returned ‘B’ loop to the LPCI mode, which required shutting the loop ‘B’ SDC suction valve (F006) and opening the loop ‘B’ suppression pool suction valve (F004). Since a light bulb was burned out on the open indicator for F006, the operator assumed that F006 was already shut, and opened F004. This opened a flow path from the reactor vessel via the ‘B’ RHR loop to the suppression pool. Approximately 37.85 m^3 (10,000 gallons) of water drained from the reactor vessel prior to automatic isolation of the RHRS on low water level. The operator attempted to reshut F004 upon receiving a low level alarm, but the valve’s MOV breaker tripped.	Operator error; misinterpretation of valve position indication. F006 “fully open” indicator light was not burning, but neither was the “fully shut” indicator. Valve was probably in a partially open position. Reason for F004 MOV breaker trip not explained.	The potential for this operator error has been eliminated in the ABWR design by providing valve interlocks. When RHR system is in the shutdown cooling mode (i.e., taking suction from the RPV), the discharge valves to the suppression pool are interlocked in the closed position to prevent inadvertent draining of the RPV. To realign to the Low Pressure Flooder (LPFL) mode, the suppression pool suction valve cannot be opened until the SDC suction valve is fully closed.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Susquehanna 183-056 April 7, 1983	Mode 3, Hot Shutdown.	During a startup test to determine the capability of the shutdown cooling mode of RHR, the ‘A’ RHR heat exchanger was valved in causing a rapid temperature decrease. As a result of RPV water volume shrinkage, the RHR automatically isolated on low reactor water level. CRD flow was used to restore level; and MSIVs were opened to decrease the vessel delta-T. CUW was established to stop stratification. RHR loop ‘A’ was restored, but a valve lineup error caused the pump miniflow valve to bypass RHR flow to the suppression pool, causing a second RHR isolation on low level. Level was restored and RHR reinitiated, but the inventory addition via condensate transfer caused another temperature decrease of 322 K (120°F) in 5 minutes, so the RHR system was isolated a third time to halt the cooldown. The system was restored again, and a fourth short isolation was received when starting the ‘B’ RHR pump.	Reactor Coolant System shrinkage caused by rapid temperature decrease. Valve lineup error caused loss of inventory to suppression pool.	See LaSalle 1 (6/11/82). RHR valve misalignments are minimized in the ABWR design by mode switches for the five operational RHR modes. Selection of a mode (e.g., SDC causes automatic valve realignments).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 2 N/A August 15, 1983	Cold shutdown. Preoperational testing prior to fuel load.	With the control rod drive system in service and the reactor water cleanup system out of service, reactor water level was being controlled by draining through the RHRS 'B' loop to the suppression pool. A new drain path was being established via the 'A' RHR loop (F004 and F006). As soon as this new drain path was lined up, the reactor vessel began draining rapidly. The event did not terminate automatically on low RPV water level isolation of RHRS, because the low level isolation signal had been bypassed by transferring control for the RHR shutdown cooling isolation valves to the remote shutdown panel. This was done intentionally to prevent inadvertent isolations of the temporary drain path. The loss of coolant event was terminated by operator action, 81.28 cm (32 inches) above the top of the fuel region (fuel had not yet been loaded).	Using an unusual valve lineup and bypassing automatic safety features.	See LaSalle 1 (6/11/82). The ABWR design has adequate safety features. However, unusual valve lineups and bypassing of safety features should be performed under strict administrative control.
LaSalle 1 83-108 September 1, 1983	Cold shutdown. RHRS operable.	RHRS operable, but shutdown cooling status not stated. RHRS pump 'A' minimum flow bypass valve (F064A) stuck open following a test. If shutdown cooling was lined up to loop 'A' then a drain path to the suppression pool existed.	Trip fingers which hold the motor operation in handwheel operation were found broken. Valve motor damaged.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 83-105 September 14, 1983	Cold shutdown.	RHR logic testing was in progress which required opening most loop B injection and spray valves: drywell spray valves (F016B and F017B), suppression pool spray valve and test return valves (F027B and F024B), and B and C loop injection valves (F042B and F042C). This lineup relied on testable injection check valve F041B to prevent reactor vessel inventory loss via injection valve F042B to the open spray and test return lines. When F042B was opened, reactor vessel inventory was rapidly lost to the drywell and suppression pool because the testable check valve was stuck open. Most of the water lost from the reactor vessel went to the suppression pool. The operator terminated the event after a 1.27-m (50-inch) level drop to about 4.06 m (160 inches) above the top of the active fuel. Total inventory loss was between 18.93 and 37.85 m^3 (5,000 and 10,000 gallons). It should be noted that no automatic isolation feature would have terminated this flow path; however, the LPCI injection line penetration is above the top of the active fuel.	The LPCI injection check valve was stuck open. Inspection of the valve revealed improper maintenance on the valve operator. The valve had been reassembled by lining up the wrong mark on the spline shaft to the air operator gears, which held the check valve 35° open. The packing gland was also too tight to permit full closure.	ABWR component design and procurement will emphasize fabrication quality and proper maintenance to minimize individual component failures. RHR logic testing does not require that RPV isolation rely on a single check valve during RHR logic testing.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Quad Cities 1 1/24/91	Cold shutdown.	The RPV level decreased 35.56 cm (14 inches) in two related events. The shutdown cooling suction valve was stroked as a maintenance check but some vent and drain valves in the loop were also open, when the SDC suction valve was open the RPV drained 12.7 cm (5 inches). The operator isolated SDC to stop the flow but when the loop was returned to service an additional 22.86 cm (9 inches) were drained from the RPV into the partially empty RHR loop.	Operator error in misaligning RHR valves.	ABWR procedures will highlight RHR system valve alignments during maintenance. The keep fill pump and pressure alarm assures a full loop.
Quad cities 2 8/17/87	Cold Shutdown. On shutdown cooling in one RHR loop, reactor water clean up (CUW) system out for maintenance.	After isolating RCW the RPV level began to increase. Operators attempted to reduce level by draining to the suppression pool using the RHR system test return valve [350A (14-inch) valve]. This resulted in rapid decrease in RPV to low level setpoint and an automatic RPV isolation.	Operator error in not following approved procedure for draining the RPV.	ABWR RHR valves are interlocked to prevent SDC suction and injection valves from being open at the same time as the suppression pool return valves.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Fermi 2 3/17/87	Hot Shutdown following loop test, one RHR loop inoperable.	SDC loop was being put in service but normal loop heatup alignment could not be used because one valve would not open [600A (24-inch) testable check valve]. A smaller [25A (1-inch)] valve was used to fill the loop but the normal 100A (4-inch) drain line caused drainage faster than the 25A (1-inch) line could fill the loop. This drained the loop but the operator could not tell. When proper SDC loop temperature was reached the operator opened the SDC suction valve to the RPV and RPV level decreased to the low level setpoint and RPV isolation occurred.	Operator error in placing SDC loop in service using unapproved procedure.	ABWR RHR system keep fill alarm would alert operator to a partially drained loop condition.
Fermi 2 8/2/87	Cold Shutdown. SDC on Division II.	During the process of shifting SDC from Division II to Division I, a RPV low level signal occurred because valves were misaligned resulting in an open flow path to the suppression pool from the RPV.	Operator error in not following proper procedure placing SDC in service.	ABWR RHR suppression pool suction and SDC suction valves are interlocked to prevent inadvertent RPV drainage.
WNP-2 5/7/85	Cold Shutdown in SDC.	While returning from SDC to standby low pressure injection mode of RHR, the operator opened the suppression pool suction valve before the SDC suction valve was fully closed. This opened a drain path from the RPV to the suppression pool resulting in a low RPV and SDC isolation.	Operator error in not knowing that stroke time for each valve is 90–100 seconds.	Suppression pool suction valve cannot be opened until SDC suction valve is fully closed.
Shoreham 7/26/85	Cold Shutdown both RHR loops in SDC mode.	While returning one RHR loop to standby, operator opened suppression pool suction valve while SDC suction valve was partially open (see WNP-2 5/7/85).	See WNP-2 5/7/85.	See WNP-2 5/7/85.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 2 9/24/85	Cold Shutdown. SDC on 'A' RHR loop.	Loop 'C' SDC suction valve remained open after previous SDC operation. Loop 'A' required a full flow test due to pump problem investigation. SDC 'A' isolated and 'A' pump aligned to suppression pool for test. This opened path from RPV to suppression pool through 'C' SDC suction valve.	Operator error in not knowing status of RHR system valves.	ABWR RHR loops are independent and cross train flow cannot occur.
Riverbend 9/23/85	Cold Shutdown.	While restoring SDC loop to standby, suppression pool suction and SDC suction valves were open at the same time.	See WNP-2 5/7/85.	See WNP-2 5/7/85.
Susquehanna 2 4/27/85	Cold Shutdown.	While placing 'A' SDC on line a path was open from the RPV to the main condenser. RPV level dropped 88.9 cm (35 inches) resulting in RPV low level signal and isolation of SDC.	Operator error improper valve lineup.	ABWR procedures will clearly describe proper valve lineups.
Susquehanna 1 5/10/85 5/20/85	Cold Shutdown.	SDC pump miniflow valve failed open allowing water to flow from RPV to suppression pool.	Valve failure.	SDC would isolate on low RPV level.
WNP-2 8/23/84	Cold Shutdown.	While warming up SDC loop, an isolation signal occurred on high SDC flow. Operator did not notice and loop drained to the radwaste system. When operator placed loop in service water drained from RPV into empty SDC loop.	Operator error. SDC loop isolation not alarmed in control room.	The keep fill alarm would alert the operator to a partially drained RHR loop.
LaSalle 1 9/14/83	Cold Shutdown.	RHR loop in test mode with several valves open. Loop check valve depended upon to isolate RPV. Check valve failed open due to misassembly and improper packing gland installation.	Maintenance error.	ABWR RHR system tests would not require all valves be open and rely on check valve to isolate the RPV.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 2 9/24/84	Cold Shutdown.	Operator attempted to lower suppression pool level to radwaste but loop was in SDC mode and resulted in water diversion from RPV to radwaste.	Operator error.	RHR system drain to the radwaste system contains two valves in series that automatically close on low RPV level.
Pilgrim 81-064 December 21, 1981	Mode 5, Refueling RHRS in operation. Coolant temperature at 294 K (70°F).	While performing maintenance on a feeder transformer, a live transfer of power was attempted. Mal-operation of a power breaker de-energized a vital instrument panel, causing two shutdown cooling valves (MO-47 and MO-48) to close on receipt of a reactor high pressure isolation signal. The ‘C’ RHRS pump should have tripped immediately when its suction valves shut, but failed to do so. After about 5 hours, when the process computer was returned to service, abnormal heat exchanger temperatures alerted operators to a problem. At this time, the ‘C’ RHRS pump was observed to be running with both suction valves shut. The ‘C’ pump was tripped, the valves opened, and the ‘A’ pump started to restore shutdown cooling.	Electrical contacts in the pump trip logic were corroded to the extent that they seized in the open position. ‘C’ RHRS pump, therefore did not trip when the suction valves left their full open position. Inadequacies in the implementation of administrative controls for shift turnover, valve lineup checks, and board checks aggravated the situation. Extensive maintenance activities distracted operators.	See Peach Bottom 3 (1/8/79) and LaSalle 1 (6/11/82).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Susquehanna 1 83-030 February 16, 1983	Mode 4, Cold Shutdown. RHRS in operation on loop 'A'.	The RHRS was operating in the shutdown cooling mode. A Division I isolation signal to the inboard isolation valve to the RHRS caused a loss of shutdown cooling. The system was reestablished by resetting the signals. A second occurrence was experienced within an hour.	The Reactor Protection System (RPS) was operating on alternate power supplies while the RPS MG set was undergoing maintenance. Spurious trips of the RPS alternate power supply breakers caused isolation signals.	Loss of power does not cause isolation of SDC in the ABWR design. The multiplexed safety system logic will only cause isolation if a valid isolation condition existed.
Susquehanna 1 83-060 April 11, 1983	Mode 4, Cold Shutdown. RHR in operation on loop 'B'.	An RPS actuation caused RHR loop 'B' operating in the shutdown cooling mode to isolate. RHR pump 'D' tripped twice on attempts to restart. RHR cooling was established again on loop 'B' using pump 'B'.	RPS actuation caused by an inadvertent breaker trip (bumped by a construction worker). The restart trips are believed to be due to a faulty shutdown cooling flow switch.	See Susquehanna 1 (2/16/83).
Grand Gulf 83-069 May 23, 1983	Mode 4, Cold Shutdown. (During initial plant startup phase).	Following electrical maintenance during which some shutdown cooling motor-operated valves were blocked open, power was restored, and the valves were unblocked. The valves isolated as a result of a previously existing isolation signal from the valve isolation logic, causing a loss of both shutdown cooling loops.	The power supply fuses to the isolation logic had not been replaced following completion of a design change.	ABWR solid state logic minimizes use of fuses and logic testing is easier such that these types of operator errors will be reduced.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Grand Gulf 83-119 August 18, 1983	Mode 4, Cold Shutdown. RHRS loop 'A' in operation. (During initial plant startup phase.)	Both RHR shutdown cooling loops isolated on two occasions during attempts to start a control room air-conditioning compressor. The systems interaction was due to a common power source to the compressor and to leakage detection logic circuitry, which caused the isolation.	The solid state trip unit for the common 480V trip breaker had failed.	ABWR has three independent (both physically and electrically) RHR systems. No common power supplies between RHR systems exist.
Grand Gulf 83-137 September 1, 1983	Mode 4, Cold Shutdown. RHRS loop 'A' in operation. (During initial plant startup phase.)	The RHRS isolated after shifting the RPS power supply to an alternate source. The alternate supply breaker tripped, causing an isolation of shutdown cooling.	The distribution transformer on the unregulated RPS alternate power source was subject to transients.	See Susquehanna 1 (2/16/83).
Grand Gulf 83-193 December 27, 1983	Mode 4, Cold Shutdown.	During an instrument surveillance on the isolation logic for shutdown cooling, the outboard suction valve (F008) closed, isolating both loops of the SDC system. The system was returned to service in 49 minutes.	The cause of the isolation was a tip breaking off a minitest clip used for jumpering.	See Grand Gulf (8/18/83). ABWR solid state logic eliminates need for test jumpers. Surveillance is automated to reduce chance of operator error.
Susquehanna 1 83-172 December 30, 1983	Mode 5, 0% Power.	During the Unit 1 - Unit 2 tie-in outage, one of the RPS 'B' breakers tripped, closing SDC inboard and outboard isolation valves. Reactor coolant recirculation was established through the fuel pool cooling system.	The cause of the trip was a failed breaker.	See Susquehanna 1 (2/16/83).
Hatch 2 September 19, 1986	Mode 4, Cold Shutdown.	Received a low RPV water level signal while valving out a RPV level indicator. This resulted in a scram signal and isolation of SDC. SDC was restored in 10 minutes.	Personnel error in not placing level transmitter in bypass before valving out detector.	ABWR procedures will clearly specify required maintenance steps and precautions to preclude inadvertent SDC isolation.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Hatch 2 September 21, 1986	Mode 4, Cold Shutdown.	Lost SDC for 1.5 hours due to inadvertent RHR suction valve isolation during a surveillance test.	Surveillance procedure required removal of instrument links instead of jumpering them out. When links were opened, a RHR valve isolation signal was initiated.	ABWR solid state logic does not require the use of jumpers to complete circuit logic checks.
Perry 1 October 24, 1986	Mode 4, Cold Shutdown.	While transferring RPS power to an alternate bus to complete RPS MG set maintenance, a voltage transient occurred which resulted in isolation of SDC.	Inadequate procedure for transferring power between buses.	See Susquehanna 1 (2/16/83).
River Bend 1 October 28, 1986	Mode 4, Cold Shutdown.	SDC valve was inadvertently closed when technician accidently grounded a portion of the valve control circuitry during a surveillance test. The ground caused a blown control circuit fuse which resulted in a valve closure signal.	Personnel error.	See Susquehanna 1 (2/16/83).
Perry 1	Cold Shutdown.	SDC isolated due to loss of power to RPS bus. RPS was being powered by alternate power since MG set was in maintenance.	Voltage fluctuation due to starting one of the plant's circulation water pumps, caused electrical protection devices (EPAs) to trip resulting in loss of power to the RPS.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Clinton 1 January 22, 1987	Mode 4, Cold Shutdown.	While performing a reactor coolant system hydrostatic leak test. An isolation of SDC occurred due to high system pressure.	The breaker controller for the high pressure interlock RHR valve was racked out prior to the test to prevent valve closure. Following the test, the trip function was not reset prior to racking in the breaker. When the breaker was racked in the valve closed due to the locked-in high pressure signal.	See Hatch 2 (9/19/86).
Peach Bottom 2 March 28, 1987	Mode 5, Refueling.	Isolation of SDC occurred during maintenance on emergency bus relays.	Maintenance procedure called for pulling fuses prior to replacement of certain relay coils. When one of the required fuses was pulled, the high pressure RHR interlock coil was de-energized. This resulted in isolation of SDC.	See Susquehanna 1 (2/16/83).
WNP-2 April 21, 1987	Mode 5, Refueling.	SDC isolated when an isolation control relay for a non SDC function was de-energized for maintenance.	The neutral wire for several relays, including the SDC relay, were all connected together. Lifting the neutral to one relay caused a loss of power to all relays with a common neutral.	ABWR solid state is less susceptible to this type of failure. Maintenance bypass does not require the lifting of leads.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Hatch 1 April 22, 1987	Mode 3, Hot Shutdown.	While placing a SDC loop in service, RPV level dropped from 157.5 to 7.6 cm (62 to 3 inches).	SDC loop was only partially full prior to placing in service.	See Hatch 2 (9/19/86).
Hatch 1 June 7, 1987	Mode 5, Refueling.	SDC isolated when power was lost to the RPS bus.	RPS MG set output breaker inadvertently tripped.	See Susquehanna 1 (2/16/83).
Perry 1 July 4, 1987	Mode 4, Cold Shutdown.	SDC isolated when power was removed from the RPS bus for a surveillance test.	Procedure did not recognize the impact on SDC of removing power from the RPS bus.	See Susquehanna 1 (2/16/83).
Peach Bottom 2,3 August 16, 1987	Mode 4, Cold Shutdown.	SDC isolation occurred when the normal offsite power supply was lost and a transfer to an alternate source temporarily de-energized electrical buses.	The cause of the loss of offsite power was not included in the report.	See Susquehanna 1 (2/16/83).
Peach Bottom 2 August 28, 1987	Mode 4, Cold Shutdown.	SDC isolated during maintenance on electric circuits.	SDC isolation coil inadvertently de-energized during maintenance.	See Susquehanna 1 (2/16/83) and WNP-2 (4/21/87).
Susquehanna 1 September 13, 1987	Mode 4, Cold Shutdown.	While transferring SDC from the ‘A’ to the ‘C’ RHR pump, SDC isolated.	A spurious high RHR flow signal caused the SDC isolation.	ABWR solid state logic requires two-out-of-four signal to actuate a safety function.
Peach Bottom 2 September 16, 1987	Mode 4, Cold Shutdown.	SDC isolated for 15 minutes.	Loss of power to a MCC.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Perry 1 September 29, 1987	Mode 4, Cold Shutdown.	SDC isolated during a pressure transmitter response time test.	Personnel error in allowing pressure signal from test instrument to exceed SDC high pressure isolation set point.	See Hatch 2 (9/19/86).
Pilgrim October 6, 1987	Mode 5, Refueling.	SDC isolated on loss of power to 480V bus which supplies power to the isolation valve.	Cause for loss of power not reported.	See Susquehanna 1 (2/16/83).
Pilgrim October 15, 1987	Mode 4, Cold Shutdown.	SDC isolated during maintenance on primary containment isolation system.	An incorrect lead was lifted which generated a false high reactor pressure signal.	See WNP-2 (4/21/87).
Susquehanna November 1, 1987	Mode 5, Refueling.	SDC isolated when RPS power supply was transferred between alternate sources.	Momentary loss of RPS power.	See Susquehanna 1 (2/16/83).
Grand Gulf November 30, 1987	Mode 5, Refueling.	SDC isolated during maintenance on power buses.	A temporary loss of power occurred when bus was re-energized following maintenance.	See Susquehanna 1 (2/16/83).
Peach Bottom 2 December 6, 1987	Mode 4, Cold Shutdown.	SDC isolated due to initiation of reactor scram signal.	Technician caused a scram signal to be generated during an ATWS logic pressure switch calibration.	See Hatch 2 (9/19/86).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Nine Mile Point 2 February 1, 1988	Mode 4, Cold Shutdown.	SDC isolated during maintenance on RPV level sensor.	Technician caused a pressure surge in the instrument line which resulted in a high RHR system pressure signal to be generated.	See Hatch 2 (9/19/86).
Pilgrim February 2, 1988	Mode 4, Cold Shutdown.	SDC isolation signal generated during maintenance on emergency parameter information computer.	Personnel error during maintenance.	See Hatch 2 (9/19/86).
WNP-2 May 30, 1988	Mode 4, Cold Shutdown.	SDC isolated during refueling outage.	Maintenance personnel pulled wrong set of fuses.	See Grand Gulf (5/23/83) and Susquehanna 1 (2/16/83)
Peach Bottom 2 July 29, 1988	Mode 4, Cold Shutdown.	SDC isolated during maintenance on PCIS logic circuitry.	Inadequate procedure. SDC isolation logic should have been blocked as part of maintenance task.	See Hatch 2 (9/19/83).
Nine Mile Point 2 October 25, 1988	Mode 4, Cold Shutdown.	SDC isolated during modification work on a RPS cabinet.	Technician inadvertently grounded the RPS 24 VDC power supply.	See Susquehanna 1 (2/16/83).
FitzPatrick October 31, 1988	Mode 5, Refueling.	SDC isolated following a loss of two offsite power lines and a 120 VAC UPS.	Loss of RPS power caused SDC isolation.	See Susquehanna 1 (2/16/83).
Peach Bottom 2 December 6, 1987	Mode 4, Cold Shutdown.	SDC isolated due to initiation of reactor scram signal.	Technician caused a scram signal to be generated during an ATWS logic pressure switch calibration.	See Hatch 2 (9/19/86).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Nine Mile Point 2 February 1 1988	Mode 4, Cold Shutdown.	SDC isolated during maintenance on RPV level sensor.	Technician caused a pressure surge in the instrument line which resulted in a high RHR system pressure signal to be generated.	See Hatch 2 (9/19/86).
Pilgrim February 2, 1988	Mode 4, Cold Shutdown.	SDC isolation signal generated during maintenance on emergency parameter information computer.	Personnel error during maintenance.	See Hatch 2 (9/19/86).
WNP-2 May 30, 1988	Mode 4, Cold Shutdown.	SDC isolated during refueling outage.	Maintenance personnel pulled wrong set of fuses.	See Grand Gulf (5/23/83) and Susquehanna 1 (2/16/83)
Peach Bottom 2 July 29, 1988	Mode 4, Cold Shutdown.	SDC isolated during maintenance on PCIS logic circuitry.	Inadequate procedure. SDC isolation logic should have been blocked as part of maintenance task.	See Hatch 2 (9/19/83).
Nine Mile Point 2 October 25, 1988	Mode 4, Cold Shutdown.	SDC isolated during modification work on a RPS cabinet.	Technician inadvertently grounded the RPS 24 VDC power supply.	See Susquehanna 1 (2/16/83).
FitzPatrick October 31, 1988	Mode 5, Refueling.	SDC isolated following a loss of two offsite power lines and a 120 VAC UPS.	Loss of RPS power caused SDC isolation.	See Susquehanna 1 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
FitzPatrick November 9, 1988	Mode 5, Refueling.	SDC pump stopped when SDC isolation valve left its open position.	Momentary loss of power to RPS caused SDC valve to start closing. Interlock of SDC isolation valve and pump caused control breaker to open.	See Susquehanna 1 (2/16/83).
Fermi 2 January 10, 1989	Mode 4, Cold Shutdown.	SDC isolated when Div. 1 ESF power was lost.	Loss of power cause not reported.	See Susquehanna 1 (2/16/83).
Clinton January 10, 1989	Mode 5, Refueling.	SDC isolated during testing of RCIC logic.	While attempting to jumper out the SDC isolation signal, a technician inadvertently grounded the RPV low level circuit. This caused a fuse to blow and SDC to isolate.	See Grand Gulf (5/23/83) and Susquehanna 1 (2/16/83).
Nine Mile Point January 22, 1989	Mode 4, Cold Shutdown.	SDC isolated during a surveillance test of the reactor building high temperature isolation signal.	Test procedure specified the wrong isolation signal be actuated.	See Hatch 2 (9/19/86).
Hope Creek March 1, 1989	Mode 4, Cold Shutdown.	During performance of a surveillance test, the SDC injection valve closed resulting in a loss of SDC.	Procedural error. Leads were lifted to allow completion of RHR logic test without valve actuations. The lead for the RHR injection valve was inadvertently left off the list of leads to be lifted.	See Hatch 2 (9/19/86) and Hatch 2 (9/21/86).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
River Bend March 25, 1989	Mode 5, Refueling.	SDC cooling isolated when 120 VAC divisional logic was de-energized.	Maintenance personnel de-energized logic power to complete work on the reactor plant sampling system.	See Susquehanna 1 (2/16/83).
River Bend March 29, 1989	Mode 5, Refueling.	SDC isolated due to loss of RPS power.	A jumper fell off during installation causing a ground of RPS power and a blown fuse in the RPS power supply.	See Hatch 2 (9/21/86) and Susquehanna 1 (2/1/6/83).
Grand Gulf April 26, 1989	Mode 4, Cold Shutdown.	RHR pump tripped during surveillance test of RCIC trip throttle valve.	Technician lifted DC power lead for RCIC throttle valve but did not realize that the RHR pump “no suction path” trip logic was also on the circuit. When the lead was lifted, the RHR pump tripped.	See Hatch 2 (9/21/86).
River Bend April 27, 1989	Mode 5, Refueling.	SDC isolated during a surveillance test of manual scram function.	Lead became disconnected during test and grounded out the RHR high pressure interlock circuit. This caused the isolation valve to close.	See Hatch 2 (9/21/86).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 1 77-045 July 28, 1977	Mode 3, Hot Shutdown. Plant cooldown in progress. Temperature at 462 K (372°F).	A reactor cooldown was in progress following a scram. With reactor water temperature at 462 K (372°F), preparations were commenced for placing RHRS loop ‘A’ in shutdown cooling. RHRS booster pumps were started in conjunction with the 1B nuclear SW pump. A gasket ruptured on the RHR service water system as it was being placed in shutdown cooling. Water was observed spraying from the overhead of the 6.1-m (20-ft.) elevation in the reactor building. The 1B loop of RHRS was placed in service at 436 K (325°F). When attempting to place the RHRS ‘1B’ loop in shutdown cooling, it was found that the inboard shutdown cooling suction valve would not open, due to a false signal from a pressure switch.	Ruptured flange gasket on RHRSW loop 1A heat exchanger outlet valve, causing spray-induced electrical damage.	See Peach Bottom 3 (1/8/79). ABWR uses analog transmitters instead of pressure switches for actuation circuits, so this type of failure would not occur in the ABWR.
Brunswick 2 78-036 April 3, 1978	Mode 3, Hot Shutdown. Plant cooldown in progress.	After a reactor shutdown, while establishing shutdown cooling, the shutdown cooling outboard suction valve (F008) would not open remotely. Valve was opened manually and reactor placed in cold shutdown.	Electromechanical brake on valve operator failed, causing valve to bind and the motor operator to draw excessive current when energized.	See Peach Bottom 3 (1/8/79). The current level of the ABWR design does not generally address detail component features. But it is expected that as is the case for operating plants, MOVs will include handwheels to mitigate events such as this.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 2 78-052 June 3, 1978	Mode 3, Hot Shutdown. Plant cooldown in progress.	During normal shutdown and cooldown, RHRS shutdown cooling valve located inside the containment (F009) would not open from the control room. This valve must be opened before the reactor can be placed in cold shutdown. Entry into the drywell via the personnel air lock was unsuccessful. Entry into the drywell was made through the CRD hatch and the RHRS valve was manually opened.	Cause for valve failure not reported. Personnel air lock inner door would not open due to sticky gaskets, caused by large amount of compressive force applied to gaskets by strongback installed 2 days earlier for test. Strongback removed on day of event.	See Peach Bottom 3 (1/8/79).
Brunswick 2 78-074 November 12, 1978	Mode 3, Hot Shutdown.	Reactor steam dome high pressure switch would not reset and would not allow RHRS valve (F008) to open for shutdown cooling at a reactor pressure of 0.80 MPa.	Sticking microswitch caused instrument failure.	See Brunswick 1 (7/28/77).
Brunswick 2 81-019 February 14, 1981	Mode 3, Hot Shutdown. Plant cooldown in progress.	Following a reactor shutdown, while attempting to place RHRS shutdown cooling into service, the RHR supply inboard isolation valve (F009) would not open electrically. Burned motor windings prevented the valve motor from opening the valve. Valve was manually opened and RHRS shutdown cooling placed in service. Cold shutdown reached 8 hours after opening valve.	Thorough investigation revealed no cause for failed motor windings.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 2 81-070 July 18, 1981	Mode 3, Hot Shutdown. Plant cooldown in progress.	While attempting to place RHRS shutdown cooling into service, RHRS shutdown cooling supply inboard isolation valve (F009) would not open on a remote signal. Valve was manually opened, RHRS shutdown cooling placed in service and cold shutdown achieved in 8 hours.	Loose fastener on one of the overcurrent devices in the valve motor breaker, resulting in an overcurrent condition on two of the motor phases, tripping the breaker.	See Peach Bottom 3 (1/8/79).
LaSalle 1 82-034 June 5, 1982	Mode 3, Hot Shutdown at 380 K (225^{o}F). (During initial plant startup phase.)	When lining up for shutdown cooling operation, the RHR shutdown cooling isolation valve (F009) would not open due to an isolated RHR pump suction flow switch.	Flow switch had been isolated to perform calibration check; maintenance tech failed to unisolate instrument after test.	See Brunswick 1 (7/28/77).
Monticello 82-009 September 2, 1982	Mode 3, Hot Shutdown. Plant cooldown in progress.	During startup of shutdown cooling for a refueling outage, the RHRS outboard shutdown cooling isolation valve (MO-2030) motor failed.	Relaxing torque switch problem, which caused continuous close signal to jam the valve gate into the seat.	See Peach Bottom 3 (1/8/79).
LaSalle 1 83-142 November 4, 1983	Mode 3, Hot Shutdown.	The RHR shutdown cooling suction inboard isolation valve (F009) could not be opened either by the motor operator or manually. The unit was shutting down for planned maintenance.	During the last operating period, the valve was manually seated to stop leakage. With the plant at lower temperature, the valve would not open. Failure was attributed to high differential temperatures resulting in thermal contraction and pinching of the disk wedge into the valve seat.	See LaSalle 1 (6/11/82). ABWR has 3 RHR systems. One of the two remaining SDC loops would be available to bring the plant to cold shutdown.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Browns Ferry 1 84-012 February 14, 1984	Mode 3, hot shutdown. Plant cooldown to cold shutdown in progress.	While cooling down to cold shutdown following a manual scram, the inboard RHR shutdown cooling isolation valve (FCV-1-74-78) failed to open, making it impossible to achieve cold shutdown using normal shutdown cooling. An ALERT was declared, and the plant brought to cold shutdown through continued normal cooldown to the main condenser, and the use of control rod drive pumps and RWCUS as alternate inventory addition and heat removal systems. Since the stuck shut suction valve was inside containment, a containment entry was necessary to open the valve manually. It took approximately five hours to de-inert the drywell to permit entry, and another fours hours to open the stuck valve and establish shutdown cooling, after which the ALERT was cancelled. Additional alternate means of heat removal were available.	‘B’ phase winding of motor operator had failed. Apparently the gate had stuck in the valve seat and the motor could not generate enough torque to open the valve. Further investigation revealed that the ‘close’ torque switch setting was set higher than the manufacturer’s recommended value (2.5 vice 2.0). This over-tightening probably contributed to the stuck valve.	See LaSalle 1 (11/4/83) and Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Dresden 3 May, 1978	Mode 4, Cold Shutdown. RHRS in operation at 344 K (160°F).	An inadvertent heatup and pressurization was caused by a valve lineup error during containment leak rate testing. About 18 hours after reaching test pressure, reactor vessel flange temperature was discovered to be at approximately 422 K (300°F) and increasing. One loop of shutdown cooling was in service recording a temperature of approximately 344 K (160°F). The RHRS heat exchanger shell temperature and vessel flange temperature should have been equal. Investigation revealed that the recirc pumps were off and recirc loop suction and discharge valves were open. This lineup resulted in the majority of RHRS flow circulating through the recirc loop and not the core. The vessel heatup and pressurization caused a temperature and pressure increase in the drywell. The computer program used to calculate the containment leak rate was using shutdown cooling temperature to indicate conditions inside the vessel. The computer misinterpreted vessel conditions and concluded there was a large inleakage condition.	Valve lineup error. Post maintenance testing of a recirc pump MG set required a recirc pump test run. The motors were uncoupled from the recirc pumps for the test. The motors would not start because pump/valve interlocks gave a trip signal to the pump motor since the suction and discharge valves were closed. Consequently, maintenance personnel opened the valves to perform the test. This permitted shutdown cooling flow to bypass the core via the recirc loop, causing the inadvertent heatup and pressurization.	See LaSalle 1 (6/11/82). ABWR does not have external recirc pumps or valves. Reactor internal pumps (RIPs) supply recirc flow so this event could not occur in the ABWR.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Hatch 1 80-057 May 25, 1980	Mode 4, Cold Shutdown. RHRS in operation.	With the reactor in the shutdown mode during testing, the shutdown cooling suction valve for the 'B' RHRS pump (F006B) failed to open. The 'B' pump was declared inoperable. Since the 'A' division of RHRS was out for maintenance, both pumps in the 'B' division were required to be operable.	Faulty auxiliary contact block. The normally closed relay contact was found stuck in the open position.	ABWR has three RHR loops, failure of loop 'B' with loop 'A' in maintenance could be mitigated by using loop 'C'. The CUW system, FPC, and main condenser can also be used for DHR under certain plant conditions.
Dresden 3 80-047 December 21, 1980	Mode 4, Cold Shutdown. RHRS system in operation.	Shortly after achieving cold shutdown, with recirc pumps off, CUW system isolated, and with one loop of shutdown cooling system in operation, it was noted that reactor vessel pressure was 1.136 MPa while recirc loop temperature was 341 K (155°F). Primary containment integrity specifications had been violated and both the HPCI and isolation condenser systems were out of service. A second shutdown cooling loop was placed in operation to achieve greater vessel flow, and to eliminate temperature stratification. When the mixing occurred, recirc loop temperature temporarily exceeded 373 K (212°F). Pressure and temperatures were reduced when the second loop was placed into service. The reactor pressure was above 0.72 MPa for about 1.25 hours.	Procedures were inadequate to address temperature stratification in reactor vessel with recirc pumps off and low shutdown cooling flow. Analysis in NSAC-27 also indicated a lower than normal reactor vessel water level contributed to the event by precluding core natural circulation.	See LaSalle 1 (6/11/82).
Dresden 2 83-052 June 21, 1983	Mode 3, hot shutdown. Plant cooldown in progress.	During preparation for placing shutdown cooling in service, a shutdown cooling return valve (MO-5A) failed to open.	The valve stem packing leakage from a nearby valve shorted out the valve operator motor.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
LaSalle 1 83-096 August 24, 1983	Mode 4, Cold Shutdown. RHRS loop 'B' in shutdown cooling operation.	During shutdown cooling operation on RHRS loop 'B', the 'B' heat exchanger discharge valve (F003B) failed to open. Most or all RHR flow was allowed to bypass the heat exchanger, and the heat exchanger outlet temperature increased from 333 to 359 K (139 to 186°F) over a three hour period. The inlet temperatures similarly increased. After three hours of attempts to open the shut valve, the 'B' RHR loop was secured and the 'A' loop started. 'B' loop temperature indication had not been accurate because of low flow conditions and temperature element placement, so actual reactor coolant temperature was higher. 'A' loop heat exchanger inlet temperature reached 370 K (207°F) (violating cold shutdown limits). The RPV head drain indicated a maximum temperature of 378 K (220°F).	The 'B' heat exchanger valve breaker was defective. The measured 'B' heat exchanger temperatures were concluded to be inaccurate due to temperature element location.	Placement of RHRS temperature detectors accurately reflect RCS temperatures if proper flow rates exist. See Peach Bottom 3 (1/8/79) for discussion of component quality and redundancy of DHR capability.
LaSalle 1 83-147 November 12, 1983	Mode 4, Cold Shutdown.	The 'B' RHR heat exchanger outlet valve (F003B) failed to open either by the motor operator or manually. The 'A' loop of RHRS was operable to control decay heat, but one of the two RHR SW pumps cooling the 'A' loop was inoperable.	It is believed that the valve became inoperable in the closed position due to water trapped in the body/bonnet cavity above the disk/seat ring seals. The cavity does not have a mechanism to vent entrapped water.	See Peach Bottom 3 (1/8/79).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Hatch 1 79-050 July 25, 1979	Mode 4, Cold Shutdown. RHRS in operation on loop 'A'.	While in the shutdown cooling mode, the 1C RHRS pump was found to have an excessive leak at the mechanical seal. The pump was removed from service to repair the seal. Both RHR pumps in the 'B' RHRS loop were out of service for hanger repairs. The 1C RHRS pump was returned to the shutdown cooling mode, the 1C RHRS pump was found to have an excessive leak at the mechanical seal. The pump was removed from service to repair the seal. The 1C RHRS pump was returned to service on July 27, 1979.	Ruptured seal in 1C RHRS pump.	ABWR technical specifications will be based on risk associated with shutdown mode and decay heat loads. Under certain conditions to minimize risk, at least two divisions of RHR or multiple alternate methods of DHR will be required to be operable.
Hatch 1 79-051 July 26, 1979	Mode 4, Cold Shutdown. RHRS in operation.	While performing design changes, control power cables to the RHRS outboard isolation valve (F008) were disconnected and cut with the valve in the open position. The inboard isolation valve (F009) had been made inoperable to allow modifications to be made to it. One of these valves is required for isolation of both divisions of the RHRS.	Personnel error in making the modifications to the operable RHRS isolation valve instead of the inoperable valve.	The three ABWR RHR systems are independent of each other. No common components, outside the RPV, exist which would impact more than one RHR division.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 2 80-107 December 8, 1980 80-112 December 9, 1980	Mode 4, Cold Shutdown. RCS temperature at 347 K (165°F). RHRS in operation on 'A' loop.	On December 7, RHRSW was secured in the 'A' loop to repair a leak on a 2.54 cm (1 inch) pipe to the RHRSW radiation monitor. Shutdown cooling was lined up to the 'A' loop with an RHRS pump running (to recirc the vessel water volume without heat removal). Both reactor recirc pumps were secured. 45 minutes were estimated to complete SW repairs. However, repairs were completed in 3 hours. RCS temperature at this time approached 373 K (212°F) with a local maximum of 376 K (217°F). The reactor head vents were open with atmospheric pressure in the vessel. SW was restored and shutdown cooling was initiated. Primary coolant temperature decreased to normal levels approximately 30 minutes after repairs were complete. Shutdown cooling was not lined up in loop 'B' because it was expected that loop 'A' would be back in service prior to approaching 373 K (212°F), and because there were possible leaks on a room cooler and inoperative 'B' loop pump suction valve motors. (Continued on following page)	In both events, maintenance was not completed in expected time. In the first event, loop B was available but not used, due to potential leaks on a room cooler and the requirement for manual valve operation due to inoperative pumps suction valve motors. In the second event, securing RHRS pumps while maintenance was in progress caused loss of representative temperature indications due to low flow and lack of vessel recirculation. Control room operators did not recognize the heat up rate. Failure to plan and promptly implement contingency plans for the possibility of unexpected delays in maintenance also contributed to the problem.	ABWR has three divisions of RHR. In this case, loop 'C' could have been used. See LaSalle 1 (6/11/82) for discussion of ABWR procedures.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 2 80-107 December 8, 1980 80-112 December 9, 1980 (Continued)		On the next day, the conventional and nuclear SW systems were secured to repair the 2A conventional SW pump discharge check valve. RCS temperature was initially <322 K (<120°F). Approximately 2 hours later, RHRS pumps were secured to reduce coolant heat input from the pumps. Approximately 4.5 hours later when the system was restored, the average RCS temperature was over 373 K (212°F) with a local maximum of 398 K (256°F). Again, vessel head vents were open during the event.		
Peach Bottom 2 81-031 May 18, 1981	Mode 4, Cold Shutdown. RHRS in operation.	With the unit shutdown for maintenance, shutdown cooling was secured to permit maintenance of a shutdown cooling suction isolation valve. RCS temperature exceeded 373 K (212°F) before cooling was reestablished. Temperature exceeded 373 K (212°F) for about 2.5 hours. Primary containment integrity requirements were not met during this period.	Lack of timely coordination between operations and maintenance personnel.	See Brunswick 2 (12/8/80).
Hatch 2 82-030 April 20, 1982	Mode 5, Refueling. RHRS in operation on loop 'A'.	The 'A' loop flow indicators for both RHRS and RHRSW systems were noticed to be inoperable. Investigation revealed that the indicators and controller for the RHRSW heat exchanger pressure control valve were de-energized. The 'A' loop RHRS and 'A' RHRSWS were declared inoperable and fuel movement was suspended.	Sliding links were opened by maintenance personnel while performing a wiring change.	See LaSalle 1 (6/9/82 and 6/11/82).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Hatch 2 82-042 April 27, 1982	Mode 4, Cold Shutdown, with 'A' loop of shutdown cooling service.	The RHR and RHRSW flow indicator for the 'A' loop in shutdown cooling were inoperable. The 'A' loop was declared inoperable. The 'B' loop was already inoperable for the leak rate testing.	The spring clips on the fuse block energizing the 'A' loop RHR and RHRSW flow indicators were loose.	See Brunswick 2 (12/8/80) and Hatch 1 (7/25/79).
Browns Ferry 1 77-003 January 4, 1977	Mode 4, Cold Shutdown. RHRS in operation on loop 'A'.	The radiation monitor on the RHRSW discharge line from 1A RHRS heat exchanger showed an increasing radiation level, approximately 1 hour after being placed in service. Heat exchanger service water effluent was sampled and found to be in excess of release limits. The 1C RHRS heat exchanger was then placed in service, approximately 5 hours after the initial radiation alarm.	Leaking inner head gasket in heat exchanger, due to loose stud bolts. Delay in leak isolation due to failure to acknowledge alarm, and communications misunderstanding over the actual release rate occurring.	See LaSalle 1 (6/9/82 and 6/11/82).
Brunswick 2 80-030 April 12, 1980	Mode 4, Cold Shutdown. RHRS in operation.	During inspections, the 2B RHRS heat exchanger baffle plate was found to be partially buckled near the bottom where it fitted into the groove of the channel cover. The plate was 21.59 cm (8.5 inches) off-center, and welds up each side were pulled loose within the waterbox. Approximately 20 - 25-cm (8 - 10-inch) thick accumulation of marine growth shells were found in the inlet side of 2B heat exchanger waterbox, and about the same in 2A heat exchanger inlet waterbox, although the 2A baffle plate was not damaged. The buckling created a service water bypass flow path from the heat exchanger inlet to outlet bypassing the tubes.	Excessive differential pressure across the baffle plate due to an accumulation of marine growth shells in the heat exchanger.	ABWR procedures to minimize marine growth have been modified to ensure this type of event does not occur. Intermediate RCW system loop provides clean water to the RHR heat exchanger. Also, alternate methods of DHR such as the main condenser, FPC, and CUW can be used under certain plant conditions.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Brunswick 1 81-032 April 19, 1981	Mode 4, Cold Shutdown. RHRS in operation.	During inspection of the 1B RHRS heat exchanger, it was found that the heat exchanger baffle plate was displaced about 23 cm (9 inches), creating a direct SW flow path from inlet to outlet, bypassing the tubes. During repair of the 1B heat exchanger, a loss of cooling was experienced immediately following the starting of a second RHRSW pump on the 1A heat exchanger. Alternate cooling was established with the RHRS system by flow from the vessel, through the fuel pool coolers and the CST. Vessel temperature remained below 350 K (170°F). The 1A heat exchanger was also found to have a displaced baffle plate.	Failure of plate welds, resulting from excessive differential pressure across the plate. Excessive differential pressure attributed to blockage of the tubes by marine shells accumulating in the heat exchanger. The SW chlorination system had been out of service for an extended period.	See Brunswick 2 (4/12/80).
Brunswick 2 81-049 May 6, 1981	Mode 1, 76% power.	As a result of problems with Unit 1 RHRS heat exchangers, a special inspection of Unit 2 RHR HXs was conducted at power. Heat exchanger 2B was damaged and plugged by marine shell buildup. The divider plate was found buckled about 7.6 cm (3 inches). (It had been replaced in 1980—see LER 80-030 above.) The heat exchanger had blocked and obstructed tubes. Heat exchanger 2A was undamaged with no divider plate buckling, but was substantially blocked by shells.	Same cause as for Unit 1 above.	See Brunswick 2 (4/12/80).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Browns Ferry 1/2/3 81-047 August 22, 1981	Units 1 & 3, Mode 4, Cold Shutdown. Unit 2, Condition 1, 91% Power.	The A2 RHR service water/EECW pump discharge line air vent valve failed, resulting in the flooding of 'A' RHRS service water/EECW pump room to a depth of approximately 198 cm (6 1/2 ft), rendering A1, A2, and A3 RHRSW/EECW pumps inoperable. Consequently, the 'A' RHRS heat exchangers for the 3 units became inoperable. (The RHRSW/EECW system is common to all three units.)	The 'A2' pump discharge air vent valve failed to seal because of a broken float guide, causing the float to misalign with the seat.	ABWR is a single unit design and failures will not propagate to other plants. If more than one ABWR is at a site, cross connected systems between units will not be allowed. In addition, ECCS divisional rooms contain water tight doors such that flooding would be contained within the room and only affect one division. Floods in other reactor building rooms are mitigated by raised sills, floor drains, and operator action in response to flood alarms.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Browns Ferry 3 83-004 January 16, 1983	Mode 4, Cold Shutdown RHRS in operation on loop 'B'.	RHR heat exchanger 3D leaked reactor coolant into the RHRS service water in excess of Technical Specification limits. The 'D' heat exchanger and pump were removed from service and the 'B' heat exchanger and pump placed in service. Approximately 8 hours later an alarm was received on the SW effluent monitor. The 'B' heat exchanger and pump were removed from service. (The 'A' and 'C' heat exchanger and pumps were inoperable due to a bent stem on their common injection valve.) Thus there was a complete loss of RHR shutdown cooling capability. The RCS temperature increased from 360 to 373 K (188 to 211°F) in approximately 45 minutes. Reactor heat removal was provided by steaming to the main condenser, and by coolant makeup from the CRD and CUW systems.	Twelve dented tubes were found in the 'D' heat exchanger. One of the dented tubes was leaking. The 'B' heat exchanger did not actually leak, but had to be isolated until it could be confirmed to not be leaking. The 3B and 3D heat exchangers share a common radiation monitor.	ABWR has three independent RHR loops. The probability of losing all three loops due to component failures is very low. Even if loss of all RHR were to occur, DHR could be completed using the main condenser, CUW, FPC, CRD, HPCF, condensate, or fire protection water systems depending on plant conditions.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 3 81-014 September 2, 1981	Mode 4, Cold Shutdown.	While in cold shutdown near the end of an extended refueling outage, difficulty in maintaining reactor water level was encountered when the control rod drive water system was removed from service to support plant testing. Vessel level decreased to about 152 cm (60 inches). A feedwater inlet valve was then opened slightly to supply makeup water to the vessel. (Pumping source not stated in LER, and uncertain because turbine driven feed pumps are unusable in cold shutdown. Source was probably condensate pump.) Vessel level was recovered to 229 cm (90 inches) and the feedwater valve closed. Leakage through the valve occurred and level increased above the main steam line nozzles. As a result of the loss of the reactor vent path (main steam lines to condenser), the reactor pressurized to about 0.322 MPa for about 35 minutes (MSIVs assumed to be shut to prevent flooding of steam lines). To decrease vessel level, water was transferred from the vessel to the torus. Later during an attempt to obtain a tight shutoff of the feedwater inlet valve, the reactor was again pressurized to about 0.646 MPa for 30 minutes. The reactor head vents were opened to depressurize the vessel.	Operator failed to recognize that he was losing primary system inventory when the CRD water system was removed from service. Incomplete closure of feedwater inlet valve MO-3-2-29B caused level increase above main steam line nozzles and subsequent pressurization.	See LaSalle 1 (6/11/82).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Browns Ferry 2 83-005 February 16, 1983	Mode 5, Refueling.	During preparation for a containment integrated leak rate test (ILRT), a spurious low reactor vessel water level signal was initiated, apparently due to improper operation of a high drywell pressure switch drain. The combination of low water level and high drywell pressure signals started four core spray pumps, four RHR pumps, and eight diesel generators. The RHR system was secured before injection into the vessel occurred. However, a total of 167 m^3 (44,000 gallons) of water were injected into the vessel from the torus via the core spray system, which caused spillage into the drywell sumps via an open head vent, and put some water into the steam lines. The vessel head was in place with the head fastening nuts not installed.	Cause not reported in LER.	See LaSalle 1 (6/11/82). ABWR has complete divisional separation in a two-out-of-four logic network that prevents spurious initiation signals from single event errors.

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Peach Bottom 3 83-007 March 3, 1983	Mode 5, Refueling.	During a refueling outage, an inadvertent initiation of two RHR pumps in the LPCI mode caused an injection of 246 m^3 (65,000 gallons) of water from the torus into the reactor vessel. Since the unit was in refueling with the reactor cavity flooded, most of the water overflowed onto the fuel floor, and down the main hatchway to El. 41,150 mm (135 feet), where approximately 0.189 m^3 (50 gallons) flowed out the building under the railroad door and into the storm drain system. The initiation was a false low water level signal which was present for less than 3.5 seconds. The signal started all operable diesel generators, tripped and isolated recirc pumps, tripped HPSW pumps, and started 2 RHR pumps. Diesel generator starts and the large number of spurious alarms distracted operators from verifying reactor water level until about 4 minutes after actuation, at which time the pumps were tripped and injection valves closed. Personnel exited the area, and no personnel exposures resulted from the flooding. The total dose associated with the subsequent cleanup effort was less than 0.02 person-Sievert. Total release was estimated at 11.69 megabecquerel.	The spurious low water level signal was caused by a pressure surge in the reference leg of the 2B Yarway instrumentation loop during surveillance testing.	See LaSalle 1 (6/11/82) and Browns Ferry 2 (2/16/83).

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Vermont Yankee March 9, 1989	Mode 5, Cold Shutdown. RHRS in operation on loop 'B'.	With loop 'B' of RHR in SDC mode and loop 'A' out of service for maintenance, 'A' and 'C' RHR pump motor breakers were racked out for maintenance. System logic then causes the mini-flow valves for these pumps to open. Following maintenance, the 'A' and 'C' SDC suction valves were manually stroked open per procedure. This opened a drain path from the RPV to the suppression pool. Reactor cavity level dropped approximately 554 to 183 cm (218 to 72 inches) above top of active fuel.	Improper use of procedures.	In the ABWR design, racking out the RHR pump breakers does not result in the mini-flow valves opening. See LaSalle 1 (6/11/82) for discussion of ABWR procedures.
Susquehanna 1 February 3, 1990	Mode 4, cold Shutdown. RHRS in SDC mode using loop 'A'.	With reactor coolant temperature approximately 264 K (125°F), the RHR system was removed from service to perform a test of the RPS electrical protection assembly (EPA) breakers. RHR must be secured during this test because opening the EPA breakers causes isolation of SDC. Following testing, difficulty was experienced in closing some of the EPA breakers to energize the RPS. This delayed reestablishing SDC. Reactor coolant temperature increased to 396 K (253°F) and pressure increased to 0.232 MPa before SDC was restored.	Excessive time to complete maintenance.	Alternate means of DHR could be used including main condenser, venting steam to the suppression pool through SRVs and suppression pool cooling. See Susquehanna 1 (2/16/83).
Quad Cities 2 4/2/92	Mode 4, Cold Shutdown.	SDC was lost for two hours and twenty minutes due to loss of power to 1E buses.	Inadvertent actuation of fire protection deluge system.	See Susquehanna 1 (2/16/83)

Table 19QC-2 Decay Heat Removal Precursors (Continued)

Event Category: Losses or Degradation of RHRs Due to Loss of Coolant from Reactor Vessel

Plant LER/date	Initial Plant Conditions	Event Description	Reported Cause	Applicable ABWR Feature
Washington Nuclear Plant 2 May 1, 1988	Cold Shutdown with RHR 'B' in SDC mode.	Operators were in the process of changing the operating SDC loop from 'B' to 'A'. The procedure called for closing the loop 'B' SDC suction valve and then open the loop 'B' suppression pool suction valve. The operator did not wait until the SDC suction valve completely closed before opening the suppression pool suction valve. The stroke time on each of these valves is 120 seconds. Both valves were partially open for 40 seconds and resulted in about 37.85 m^3 (10,000 gallons) of water draining from the reactor cavity to the suppression pool. Draindown was automatically terminated on low RPV level when SDC was isolated.	Improper operator action.	The ABWR suppression pool suction valve cannot be opened until the SDC suction valve is fully closed.
River Bend April 19, 1989	Cold Shutdown.	Work was being performed on the standby service water (SSW) supply and return valves. As these valves are unisolatable, freeze seals were being used to isolate the valves. One of the freeze seals failed and caused approximately 56.78 m^3 (15,000 gallons) of water to flood the Division II ECCS power supply room. Electrical faults resulted in loss of power to RPS bus 'B'. This caused containment isolation and loss of SDC.	Improper freeze seal implementation.	ABWR freeze seal procedures will include adequate administrative controls to minimize freeze seal failures. Analysis have been completed to ensure that flooding in the ABWR will not result in loss of ECCS or RPS power supplies.

19R Probabilistic Flooding Analysis

19R.1 Introduction and Summary

The ABWR has been designed to withstand the effects of postulated flooding internal to the plant. This appendix discusses the capabilities of the ABWR to withstand internal flooding (e.g., service water, suppression pool line breaks).

Results of the ABWR probabilistic flood analysis show:

(1) The only buildings where potential flooding could damage safety-related equipment or cause plant transients are the turbine, control, service and reactor buildings. The radwaste building does not contain safety-related equipment and flooding cannot affect safety-related equipment in other buildings. Failure of seals in the radwaste tunnels between buildings was determined to result in several orders of magnitude lower core damage frequency than direct flooding due to pipe breaks in each building and was not included in the flooding event trees.

(2) The flood concern for the turbine building is water filling up the condenser pit and flowing into the service building tunnel which is the access path to the reactor and control buildings. The reactor and control buildings contain safe shutdown equipment. The turbine building has the potential to be flooded by two unlimited sources: circulating water and turbine service water. The condenser pit contains redundant water level sensors (in a two-out-of-four logic) which send an alarm to alert the operator to potential flooding and automatically trip the circulating water system (CWS) pumps and close CWS isolation valves. In the unlikely event this automatic protection fails and the operator fails to take any action, potential flood waters would still be prevented from reaching the service building. Potential flood waters would be expected to exit the turbine building through the non-watertight truck entrance door. Also, there is a normally closed and alarmed door separating the turbine and service building access tunnel. If this door were to open due to water pressure from the flood, watertight doors at the entrances to the reactor and control buildings from the service building should prevent damage to safety-related equipment. Turbine service water (TSW) breaks must be manually mitigated by either tripping the pumps, or closing valves, or opening the truck entrance door. Sufficient time is available to complete these actions (greater than several hours) due to the relatively low TSW flow and the large size of the turbine building. CWS breaks dominate the CDF so no TSW event trees were completed. Thus, no impact on plant safety is expected from potential turbine building flooding. The estimated core damage frequency from turbine building flooding is extremely small for a plant with a low power cycle heat sink (PCHS) and is slightly higher for a high PCHS.

(3) The control building could potentially be flooded by the reactor building service water (RSW) system which is an unlimited source or by breaks in the Fire Water System. The control building has six floors but floor drains and stairwells would direct all potential flood waters to the bottom floor where the safety-related reactor building cooling water (RCW) system components are located. There are three divisions of RCW/RSW in physically separate rooms with watertight doors. The RCW/RSW rooms in the control building lower level contain two sets of water level sensors in each division in a two-out-of-four logic. The first set of sensors send an alarm signal to the operator at 0.4 meter. The second set of sensors are actuated at 1.5 meters and send an alarm signal to the operator and trip the RSW pumps and close RSW system isolation valves in the affected division. Water remaining in the lines between the control building and the ultimate heat sink could be siphoned or drained into the control building. The water pumped into the control building prior to isolation of the RSW system and the water drained in from the RSW line outside is limited to affecting only one RCW division. The two other safety divisions (or alternate means) would remain undamaged and able to be used to achieve safe shutdown if necessary. The estimated core damage frequency from RSW flooding is extremely small.

Fire Water System breaks could cause flooding in all three safety divisions on a given floor since doors separating the divisions do not have sills. Floor drains and other floor openings in all three divisions ensure that postulated fire water breaks, if unisolated, will be directed to the first floor. The CDF for fire water flooding in the Control Building is extremely small.

The total control building flooding CDF is extremely small.

The reactor building is adequately protected from flooding concerns by the following:

(a) Inside secondary containment, extensive flooding sources in ECCS divisional rooms at the lowest elevation are limited to impacting no more than one safety division by watertight doors. Extensive flooding in non-divisional rooms (i.e., corridors) is prevented from entering divisional rooms by watertight doors. In addition, the corridor volume is large enough to contain the largest flood source (Suppression pool). At higher elevations, potential flooding in systems such as Fire Water is directed to the first (bottom) level by floor drains and stairwells. The CDF for flooding inside secondary containment is extremely small.

(b) Outside secondary containment, floor drains direct all flood sources to the sumps on floor B1F. If the sump pumps fail or flood rates exceed sump pump capacity, a sump overfill line directs water to the corridor of floor B3F inside secondary containment where it can be contained as discussed above. Emergency diesel generator lube or fuel oil leaks are contained within the individual rooms until a portable pump can be brought in to remove the oil. The estimated core damage frequency for reactor building flooding outside secondary containment is extremely small.

(c) The total reactor building flooding CDF is extremely small.

(4) The estimated total core damage frequency from internal flooding is very small for a low PCHS and slightly higher for a high PCHS. This low risk level is attributable to the relatively low probability of large internal floods and the physical separation of certain safety equipment in the ABWR design. It is highly unlikely that a single flood can result in loss of more than one safety division. Where there is a potential for large flood sources to affect equipment in more than one division, instrumentation for detecting the flood and isolating the flood source is provided. The two remaining safety divisions and alternate core cooling and decay heat removal features (e.g., AC independent water addition, power conversion system) give high assurance of achieving safe shut down.

19R.2 Scope of Analysis

The ABWR flooding analysis covers all phases of plant operation. It addresses all potential flooding sources and their impact on safe shutdown of the plant. The effect on safety systems that are required to achieve and maintain safe shutdown is covered.

The analysis is completed in three steps. First, a listing is completed of all internal water sources and the buildings that they serve. This list is then screened to determine the sources and buildings that have a potential to prevent safe shutdown.

Following the screening analysis, the ability of the plant to achieve safe shutdown is analyzed both deterministically and probabilistically. The deterministic analysis describes plant features that are designed to either prevent or mitigate potential flooding concerns. This analysis focuses on plant features such as physical separation of buildings and rooms within buildings, isolation mechanisms to limit flooding, and the ability of the plant to contain potential flood waters due to room size and sump pumps. The intent of the deterministic analysis is to show that, for all postulated water sources, the ABWR design features can, with realistic operator actions, successfully achieve safe shutdown.

The probabilistic flooding analysis involves the use of event trees to evaluate the frequency of core damage for pipe breaks in various systems and buildings. Pipe breaks for each building of concern are evaluated and shown to have a negligible contribution to core damage frequency.

The results of the analysis are presented in terms of insights gained from the study and interface requirements that came out of the study which will be used as input for the inspections, tests, analysis and acceptance criteria (ITAAC), reliability assurance program (RAP), and emergency procedure guideline programs. Lastly, the main conclusions from the flooding analysis are presented which support the ABWR's capability to withstand postulated internal floods.

19R.3 Screening Analysis (Water Sources and Buildings)

In order to focus the flooding analysis on buildings and water sources that have the potential to cause flooding concerns, a screening analysis was completed to eliminate sources and buildings that, for various reasons, do not require further analysis.

The screening analysis was carried out for each of the buildings. From a safe shutdown perspective, the radwaste building does not contain any equipment that is required for safe shutdown and because of physical separation, flooding cannot affect safe shutdown equipment in other buildings. Therefore, the radwaste building was not evaluated further for flooding concerns. Failure of seals in the radwaste tunnels between buildings was determined to result in several orders of magnitude lower core damage frequency than direct flooding due to pipe breaks in the buildings and was not included in the flooding event trees. Adequacy of these seals should be confirmed by the COL applicant. The turbine building does not contain any safe shutdown equipment but a flood could cause a turbine trip which is an accident initiator. Also, the turbine building is next to the service building which is the access to the reactor and control buildings and so flooding between the two buildings must be considered. The reactor and the control buildings contain safe shutdown equipment (e.g., RHR, RCIC, HPCF, RSW, Class 1E batteries). The flooding analysis will thus focus on the turbine, control, service and reactor buildings, all of which either contain safety-related equipment or where flood damage could result in plant transients.

The sources of water in the ABWR are shown in Table 19R-1. As will be shown later, some of the smaller water sources (e.g., HVAC) can be eliminated due to insufficient volume to cause flooding concerns (i.e., damage safety-related equipment).

Potential flooding in the main steam tunnel and inside the drywell are adequately addressed in the LOCA discussion included in the full power PRA (Appendix 19D) and will not be further discussed in this appendix. In addition, the spent fuel pool is a seismic Category I structure that is fully lined and does not contain any drain lines. Therefore, flooding due to leaks in the spent fuel pool was also not considered in the study.

19R.4 Deterministic Flood Analysis

This subsection summarizes the physical design features of the ABWR that are capable of mitigating the effects of potential floods. A more detailed discussion of ABWR flooding features is contained in Tier 2 Subsection 3.4. The analysis will focus on the turbine, control, and reactor buildings.

19R.4.1 Analysis Assumptions

The following general assumptions apply to all buildings in this deterministic flooding analysis:

(1) In moderate energy piping larger than nominal one inch diameter, leakage cracks are postulated to occur in accordance with ANSI/ANS 56.11, "Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants."

(2) No credit is taken for operation of the drain sump pumps although they are expected to operate during some of the postulated flooding events.

(3) When flooding can be identified and terminated by operator action from the control room, a 10-minute response time is assumed. If flooding is identified only visually and subsequent local or control room action is required to mitigate the flood, a 30-minute response time is assumed.

(4) A single active failure of flood mitigating systems is assumed following the flood.

19R.4.2 General Design Features

In each ABWR building with a potential flooding concern, there are common design features that are capable of mitigating potential floods. These features include:

- Wall and floor penetrations for cables and pipes
- Automatic pump trips on high water level or high system flow rate in a room
- Room barriers that are capable of containing water within a room or preventing water from entering another room
- Alarmed watertight room doors to ensure integrity
- Floor drains
- Pipes within or between rooms contained in pipe chases

These features are described in more detail below.

19R.4.2.1 Penetrations

Whenever an electrical cable, pipe, or HVAC duct must pass through a wall or a floor separating areas of different safety-related divisions, a penetration seal is provided to ensure the integrity of the room.

Cable tray penetrations are furnace tested to 1089 K (1900°F) and subjected to a hose stream test [38.1-cm (1.5-inch) hose operating at 0.618 MPa]. This ensures that flooding by hot sources such as reactor water cleanup (CUW) will not cause failure of the penetration.

Piping penetrations have been minimized throughout the plant to reduce the potential for loss of barrier integrity. No high pressure or high temperature piping lines penetrate walls or floors separating two different safety divisions. Piping penetrations are qualified to the same differential pressure requirements as the walls or floors they penetrate.

HVAC ducts have also been minimized throughout the plant. In areas where isolation is essential (e.g., secondary containment), motor operated positive shutoff valves are provided in the HVAC duct.

19R.4.2.2 Automatic Pump Trips

Some rooms contain level sensors to detect the presence of water in the room. In general, one set of level sensors alerts the operator of a potential flooding condition and a second (higher level) set of sensors actuate to trip pumps that could be causing the water level to increase in the room. These sensors are safety or non-safety grade (depending on the application), diverse, and typically arranged in a two-out-of-four logic.

Some systems (e.g., fire water) have high flow rate sensors to detect leaks in the system. In this case, the operator would be warned of a high flow rate instead of high room water level. Appropriate action would then be taken to isolate the leak.

19R.4.2.3 Room Barriers

Except in primary containment and the control room complex, divisional areas are separated from each other by 3-hour rated fire barrier walls and floors.

These walls are made of concrete and are at least 15.24 cm (6 inches) thick. They are designed to ensure that fires are not propagated between safety-related divisions and thus act as effective flood barriers. As with penetration seals, the fire barriers are flame and hydrostatically tested to ensure their high temperature performance and thus will not fail due to flooding by hot water sources.

19R.4.2.4 Watertight Doors

ECCS equipment rooms on the first floor of the reactor and control buildings have watertight doors. Also, external entrances to the control and reactor buildings below grade level have watertight doors. The entrance to other divisional rooms have fire rated doors. These doors are normally closed and are included in the security surveillance system. These doors can be opened only with a card key and if left open security personnel will be alerted immediately. This system gives high assurance that the divisional separation will not be breached due to a door being inadvertently left open. The alarm system can detect if a watertight door is closed but not if it is dogged. A once per shift walkdown will ensure that watertight doors remain dogged when not in use.

In cases where a fire door must be opened due to maintenance or surveillance activities, administrative controls are implemented to require that a watch be posted near the door until the activity is completed and the door returned to its normally closed position. In the event of a flood in an ECCS room when the other two ECCS rooms are opened, plant procedures should direct operators to ensure that at least one of the two unflooded room doors be closed before opening the door to the flooded ECCS room.

19R.4.2.5 Floor Drains

The reactor and control buildings contain floor drains to direct potential flood waters to rooms where sumps and sump pumps are located. The drain system is sized to withstand breaks in the fire water system which is the most probable flood source for these two buildings. Sizing of the drain system will include provisions for plugging of some drains by debris.

The drain system will be designed so that floor drains in ECCS rooms will be connected to the corresponding divisional sump in the ECCS rooms on the first floor. Non-divisional rooms will drain to the non-divisional sumps on appropriate floors.

Floor B1F of the reactor building has overfill lines on the non-divisional sumps outside secondary containment. If the sump pumps fail or the flow rate exceeds the sump pump capacity, the lines will direct water to the non-divisional corridor of the first floor (B3F) inside secondary containment. A water seal is provided to maintain secondary containment integrity.

19R.4.2.6 Pipe Chases

When pipes are run between certain buildings(e.g., main steam feedwater lines between the reactor building and the turbine building) they are encased in sections called “chases.” The chases are capable of retaining the water or steam that may be deposited due to a pipe break or leak. In essence, the pipe chase acts as a divisional barrier similar to a fire wall or floor.

19R.4.2.7 Equipment Mounting

All electrical equipment is mounted 20.32 cm (8 inches) off the floor to help protect against damage from potential flood sources.

19R.4.2.8 Electrical Equipment Design

Electric motors are all of drip proof design and motor control centers have NEMA Type 4 enclosures. Both of these features protect electrical equipment from water spray and dripping water from above.

The above general design features all contribute to limiting the risk due to potential internal flooding in the ABWR. The following discussion addresses specific features of the turbine, control, and reactor buildings that prevent or mitigate postulated ABWR internal floods.

19R.4.2.9 Shutdown Considerations

During shutdown, increased maintenance activities introduce the potential for flooding to impact more than one safety division. As discussed in SSAR Subsection 19Q.7, it is recommended that maintenance during shutdown be completed on one division at a time. The recommended shutdown configuration is: one RHR division and support systems operating, one safety division administratively controlled to not be in maintenance and its barriers intact, and the third division undergoing maintenance.

In this configuration, flooding in any division will still result in at least one division being available for decay heat removal. If flooding occurred in the intact division, the water would be contained in that division and the operating division would continue to supply decay heat removal capability. The watertight doors are designed to stop water from entering the room (i.e., the door seals seat with water pressure from flooding external to the room) but only small leakage is expected past the seals from flooding in the ECCS room. If flooding occurred in either of the two other divisions, even if barriers in both divisions were breached for maintenance, the intact division would be available.

Due to maintenance activities there is also greater potential for debris clogging floor drains and preventing proper drainage of flood waters. The ABWR contains an adequate number of floor drains that if some were to become plugged the remaining drains would be available to direct water to the sump tanks. Some backup of water may occur but equipment mounted at least 20.32 cm off the floor would ensure that no equipment was damaged. Also, normal housekeeping tasks required by NRC regulations would keep debris to a minimum.

19R.4.3 Turbine Building

There is no safety-related equipment located in the turbine building. It is included as part of the detailed flood analysis because it contains non-safety-related equipment (e.g., condenser, condensate pumps) that could be used to achieve safe shutdown if required, a turbine building flood could result in a turbine trip which is a transient initiator, and because it is connected to the control and reactor buildings through the service building access tunnel. Since the control and reactor buildings contain safety-related equipment, interbuilding flooding must be addressed.

The water sources contained in the Turbine Building are shown in Table 19R-1. These are:

- Circulating water
- Turbine service water
- Turbine cooling water
- HVAC normal cooling water
- Fire water
- Make up water (condensate)
- Reactor feedwater

The flood concern in the turbine building is filling up the condenser pit until it overflows, at which point water has the potential to enter the service building. Of all the turbine building water sources, only the circulating water or turbine service water (both unlimited sources) are capable of flooding the turbine building and causing a flood concern.

If either the circulating or turbine service water systems were to develop a leak and flood the turbine building, several features exist to mitigate the consequences of the flood. There are three circulating water pumps and two turbine service water pumps supplying water to the turbine building. Each pump has an associated motor operated isolation valve. The condenser pit has redundant water level sensors arranged in a two-out-of-four logic. If flooding were to occur, the level sensors would alert the control room operator, trip the CWS pumps and close CWS valves. For breaks in the TSW system, adequate time (greater than several hours) is available for operator action to trip pumps, or close isolation valves, or open the truck entrance door.

If, for some reason, failure of one or more pumps or valves allowed the water level to rise up to the top of the condenser pit and start flooding the grade level, the following additional flood protection features can mitigate the flood. At one end of the turbine building there is a non-watertight truck entrance door. Water would be expected to flow under and around this door and exit on the ground outside the turbine building. In addition, between the turbine building and service building is a normally closed and alarmed door. It is opened for passage and immediately closed. If the door is not closed immediately, the operator in the control room and plant security would be notified and a security guard sent to investigate. This door has an 20.32-cm (8-inch) step up from the turbine building to divert water away from the service building.

From the above, it can be concluded that the ABWR contains adequate mitigating features such that flooding in the turbine building would not prevent safe shutdown of the plant.

19R.4.4 Control Building

The control building contains safety-related equipment that could be used to achieve safe shutdown. Potential flooding of the control building could thus negatively impact the plants ability to reach and maintain safe shutdown.

Of the five sources of water in the control building listed in Table 19R-1, three of them (RCW, HNCW, and HECW) are relatively small and available room volumes and floor drains are adequate to mitigate flooding from these sources. Fire water is located on all floors (typically in the corridors not in equipment rooms). The fire water system flowrate is low and the system contains a flow alarm to alert the operator to a potential flooding condition. Adequate time would be available to locate and isolate fire water system leaks before any safety-related equipment would be damaged.

The main flooding concern in the control building are potential leaks in the reactor service water (RSW) system which is an unlimited source. Leaks in the RSW system could cause flooding damage to the reactor building cooling water (RCW) pump motors which are located on the bottom floor (i.e., -13150 mm level). The three divisions of RCW are physically separated in rooms with watertight doors and each RCW/RSW room is equipped with a sump pump.

All floors above the bottom floor contain divisionally separated floor drains. These floor drains direct any water in the rooms to the bottom floor where sump pumps are located. Safety-related equipment located in upper floor rooms (e.g., electrical control panels, emergency HVAC, divisional batteries) will be protected from flooding damage by equipment installed 20.32 cm (8 inches) from the floor and floor drains which will divert accumulated water to the bottom floor. Figure 19R-1 shows an elevation view of the control building.

The RCW/RSW rooms contain two sets of diverse safety grade level sensors in a two out of four logic. The first set is located at 0.4 meters from the floor and is intended to alert the control room operator to investigate for the presence of water in the RCW/RSW rooms. The second set of sensors are located at 1.5 meters and informs the control room operators that a serious condition exists that needs immediate attention. In addition, the upper level sensors trip the RSW pumps and close motor operated isolation valves in the RSW system of the affected division.

Anti-siphon capability (e.g., vacuum breakers, air breaks) is included to prevent continued flooding in the event that the RSW pump is tripped but the isolation valves do not close. Figure 19R-2 depicts the RSW system. Given that the pumps have tripped, actuation of the anti-siphon capability will terminate the flood. The ABWR UHS cannot gravity drain into the control building.

From the above, it is concluded that the only flooding concern in the control building is a leak in the RSW system that threatens the RCW system motors in the RCW/RSW rooms. If the upper level sensor alarms, it is a clear indication of a major RSW system leak in the RCW/RSW room.

The following assumptions are used in this “worst case” control building flood:

(1) The ultimate heat sink (UHS) is at an elevation higher than the control building RCW/RSW rooms such that siphoning of UHS water through the RSW system to the RCW/RSW rooms is possible.

(2) There is a maximum of 4000 meters of pipe (2000 each for supply and return) between the UHS and the RCW/RSW room which can be discharged to the RCW/RSW room following RSW pump trip.

(3) The size of the RSW crack is about 103 cm^2 (16 in^2) per ANSI/ANS-58.2 and BTP MEB 3-1.

(4) The leak occurs in the RCW/RSW room.

(5) No operator action was assumed.

The results of this “worst case” control building flood are:

(1) A leak occurs in the RCW/RSW room with the RSW pump running and the lower level sensor alarms at 0.4 meters.

(2) The water level continues to rise and reaches the high level sensor. The RSW pumps in the leaking division are tripped at 1.5 meters.

(3) Water flows into the RCW/RSW room from the 4000 meters of RSW pipe outside the control building.

(4) No water leaves the flooded room and only one division of RCW is affected.

From the above, it is concluded that there are no flooding concerns in the control building because most sources of water are either not large enough or leak at small enough rates that no equipment damage could reasonably occur. The only potential water source of concern is the RSW system but automatic isolation of a potential leak would occur and only one division of RCW would be affected. The reactor could be brought to safe shutdown using equipment from the other two divisions.

19R.4.5 Reactor Building

The reactor building flooding analysis will be presented in two parts: flooding inside and outside secondary containment. For both cases flooding concerns on each floor will be discussed as appropriate.

From Table 19R-1 it can be seen that there are several sources of water in the reactor building. Of these sources, fire water is the main concern (i.e., capable of damaging electrical equipment) on all floors outside secondary containment and on floors B1F and above inside secondary containment. Inside secondary containment on floors B3F and B2F potential flooding from breaks in the suppression pool and condensate make up lines are the major flood concerns. The other water sources are smaller and thus are of lesser concern. The following flooding discussion will thus focus on potential flooding damage from suppression pool and fire water breaks. Potential breaks in the emergency diesel generators fuel and lube oil systems will be treated separately. Table 19R-2 summarizes the reactor building flood sources and safety-related equipment for each floor.

Inside Secondary Containment

Floor B3F

This is the lowest floor in the reactor building and is entirely within secondary containment. The equipment that could be damaged by potential flooding are ECCS equipment (e.g., HPCF, RCIC, RHR), the CRD hydraulic control units, and the CRD pumps. (Figure 19R-3).

Flooding on this floor could occur from breaks in lines attached to the suppression pool or condensate storage tank (CST) and result in water accumulation in one of the three ECCS divisional rooms or within the divisional corridors. The ECCS rooms each contain watertight doors and have individual sump pumps. Flooding inside these rooms would result in loss of the ECCS function for that division. Suppression pool flooding in an ECCS room will reach an equilibrium level below the ceiling of each ECCS room. The watertight doors open into the corridor so that some water in the ECCS room may leak past the door seal into the corridor.

In the divisional corridor, maximum flooding could occur from line leaks associated with the suppression pool cleanup system. In addition, breaks in fire water standpipes on B3F or other floors inside secondary containment would accumulate in the B3F corridor. This is because all inside secondary containment floor drain lines are routed to the B3F corridor. The corridor volume is large enough to contain all of the water from the suppression pool or CST that could enter the corridor. The ECCS watertight doors would prevent any damage to ECCS equipment. The corridor sump pump alarms would alert the operator to flooding in these areas.

Flooding could also occur in the HPCF and RCIC rooms due to a leak in the line to the CST. The CST volume is less than the suppression pool but since it is located at a higher elevation, more water could potentially enter the reactor building (i.e., flood volume not limited by water level equilibrium conditions). The operator could close the CST isolation valve from the control room based on ECCS room sump pump operation and indication of a decreasing water level in the CST. It is expected that the operator would close the CST isolation valve before the low CST level was reached. If not, the ECCS system is designed to automatically shift HPCF or RCIC suction to the suppression pool. In this case, the normally closed suppression pool suction valve would be under water and not expected to open. Even if the suction valve were to open, suppression pool water would fill the ECCS room and flow on to the floor of B2F where it would return to the B3F corridor of the same division via floor drains. The volume of the ECCS Room and the divisional corridor are sufficient to contain the flood water from both the CST and suppression pool.

Floor B2F

Inside secondary containment potential flooding on this floor could occur from the same sources as on B3F (i.e., suppression pool and fire water). A leak in the ECCS chase in each of the divisional valve rooms would cause water to flow down floor drains to the ECCS divisional room on B3F and be processed as previously discussed. Flooding in other areas would be routed through floor drains to the divisional corridor in B3F (Figure 19R-4).

Floors B1F-4F

All inside secondary containment flooding sources on floors B1F-4F would be routed through floor drains to the corridor of B3F and mitigated as discussed above for flooding on B3F.

CUW Line Breaks

The effects of an unisolated reactor water cleanup (CUW) break were analyzed to determine the potential impact on ECCS equipment. The specific effects considered were the possibility of a CUW break rupturing an ECCS wall due to pressure, and the possibility of a CUW break flooding an ECCS room.

The analysis was based on the ABWR secondary subcompartment pressurization analysis (SSPA) model, which postulates a break in each subcompartment through which a CUW high energy line passes. For each postulated break, pressure and temperature transients for each subcompartment were determined using the same methodology as used for compartment pressurization analyses reported in SSAR Subsection 6.2.3.

Since there are no common walls between ECCS and CUW quadrants, the pressure in the El(-)8200 mm corridor is the only CUW break source that could rupture an ECCS wall. The worst-case CUW break was determined from the SSPA to be a 200 mm double-ended break in the El(-)8200 mm pump rooms. The worst-case break in this analysis was defined as the break which will result in the highest pressure in the El(-)8200 mm Division B corridor. The entire corridor at elevation (-)8200 mm is modeled as one volume, assuming that divisional separation doors (at this elevation) are open and remain open during the high energy line break events (Figure 19R-3).

Break flow is comprised of flow from the reactor side (upstream of break) and from the balance-of-system (BOS) side (downstream of break to check valve). Reactor side break flow is modeled in two distinct phases: a period of unsteady flow called the inventory depletion period followed by steady, critical flow choked at flow venturi FE-001 (Figure 5.4-12, Sheet 1 of 4) inside the primary containment. BOS break flow consists of inventory depletion period flow only since check valves isolate this side of the break from feedwater, the downstream pressure source. The analysis conservatively assumes the complete BOS volume of water, including heat exchangers and filter-demineralizers, will flow out of the break. Steady critical flow is calculated using the Moody Homogenous Equilibrium Critical Flow model.

Analysis results showed that the maximum pressure and temperature values for the EL(-)8200 mm corridor during the worst-case CUW break are 0.028 MPaG and 381 K (107.9°C) respectively. These values are below the design pressure and temperature conditions (Tables 6.2-3 and 3I.3-15).

The volume of water released from the worst-case CUW break was determined to be 439 cubic meters, based on the density of water at 381 K (107.9°C). This calculation also assumed that the operator depressurizes the reactor 30 minutes after the break terminating the flood. Assuming conservatively that the Division B corridor contained all the released water. This volume of water will fill the corridor to a level of approximately 1.4 m. The ECCS watertight doors will ensure that no water enters any of the ECCS rooms. If the break were to occur during shutdown, administrative procedures ensure that at least one ECCS division will be available.

In view of the above, a CUW line break is not expected to cause failure of any ECCS walls. Also, the flood volume will be contained within the corridor.

Outside Secondary Containment

Flooding sources outside secondary containment are dominated by fire water leaks. Areas outside secondary containment start at level B1F (i.e., B3F and B2F are entirely within secondary containment.) B1F contains two sump pumps, one each in the Division B and C areas. Floors 1F-4F contain floor drains which all terminate on floor B1F.

The following discussion addresses specific flooding concerns on each floor outside secondary containment.

B1F

B1F contains the three emergency electric rooms. Flood damage in these rooms would affect power supplies to safety-related equipment. The major flooding source is fire water contained in standpipes in Division B and C areas outside the electrical rooms and in the clean access path near the entrance to all three divisional areas (Figure 19R-5). There is no water source in the Division A area or in the emergency electrical rooms of Division B or C.

Fire water breaks in Division B or C would be mitigated by the sump pumps in each area in conjunction with operation of sump overfill lines if necessary. The sump overfill lines are installed to mitigate the result of sump pump failures or flood rates in excess of sump pump capability [i.e. $0.0091 m^3/s$ (150 GPM)]. The sump overfill feature is a drain line through the B1F floor to the corridor of floor B3F inside secondary containment. The line contains a water filled loop seal which acts as the secondary containment boundary. If the water level were to rise above the sump, it would enter the overfill line and be directed to the B3F corridor. The flood would then be mitigated as previously discussed.

For a fire water break in the clean access area (i.e., outside entrance to all three divisional areas), the water could flow under the Division B and C fire doors and enter the sumps in those areas or, if necessary, the sump overfill lines as discussed above. The Division A room does not contain a sump but all equipment is mounted 20.32 cm (8 inches) off the floor and flood levels will not damage the equipment.

1F (Grade)

Fire Water flooding on this floor would be routed by floor drains to B1F and mitigated as discussed above. The emergency diesel generator (EDG) rooms could be flooded by fuel or lubricating oil. In order to preclude the possibility of oil plugging the other floor drains, the EDG rooms can contain any potential oil spills until a portable pump can be used to remove the oil. Figure 19R-6 is a schematic of floor if showing equipment that could be damaged by floods.

2F-4F

Flooding on these floors would be routed by floor drains either to B1F or the EDG rooms (for fuel oil or lubricating oil leaks) and mitigated as discussed above.

Summary of Reactor Building Deterministic Flooding Analysis

Flooding in the reactor building can be mitigated for all postulated flood sources by the following features:

(1) Inside secondary containment-floor drains in all floors above the first floor direct potential flood water to either the non-divisional corridor or, for ECCS line breaks, into the ECCS room on the first (B3F) floor. For flooding outside the ECCS rooms in the non-divisional corridor, watertight doors on each ECCS room prevents water from damaging electrical equipment. The corridor volume is large enough to contain any postulated water source. Flooding inside an ECCS room would only damage electrical equipment in the affected division. For large flooding sources leaking into an ECCS room, the water would eventually leak out into the corridor either:

   (a) Past the watertight door seals (since the doors open into the corridor)

   (b) Continue flooding up to the valve room on floor B2F and then flow under the fire door and down the floor drain to the B3F corridor.

(2) Outside secondary containment-floor drains in all floors above B1F route potential flood sources either to the sumps on B1F or, in the case of EDG oil leaks, to the EDG room. The sump pumps would deliver the water to the plant draining system. If the sump pumps fail or if flooding exceeds sump pump capacity [0.0091 m^3/s (150 GPM) per pump], sump overfill lines route the water to the B3F corridor inside secondary containment. The corridor can safely contain the flood waters. EDG oil flooding is contained within the EDG room for later removal with a portable pump.

From the above, it is concluded that all postulated internal flooding can be mitigated by the ABWR design. No more than one safety division of electrical equipment would be affected and the plant would be able to achieve safe shutdown using either of the two remaining safety divisions or other features (e.g., feedwater, condensate, AC independent water addition system).

19R.5 Probabilistic Flood Assessment

19R.5.1 Introduction

The objective of the ABWR internal probabilistic flood analysis is to identify and provide a quantitative assessment of the core damage frequency due to internal flood events. Internal floods may be caused from large leaks due to rupture or cracking of pipes, piping components, or water containers such as storage tanks. Other possible flooding causes are the operation of fire protection equipment and human errors during maintenance. The spraying or dripping of water from high energy pipe breaks or fire protection equipment onto safety equipment are also considered in the analysis.

The internal flooding event may contribute to core damage frequency by:

(1) Initiating an accident sequence which in combination with the probability of random failure events could lead to core damage, and / or

(2) Disabling safety equipment required to achieve plant safe shutdown.

Therefore, both types of contributions are identified in the evaluation of internal flooding.

19R.5.2 Methodology

Event tree analysis is used to estimate core damage frequency due to internal flooding. The information developed in the deterministic phase of internal flooding analysis is used to construct event trees. Each node in an event tree diagram is dependent on the occurrence of previous events. Therefore, the event tree approach allows the dependence among the flooding initiating event and the success or failure of flooding detection, mitigation, and safe shutdown operation events to be combined properly. Thus, the probability of a specific flooding sequence will be the product of all system failure probabilities in the sequence.

In the ABWR probabilistic flooding analysis, one or more event trees are constructed for each building of flooding concern which is identified in the deterministic flood analysis (i.e., turbine, control, and reactor buildings). Floods in the remaining buildings were eliminated from the study based on a screening analysis. Each node in the event tree represents a different stage in the flooding progression. The first stage depicts the flooding initiating event. The existing nuclear power plant operating data and the data in the Hatch internal flood analysis are used to assess the ABWR flooding initiation frequency (Table 19R-3) for each building of concern. The subsequent nodes represent the success or failure of flooding detection and mitigation features. The data in Table 19R-4 is used to evaluate these events. The final node represents the plant safe shutdown operation. Success or failure at this point may lead either to a safe shutdown or to a core damage event. The probability of failure of this event is dependent upon the availability of systems (which survive internal flooding events). The conditional core damage given failure of the specific mitigating systems is obtained from the ABWR full power PRA. Table 19R-5 lists the conditional probabilities for failure to safely shut down the plant given the loss of various ECCS and BOP systems. The table also explains the basis for each conditional probability.

Since the internal flooding contribution to the ABWR core damage frequency is expected to be very small, a bounding analysis approach is adapted in this study to simplify the computation. The following assumptions are used to construct and quantify the event trees:

(1) Any flooding event in a given building is assumed to be the worst case flood possible (i.e., a double ended shear of the largest pipe). This is a very conservative assumption because, in general, most floods result in leaks, not double ended breaks.

(2) When a flooding event progresses to fail equipment in a safety division, the complete division is assumed to have failed.

(3) Given the failure of a safety division, there is a conditional probability that plant shutdown using the other two divisions may result in core damage. This conditional probability of core damage has been evaluated using the ABWR full power PRA model. It has been determined from that PRA that the probability of core damage following loss of division 2 or 3 is equal and also greater than that for the loss of division 1. Conservatively, the division 2 (or 3) core damage probability was used in the study irrespective of which division was damaged by the postulated flood. Other conditional core damage probabilities (e.g., turbine trip without bypass) were also taken from the full power PRA.

(4) When the plant is shutdown, at least one division of equipment will be administratively controlled to ensure that all systems are available (i.e., not in maintenance). This is in addition to the operating division. (See Subsection 19Q.7 for a discussion of the ABWR shutdown maintenance recommendations). Flooding in the intact division will be contained and will not affect the operating division and flooding in other divisions will not affect the intact division due to the presence of watertight doors and other flood barriers.

19R.5.3 Turbine Building

The turbine building does not contain any safety-related equipment. But the flooding of the turbine building can initiate a reactor trip and may impact the safe shutdown of the plant if the water reaches the control building through the service building access tunnel. There are several water sources listed in Table 19R-1 that may leak into the turbine building. Only the two unlimited water sources (circulating water and turbine service water) are capable of flooding the turbine building and threatening safety equipment in the control building.

The circulating water system (CWS) has three pumps and each pump has an associated motor operated isolation (shutoff) valve. The turbine service water (TSW) system has three pumps and three motor operated isolation valves. For a high power cycle heat sink plant design (i.e., the heat sink is at an elevation higher than grade level of the turbine building), an additional isolation valve is installed in each line. All of these are classified as non-safety grade equipment. If a large pipe break develops either in the CWS or TSW piping and initiates flooding in the turbine building, it is necessary either to trip all of the pumps (for a low heat sink) or to close all of the valves of the associated system to terminate the flood. Four redundant safety grade water level sensors (operating in a two-out-of-four logic) in the condenser pit of the turbine building will generate a signal to alert the control room operator and trip all pumps and close all isolation valves in the CWS. TSW breaks must be manually mitigated but, due to the lower flow rate (Compared to CWS), sufficient time is available to trip the pumps or close isolation valves from the control room. A turbine trip and reactor shutdown will be initiated as a consequence of turbine building flooding.

If one or more pumps fail to trip or its associated valve fails to close, the water level may rise up to the top of the condenser pit and reach grade level. If the operator received an alarm from the level sensors, even though the automatic protective features failed, the operator could open the truck entrance door (roll up type door) to allow the flood water to exit the building. If the operator does not receive an alarm, it is assumed that insufficient time will be available for the operator to open the truck door for a CWS break before the water level would effectively cause binding of the door and prevent opening. For TSW breaks, greater than 2 hours is available to open the door.

Even if the door could not be opened, leakage past the door could be sufficient to keep the flood level below the bottom of the door entering into the service building. There is a 20.32-cm (8-inch) step up from the turbine building to the service building door. If the flood level were to increase above 20.32 cm (8 inches), the service building door is a normally closed alarmed door that will offer resistance to flooding. If the door remains closed (it opens into the service building), the flood rate into the service building would be low enough that personnel in the service building would discover the flooding. There would be sufficient time to mitigate the flood before any damage to safety-related equipment could occur because the service building must flood before water could start to enter either the reactor or control buildings.

If the service building door fails open, the flood rate into the service building could be high enough to flood the service building to a significant level. Since the service building is the main entrance to the plant, personnel would hear or see the flood water and alert operators in the control room. Operator action could then be taken to manually trip the CWS or TSW pumps or close CWS or TSW valves. This is assuming that the level sensors failed but control circuitry for pump trip / valve isolation was still available.

If these actions failed, the flood waters would fill up the service building and could potentially enter the control or reactor buildings through several external normally closed watertight doors. On the first floor of the service building there is a watertight door which allows entrance to the reactor building cooling water (RCW) heat exchanger rooms. Failure of this door could allow the flood waters to damage equipment in all three safety divisions and potentially the battery room on the next level. If the watertight door to the RCW rooms does not fail, the water level would rise up in the service building to the next level where there are two watertight doors, one to the battery rooms of the control building and another to the reactor building clean access area. Failure of the watertight door to the battery rooms is assumed to result in core damage as loss of all DC (batteries and battery chargers) will occur. DC power is required for control of safe shutdown systems or to depressurize and use non-safety-related makeup sources such as condensate or AC independent water addition systems. Failure of the watertight door to the reactor building clean access area could result in damage to all three electrical divisions. If none of these watertight doors fail, flooding could continue to the next level where a normally open watertight door allows access to the control room area. Given the extensive flooding which had occurred to this point, the operators would have sufficient time and warning to close this watertight door. If the door failed or the operators failed to close it, no core damage should occur because automatic initiation of safety systems such as the high pressure core flooder would ensure that the core remained covered with water. Continued flooding would then reach grade level where the water could exit the service building through the main entrance. It is assumed that failure of any of the external watertight doors (except the control room door) results in core damage.

Figures 19R-7 and 19R-8 are event trees which describe the turbine building flooding for low and high Power Cycle Heat Sink (PCHS) configurations, respectively. The accident progression due to a large pipe break in the CWS (the worst case flooding) is described in the event tree. As the CWS break is bounding, no TSW flooding event trees were developed. The success or failure of each flood mitigating feature in the event tree diagram may have a significant impact on the result of accident progression. The event trees in Figures 19R-7 and 19R-8 are described as follows:

(1) A large CWS pipe break occurs in the turbine building (flooding initiator).

(2) Four redundant safety grade water level sensors (operating in two-out-of-four-logic) in the condenser pit of the turbine building detect and alert control room operators about flooding (detection).

(3) The bus breaker and / or pump breakers of CWS pumps open and trip all three pumps (flooding prevention for low PCHS). Although siphoning could occur if the PCHS was higher than the bottom of the condenser pit, the siphon could not cause flooding to grade level. Therefore, the flood would be contained within the turbine building. In case of high PCHS, the success probability of this feature is assumed to be zero.

(4) CWS isolation valves close (flooding prevention for high or low PCHS).

(5) If the water level sensors alerted the operator to the flooding condition but the automatic flood protection features failed (CWS pump trip and valve isolation), time may be available for an operator to open the roll up truck

entrance door to ensure that flood waters would exit the turbine building to the ground outside. If the water level sensors failed, it is assumed that by the time the operator becomes aware of the flooding condition that the water level will have reached the truck door and the water pressure against the door will not allow it to be opened.

(6) The roll up truck entrance door is not watertight and it is expected that it will leak if flooding occurs. The door may not fail open but it will buckle and could leak at a rate high enough to keep the flooding level below the level of the service building door [20.32 cm (8 inches)].

(7) The service building door is a normally closed and alarmed security door. It is not watertight but it should give significant resistance to flooding. If the door remains closed, the flood rate into the service building will be low.

(8) The control room operator can prevent flood damage to safety-related equipment by manually tripping the CWS pumps or closing the valves. It is assumed that if automatic features failed (given that the sensors did not fail) that control room actuations would also fail. If the sensors failed though, it may be possible to manually close the valves or trip the pumps from the control room once the operator is aware of the flooding condition. The probability of success is higher if the sensors did not fail because the operator would receive two indications of flooding: early in the scenario from the sensors in the turbine building and later from personnel in the service building if the flood were to propagate to that point. In either case, the watertight doors in the control and reactor buildings can prevent damage to safety-related equipment.

(9) Once the flood is terminated, the plant is manually shutdown using equipment not damaged by the flood. Failure to terminate the flood and any external watertight door failure is assumed to result in core damage.

The description of flooding for a high PCHS is the same as for a low PCHS except that the pump tripping feature is not credited.

The core damage frequency for turbine building flooding is extremely small for a low PCHS and slightly higher for a high PCHS.

19R.5.4 Control Building

The control building contains safety-related equipment and the potential flooding of the control building could impact the ability of the reactor to shutdown. The major flooding source in the control building is reactor service water (RSW) which is used to remove the heat from the RCW heat exchangers. The control building could potentially be flooded by the RSW system which is an unlimited water source. Unisolated breaks in the fire water system could cause inter-divisional flooding since doors separating safety divisions do not have sills. The control building has six floors (Figure 19R-1) but floor drains and stairwells would direct all potential flood waters which could potentially impact safe shutdown equipment to the bottom floor (-8,200 mm level).

19R.5.4.1 RSW Line Breaks

The RSW system is the only unlimited water source that could cause substantial flooding in the control building (Table 19R-1). It is highly unlikely that RSW flooding could damage more than one safety division. But the occurrence of several unlikely random failures and operator errors could result in flooding damage to equipment in all three RCW divisions.

The safety-related RCW motors are located on the -8,200 mm elevation (the lowest level of the control building) in three RSW/RCW rooms which are physically separated from each other by concrete walls and watertight doors. Each RSW/RCW room is also equipped with a sump pump.

Each of the three RSW divisions has two safety grade pumps, safety grade motor-operated isolation (shutoff) valves, and anti-siphon capability (e.g., vacuum breaker) (Figure 19R-2). During normal operation, one pump in each division is operating and the other pump is in standby. If a large leak or a pipe break develops in any one of the RSW/RCW rooms, tripping the pump and closing the associated valves in the affected division will stop the flooding. If the RSW pump trips but isolation valves fail to close, then the anti-siphon capability prevent continued flooding. Four redundant safety grade water level sensors (operating in a two-out-of-four logic) at the lower level (0.4 meter) of the control building will generate a signal to alert the control room operator. If the control room operator fails to take appropriate action to stop the water flow, the second set of level sensors will actuate when the water reaches the 1.5 meter level of the room. At this level, the sensors (operating in two-out-of-four logic) not only send an alarm signal to the operator but also trip the affected RSW pump and close all the isolation valves. The upper level sensors are diverse from the lower level sensors.

It is assumed that only one division of RCW is lost if the affected RSW pump trips or the isolation valves close. In case the level sensors fail to detect the flood, the water will rise to the second floor level and may start flowing into the other two remaining divisional RCW/RSW rooms. The level sensors in these two divisional rooms will generate a signal to alert the operator about the flood. If the sensors in the first division failed, the sensors in the other divisions are assumed to fail with a high probability to account for common cause failures (CCF). Only one division is assumed lost if the operator is successful in isolating the flood, otherwise the loss of all three safety divisions is possible.

Flooding in the RSW pump house was not addressed because the ultimate heat sink including the RSW pump house is outside the scope of GE supply. The COL applicant must complete a plant specific probabilistic analysis of flooding in the RSW pump house.

The event tree in Figure 19R-9 describes the accident progression for a control building RSW flood. A large pipe break in the RSW in the RSW/RCW heat exchanger room is considered to be the worst case flooding in the control building. The description of events shown in Figure 19R-9 follows:

(1) A large RSW pipe break occurs in the RCW/RSW room in the control building (flooding initiator).

(2) Four redundant safety grade water level sensors located at the 0.4 m level detect and alert the control room operator about flooding (detection).

(3) The operator investigates the presence of water and isolates the flooding by tripping the affected pump and/or closing the isolation valve (flooding prevention).

(4) If the first level of detection fails or the operator fails to isolate the flowing water, then water continues rising in the room and the second set of diverse sensors located at 1.5 meters detects the water and trips the affected pump and closes all motor operated valves in the RSW system. Meanwhile the signal alerts the control room operator of the flooding condition (flooding prevention).

(5) If the operator is successful in isolating the flooding, one safety division is assumed lost, otherwise the loss of all three safety divisions may occur (flooding mitigation).

(6) The pump breaker of the affected RSW pump opens to trip the pump and/or the isolation valves close automatically (flooding isolation).

(7) In the unlikely event that the flood is not mitigated by automatic means or operator action, the water rises to the second floor level and starts flowing into the other two remaining RCW/RSW rooms. The first set of level sensors in these two divisional rooms detects the water and alerts the operator the third time (flooding detection). This operator action is considered separately from the previous high water alarm action because the alarm would occur approximately 45 minutes later and be annunciated as occurring from a different division.

(8) Reactor safe shutdown using available equipment (reactor shutdown).

The core damage probability for an RSW flood is estimated to be extremely small.

19R.5.4.2 Fire Water System Breaks

The ABWR fire water system is a moderate energy system that is designed to withstand a 0.3g seismic event. The system is very rugged and large breaks of the eight inch header piping are not expected. The most probable failure mechanism for the fire water piping would be a crack which would not propagate to a large break because of the low pressure of the system (approximately 0.69 MPa). In keeping with the bounding analysis methodology used in other parts of the flooding PRA, a large break (0.086 m^3/s) will be assumed. The frequency of this bounding case large break of the fire water piping is small. This value was obtained from a review of the Limerick Generating Station flooding PRA for a fire suppression system pipe double ended shear.

Figure 19R-10 is the event tree for fire water system flooding in the Control Building. The system unavailabilities are taken from the ABWR full power PRA and the operator failure probabilities are based on methods used in Chapter 10 of Swain and Guttman given the many sources of information available indicating a fire water system break and the simple action of stopping the fire water pumps.

Fire water standpipes are located on all floors of the Control Building. A large break on any upper floor will result in a 0.086 m^3/s flood which will be directed by the floor drain system to the RCW rooms on the first floor.

The ABWR does not contain sills on doors between safety divisions and fire doors can have up to a 1.9 cm gap at the bottom per National Fire Protection Association (NFPA 80) requirements. The floor drain system, although not finalized yet, will be designed so that this break flow can be accommodated taking into account all the drain lines in the three safety divisions. In addition, water may flow under the fire doors and down stairwells and elevator shafts. Due to the available drainage sources, the water level on any upper floor will not exceed 20.32 cm which is the minimum height that all water sensitive equipment must be mounted from the floor (ABWR Tier 2 Section 3.4). Therefore, no damage to equipment on any of the upper floors will occur due to emersion in the flood water. Spray onto safety-related equipment is not a concern because all fire water system flow will be directed to the three RCW rooms on the first floor.

Following a break in the fire water piping, the operator will receive indication that the fire water system pump(s) have started due to low system header pressure. Within a few minutes, he will also receive indication of excessive sump pump operation in the Control Building. As no fire alarm will accompany the fire water system actuation, the operator would send someone to confirm the nonexistence of a fire in the Control Building. Water flowing past hose stations causes actuation of audible alarms in each hose station. These alarms will alert personnel to actuation of the fire suppression system and help direct them to the break location. Once it has been determined that no fire exists, an operator can trip the fire pumps locally and close manual valves, if necessary, to terminate the flood. It is estimated that it will take a minimum of 30 minutes to isolate the flood (i.e., no credit for operator action within 30 minutes).

Based on the size of the three RCW rooms and the maximum fire water flow rate, it will take over one hour to flood the three RCW rooms up to the bottom of the RCW motors (minimum of 400 mm from the floor). If the flood is not terminated in approximately one hour, all three divisions of RCW are assumed lost. This assumes a double ended shear of the fire water piping. If a design basis crack (ANSI/ANS 56.11) were to occur (the more probable occurrence) it would take approximately 10 hours for water to accumulate to a level of 400 mm.

Given that three divisions of ECCS are lost due to the loss of RCW, the plant must then be manually shutdown per Technical Specifications using feedwater/condensate and the main condenser. After isolation of the fire water system flood, fire water would also be available for make up if required. RCIC does not require direct cooling by RCW, so it would be available for make up during this event.

The CDF for fire water flooding in the control Building is extremely small.

19R.5.5 Reactor Building

The reactor building contains safety-related equipment and the potential flooding of the reactor building could impact the safe shutdown of the plant. From the flooding stand point, the reactor building is divided into two parts:

(1) Inside secondary containment

(2) Outside secondary containment

19R.5.5.1 Flooding Inside Secondary Containment

The major flooding sources inside of the secondary containment are the suppression pool, condensate makeup and fire water. The rest of the potential flooding sources are listed in Table 19R-1. The lowest floor of the reactor building (B3F) is entirely within the secondary containment. The safety-related equipment that can be damaged by potential flooding are in the three divisional ECCS rooms. Each of these rooms have watertight doors and individual sump pumps. The flooding inside these rooms would result in loss of ECCS equipment (e.g., HPCF, RCIC, RHR).

Flooding on this floor (B3F) could occur due to a pipe break attached to the suppression pool and result in water accumulation in one of the three ECCS divisional rooms or within the non-divisional corridor. The ECCS rooms are large enough to contain the water that could enter from a suppression line break (described in Subsection 19R.4.5). The watertight door of the affected ECCS room opens into the corridor and the corridor sump pump alarms alert the operator to the flooding in this area. The watertight door of the unaffected ECCS room prevents the water in the corridor from entering the other ECCS rooms. Therefore, flooding on this floor can not impact more than one division due to:

(1) The corridor volume is large enough to contain the largest flood source (suppression pool)

(2) The watertight doors prevent the water from entering the unaffected ECCS rooms. Common cause failure of the watertight doors is addressed.

At higher elevations, the potential flooding due to the largest water source (fire water) is directed to the B3F corridor (the lowest level) by floor drains and stairwells. The corridor volume is large enough to contain any one of the upper level water sources.

The flooding sources outside secondary containment are dominated by fire water leaks. Areas outside secondary containment start at level B1F. This level contains two sump pumps, one each in the Division B and C areas. Floors 1F through 4F contain floor drains which all terminate on floor B1F. Floor drains in all floors above B1F route potential flood sources to the sumps on B1F. The sump pumps would deliver the flooding water to the plant drainage system. If the sump pumps fail or if flooding exceeds sump pump capacity, sump over-fill lines route the water to the B3F corridor inside secondary containment. The corridor can safely contain the flood waters. In the case of emergency diesel generator (EDG) oil leaks, the EDG room can contain the spill until portable pumps can be brought in to remove the oil. As described in detail in Subsection 19R.4.5, the flooding outside secondary containment can not impact more than one safety division.

The worst case reactor building flooding could potentially occur in the HPCF or RCIC rooms (inside secondary containment) due to a leak in the line to the condensate storage tank (CST). If the operator fails to close the CST isolation valve (in spite of sump pump alarm and the indication of CST water level decreasing), the suction line of the affected ECCS system automatically realigns to the suppression pool on low CST level. In this case, the normally closed suppression pool suction valve would be under water and not expected to open. Even if the suction valve were to open, the suppression pool water would fill the ECCS room and flow to the floor of B2F where it would return to the B3F corridor via floor drains. The volume of the ECCS room and the divisional corridor are sufficient to contain the flood water from the CST or the suppression pool.

If a leak were to occur during shutdown, some of the ECCS rooms may be open for maintenance. ABWR procedures specify that one safety division will be maintained intact at all times during shutdown. If a leak were to occur in the intact division, the operator would be directed to close the ECCS door to an unaffected division before attempting to mitigate the flood in the protected division. The ABWR Technical Specifications require at least two ECCS divisions be operable during shutdown except under certain conditions in mode 5 (reactor cavity flooded). Thus, in general, one other safety division would be available to maintain decay heat removal. Also, non-safety grade equipment (e.g., condensate, AC independent water addition system) can be used if necessary for decay heat removal. Appendix 19Q discusses the ABWR decay heat removal reliability during shutdown.

Figures 19R-11, 19R-12, and 19R-13 are the event trees for flooding in the reactor building. The figures describe flooding on floor B3F inside an ECCS room, flooding on B3F in the divisional corridor, and fire water flooding outside secondary containment, respectively. These are the three worst case reactor building floods.

Flooding inside an ECCS room due to a leak in the suppression pool suction line upstream of the isolation valve results in an unisolable suppression pool leak (Figure 19R-11). The first indication of a flood will be actuation of the ECCS sump pump. No operator action to stop the flooding is possible, but the ECCS room volume is large enough to contain the amount of water that will enter the ECCS room before the water levels in the ECCS room and suppression pool reach equilibrium. Equipment in the affected ECCS room would be damaged but the watertight doors would prevent water from damaging the other two safety divisions. Failure of the watertight doors results in loss of all three ECCS rooms and only the AC independent water addition is available for shutting down the plant. The core damage probability for this event is extremely small. As mentioned in Subsection 19R.4.5, a break in a CST line could potentially allow more water to enter the rooms than a suppression pool line break, but the leak could be isolated and the results are the same (i.e., loss of only one ECCS division). Therefore, the core damage probability will be lower than for an unisolable suppression pool line break and no event tree was completed for a CST line break.

Flooding in the non-divisional corridor from a break in the suppression pool line is described in Figure 19R-12. The sumps high water alarm would alert the operators to the presence of a flood. In this case, the flooding can be stopped by appropriate operator action. If the operator fails to stop the flooding, the corridor volume is large enough to contain the amount of water that could enter from the suppression pool. The watertight doors on the ECCS rooms will prevent any damage to safety equipment and the plant can be safely shutdown using the three divisions of safety equipment. The core damage probability for this event is extremely small.

In the Reactor Building, fire water flooding inside secondary containment is bounded by flooding due to postulated breaks in lines from the suppression pool or condensate storage tank (CST). The maximum water level inside secondary containment is approximately the same for breaks in the fire water system, CST, or lines from the suppression pool since they each are capable of supplying the same volume of water but it would take over seven hours to drain the fire water system. By contrast, a break in a line from the suppression pool would result in an equilibrium level in the corridor in approximately one hour and it would take over five hours for the CST to drain. Even if the fire water flood was not isolated within 7 hours, the watertight doors on the ECCS rooms would prevent damage to any safety-related equipment. Therefore, fire water system flooding inside secondary containment will have a very low CDF and since is dominated by breaks in lines from the suppression pool, a separate event tree was not completed.

19R.5.5.2 Flooding Outside Secondary Containment

Flooding outside secondary containment could affect the emergency electric motor control rooms and other equipment for all three safety divisions. The flooding concern is a break in a fire water standpipe. For breaks in the fire water system outside secondary containment, the situation is similar to flooding in the Control Building. Figure 19R-13 is the event tree for a fire water system flood outside secondary containment. Floor B1F is the lowest floor outside secondary containment and floods on all floors above this are directed to B1F via floor drains. Floor B1F contains two sumps outside secondary containment. The overfill lines direct water in excess of the sump pump capacity to the corridor of the first floor (B3F) inside secondary containment. As the sump pump capacity will not handle a full header break, some water will enter the overfill lines and flow into the B3F corridor. The overfill lines may not be able to pass full fire water system flow either, so water will flow under fire doors and may enter the three Class 1E electrical equipment rooms. Equipment in these rooms is raised at 20.32 cm off the floor and it is estimated that will take over one hour for the flood to damage equipment in these rooms. Water could also flow down the stairwells to the first floor but this was conservatively excluded.

The initiating events frequency will be very low and similar to the Control Building since the same pipe and configuration are used and the length of piping is similar. The time for operator action is approximately the same as for the Control Building (one hour) and the effect of not isolating the flood is loss of the three ECCS divisions due to loss of power. RCIC will not be affected by the power loss and thus is assumed to be available for this scenario. As in the case of the Control Building, feedwater/condensate, the main condenser, and fire water (after isolation of the break) will be available. The CDF for fire water system flooding outside secondary containment is extremely small.

19R.6 Results and Interface Requirements

This subsection summarizes the results of the ABWR probabilistic flooding analysis including insights gained from the analysis, important flooding design features, operator actions to prevent/mitigate potential flooding, system reliability goals to ensure validation of probabilistic core damage estimates, and the conclusions reached on the ability of the ABWR to withstand postulated internal flooding.

19R.6.1 Results

The results from the ABWR probabilistic risk analysis are shown in Table 19R-6 for the turbine, control and reactor buildings. This conservative bounding analysis shows that the CDF for internal flooding is very small and is less than the total plant CDF .

19R.6.2 Insights Gained from Analysis

Completion of the ABWR probabilistic flooding analysis has led to the following insights on the flooding mitigation capability of the ABWR:

(1) The ABWR due to its basic layout and safety design features is inherently capable of mitigating potential internal flooding. Safety system redundancy and physical separation for flooding by large water sources along with alternate safe shutdown features in buildings separated from flooding of safety systems give the ABWR significant flooding mitigation capability. Also, fire protection features such as floor and wall penetrations and fire barriers help to contain potential flood sources.

(2) Due to the inherent ABWR flooding capability discussed above, only a small number of flooding specific design features must be relied on to mitigate all potential flood sources. The flood specific features are: watertight doors on control and reactor building entrances, ECCS rooms, and RCW rooms; floor drains in reactor and control building; RSW pump trip, isolation valve closure and actuation of anti-siphon capability on high water level in the RCW rooms; CWS pump trip and valve closure on high water level in the condenser pit; and sump overfill lines on floor B1F of the reactor building.

(3) All postulated floods can be mitigated without taking credit for operation of sump pumps.

(4) While timely operator action can limit potential flood damage, all postulated floods can be adequately mitigated (from a risk perspective) without operator action.

19R.6.3 Important Design Features

Table 19R-7 lists the features of the ABWR design that contribute to its ability to withstand postulated flooding. The list includes general features as well as specific features in each building for identified potential flood sources. Also included is the rationale for how each feature could prevent/mitigate flood damage. The table also identifies the new features that were added as a result of the flooding PRA.

The features that are considered important to mitigate flooding in the ABWR are discussed in Subsection 19.8.5.

19R.6.4 Operator Actions

From a flooding perspective there are several operator actions that, if taken in a timely manner, could mitigate the effects of potential internal flooding. These are:

(1) Isolation of flood sources following detection by sump pump operation and alarms or floor water level detectors. Potential flood sources as listed in Table 19R-1 can, in general, all be isolated by appropriate operator actions.

(2) A leak in the suppression pool line upstream of the HPCF/RCIC suction valve cannot be isolated but operator action to ensure other divisional watertight doors are closed could prevent damage to equipment in more than one safety division.

(3) Open doors or hatches to divert water from safety-related equipment following postulated floods.

(4) Close watertight door at entrance to the control room area if floods in the turbine building result in service building flooding.

(5) Open the truck entrance door in the turbine building to prevent a CWS or TSW break from entering the service building. This will prevent external flooding of the control and reactor buildings from a CWS or TSW line break.

In the PRA, operator action of responding to a flood alarm has been modeled. Floods in the turbine, control and reactor buildings result in alarms in the control room. It is assumed that flood procedures exist and operators are well trained to respond to flooding events. The operator failure probability depends upon the time available for taking action and are conservative values based on engineering judgment. The operator actions are not important in the sense that automatic actions will prevent core damage. However, timely operator action could limit the consequences of flood events.

19R.6.5 Reliability Goals (Input to RAP)

The results of the probabilistic flooding analysis indicate that the following equipment is important to reducing the risk due to internal floods and should be included in the ABWR reliability assurance program:

Equipment	Reliability Goal
Watertight doors	*
Sump level switches	*
Pump trip	*
Isolation valve closure	*
Anti-siphon capability	*

* Not part of DCD (refer to SSAR)

19R.6.6 Conclusions

The conclusions from the ABWR probabilistic flooding analysis is that the risk from internal flooding is acceptably low. The estimated core damage frequency from all internal flood sources is very small for a low PCHS and slightly higher for a high PCHS.

The ABWR is inherently safe regarding internal flood events and no operator actions are required to mitigate postulated floods although timely operator action can reduce damage to equipment and flood severity. All potential floods have been analyzed and it has been shown that the plant can be safely shutdown with low risk to plant personnel and the general public.

Table 19R-1 Sources of Water

Source	Capacity	Flow Rate	Turbine Building	Control Building	Reactor Building	Service Building	Radwaste Building
Reactor Service Water (RSW)	Unlimited	499.67 liters/sec/div. (7,920 GPM/div.)		X			
Turbine Service Water (TSW)	Unlimited	12,618 liters/s/pump (15,000 GPM/Pump) (3 pumps)	X				
Circulating Water (CW)	Unlimited	12,618 liters/s/pump (200,00 GPM/pump) (3 pumps)	X				
Fire Water	1,249,182 liters/tank (330,000 gal/tank) (2 tanks)	9.46 liters/s/2 pumps (150 GPM/2 pumps)	X	X	X	X	X
Reactor Building Cooling Water (RCW)	257,407 liters/div. (68,000 gal/div)	360.87 liters/s (A,B) (5,720 GPM (A,B)) 305.36 liters/s (C) (4,840 GPM (C))		X	X		X
HVAC Normal Cooling Water (HNCW)	113,562 liters (30,000 gal)	106.94 liters/s (1,695 GPM)	X	X	X	X	X
HVAC Emergency Cooling Water (HECW)	113,562 liters (30,000 gal)	7.57 - 13.88 liters/s (120–220 GPM) (Chilled) 21.51 -35.58 liters/s (341–564 GPM) (Condenser)		X	X	X	X
Makeup Water (Condensate)	2,108,468 liters (557,000 gal)	104.10 liters/s (1,650 GPM)	X		X		X
Makeup Water (Purified)	757,080 liters (200,000 gal)	19.43 liters/s (308 GPM)			X		
Turbine Cooling Water (TCW)	378,540 liters (100,000 gal)	1829.61 liters/s (29,000 GPM)	X				

Table 19R-1 Sources of Water (Continued)

Source	Capacity	Flow Rate	Turbine Building	Control Building	Reactor Building	Service Building	Radwaste Building
Feedwater	757,080 liters (200,000 gal)	2119.82 liters/s (33,600 GPM)	X		X		
City Water	Unlimited	12.62 liters/s (200 GPM)				X	
Suppression Pool	3,579,754 liters (945,674 gal)				X		

Table 19R-2 Reactor Building Floor Descriptions

Designation	Elevation (mm) (TMSL)*	Safe Shutdown Equipment	Potential Flooding Sources
100 (B3F)	-8200	HPCF, RHR (LPFL), RCIC	Condensate Storage Tank (CST), Suppression Pool (SP), Fire Water (FW), Reactor Building Cooling Water (RCW), Purified Makeup Water (MP)
200 (B2F)	-1700	Instrument Racks (e.g., RCIC)	CST, SP, FW, RCW, MP
300 (B1F)	4800	IE MCCs, Remote Shutdown Panel	FW, RCW, CST, MP
400 (1F)	12300 (Grade)	EDGs and EDG Control Panel, and Valve Rooms Oil (LO), EDG Fuel Oil (FO)	FW, RCW, CST, MP, EDG Lube
500 (2F)	18100	EDG Control Panels and Cooling Fans	FW, RCW, MP
600 (3F)	23500	EDG Auxiliaries (e.g., Air Compressor, Fuel Oil Tank), Standby Liquid Control (SLC)	FO, FW, RCW, SLC, MP
700 (M4F)	27200	Emergency HVAC	HVAC, FW, RCW,MP
800 (4F)	31700	RCW Surge Tank, EDG Exhaust Fan	RCW, FW, MP

* Typical mean seawater level

Table 19R-3 ABWR Flood Frequency*

Flooding Location (Equivalent ABWR)	Generic Flooding Frequency	NUREG/CR-2300	Hatch IPE	ABWR†
Turbine Building‡	2.6E–02	3.3E–02	2.3E–02	
• Circulating Water		2.8E–02		
• Turbine Service Water		4.9E–03		
Reactor Building	3.8E–02	2.4E–02	3.9E–02	
• Inside Sec. Cont.				
– ECCS Room				
– Corridor				
• Outside Sec. Cont.				
Control Building			1.2E–02	

* Per reactor year
† Not part of DCD (refer to SSAR).
‡ Total for CWS and TSW breaks

Table 19R-4 Reliability Data for ABWR Probabilistic Flood Analysis

Failure Rate Data and Common Cause Factor		
Component, Element	**Failure Mode**	**Failure Rate (per demand except as noted)***
1. Level Sensors	Fail to operate Fail to operate (standby)	
2. Isolation Valve	Fail to close	
3. Motor Driven Pump	Fail to trip pump (Breaker fails to open)	
4. Operator Fails to Act	Available time < 12 min Available time < 30 min Available time < 1 h Available time > 1 h	
5. Common Cause Factor (multiple Greek letter)	Beta Gamma Delta Others	
6. Over Fill Line	Clogged	
7. Sump Pump	Exceeding the design capacity	
8. Anti-siphon Capability	Fail to operate	
9. Watertight Doors	Fail to stay closed Common cause	

* Not part of DCD (refer to SSAR).

Data obtained from the following references:

1. EPRI ALWR Utility Requirement Document
2. "Handbook for Human Reliability Analysis with Emphasis on Nuclear Power Applications", NUREG-CR-1287
3. GE Reliability Data Manual Used for Probabilistic Risk Analysis

Table 19R-5 Conditional Failure Probability of Safe Shutdown

Conditional Event	Failure Probability of Safe Shutdown (per demand)*
1. One ECCS Division Unavailable	
2. Two ECCS Divisions Unavailable	
3. All ECCS Divisions Available	
4. Three ECCS Divisions and Power Conversion System Unavailable	

* Not part of DCD (refer to SSAR).

Table 19R-6 Internal Flooding Core Damage Frequency (CDF)

	CDF (per reactor year)	
Building	Low PCHS*	High PCHS*
Turbine		
Control		
Reactor		
Total		

* Not part of DCD (refer to SSAR).

Table 19R-7 ABWR Features to Prevent/Mitigate Flooding

Feature	Benefit
General Features	
Floor/wall penetrations are sealed.	Restricts water flow from entering or leaving divisional rooms.
Alarmed doors.	Alert operator to loss of room integrity.
Pipe chases.	Contains potential flooding of high pressure systems (e.g., feedwater and main steam in tunnel between reactor and control buildings).
High pressure or high temperature lines not routed through floors or walls separating two different safety divisions.	Reduces potential for high energy pipe break to affect more than one safety division.
Room sump pumps and alarms.	Remove water from rooms to protect equipment and to alert operators to potential flooding conditions.
Floor drains.	Directs water from room to lower floors for removal by sump pumps or other means.
Room water level alarms, pump trips, and isolation valve closures.	Alert operator to potential flooding and isolate flooding source(s).
Fire water flow initiation alarmed in the control room.	Alerts operator to initiation of fire water flow and possible flood concern.
*Motor control centers have NEMA Type 4 enclosures.	Protects MCCs from damage due to water from pipe break or fire hose.
Drip proof motors.	Protects motors from falling liquids.
Turbine Building	
*Four water level sensors in condenser pit in two-out-of-four logic. When actuated they alarm and trip circulating water pumps and close isolation valves. They will also alert the operator to other floods such as TSW.	Isolates flooding source and alerts operator to potential flooding.
Non-watertight truck entrance door.	Expected to allow any internal flood water to leak out of turbine building to the outside and thereby not allow water to reach control or reactor buildings.
Normally closed and alarmed fire door between turbine building and service building tunnel (access to control and reactor buildings).	Restricts potential flood water in turbine building from entering control or reactor buildings via the service building.

Table 19R-7 ABWR Features to Prevent/Mitigate Flooding (Continued)

Feature	Benefit
A 20.32-cm (8-inch) step up to the door between turbine building and service building.	Direct flood water away from service building back into the turbine building thus preventing potential flooding in reactor and control building.
Control Building	
RCW/RSW rooms and entrances to control building from the service building have watertight doors.	Prevent flooding in one division from affecting other divisions and external flooding from the service building due to potential CWS floods in the turbine building.
Floor drains route water to first floor (RCW/RSW rooms).	Protects equipment in rooms from water damage and directs water to sump pumps.
RCW/RSW rooms have sump pumps.	Remove flood water from room to prevent damage to equipment.
RCW/RSW room floor water level sensors alarm at 0.4 meter and trip RSW pumps and close isolation valves at 1.5 meters in affected division.	Alert operator to RCW leak and shutoff RSW supply if flooding were to continue.
*Maximum of 4000 meters of pipe between RSW pump house and the RCW/RSW room (Figure 19R-2).	Limits volume of water which could be drained into RCW/RSW room following RSW pump trip during flooding. This plus high level trip of RSW pump limits maximum RCW room flood level such that only one division of RCW would be affected.
*RSW system anti-siphon capability.	Ensures termination of flood if pump trips but isolation valves do not close.
Reactor Building	
Rooms on floors B2F-4F have floor drains. Inside containment floor drains collect on floor B3F. Outside containment floor drains (except EDG room) collect on floor B1F. EDG room oil leaks are contained in the room.	Potential flood waters routed to rooms with sump pumps for processing by plant waste system. EDG potential oil flooding has relatively small volume and the oil is contained until a portable pump can be brought in to remove the oil.
*Wall penetrations through ECCS rooms on floor B3F must be above specified high water mark or sealed and tested to prevent leakage.	Prevent leakage into ECCS divisional rooms from potential flooding in corridors.
Fire water not routed through ECCS rooms.	Reduces probability of flood from broken fire water lines impacting ECCS equipment.
Equipment in all rooms mounted 20.32 cm off the floor.	Enhance ability to survive potential flooding.

Table 19R-7 ABWR Features to Prevent/Mitigate Flooding (Continued)

Feature	Benefit
High level alarms in room sumps.	Alert operator to potential flooding.
Steamline tunnel pipe chase.	Contains potential flooding from high pressure main steam and feedwater.
ECCS rooms (Floor B3F) have watertight doors. Doors open into corridor.	Prevent flooding in corridor from entering ECCS Rooms. Contains leaks in ECCS room to limit damage to one safety division (small leakage past door seals may occur).
Entrances to the reactor building control room and clean access areas from the service building have watertight doors.	Prevents flooding of these areas due to turbine building CWS leaks that propagate to the service building.
Floor B3F corridor volume adequate to contain single largest flood source (suppression pool).	Largest flood mitigated without need to rely on active components (sump pumps).
*CRD rooms have non-watertight doors. Doors open into corridor.	Restricts water in corridor from entering room. Flooding in CRD room will leak out into corridor.
*Sumps on floor B1F outside secondary containment have overfill lines to B3F corridor.	Ensure adequate mitigation for outside containment flooding if sump pumps fail or flood rate exceeds sump pump capacity.

* New Feature

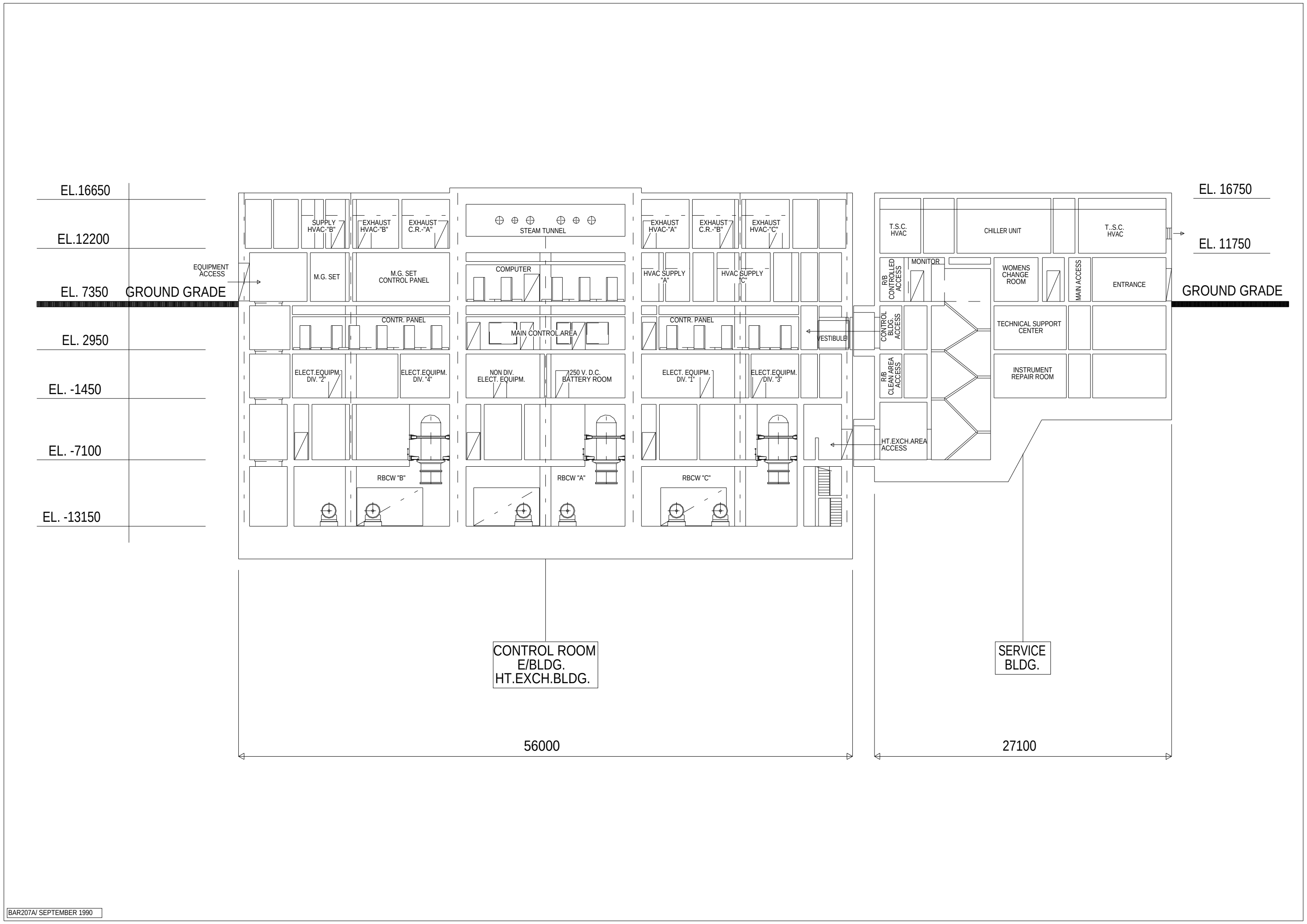


Figure 19R-1 Control Building

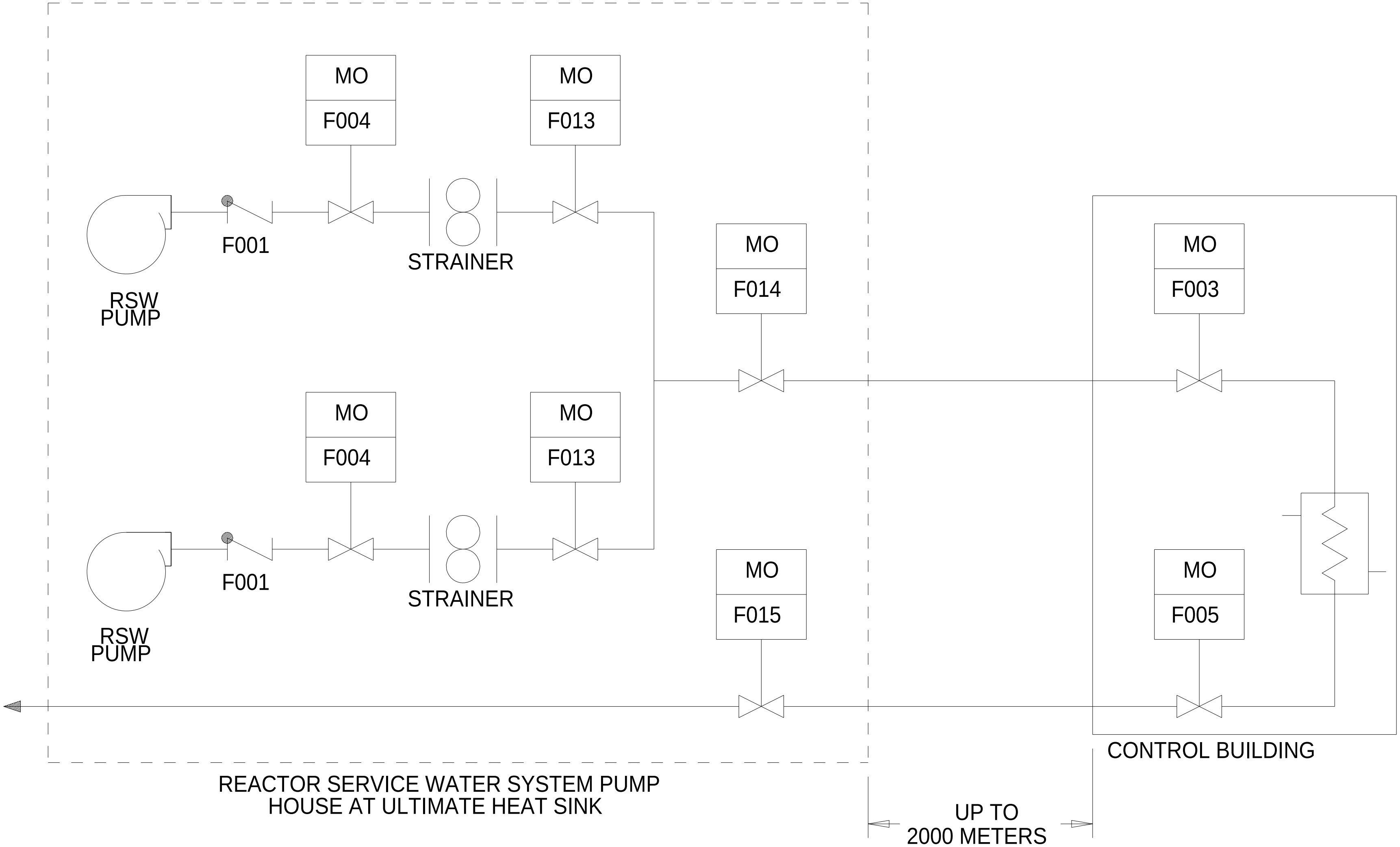
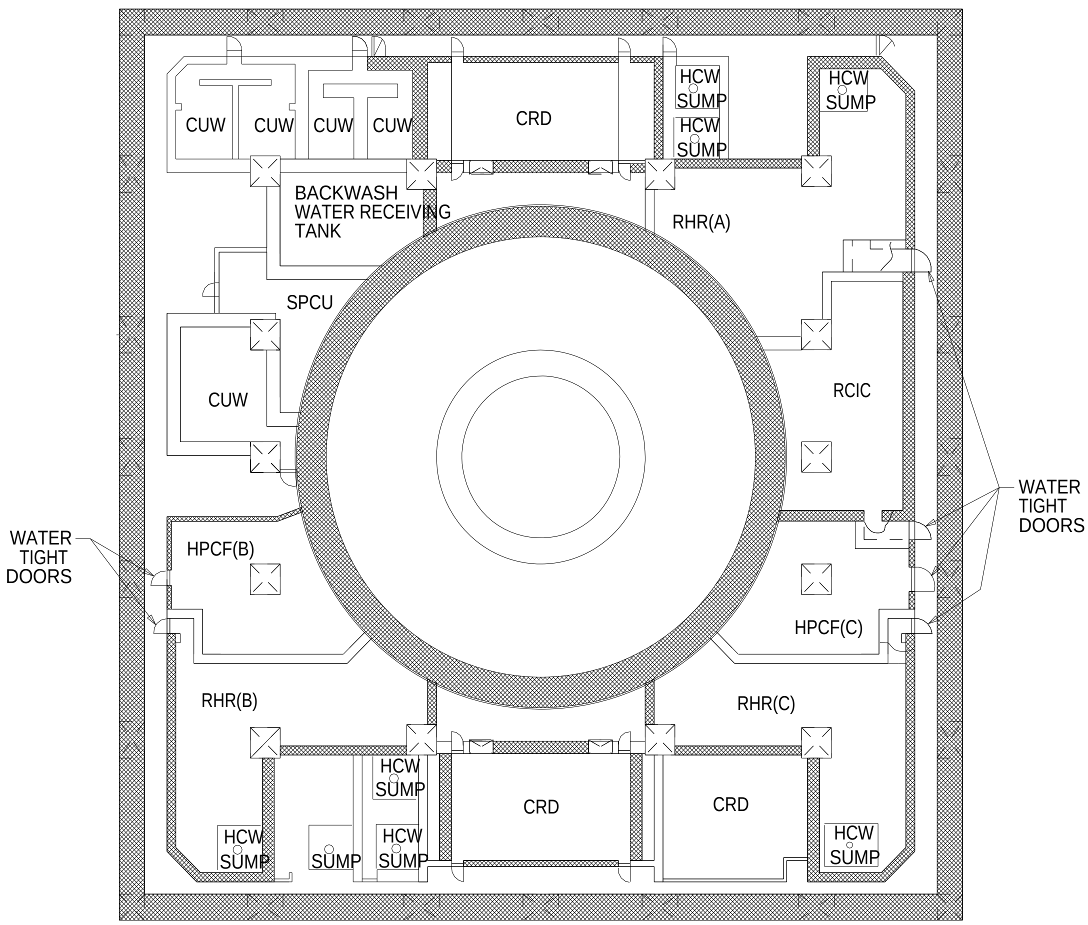
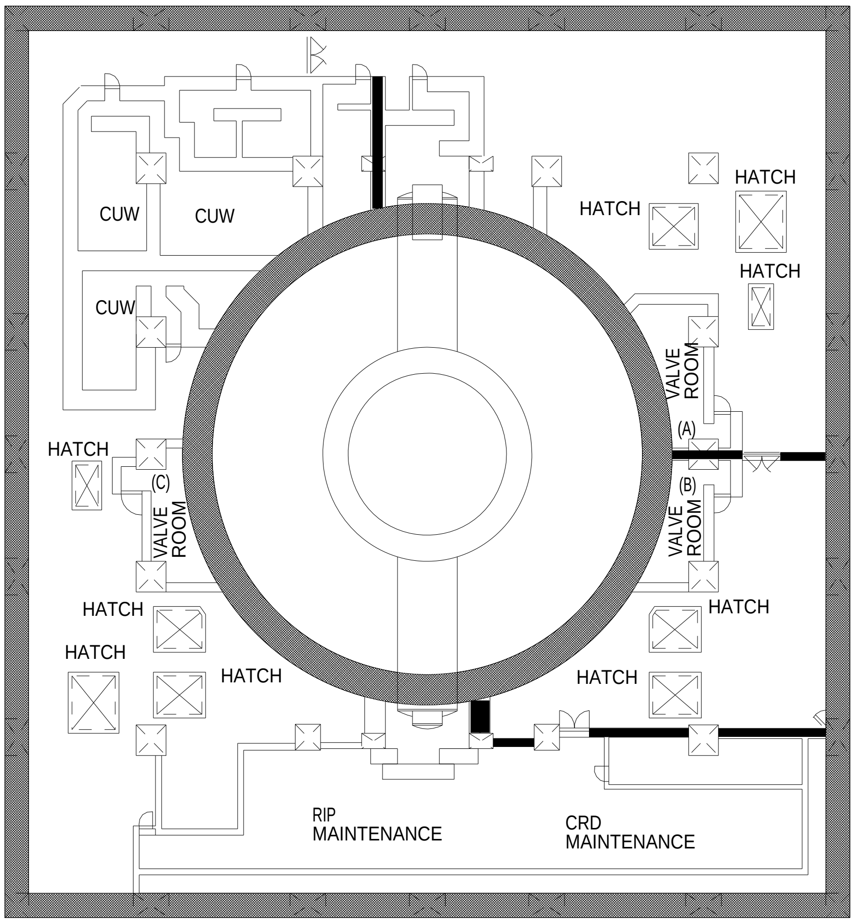


Figure 19R-2 Reactor Service Water System



1. " " DENOTES FIRE RATED DOORS.
2. COLUMN DIMENSIONS ARE 1.6m X 1.6m (TYPICAL).
3. FLOOR SLAB THICKNESS IS 0.5m.
4. MAIN BEAM DIMENSIONS ARE 1.4m X 1.8m.
5. ALL DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

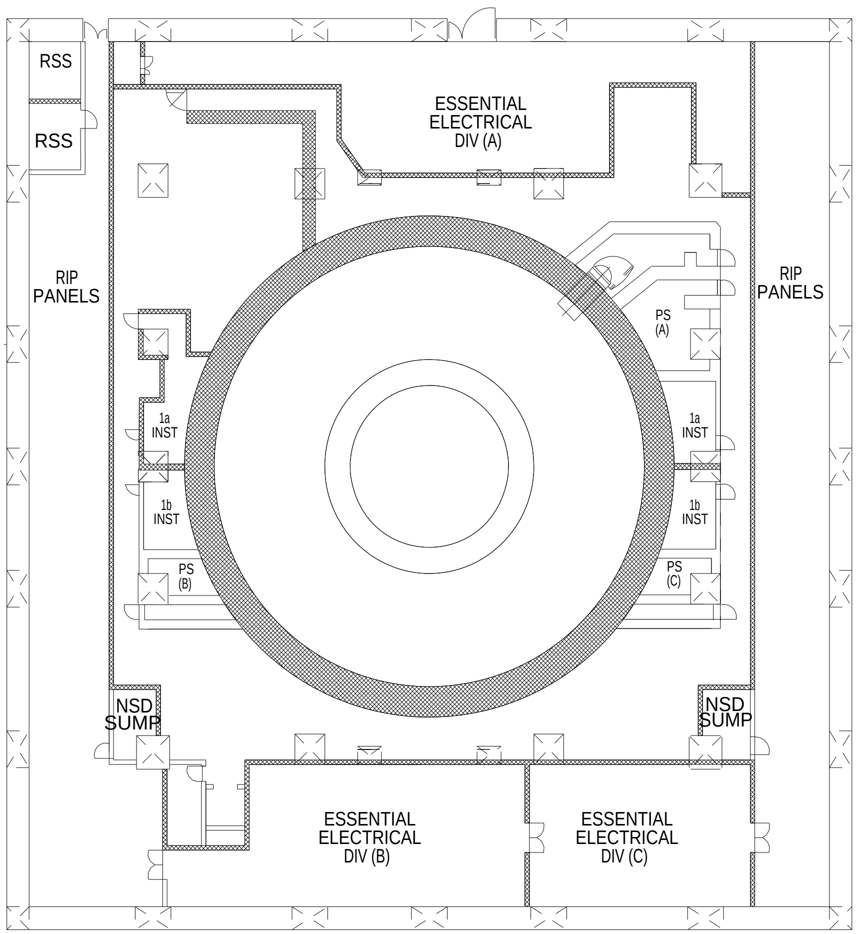
Figure 19R-3 Reactor Building Arrangement—Elevation –8200 mm (B3F)



NOTES:

1. " " DENOTES FIRE RATED DOORS.
2. COLUMN DIMENSIONS ARE 2.0m x 2.0m (TYPICAL).
3. FLOOR SLAB THICKNESS IS 0.6m.
4. MAIN BEAM DIMENSIONS ARE 1.5m X 1.8m.
5. ALL DIMENSIONS ARE IN mm UNLESS OTHERWISE INDICATED.

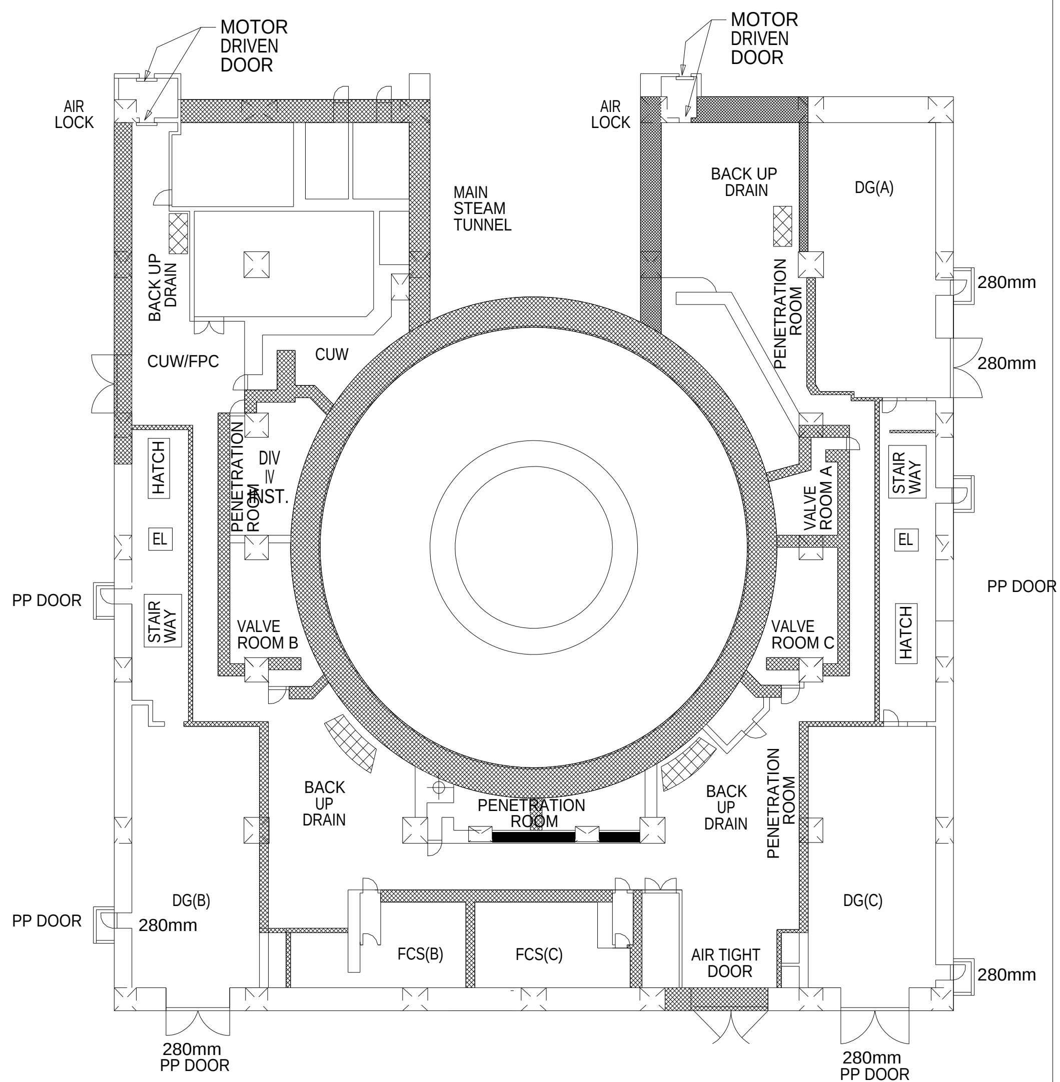
Figure 19R-4 Reactor Building Arrangement—Elevation –1700 mm (B2F)



NOTES:

1. " " DENOTES FIRE RATED DOORS.
2. COLUMN DIMENSIONS ARE 1.8m X 1.8m (TYPICAL).
3. FLOOR SLAB THICKNESS IS 0.6m.
4. MAIN BEAN DIMENSIONS ARE 1.2m x 1.8m.

Figure 19R-5 Reactor Building Arrangement—Elevation 4800 mm (B1F)



NOTES:

1. " " DENOTES FIRE RATED DOORS.
2. COLUMN DIMENSIONS ARE 1.6m X 1.6m (TYPICAL).
3. FLOOR SLAB THICKNESS IS 0.5m.
4. MAIN BEAM DIMENSIONS ARE 1.4m X 1.8m.

Figure 19R-6 Reactor Building Arrangement—Elevation 12300 mm (1F)

Figure 19R-7 Turbine Building Flooding (Low PCHS)

Not Part of DCD (Refer to SSAR)

Figure 19R-8 Turbine Building Flooding (High PCHS)

Not Part of DCD (Refer to SSAR)

Figure 19R-9 RSW Control Building Flood

Not Part of DCD (Refer to SSAR)

Figure 19R-10 Fire Water Flood in the Control Building

Not Part of DCD (Refer to SSAR)

Figure 19R-11 Reactor Building Flooding in ECCS Room

Not Part of DCD (Refer to SSAR)

Figure 19R-12 Reactor Building Flooding in Corridor

Not Part of DCD (Refer to SSAR)

Figure 19R-13 Fire Water Flood in the Reactor Building Outside Secondary Containment

Not Part of DCD (Refer to SSAR)