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Washington, D.C. 20555

Attention: Richard Borchardt, Acting Director
Standardization Project Directorate

Subject: **Submittal of Amendment 28 to GE's ABWR SSAR**

Reference: Submittal of Amendment 28, Proprietary Information, to GE's ABWR SSAR, MFN No. 075-93, dated May 14, 1993

Dear Mr. Borchardt:

Enclosed are thirty-four copies of selected sections of Chapter 19, Response to Severe Accident Policy Statement, of the Standard Safety Analysis Report (SSAR) for the Advanced Boiling Water Reactor (ABWR). Amendment 28 incorporates the majority of transmittals resolving DFSER issues pertaining to Severe Accident and PRA. The remaining sections will be provided in a near future amendment. At that time, the entire Chapter 19 will be provided in the single column format (similar to SBWR).

Please note that all or parts of the following sections contain information that is designated as General Electric Company proprietary information: 19F.1,19F.2, 19F.3, 19FA.0, 19FA.1, and 19FA.2. The proprietary text is being submitted under separate cover.

Sincerely,

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ABWR SSAR

Amendment 28 - Change Instruction

The following Chapters and Appendixes have been changed, please make the specified changes in your SSAR.

REMOVE	ADD
CHAPTER 19	
19 TOC Thru Section 19.6	19 TOC Thru Section 19.13
APPENDIX 19D	
19D TOC	19D TOC
19D.3 TOC Thru Section 19D.5	19D.3 TOC Thru Section19D.5
Section 19D.8	Section 19D.8
Add	Section 19D.9
APPENDIX 19E	
19E.2 TOC Thru Section 19E.3	19E.2 TOC Thru Section 19E.3
Add	Section19EA Thru Section 19EB
APPENDIX 19F	
Section 19F.4-1	Section 19FA.1
APPENDIX 19H	
Section 19H	Section 19H

REMOVE	ADD
APPENDIX 19K	
Section 19K	Section 19K
APPENDIX 19J	
Section 19J.4	Section 19J.4
APPENDIX 19L	
Section 19L	Section 19L
APPENDIX 19M	
Section 19M	Section 19M
APPENDIX 19P	
Add	Section 19P

CHAPTER 19

TABLE OF CONTENTS

CHAPTER 19

TABLE OF CONTENTS (Continued)

CHAPTER 19

TABLE OF CONTENTS (Continued)

CHAPTER 19

TABLE OF CONTENTS (Continued)

CHAPTER 19

TABLE OF CONTENTS (Continued)

CHAPTER 19

TABLE OF CONTENTS (Continued)

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 1) and in response to the ABWR Licensing Review Bases (Reference 3) which would be used for NRC review of the ABWR Standard Plant design. Response to the CP/ML (Construction Permit/Manufacturing License) Rule (Reference 2) 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 3. Appendix 19E through 19J support the probabilistic risk assessment. Appendix 19K indentifies 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 P provides information about the consideration of additional design modification to reduce the residual risk of severe accidents. Finally, Appendix R 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 3. 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 which includes 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 1.6E-7 per year.

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 was estimated at 1.1E-9 per year. Only 0.1 percent of these cases resulted in core damage.

The probability of exceeding 25 rem whole body dose at one half mile from the reactor was determined to be less than 1E-9 per year. The calculated societal risk was determined to be 8.4E-13 and the calculated individual risk was determined to be 1.4E-13.

19.1.3 References

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

2. Title 10, Code of Federal Regulations, Part 50, Section 50.34(f).

3. Thomas E. Murley (NRC) letter to Ricardo Artigas (GE), August 7, 1987, *Advanced Boiling Water Reactor Licensing Review Bases*.

SECTION 19.2

CONTENTS

SECTION 19.2

TABLES

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:

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.

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.

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 this SSAR. 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 Section 19E.3. 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 2). These assumptions are those which relate to systems which are outside the scope of the SSAR 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.

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 1). 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 (Section 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 3). A realistic safety system maintenance policy, similar to that practiced by a number of operating BWRs is applied in the analysis. This policy prohibits any routine on-line maintenance that would disable a standby safety system as discussed in Appendix A of Reference 5. Thus, all routine standby system maintenance tasks are assumed to be performed when the plant is shut down.

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 1) 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 4). LOCA initiation frequencies are the same as those used in the GESSAR II PRA (Reference 5) and are based upon the Reactor Safety Study (Reference 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 on-site or off-site power; and 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 Sections 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; and 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 Section 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 2 of Section 19D.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, on-line 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 assesment 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 5) and the WASH-1400 Reactor Safety Study (Reference 6). Consideration has been given to startup and to lower power operation in the shutdown risk assesment in Appendix 19L and 19Q.

The conditions of this analysis are the conditions

applicable to a mid-life plant with an 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 code as noted in Reference 1. These analyses cover events and conditions depicted by the accident and containment event trees in Sections 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 Section 19D.2. Procedures for assessing the quantities of fission products released are discussed in Section 19E.2 and the process for evaluating the consequences of radioactive release are addressed in Section 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 Section 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. Structural analysis is contained in Appendix 19F. Melt progression analysis is contained in Section 19E.2.

The containment ultimate strength under postulated severe accident conditions is evaluated in Appendix 19F based on scale model testing, analysis and judgements.

The pressure capability of the concrete shell of the prototypical design at ambient temperature is 180 psig for the top slab region as determined from model overpressurization tests. The pressure capability is estimated to be as high as 180 psig for the

cylindrical wall as determined from other model tests. The ABWR containment of the standard plant situated in U.S. has more reinforcing steel than the prototypical design on which the test models were based. Consequently, the pressure capability of the concrete containment as determined by the tests is conservative for the standard plant. The thermal effect of the representative severe accident temperature of 500 F on the pressure capability of the concrete shell is expected to be insignificant.

The pressure capability of the drywell head which was not included in the concrete containment tests 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 mediam limit pressure is 134 psig at 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 0.005 in.) separation displacements of the sealing surfaces at 90 psig 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 500 F, for pressures below the capability pressure as:

Pressure (psig)	Leak Area (in^2)
0	0.00
45 (design)	0.00
52 (SIT)	0.00
60	1.23
70	2.77
80	4.31
90 (COPS setpoint)	5.85

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

The evaluation of the accident progression was performed using the MAAP code as described in Section 19E.2. MAAP is an integrated code which considers 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. Additional analyses were performed using separate effects models as described in Section 19E.2

Inputs of the MAAP code are the plant parameters and event sequence information. Based on this information MAAP 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:

- o Amount and type of fission product release.
- o Behavior of the fission products after release from the plant.
- o 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 Section 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 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 PRA's 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 Section 19E.3. These results in terms of the risks of early fatality for individuals within one mile of the site boundary and increase in risk of cancer fatality for individuals within 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 25 rem at one half mile at 1.0E-6 probability.

19.2.5 References

1. *Advanced Light Water Reactor Requirements Document; 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.

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

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

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

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

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

TABLE 19.2-1

KEY PRA ASSUMPTIONS

SUMMARY ASSUMPTIONS	REFERENCE SUBSECTION
Condensate Storage Pool Volume	19E.2.1.2.1(1)
Containment Overpressure Protection	6.2.5
Limestone Sand Concrete	19E.2.1.2.1(3)
Battery Loading Profiles for Station Blackout	19D.4.2.9, 19E.2.1.2.1(3)
RCIC Room Temperature Less Than Equipment Design Temperature	19D.4.2.9, 19E.2.1.2.1(5)
Control Room Temperature Less Than Equipment Design Temperature	19E.2.1.2.1(6)
Suppression Pool Makeup within 24 hours	19D.5.7.6
Preliminary Design of Containment Drywell Head and Major Penetrations	19F.3.1.2, 19F.3.2.2
Reactor Service Water System Definition	19D.6.4.2
AC-Independent Water Addition System	5.4.7, 19.3.1.5.2
Lower Drywell Flooder System	9.5.12
Gas Turbine	9.5.11, 19.3.1.5.1

TABLE 19.2-1

KEY PRA ASSUMPTIONS (Continued)

SUMMARY ASSUMPTIONS	REFERENCE SUBSECTION
Category I Building Structure (other than reactor building) Fragility	19H.3.3
Non-category I Building Structure Fragility	19H.3.4
No Seismic-induced Soil Failure	19H.3.1
Interconnecting Piping Capacity Against Differential Building Displacement	19H.3.1
ECCS Test and Surveillance Intervals, Same as GESSAR	19.3.1.6
ECCS equipment room sump drain system includes AC-independent means to prevent sump pump from back-flooding into adjacent ECCS equipment room	19E.2.3.4

SECTION 19.3

CONTENTS

SECTION 19.3

CONTENTS (Continued)

SECTION 19.3

TABLES

SECTION 19.3

FIGURES

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 section describes the accident sequence initiating events documented in Section 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 two events per year. This total consists of one planned reactor shutdown and one transient which results in reactor scram. The expected number of planned shutdowns is based upon a 1985 analysis of operating plant data (Reference 1). The frequency of transients is a design requirement prescribed in the Advanced Light Water Reactor (ALWR) Requirements Document (Reference 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 Section 19D.3.

LOCA initiation frequencies are the same as those used in the GESSAR II PRA (Reference 3) and are based upon the Reactor Safety Study, WASH-1400 (Reference 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

19.3.1.2.1 System and Component Failure Data

Failure data used in the ABWR PRA have, for the most part, been taken from internal GE data and failure information compiled in the GE Failure Rate Data Manual (Reference 5). This manual consists of information compiled from a number of nuclear as well as other sources. Specific failure rates used in the PRA and their references accompany the fault trees presented in Section 19D.6.

19.3.1.2.2 Dependent Failure Treatment

Dependent failures have been addressed as integral parts of the overall PRA system and functional performance evaluations. These failures are primarily significant from the standpoint of functional redundancy. Common cause failures have been explicitly addressed from the standpoint of multiple component failures within systems and in addition have been accounted for as consequences of human error. Interdependencies have been rigorously treated in the fault and event tree analyses. Dependent failure treatment is discussed in greater detail in Section 19D.8.

19.3.1.2.3 Human Error Prediction

Human error probabilities used in this analysis are presented in the applicable component failure rate data tables which accompany each system fault tree presented in Section 19D.6 as well as the tables which document branch point values for each accident sequence event tree in Section 19D.4. They were taken predominately from the GESSAR II PRA for which they were collected from various other sources and modified, as appropriate, for the GESSAR application. Most of these values were derived from the Swain and Guttmann Handbook of Human Reliability (Reference 6). More recent studies suggest that these values may be somewhat conservative. Their application in the ABWR PRA analyses is judged to be acceptable. Section 19D.7 provides additional information on the treatment of human error.

19.3.1.3 Accident Sequence Analysis

19.3.1.3.1 Success Criteria

This section 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 section 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 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 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 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 turbine driven and also provides high pressure water make-up to the RPV as long as steam is available at pressures greaten than 50 psid for the RCIC turbine. The RCIC has a lower flow capacity than the HPCF or RHR pumps and 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 make-up 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 make-up water for these pumps are the main condenser hotwell and the condensate storage tank. Sufficient make-up 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 make-up water and electrical power. Sufficient make-up 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 tank 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.

(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 (see 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 RWCU 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 (4) 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 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 (RWCU) 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 RWCU non-regenerative heat exchanger.

(c) RPV Pressure Relief

A pressure of 150 percent of the reactor-coolant pressure-boundary design pressure (1250 psig), the faulted limit, was chosen as the criterion 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 or safety/relief valves represent the two success paths in the ABWR over-pressure 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 1875 psig.

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 or SLC injection. Work performed by General Electric and Idaho National Engineering Laboratory (Reference 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 system 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. Over-pressure 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 1875 psig.

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 dumped 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 remaining 6 RIPs will trip. The water level will continue to decrease, initiating the RCIC and two HPCF systems about one minute into the transient. Shortly thereafter, the operator must inhibit the ADS valves from opening. 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 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, or

All the SRVs are closed and the drywell pressure remains below 0.014 Mpa-g (2 psig).

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 operator will run the feedwater 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, he will trip the RIPs, run the feedwater back and initiate SLC. 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 the operator will have ample time to initiate SLC.

(iv) Loss of Offsite Power

The loss of offsite power (LOSP) initiated ATWS is similar to the MSIV closure transient, LOSP generates a scram signal and causes the MSIVs to close. Additionally, 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 operator must inhibit the ADS actuation and begin the SLC injection 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 Section 19D.5 to establish final core states. Separate trees have been developed, as shown in Figures 19D.4-1 through 19D.4-15 of Section 19D.4, 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 Section 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 devel

oped 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 Section 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 Section 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 Section 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 Section 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 Section 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.

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 dependent on successful performance of the gas turbine generator or RCIC system.

19.3.1.5 Results in Perspective

The estimated core damage frequencies of order 1E-07 presented above are extremely low and indicate the expectation of a single core damage event in ten million reactor years of operation. 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. These are judged to be reasonable assumptions when the plant CDF is on the order of 10^{-5} to 10^{-6}. If, however, the calculated plant CDF is of the order of 10^{-7}, such omitted failure modes could become significant contributors to the expected core damage frequency.

In addition, although this 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 information to the extent possible to the ABWR PRA. As a result, ABWR ECCS system test and surveillance intervals were assumed the same as GESSAR. 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 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 credit is taken for the firewater system in the accident event trees. 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 nine 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 Section 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 nine subclasses of event are modeled in the containment event trees (CET). 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 Section 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 nine 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 Section 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 Section 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:

- o Type of initiating event (transient, LOCA, etc.).
- o Relative times of core melt and containment failure.
- o 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 removal of radioactive material released in the containment atmosphere during 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, 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 (CET). 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, on-site 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 code. A detailed description of the analyses is given in Section 19E.2.

A typical core melt sequence may include the following steps:

a) Core melt in the RPV

b) RPV failure.

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

d) Evaporation of water in LDW producing steam.

e) Core-concrete interaction producing non-condensible gases.

f) 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.

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

h) 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 in f) above is an unique feature of ABWR containment configuration which allows for core melt arrest without the use of active components.

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

RPV failure:	1.8 hours
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 Section 19E.2.

19.3.2.4 Containment Structural Capability

The ABWR containment design pressure is 45 psig. 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, along with the results of the tests performed in Japan on a scaled containment model, 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 500°F, (which is a typical temperature for most accident sequences), the drywell median ultimate strength is evaluated to be 134 psig.

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. >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 52 psig.

The following failure modes are explicitly modeled in the CETs.

(1) Containment leakage occurs when the temperature exceeds 500°F and pressure exceeds 52 psig.

(2) Drywell head failure occurs when the containment pressure exceeds 134 psig and temperature is below 500°F.

(3) Containment high temperature failure occurs when the containment experiences a very high temperature (>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 are, therefore, 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 steam line 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 and one other small line wetwell-drywell line, not including these paths in the CET models explicitly will affect the risk results by no more than 10% of the total risk. Only these lines 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, on-site and offsite power has been modeled in the CETs.

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

P_f = 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 carryout 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 Section 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 off-site 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 nine 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 of 1.6E-7 per year for internal events. These results, together with the associated source terms, form the input for the consequence analysis in Section 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 Section 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 fro 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 system, 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 sucessfully evacuate a region on about 24 hours (Reference 4, Appendix 6J).

Based on the forgoing, four times frames were selected in determining the time 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 Section 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 meterological 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

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

2. *Advanced Light Water Reactor Requirements Document, Appendix A: PRA Key Assumptions and Groundrules*, August 1988 Draft, Page D4, Electric Power Research Institute.

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

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

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

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

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

TABLE 19.3-1

INITIATING EVENT FREQUENCIES

Initiating Event	Frequency Per Reactor Year	
Manual Shutdown		1.0
Isolation/Loss of Feedwater		0.20
MSIV Closure	0.04	
Loss of Condenser Vacuum	0.06	
Press. Reg./Bypass Valves Closed	0.06	
Loss of Feedwater	0.09	
Non-Isolation Event (Trip with bypass)		0.68
Inadvertent (Stuck) Open Relief Valve		0.01
Loss of Offsite Power		0.10
Less than 30 minutes	0.0579	
30 Minutes to 2 Hours	0.0246	
2 to 8 Hours	0.0158	
Greater than 8 Hours	0.0017	
Small LOCA		0.0012
(Liquid Break <0.00545 ft^2) (Steam Break <0.3 ft^2)		
Medium LOCA		0.00067
(Liquid Break 0.00545 ft^2)		
Large LOCA		0.00021
(Liquid Break 0.3 ft^2 or greater) (Steam Break 0.3 ft^2 or greater)		

TABLE 19.3-2

SUCCESS CRITERIA TO PREVENT INITIAL CORE DAMAGE FOR TRANSIENT AND LOCA EVENTS WITH RPS SCRAM

CORE COOLING:

Event	Success Criteria
Large Liquid LOCA ($> = .3$ ft^2)	HPCF-B or C or LPFL$^{(1)}$-A or B or C
Large Steam LOCA ($> = .3$ 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 ($< .3$ ft^2) ($> .00545$ ft^2)	HPCF-B or C or ADS3$^{(3)}$ + LPFL$^{(1)}$ - A or B or C
Small Liquid LOCA ($> = 0.00545$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$^{(2)}$ ADS8$^{(6)}$ + 1 Fire Protection System Pump

TABLE 19.3-2 (continued)

SUCCESS CRITERIA TO PREVENT INITIAL CORE DAMAGE FOR TRANSIENT AND LOCA EVENTS WITH RPS SCRAM

CORE COOLING (Cont.):

Event	Success Criteria
Small Steam LOCA (<.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)

LONGTERM HEAT REMOVAL:

Event	Success Criteria
All Transients or Small Liquid LOCA	RHR-A or B or C(7) or Normal Heat Removal(8) or RWCU(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:

Event	Success Criteria
Isolation Events	6 Safety/Relief Valves
Non-Isolation Events	3 Turbine Bypass Valves or 2 Turbine Bypass Valves + 2 Safety/Relief Valves or 1 Turbine Bypass Valve + 4 Safety/Relief Valves or 6 Safety/Relief Valves

19.3-2 (continued)

SUCCESS CRITERIA TO PREVENT INITIAL CORE DAMAGE FOR TRANSIENT AND LOCA EVENTS WITH RPS SCRAM

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 pump needs sufficient steam generation at or above the required minimum pressure to drive the turbine. 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 low 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 (RWCU) 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 RWCU 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

PRESSURE RELIEF:

Initiator	Success Criteria
Isolation Initiators	15 Safety/Relief Valves
Non-Isolation Iniators	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

TABLE 19.3-3 (continued)

SUCCESS CRITERIA AND REQUIRED OPERATOR ACTIONS FOR ATWS EVENTS

CORE COOLING:

Initiator	Success Criteria
All events with rod insertion	See 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

TABLE 19.3-3 (continued)

SUCCESS CRITERIA AND REQUIRED OPERATOR ACTIONS FOR ATWS EVENTS

Initiator	Success Criteria	Time of Operator Action
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
	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.	4.25E-08
IB-1	Station blackout events (short term) with RCIC failure.	2.57E-08
IB-2	Station blackout events with RCIC available for core cooling for approximately eight hours.	1.62E-08
IB-3	Station blackout events (long term) with RCIC failure.	8.86E-10
IC	ATWS events without boron injection coupled with loss of core cooling.	2.18E-14
ID	Transients followed by loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling.	6.95E-08
II	Transient, LOCA, and ATWS (with boron injection) events, with successful core cooling but with possible failure of containment.	1.10E-10
IIIA	Small or medium LOCAs with failure of high pressure core cooling followed by failure to depressurize the reactor.	3.87E-10
IIID	LOCAs followed by loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling.	2.10E-10
IV	ATWS events without boron injection but with core cooling available.	1.66E-10
TOTAL		1.56E-07

TABLE 19.3-5

FREQUENCY OF CORE DAMAGE BY INITIATING EVENT

Initiating Event	Description	Frequency (Events/year)	% of Total
TM	Reactor Shutdown	1.14E-08	7
T[illegible]	Non-Isolation (Turbine Trip)	6.82E-09	4
TIS	Isolation/Loss of Feedwater	1.69E-08	11
TE2	Loss of Offsite Power for Less Than Two Hours	4.46E-09	3
TE8	Loss of Offsite Power for Two to Eight Hours	2.87E-09	2
TEO	Loss of Offsite Power for More Than Eight Hours	1.69E-09	1
BE2	Station Blackout for Less Than Two Hours	6.67E-08	43
BE8	Station Blackout for Two to Eight Hours	2.57E-08	17
BE0	Station Blackout for More Than Eight Hours	1.71E-08	11
TIO	Inadvertent Open Relief Valve	1.22E-09	1
S2	Small Break LOCA	2.55E-10	<1
S1	Medium Break LOCA	3.42E-10	<1
S0	Large Break LOCA	1.24E-13	<1
ATWS	Anticipated Transient Without Scram	166E-10	<1
TOTAL		1.56E-07	

TABLE 19.3-6

FREQUENCY OF FISSION PRODUCT RELEASE

Time of Release	Release Frequency	
No Release	1.34E-07	
	Release via Rupture Disk	**Release via Drywell**
> 24 hours	2.1E-08	3.9E-10
16 to 24 hours	1.1E-10	3.6E-11
8 to 16 hours	0.0	0.0
< 8 hours	7.3E-11	3.3E-10

TABLE 19-7

ESTIMATED FREQUENCY OF CORE DAMAGE BY INITIATING EVENT ASSUMING INCORPORATION OF GAS TURBINE GENERATOR WITH UNAVAILIBILITY OF 5 PERCENT

Deleted

TABLE 19.3-8

ESTIMATED FREQUENCY OF CORE DAMAGE BY ACCIDENT CLASS ASSUMING INCORPORATION OF FIRE WATER SYTEM CONNECTION WITH UNAVAILIBILITY OF 10 PERCENT

Deleted

TABLE 19.3-9
ESTIMATED FREQUENCY OF CORE DAMAGE BY ACCIDENT CLASS ASSUMING:
1) INCORPORATION OF GAS TURBINE GENERATOR WITH UNAVAILIBILITY OF 5 PERCENT, AND
2) INCORPORATION OF EMERGENCY FIRE WATER CONNECTION WITH UNAVAILIBILITY OF 10 PERCENT IN LOW PRESSURE SEQUENCES

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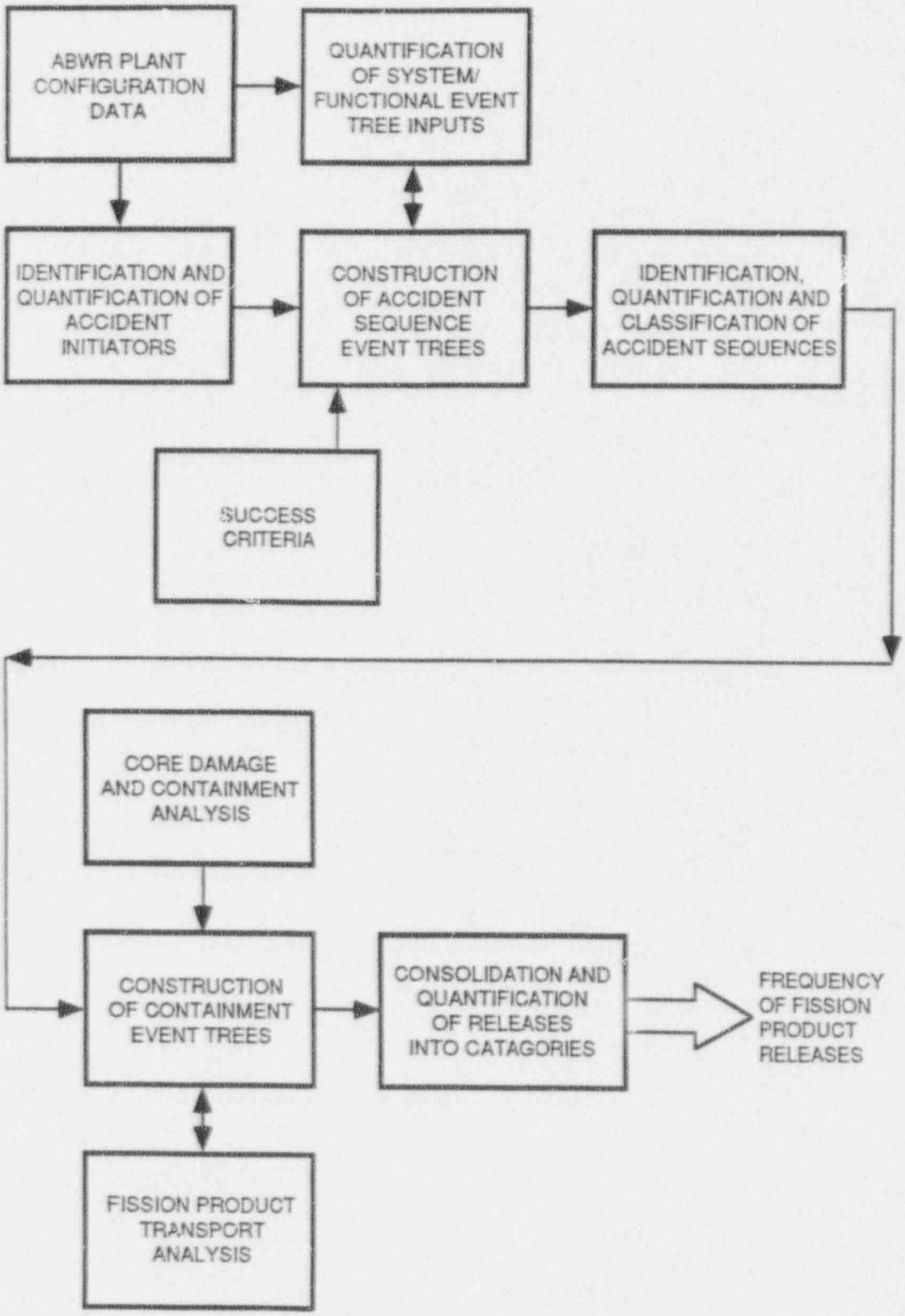


Figure 19.3-1 OVERVIEW OF METHODOLOGY FOR ASSESSING FREQUENCY OF CORE DAMAGE AND FISSION PRODUCT RELEASES

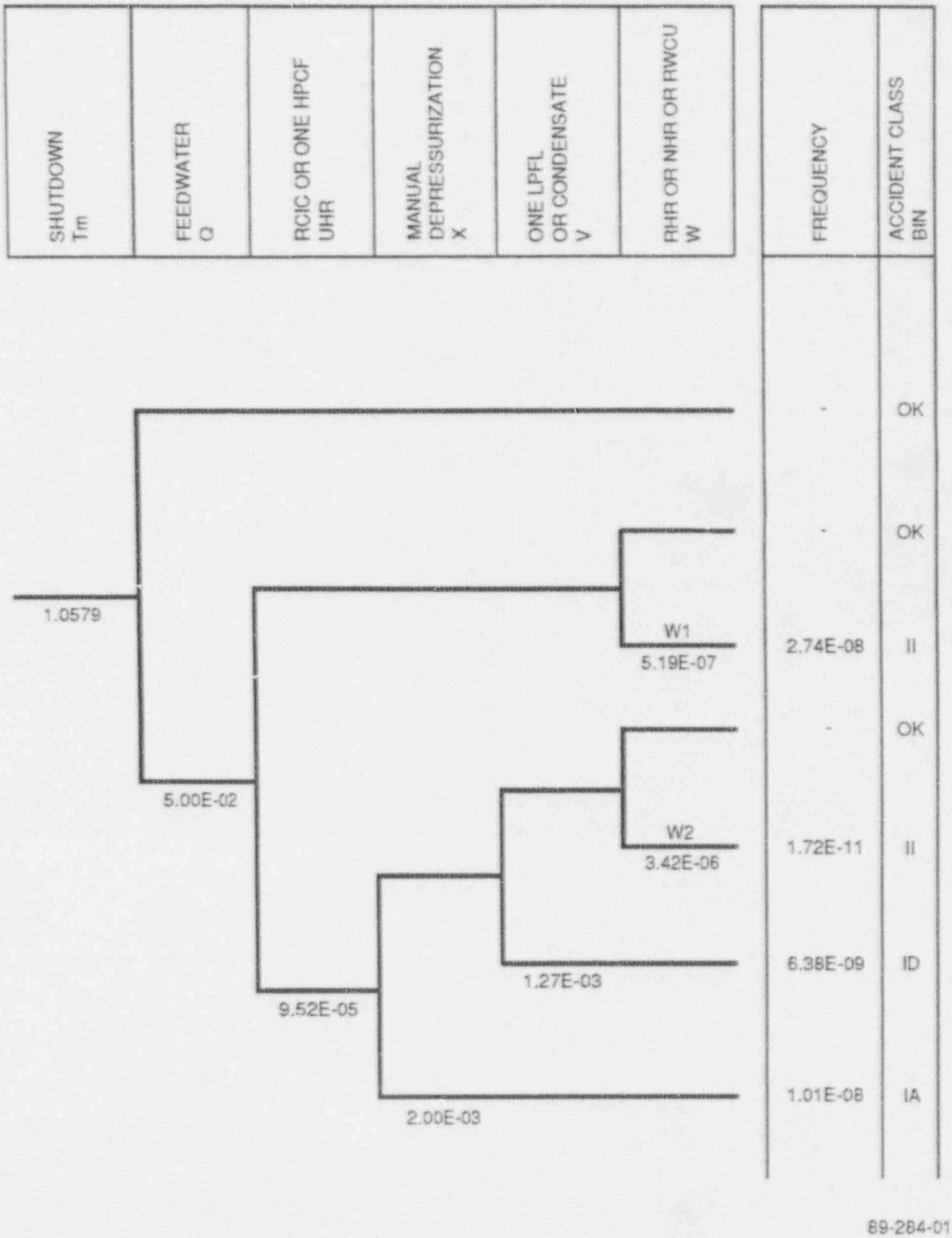


Figure 19.3-2 REACTOR SHUTDOWN EVENT TREE

SECTION 19.4

CONTENTS

SECTION 19.4

TABLES

ILLUSTRATIONS

19.4 EXTERNAL EVENT AND SHUTDOWN RISK ANALYSIS

19.4.1 External Event Review

The Advanced Light Water Reactor (ALWR Utility Requirements Document (Reference 1), contains a set of design requirements for the ALWR. The Nuclear Regulatory Commission's Severe Accident Policy (Reference 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 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 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 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 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; and 2) earthquakes. This EPRI assessment was 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 pe formed 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 section, 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 section 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 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 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 0.14 square miles. The resulting regional site strike frequencies were found not to be strongly region dependent, and therefore the maximum assessed regional value of 2.86E-05 tornado strikes per year was conservatively specified as the basis for evaluation.

Consequently, the loss of offsite power and station blackout accident event trees of Section 19D.4 were evaluated using 2.86E-05 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, and
- Offsite powe: 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 Section 19D.4.

Evaluation of these event trees on this basis yields an expected total core damage frequency due to tornado-initiated events of 1.1E-08 per year. This 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 Event Analysis

19.4.3.1 Introduction

This section discusses the background, objectives, and general approach to the seismic event 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 9). 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. Because of the high degree of uncertainty that presently exists in seismic analysis, there is generally a higher degree of consservatism applied to seismic analysis than to the analysis of internally-initiated events. The comparison of numerical results of the two types of analyses should be made with due caution to avoid arriving at erroneous conclusions.

Section 3.2 of this SSAR states the following:

> "Nuclear Island structures, systems, and components, including their foundations and supports, that must remain functional in the event of 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 of the SSAR 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 safe shutdown earthquake (SSE) on all soil conditions, 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 probabilistic risk assessment presented in the following sections 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 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, within the limits of uncertainty of the analysis, the relative degree of risk contribution from seismic events in comparison with other events.

(4) 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 event analysis that has been performed as described in the following sections 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 the guidelines of the PRA Procedures Guide (NUREG/CR-2300) and the Probabilistic Safety Analysis Procedures Guide (NUREG/CR-2815). This is the same approach used in the GESSAR II PRA and other PRAs and is depicted on Figure 19.4-1. This approach also meets the requirements of Reference 3.

This assessment of seismic-initiated core damage frequency and offsite risk consists of four primary tasks: the establishment of a seismic hazard curve, the determination of the seismic capability of critical components and structures, an assessment of the core damage frequency, and an estimate of the offsite risk. The chronology interrelationship among these tasks is as illustrated in Figure 19.4-1.

Referring to Figure 19.4-1, the first step in the analysis is to identify systems and components that are important to safety during severe accidents 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 a general

knowledge of component fragilities. The objective is to limit the size of the analysis by screening-out many components that can obviously withstand a severe earthquake without damage.

The remaining components are then subjected to component fragility analysis. 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.

A seismic hazard curve is used to represent the frequency distribution of expected earthquakes as a function of intensity for the location of the plant site.

The seismic hazard curve is then integrated with the component and structure fragilities to provide an expected frequency of failure of the components and structures. The seismic core damage frequency is determined by constructing and evaluating seismic fault trees and event trees.

The containment analysis and external consequence analysis are conducted to assess risk in much the same way as for the internal events analysis documented in Section 19D.5 and Appendix 19E.

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) 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 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 Hazard Analysis

To assess the seismic risk associated with the ABWR, the seismic hazard input must be defined. The seismic hazard analysis defines the frequency per year that levels of ground motion are expected to occur at a particular site. The seismic hazard input to the PRA consists of a complementary cumulative distribution function (CCDF) curve of expected frequency of exceedance of increasing levels of mean peak horizontal ground acceleration.

The seismic hazard curve used in this analysis is taken from Appendix 25.D of Reference 10 and is reproduced on Figure 19.4-2. Definition of the curve is given in Table 19.4-1. The curve is expressed in terms of mean peak horizontal ground acceleration. This seismic hazard curve is a bounding curve applicable in general to sites in the central and eastern United States. This hazard curve exceeds the mean hazard curves for the majority of locations in the central and eastern United States. In that regard, the seismic hazard analysis in this PRA may be conservative.

19.4.3.3 Seismic Fragility Analysis

In order to determine the capacity of the ABWR plant to resist seismic events, it is necessary to know the probability of failure of plant structures and components at varying seismic levels. This is accomplished through the development of fragility curves, which plot the probability of failure versus the seismic ground motion parameter used in the hazard analysis. A typical fragility curve is an S-shaped curve which has an increasing probability of failure at higher seismic motion (see Figure 19.4-3). The mean fragility curve is given in terms of the mean peak ground acceleration (PGA), which is the same parameter used to define the seismic hazard curves. Fragility curves are generated only for those components and structures that have been identified as potentially important to the seismic risk analysis.

The structure and component fragility curves are constructed from a median capacity factor and a logarithmic standard deviation factor representing both randomness and uncertainty. The development of the median capacities and deviation factors for the structures and components of interest is given in Appendix 19H.

To be consistent with the ground motion parameter used in the seismic hazard curve, 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:

$$A = A_m \epsilon_\gamma \epsilon_\mu$$

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 can be computed and a curve of probability of failure versus ground acceleration can be generated. Figure 19.4-4 shows a typical family of fragility curves for various confidence levels. The center solid curve represents the median fragility curve at 50% confidence level. The logarithmic standard deviation of the randomness component β_γ measures the curve slope. The logarithmic standard deviation of the uncertainty component β_μ determines the spread from the median curve.

The 95th percentile curve in Figure 19.4-4 means that there is 95% confidence that the actual fragility curve would be higher. On the other hand, the 5th percentile curve has a probability of exceedance being less than or equal to 5%.

When only a point estimate of fragility is of interest, (as 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 of the randomness and uncertainty components:

$$\beta_c = \sqrt{\beta_\gamma^2 + \beta_\mu^2}$$

The fragility curve corresponding to the median value and the associated logarithmic standard deviation can then be computed. This composite fragility curve is called the mean fragility curve, and is shown as the dashed curve on Figure 19.4-4. This curve is used as the "best estimate" fragility representation.

To generate component or structure fragility curves, first the median acceleration capacity of the component or structure is estimated, together with the associated variability. 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:

$$A_m = F A_d ,$$

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 or structure 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:

$$F = \Pi F_i .$$

The overall composite logarithmic standard deviation is the square-root of the sum of the squares of the composite logarithmic standard deviations on the individual factors of safety:

$$\beta_c = \sqrt{\Sigma \quad \beta_{ci}^2} .$$

19.4.3.3.1 Structural Fragility

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.

Detailed fragility evaluations were made for the following structures in the reactor building complex:

Reactor building shear walls

Containment

Reactor pressure vessel pedestal.

These structures were evaluated using the method described previously and using various safety factors as developed in Appendix 19H.

Detailed fragility evaluation of other Seismic Category I structures such as the control building structures could not be made since the design is not complete at this stage of certification. The control building seismic fragility of the GESSAR II Mark III standard plant design is considered achievable for the ABWR since the design requirements for the ABWR are at least as stringent as those for the Mark III standard plant, (which was designed in the late 1970's).

Similarly, the design of Non-Category I structures such as the turbine building is not currently available. The turbine building seismic fragility is estimated from the turbine building fragilities used in previous PRAs compiled in Reference 11. The estimated fragility is the data base median value and is considered achievable for the ABWR non-category I building structures.

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 were 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 systems housed within the structure.

19.4.3.3.2 Component Fragility

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

- o ABWR specific components whose fragility evaluation was made according to existing design information.
- o Generic components whose fragilities are based on data recommended in Reference 3.

(1) ABWR Specific Components

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

Reactor pressure vessel (RPV)

Shroud support

Control rod drive (CRD) guide tubes

CRD housings

Fuel assemblies

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 the evaluation of these specific components.

(a) Reactor Pressure Vessel (RPV)

The failure of the RPV due to an earthquake would result in an accident sequence similar to a large break loss-of-coolant accident, with the exception that there would 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 2-1/2" diameter high-strength anchor bolts. At an upper elevation, the RPV is laterally restrained by stabilizers which are connected to the reactor shield wall.

In this study, the RPV is assumed to fail when the support skirt fails, either by the shell gross yielding or anchor bolt breakage. The resulting RPV capacity is a lower bound since the upper stabilizer restraints are capable of providing additional resistance to seismic shock.

The critical failure mode is found to be anchor bolt failure. The median ground acceleration capacity is 5.3g with a logarithmic standard deviation of 0.33g. The individual factors contributing to the median capacity are shown in Appendix 19H.

(b) RPV Internal Components

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

The failure modes and associated median ground acceleration capacities of these components are given in Appendix 19H. The contributing factors are also given in the appendix. Note that the reference design ground acceleration is 0.15g (operating basis earthquake or OBE) since the OBE loads were used in evaluating the factors of safety.

The fuel assemblies were found to have the least seismic capacity among the RPV internal components. The failure mode is buckling of the fuel channel. The corresponding median ground acceleration capacity is 1.2g, with a logarithmic standard deviation of 0.35g.

(2) Generic Components

Detailed fragility evaluations for safety-related components other than those specific components presented above cannot be made at this stage of cer-

tification due to lack of design details.

The fragilities for generic components recommended in Reference 4 are adopted for the ABWR standard plant. These generic fragilities were chosen based on a review of prior PRAs and fragility data. These are considered achievable for the ALWRs with an evolutionary improvement in the seismic capacities of the components designed to a 0.3g safe shutdown earthquake (SSE). The generic components applicable to the ABWR are the following:

(a) Off-site Power (transformers and ceramic insulators)
(b) Cable trays
(c) Batteries and battery tracks
(d) Battery chargers/Inverters
(e) Electric equipment (chatter failure mode)
(f) Panelboards/Instrumentation panels
(g) Switchgear/Motor control centers
(h) Transformers (not off-site transformers)
(i) Diesel generators and support systems
(j) Turbine-driven pumps
(k) Motor-driven pumps
(l) Heat exchangers/Small tanks (e.g., standby liquid control tank)
(m) Air-operated valves
(n) Motor-operated valves
(o) Safety relief, manual, and check valves
(p) Hydraulic control units
(q) Accumulators
(r) Large flat-bottom storage tanks
(s) Heating, ventilation, and air conditioning ducting
(t) Air handling units
(u) Piping
(v) Buried welded steel piping

The detailed descriptions of these component categories are provided in Annex A of Reference 3. The generic fragilities are summarized in Table 19.4-2.

19.4.3.4 Evaluation of Seismic Core Damage Frequency.

The expected frequency of core damage due to seismic events was 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.

In order to quantify the fault trees and event trees, it was necessary to establish the failure probability of components and structures due to seismic shock. This was accomplished in the analysis by integrating the seismic hazard curve with the component and structure fragility curves. Although conservative, in this analysis all identical redundant components were assumed to fail simultaneously due to the seismic event.

The three inputs required to perform the seismic analysis, as shown in Figure 19.4-5, are (a) the site hazard curve, (b) component and structural fragility curves and (c) fault trees and event trees which describe the accident sequence.

The hazard curve (Subsection 19.4.3.2.1) is a function which gives the annual probability of exceeding a given peak ground acceleration.

The fragility curve (Subsection 19.4.3.3) for a component or structure defines its susceptibility to earthquake expressed as the cumulative probability of failure as a function of peak ground acceleration.

Event trees are used to describe seismic accident sequences. Each system represented on the event tree is represented by and evaluated with a fault tree. The fault tree describes the logical arrangement of basic events (component and structural) which could lead to seismically induced failure, and resulting loss of function of the system. Only those components that are critical for performing a safety function and are also somewhat susceptible to seismic acceleration are included in the fault tree model.

An accident sequence probability, corresponding to a specific event tree path, is calculated by using the following equation:

$$P = \sum_{i=0}^{n} [H(g_i) - H(g_{i+1})] \bullet \prod_{j=0} S_{ij}, \text{ (Eq.1)}$$

where i = 0 to n integration intervals, $H(g)$ = the annual probability of exceedance of a seismic ground acceleration of g, and $s_{i,j}$ is the success or failure probability (as determined by its occurrence in the event tree path of the j^{th} system in the accident sequence for the i^{th} integration interval.

The probability of failure or success of a system in the accident sequence is

failure:

$$S_{i,j} = -[(1-D_j) \prod (1-C_{i,j,k})], \text{ (Eq.2)}$$

or

success:

$$\underline{S}_{i,j} = 1 [(1-D_j) \prod (1-C_{i,j,k})], \text{ (Eq. 3)}$$

where D_j is the probability of system failure due to random and demand failures as determined in Section 19.3, and $C_{i,j,k}$ is the seismic failure probability, represented by the k^{th} cutset failure probability of the j^{th} system for th i^{th} interval.

The cutset failure probability is given by:

$$C_{i,j,k} = \prod F_{i,j,k,l}, \qquad \text{(Eq.4)}$$

where $F_{i,j,k,l}$ is the l^{th} component's seismically induced failure probability of the k^{th} cutset of the j^{th} system for an acceleration at g_{i+1}.

The component's seismically induced failure probability (from Reference 12) is:

$$F_{i,j,k,l} \cong [1.330274429\ T^5 - 1.821255978\ T^4 + 1.781477937\ T^3 - 0.356563782\ T^2 + 0.31938153\ T] \cdot \frac{1}{\sqrt{2\pi}} \cdot e^{-\frac{y^2}{2}} \quad \text{(Eq.5)}$$

where:

$$y = -\frac{\log\left(\frac{g_{i+1}}{A_l}\right)}{B_l}.$$

g_{i+1} = peak ground acceleration at upper end of i^{th} internal,

A_l = median capacity of component l,

B_l = log standard deviation of component l,

$$T = \frac{1}{(1+.2316419\ Y)}, \text{ and}$$

q_{i+1} is the peak ground acceleration at the upper end of interval i.

Equations 1 through 5 are incorporated into the Integrated Seismic Analysis Program (ISAP). This is a generalized program written to facilitate the evaluation of seismic accident sequences. ISAP has been verified against existing seismic subprograms, and complies with the analysis procedure outlined in NUREG/CR-4431 (Reference 13). As an example, consider an event tree that consists of two systems S1 and S2. (S1 or S2 may also represent a structure.)

An analysis of the supporting fault tree for systems S1 and S2 provides the following cutsets:

System S1
AB
C

System S2
D

where A, B, C and D are basic events in the fault trees. The random or demand failure probability for each system has been given as:

D_{S1} = (2,
and D_{S2} = 0.3

Applying Equations 1 through 4, the probability of an accident sequence where S1 fails and S2 succeeds (S1*$\underline{S2}$) of the i^{th} interval is given by:

$$P(S1\ \underline{S2})_i = [H(g_i) - H(g_{i+1})] \cdot \qquad \text{(Hazard)}$$

$$[1 - (1-0.2)\ (1-F_{i,A}\ F_{i,B})\ (1-F_{i,C})] \cdot \qquad \text{(System S1)}$$

$$[(1-0.3)\ (1-F_{i,D})] \qquad \text{(System S2)}$$

This provides the probability of the accident sequence for the i^{th} interval. The calculation is repeated for each of the n integration intervals, and the sum of the intervals gives the probability of the

accident sequence.

After the accident sequences were quantified, the important accident sequences were identified and grouped into seismic release categories for input to containment event trees and the containment transport analysis.

19.4.3.4.1 Event Tree Analysis

Seismic event trees were constructed for the seismic event. They include the systems needed to achieve safe shutdown. These event trees are included in Appendix 19I. A preliminary screening analysis indicated that the dominant accident sequences resulting from the earthquake would be sequences resulting from loss of offsite power, and that other accident sequences would be relatively insignificant. The reason for this is that the fragility of ceramic insulators in the switchyard is much lower than the fragility of other components and structures.

Referring to Figure 19I.3-1, the first event examined following the earthquake is structural integrity. If the reactor building fails, all safety equipment inside the building is assumed to fail resulting in a core-damaging event. Based on the strength and relatively high fragility of the reactor building, the frequency for this event is very low. For the case where their is no structural failure, the next element of concern is whether or not offsite power is lost. Because of the low fragility of ceramic insulators, there is a relatively high probability that offsite power will be lost (approximately 0.05 over the range of earthquakes). For the cases where there is no structure failure and offsite power is not lost, the frequency of core-damaging accident sequences is negligible.

When offsite power is lost because of the seismic event, the most important subsequent concern is whether or not emergency power and service water are available, since the loss of either presents a serious challenge to the plant. Given a loss of offsite power, the probability of loss of emergency power or service water is approximately 0.02 over the spectrum of earthquakes. The results of loss of offsite power followed by loss of emergency power or service water are depicted and analyzed further on Figure 19I.3-1.

If emergency power or service water are not lost, there are still potentially significant accident sequences. The next concern is whether or not there is a successful scram or alternate insertion of the control rods. Because of the possibility of physical damage to the core as a result of the earthquake, there is a small probability (approximately 0.002) that the control rods cannot be inserted. If control rods are successfully inserted, the event tree analysis is continued on another event tree (Figure 19I.3-2) representing loss of offsite power with successful scram.

If emergency power and service water are available, but the control rods cannot be inserted, the analysis is continued on an event tree representing that combination of conditions (Figure 19I.3-3). For that set of conditions, it is necessary for the operator to inhibit automatic depressurization of the reactor and initiate standby liquid control. For cases where the operator initiates standby liquid control, but cannot inhibit depressurization, the analysis is continued on another ATWS event tree (Figure 19I.3-4).

The major sequences leading directly to core damage are those sequences involving loss of offsite and emergency A.C. power. These sequences contribute 1.80E-7 per year to the total core damage frequency of 2.50E-7 per year.

Sequences resulting from loss of containment heat removal (labeled "II" in the Sequence Class column of the event trees) represent a total frequency of 4.8E-6 per year. These sequences do not result directly in core damage since there are further actions that can be taken to prevent core damage resulting from loss of containment heat removal. These sequences are transferred to and analyzed in the containment event trees presented and discussed in Appendix 19J. The total core damage frequency resulting from loss of containment heat removal is 3.2E-9 per year.

Quantification of the accident sequences in the containment event trees was performed by combining the results of the internal event fault tree analysis and the seismic fault tree analysis integrated over the range of the seismic hazard curve. Quantification was performed using the Integrated Seismic Analysis Program (ISAP) computer program.

19.4.3.4.2 Fault Tree Analysis

System fault trees developed in the events analysis of Section 19.3 were used to provide the probability of random or demand failure following the earthquake and through the time period needed to

achieve shutdown. Seismic fault trees were developed for important systems that contain components having lower fragilities. These systems include HPCF, RCIC, LPCF, RHR (suppression pool cooling), service water, electric power, and firewater. The seismic fault trees are shown in Appendix 19I. Since these fault trees are specifically for evaluation of seismically-induced failures, only those components vulnerable to seismic failure are included in the trees. Because of the assumption of complete dependence, i.e., when one component fails, all like components fail, all divisions fail at the same time and multi-divisional analysis is not necessary. Also because of this assumption, no common-cause failure analysis was needed.

Human errors are not analyzed in the fault trees, but are included in the event trees and are discussed in Appendix 19I.

For the HPCF, RCIC, and LPCF systems, the only essential components having possible vulnerability to the seismic event are the pumps (with motors), piping, and some of the valves. The RHR and service water components of interest are the pumps, piping, valves, and heat exchangers. For the fire water system, in addition to the pumps, valves, and piping, the supply tank was included in the analysis.

In addition to the ceramic insulators, the primary components of interest in the electric power system are the transformers, battery chargers, main diesel generator fuel tanks, and switchgear. Additional items included in the electric power analysis were the diesel generators, motor control centers, batteries, battery racks, and cable trays.

19.4.3.4.3 Component Failure Probability Analysis

Random and demand component failure probabilities were developed and discussed in Section 19.3. These values were not changed for the seismic analysis. Component seismic failure probabilities are functions of the component fragility and the seismic event spectrum. Since the seismic evaluation was performed for each complete accident sequence and integrated over the entire seismic spectrum, discrete component seismic failure probabilities were not developed in the analysis.

19.4.3.4.4 Accident Sequence Analysis

Accident sequences were developed using the event tree discussed in Subsection 19.4.3.4.1 and Appendix 19I. The accident sequences were classified using the basic classes defined in Section 19.3 as shown on Table 19.3-4, where Class I events are transients with loss of core cooling, Class II events are events with loss of containment cooling, Class III events are loss-of-coolant accidents (LOCA), and Class IV events are anticipated transients without scram (ATWS).

The largest contribution to seismic core damage frequency is from the Class I sequences. The total seismic CDF for Class I is 2.35E-7 per year. The Class II total seismic CDF is 4.81E-9 per year. There is no Class III seismic CDF because of the relatively high seismic fragility of the piping and other reactor pressure boundary components. The total Class IV seismic CDF is 7.34E-10 per year.

The top 15 Class I accident sequences contributing the most to core damage frequency are shown on Table 19.4-3. Of these 15 sequences, 13 involve loss of offsite power. The two remaining sequences (numbers 3 and 11 in order) involve structure failure and result directly in core damage without definition of specific equipment failures, since all housed equipment is assumed to fail.

Of the 13 sequences not involving structure failure, 10 involve failure of emergency power or service water, and 6 involve failure to insert control rods. Only one sequence not involving structure failure (number 8) does not involve either loss of emergency power, service water, or failure to scram. That sequence involves seismic loss of all high and low pressure injection.

The total frequency for the 15 top Class I accident sequences, is 2.33E-7 representing 93% of the total CDF. Sequence number 15 represents 0.3% of the total CDF and is 1% of sequence number 1.

19.4.3.4.5 Seismic Accident Subclasses

To perform post-core melt analysis, it is necessary to subdivide the basic accident classes. For the internal events analysis, the accident subclasses are shown in Table 19.3-4 and discussed in Subsection 19.3.2.2. In the seismic analysis, only Classes I, II, and IV have significant core damage frequency, and the subclasses are modified somewhat from those used in the internal events analysis.

The seismic accident subclasses are shown in Table 19.4-4.

19.4.4 Fire Protection Probabilistic Risk Assessment

A fire risk 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. The criterion for screening acceptability is that the risk of core damage from any postulated fire be less than 1.0E-06 per year. Any fire scenarios not meeting this criterion require more detailed consideration.

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-hour 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 to be less than 1.0E-06. The sum of all five was calculated to be 1.3E-06. 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 by at least an order of magnitude. It is therefore estimated that the core damage frequency resulting from fire is less than 1E-07 per year.

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 watertight door would stop water from entering the control building and nonwatertight doors at grade level should allow flood water to exit the building. The core damage frequency (CDF) for turbine building flooding is less than 1.0E-9/year for a plant with low ultimate heat sink (UHS) and approximately 3E-9/year for a plant with high UHS.

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. The CDF for control building flooding is less than 2.0E-9/year.

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 because of watertight doors on each safety division room. As was the case in the control building, floor drains and other openings will direct all flood water 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 pump 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. The CDF for reactor building flooding is less than 3.0E-9/year.

The total CDF for internal flooding is 6.0E-9/year for low UHS and 8.0E-9/year for a high UHS.

19.4.6 ABWR Shutdown Risk

The ABWR design has been evaluated for risks associated with shutdown conditions (i.e., Models 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 events is less than 1.0E-7/year. 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 combination 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

1. *Advanced Light Water Reactor Utility Requirements Document,* Chapter 1; Overall Requirements, June 1986, Electric Power Research Institute.

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

3. *Advanced Light Water Reactor Requirements Document,* Appendix A: PRA Key Assumptions and Groundrules, Final Draft, Issued 10/88, Electric Power Research Institute.

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

5. *Interim External Events Integration for the EPRI ALWR Requirements Document 9WBS 4.3.3),* DOE/ID - 10227, Donald Gene Harrison, January 1989, Advanced Reactor Severe Accident Program, U. S. Department of Energy.

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

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

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

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

10. *GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment,* 22A7007, General Electric Company, March 1982.

11. *Compilation of Fragility Information from Available Probabilistic Risk Assessments,* Campbell, R.D., Ravindra,M.K., and Bahatia, A.,LLNL, September 1985.

12. *Handbook of Math Functions,* U.S. Department of Commerce, June 1965, P. 932. Paragraph 26.2.17.

13. *Summary Report On The Seismic Safety Margins Research Program.* Cummings, G.E., NUREG/CR-4431, UCID-20549, LLNL, January, 1986.

Table 19.4-1

Seismic Hazard Curve - Table of Values

Peak Horizontal Ground Acceleration (g)	Frequency of Exceeding Per Year
0.125	1.00×10^{-3}
0.250	1.86×10^{-4}
0.375	5.03×10^{-5}
0.500	1.56×10^{-5}
0.625	7.02×10^{-6}
0.750	3.11×10^{-6}
0.875	1.56×10^{-6}
1.000	5.15×10^{-7}
1.125	5.38×10^{-8}

Table 19.4-2

Seismic Fragility Summary

Structure/Components	Failure Mode	Capacity[a] (g)	Combined[b] Uncertainty
Reactor bldg shear walls	Shear	2.8	0.45
Containment	Shear	4.3	0.44
RPV Pedestal	Flexural	7.9	0.44
Control Building	Flexural	2.0	0.50
Turbine Building	Flexural	1.0	0.50
Reactor pressure vessel	Skirt anchor bolts	5.3	0.33
Shroud support	Buckling	1.9	0.36
CRD guide tubes	Buckling	1.7	0.36
CRD housings	Plastic yielding	3.9	0.46
Fuel Assemblies	Channel buckling	1.2	0.35
Off-site power	Ceramic insulators	0.3	0.55
Cable trays	Support	2.0	0.60
Batteries and battery racks	Anchorage/LOF	3.0	0.45
Battery chargers/Inverters	LOF	1.3	0.45
Electric equipment (chatter)			
function required during event	Relay chattering	0.8	0.50
function required after event	Relay chattering	2.0	0.50
Panelboards/Instrumentation panels	Functional/Structural	3.0	0.45
Switchgear/Motor control centers	Functional/Structural	2.5	0.45
Transformers	Functional/Structural	1.5	0.45
Diesel generators & support systems	Support	2.5	0.45
Turbine-driven pumps	Anchorage	2.0	0.45
Motor-driven pumps	Anchorage/Impeller defle	1.6	0.45
Heat exchangers/Small tanks	Anchorage	2.0	0.45
Air-operated valves	Stem binding/Air line	3.0	0.60
Motor-operated valves	Operator distortion	3.0	0.60
Safety relief, manual & check valves	Internal damage	3.0	0.60
Hydraulic control units	LOF	2.0	0.50
Accumulators	Support	2.0	0.45
Large flat-bottom storage tanks	Anchorage	0.9	0.45
HVAC ducting	Support	2.0	0.60
Air handling units	Blade rubbing	2.0	0.50
Piping	Support	3.0	0.60
Buried welded steel piping	Buckling/Support	2.0	0.40

Notes:

a. Capacities are in terms of median ground acceleration.

b. Combined uncertainties are composite logarithmic standard deviations of uncertainty and randomness components.

Table 19.4-3 Highest Class I Accident Frequency Sequences

SEQ. NO.	SI	LOP	PW	LPL	C	C4	PA	PC	FCTR	UR	UH	X	V	FA	WI	HX	FREQUENCY
					EVENTS*												
1		X	X		X					X							7.98E-8
2		X	X											X			5.02E-8
3	X																4.71E-8
4		X	X							X				X			1.61E-8
5		X	X		X									X			1.13E-8
6		X	X									X					6.93E-9
7		X	X		X					X						X	4.63E-9
8		X								X	X		X				4.32E-9
9		X		X	X												3.96E-9
10		X	X							X		X					2.34E-9
11	X															X	1.99E-9
12		X	X											X		X	1.82E-9
13		X			X		X	X			X		X				9.53E-10
14		X	X							X				X		X	8.45E-10
15		X	X		X				X							X	7.48E-10

* See event trees for definition of events

Table 19.4-4 Seismic Accident Subclasses

Accident Class	Description
IA	Transients followed by failure of high pressure core cooling and failure to depressurize the reactor.
IB-1	Not used in seismic analysis.
IB-2	Station blackout events with RCIC available for core cooling for approximately eight hours.
IB-3	Not used in seismic analysis.
IC	ATWS events without boron injection coupled with loss of core cooling. Vessel failure at low pressure.
ID	Transients followed by loss of high pressure core cooling, successful depressurization, and loss of low pressure core cooling.
IE	ATWS events without boron injection coupled with loss of core cooling. Vessel failure at high pressure.
II	Transient, LOCA, and ATWS (with boron injection) events with successful core cooling but with possible failure of containment.
IIIA	Insignificant in seismic analysis.
IIID	Insignificant in seismic analysis.
IV	ATWS event without boron injection but with core cooling available.
IV-1	ATWS with one injection pump.
IV-2,3,5	ATWS with multiple injection pumps.

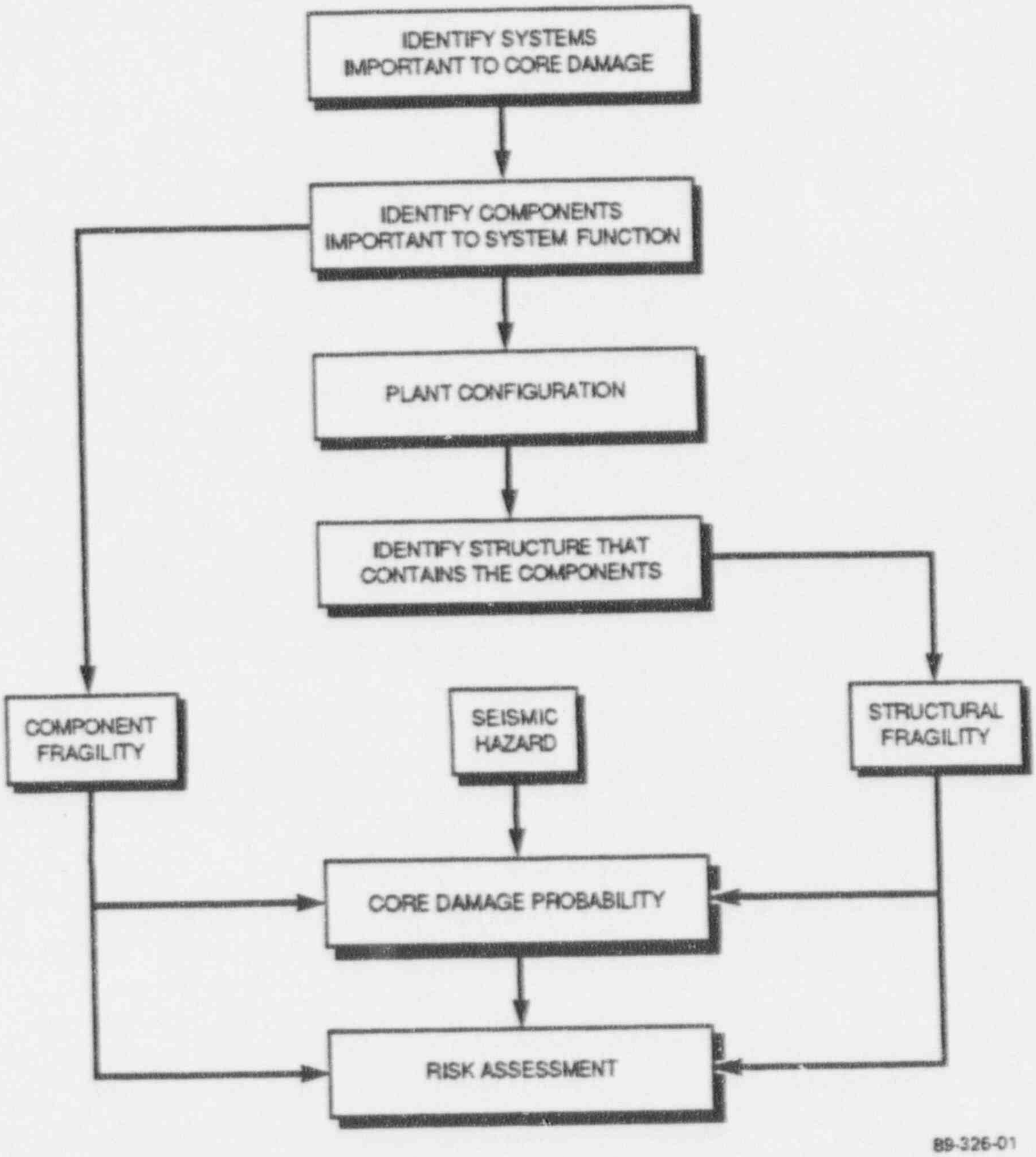
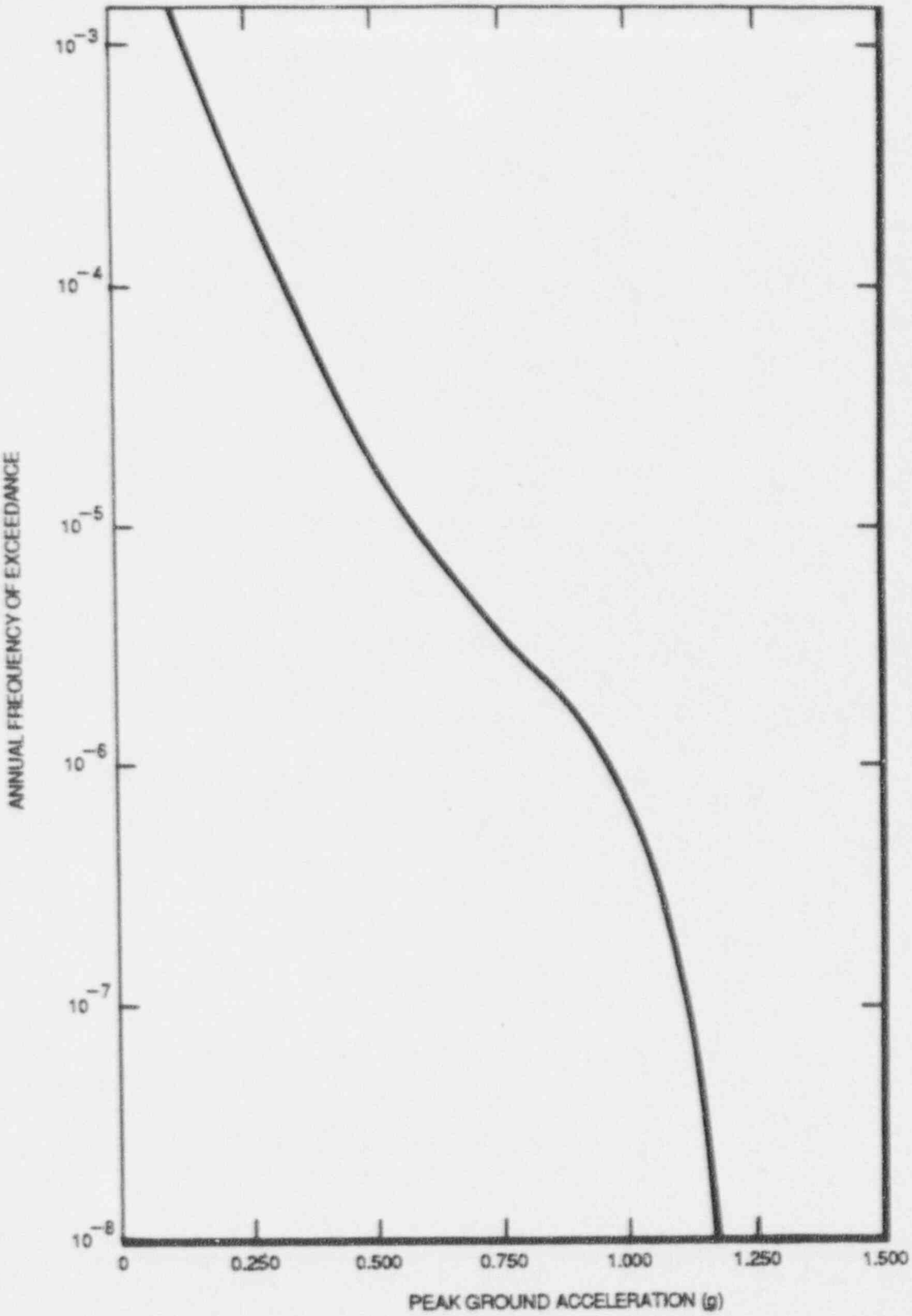


Figure 19.4–1 GENERAL APPROACH TO THE SEISMIC EVENT ANALYSIS



89-326-02

Figure 19.4–2 SEISMIC HAZARD CURVE

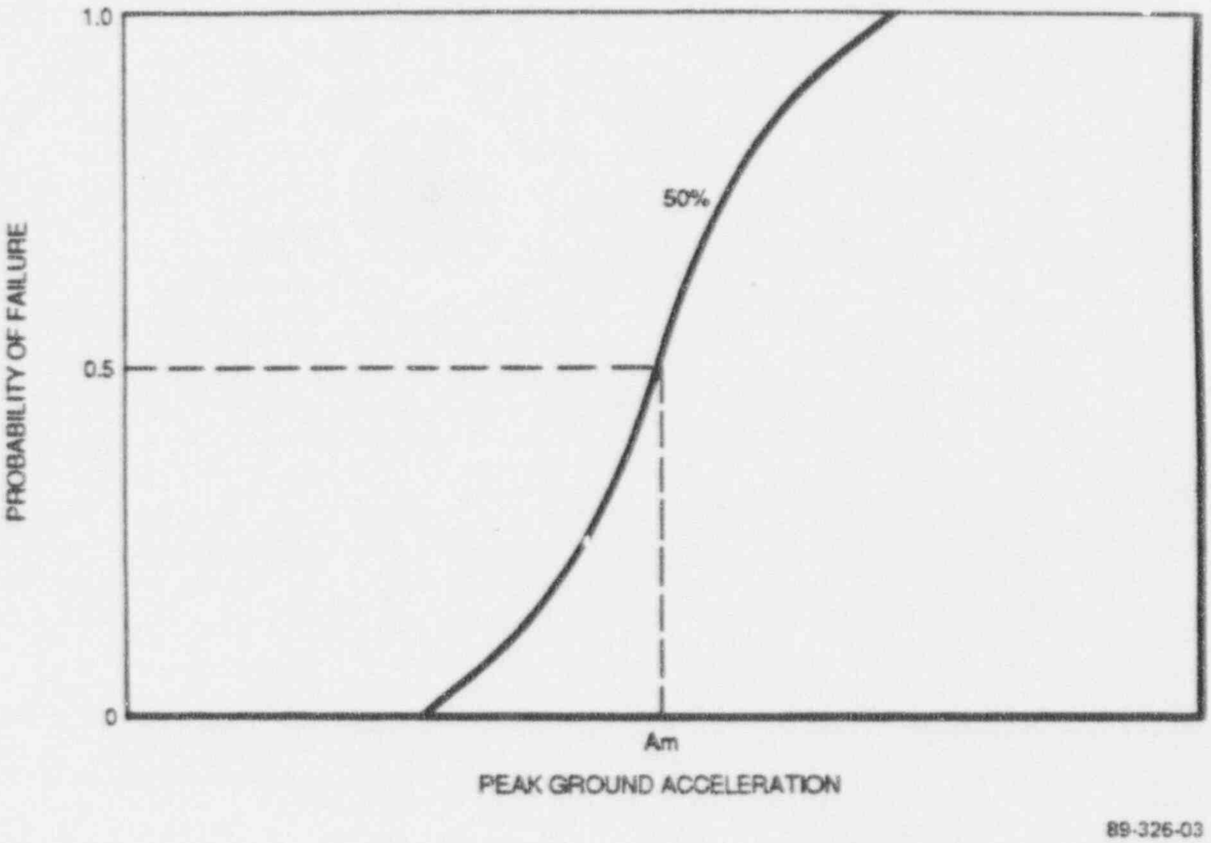


Figure 19.4–3 TYPICAL FRAGILITY CURVE

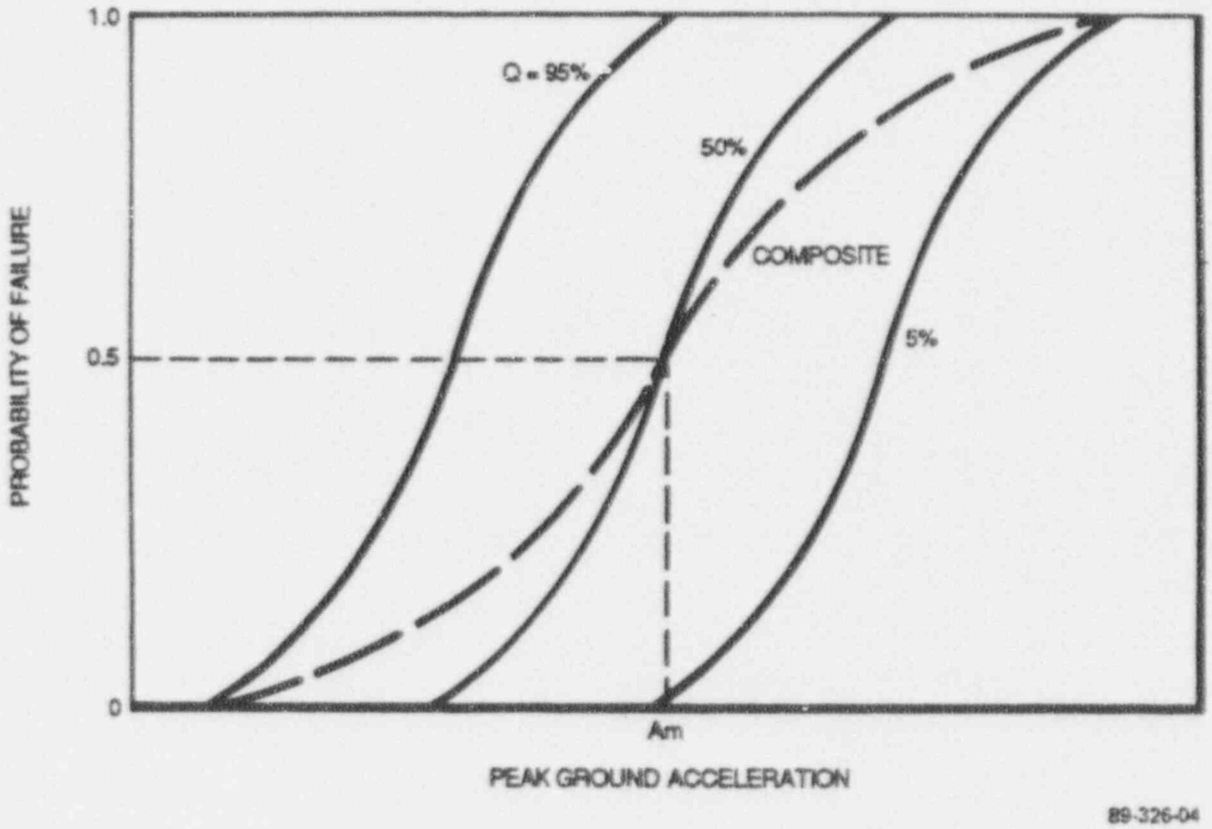


Figure 19.4—4 TYPICAL FAMILY OF FRAGILITY CURVES

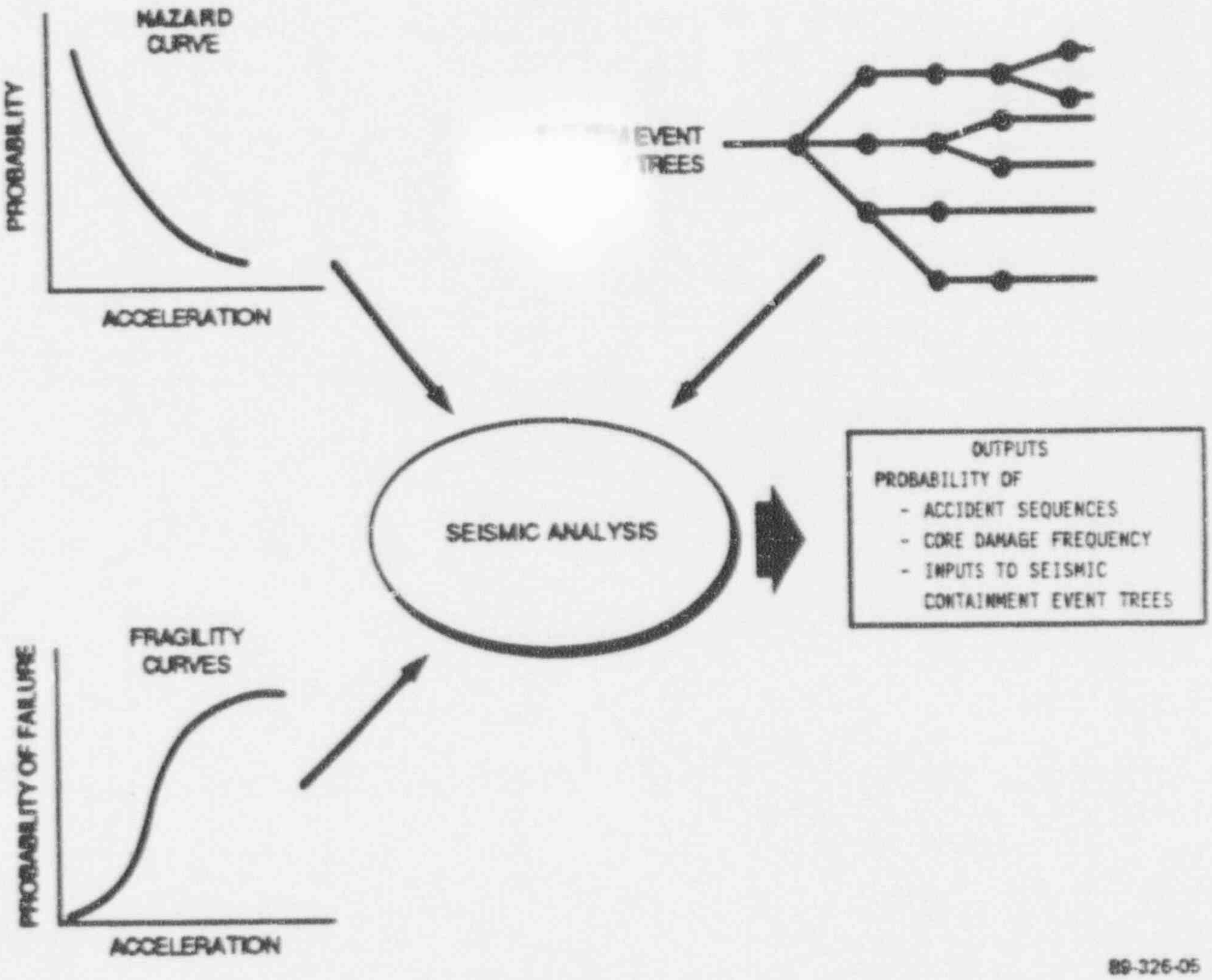


Figure 19.4-5 THE LEVEL I SEISMIC ANALYSIS

SECTION 19.5

CONTENTS

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 detemine 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 three sensitivity studies were performed; one with vessel failure at low pressure, and two with vessel failure at high pressure. In all three studies MAAP was run with both the blockage and eutectic cutoff models disabled.

For the low pressure melt sequence, 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 to decrease from 1.8 hours to 1.1 hours, and caused the time of rupture disk opening to decrease from 20.0 hours to 16.7 hours. The change in the magnitude of fission product release 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 20.0 hours for the high pressure case with passive flooder actuation and LPCI 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 90 psig (0.72 MPa). This assumes 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 (see Subsection 19E.2.4.5).

(2) For cases with leakage through the movable penetrations, 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 mean doses greater than 25 Rem at 1/2 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 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. Examining the data in Table 19.3-6, the CCFP-PI is 0.005. Therefore, the ABWR meets the containment performance goal regardless of the definition of containment failure.

SECTION 19.6

CONTENTS

19.6 MEASUREMENT AGAINST GOALS

This section summarizes the goals established in the ABWR Licensing Review Bases (Reference 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 is 1.6E-7 per year. 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.

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 about 3E-10.

Suppression pool bypass was also identified as being potentially risk significant. If the vacuum breakers or inerting lines 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 7E-11.

The total frequency is about 4E-10, almost three decades below the 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 75 psig that would be generated were this maximum hydrogen production to occur. This is substantially below the Service Level C Limit of 97 psig for the c ainment.

19.6.5 Reliable Heat Removal t 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 three general means of removing heat from the containment: normal heat removal through the main condenser, reactor water cleanup system and residual heat removal system.

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

The reactor water cleanup system can remove decay heat which is generated four hours after scram if the reactor vesse. 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 are automatically initiated by 2-out-of-4 logic for each initiating parameter (low reactor water level or high drywell pressure). Each of the four input signals to the 2-out-of-4 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 pump and one RCW heat exchanger in operation in each of the three loops. Thus each loop provides approximately one half of the total cooling capacity which is sufficient to support the RHR cooling function. When required, the other half of the cooling capability can be added by starting the standby RCW and Service Water pumps and opening one block valve. 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) the core cooling mode, 2) the suppression pool cooling mode, 3) the shutdown cooling mode, or 4) the containment spray mode.

Key components of the supporting RCW and service water systems (one half of each loop's pumps and heat exchangers as noted above) are in operation during normal plant operation. At least each month, the other half of the pumps and heat exchangers are started and the previously running service and sea water 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 Section 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 1.1E-9 per reactor year (Subsection 19.D.5.12). Only 0.1% 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 4% 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 25 rem whole body dose at 1/2 mile from the reactor is less than 10^{-9} per year, less than the 10^{-6} goal. This goal is satisfied. No attempt was made to define the term "large release" but the 25 rem 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 of 0.002 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 less than 0.0001 and that the contribution of this bypass is less than 10% 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 0.0001 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 1/2 mile of 25 rem 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 52 psig and temperature exceeds 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 100 psig at an upper drywell temperature of 500°F.

19.6.8.3 Measurement Against the Goal

From the offsite dose probability plot for (Figure 19E.3-1), the probability of exceeding 25 rem at 1/2 mile is 3E-10. Dividing by the core damage frequency (1.6E-7) gives a conditional containment failure probability of 0.002. This is 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 1.4E-13 and the societal risk is 8.4E-13. These values are about 6 decades below the numerical goals and the goals are satisfied.

19.6.10

Deleted

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

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 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 clad
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 25 rem 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 10 miles 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.

SECTION 19.7

CONTENTS

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. The K6/K7 project which followed that effort embraced in more detail the experience of TEPCO, General Electric, Hitachi and Toshiba. 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 substantually reduced the potential for LOCAs, particulary 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 reliability.

A BWR 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.

(6) 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 resulted in a 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 about one event per million reactor years of operation. 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 virtually eliminate station blackout as a contributor to core damage, decreasing the frequency by an order of magnitude.

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. Since this was a low cost system (several pipes and thermally activated valves) it was added to the ABWR design.

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 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 manu-

ally 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 onsite nonsafety power source to feed permanent nonsafety 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. However, the plant can cope with a station blackout without the CTG.

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 heatup. The fusible plug valves open when the drywell atmosphere (and subsequently the fusible plug valve) temperature reaches 260°C. 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 noncondensable 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 150 psig, contains two rupture disks, in series, which open at a pressure above the design pressure but below the Service Level C capability of the containment. 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. Most of the ac-independent water addition system is seismic Category I and has three pumping sources: ac-driven pump, direct diesel-driven pump, and a fire truck. The balance of the system consists of pipes and manually operated valves which have relatively high seismic capacity compared to many components in conventional safety systems. 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 from 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, noncondensible 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 clean-up operations, it would probably be necessary to increase contain-

ment 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 increased to increase the containment pressure capability. Basaltic concrete was added to the lower drywell to reduce the potential for non condensible gas generation which could result if core damage occurs.

As a result of the fire PRA studies (Appendix 19M) the capability to control automatic depressurization valves 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 detail were developed to reduce the potential for internal flooding to pose a significant threat. These additional features, which are shown in Table 19R.6-2 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 an RSW break, and additional floor drains, sills and doors in the reactor building to prevent floods from having significant impact.

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 non-condensible 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 ingression 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 of the 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 leads 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 being 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 Assumption 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.

SECTION 19.9

CONTENTS

19.9 COL LICENSE INFORMATION

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

The section represents the results of that review.

19.9.1 Event Specific 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 could not be isolated. Attempting to control RPV water level in the normal range 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.

Since this is a very specific event, it was judged inappropriate to complicate the symptom-based Emergency Procedure Guidelines (EPGs) with actions to mitigate the event. An event-specific procedure will be developed by the COL applicant using the following guidance:

1. If a CUW break or leak occurs (as indicated by room sump levels, high flow, temperature indication, radiation level) and successful automatic or manual isolation does not occur (as indicated by lack of closed indication on at least one of the two CUW isolation valves or lack of zero flow indication), the following actions should be taken.

2. Scram and depressurize the RPV if these actions have not occurred automatically. Attempt to close the CUW isolation valves from the main control room. Close the RPV drain line globe valve from the main control room. Control RPV water level in accordance with the EPGs if at least one of the CUW isolation valves is closed. The level should be controlled between the top of the active fuel and 15 inches above the top of the active fuel if drain line closure is not successful. (The RPV drain line connects to the CUW suction line at this elevation). If drain line closure was successful, control water level between the top of the fuel and 5 feet above the top of the fuel. (The CUW suction line is about 6 feet above the top of the fuel). Use the temperature compensated fuel zone and wide range water level indication and pumps which can be throttled (CRD, RHR, condensate pumps).

3. When practical, enter the CUW room and/or the containment and affect the necessary repairs.

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 section 19.3. 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.

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 the closed the watertight door between the turbine and service buildings;

2. Sandbag the external doors to the following:

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. Plug the diesel generator room floor drains to prevent backflow;

4. Shut the plant down; and

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

Any 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 confirmation must be deferred to the COL applicant when sufficient design detail is available. The actions specified in Section 19H.5 will be taken by the COL applicant.

19.9.5 Plant Walkdowns

A plant walkdown to seek seismic vulnerabilities will be conducted by the COL applicant in accordance with EPRI NP-6041 as noted in Section 19H.5.

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 iaddition system (including use of the fire truck) to provide vessel injection and drywell spray. Training will be included in the COL applicant's crew training program.

19.9.8 Actions To Avoid Common Cause Failures In The Essential Multiplexing System (EMUX)

To reduce the potential for significant EMUX common cause failures, (see 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.

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 RCIC can be operated for at least 8 hours. It is expected that compliance with COL license information item 8.3.4.16 will satisfy this need.

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).

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 ANSF/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 (i.e., less than one foot) and also prevent accumulation in one area flowing to another area containing equipment from a different train or division.

   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 analysis for potential flood sources and required mitigation features.

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 interlocks

- 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, 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 make-up 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 if the other two divisions are in maintenance or are potentially subject to a common fire or flood.

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-91-016,

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

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 adminsion 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 steam line 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. Maximize 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.6-11. 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 (Section 19D.7) and checked against the emergency procedure guidelines. Some of these are judged to be sufficiently important to warrant separate COL action items in this section (see Subsections 19.9.1, 19.9.7, 19.9.11). 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 19E.2.2.8 indicate the release to the atmosphere may be substantially decreased by initiating dryell 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 Operating 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.

SECTION 19.10

CONTENTS

19.10 DESIGN AND RELIABILITY 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 interface 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 to be 0.9997 in successfully supporting the ECCS injection function (single train success) and 0.991 in successfully supporting the heat removal function (two train success criterion).

An RSW isolation valve at the discharge of each pump and in the common header line on the discharge from both pumps are assumed to automatically close on a high water level (0.8 meter) in the control building RSW/RCW rooms. In addition, anti-siphon valves are located in the system to ensure that RSW flow will stop when the RSW pumps are tripped and the isolation valves remain open. The reliability assumptions for RSW system components are contained in 19R.6.5.

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 section.

In the event of an RSW line leak in the control building RSW/RCW room, floor 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. For pump tripping

alone to result in termination of the flood, anti-siphon valve(s) should be included in the RSW system design.

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.

19.10.3 Power Cycle Heat Sink Assumptions

These assumptions are noted in Table 19D.4-2. 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 Off-Site Power Assumptions

These assumptions are noted in Subsection 19D.3.1.2.4. A value of 0.1 loss of offsite power events per year was assumed, representing a 90% confidence value. 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, paragraph (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 Off-Site 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 off-site 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 19.3.1.5.2 an overall reliability for fire water injection was taken as 0.9 for transients. This reliability is controlled by operator error rather than equipment availability. It is judged that the following reliability targets (availability on demand), if satisfied will support the injection function assumed in the PRA:

fire truck: 0.9
diesel driven fire water pump: 0.9

These values should be achieved if the actions noted in the PRA input to reliability assurance (Appendix 19K) are included in the reliability assurance program.

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 rugged 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 Section 19.9.

19.11 HUMAN ACTION OVERVIEW

Several functions, previously performed manually, 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 (Section 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 (Section 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 Section 18E.2. These human factors are listed and discussed in Section 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 in the COL applicant action item list in Section 19.9 "COL License Information."

19.12 PRA 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.3-1).

The balance of the PRA was reviewed (Sections 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 (Section 19K.11).

SECTION 19.13

CONTENTS

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 less than 10^{-5} per year), 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 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 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 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 Section 19.9.

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 (Section 19D.10). The level 1 results show a mean core damage frequency of about 1.5E-7 events per year. Thus the effect of data uncertainty is relatively minor. The most important contribution to the uncertainty is that 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 (Section 19D.11). The sensitivity of the results to equipment outage times and surveillance intervals was also considered (Section 19D.9). The contribution of human errors was compared to the contribution from an operating plant and found to be substantially lower.

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 potentail 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 its high strength and lower dryell/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. The lube oil pumps within a given division of a safety related system, such as in the RHR system, are powered by the same electrical division that powers the pumps. 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, as long as the failure rate for a given equipment item includes the failure of its lubricating system.

APPENDIX 19D

TABLE OF CONTENTS

APPENDIX 19D

TABLE OF CONTENTS (continued)

APPENDIX 19D

TABLE OF CONTENTS (continued)

SECTION 19D.3

CONTENTS

SECTION 19D.3

CONTENTS (Continued)

TABLES

ILLUSTRATION

19D.3 INPUT DATA

19D.3.1 Initiating Event Frequencies

Initiating events and expected frequencies used in the ABWR PRA are presented in Table 19D.3-1. These values incorporate the design requirements prescribed in the Advanced Light Water Requirements Document (Reference 1) of a maximum of one anticipated transient per year which results in reactor scram. From the time of the scoping analyses documented in WASH-1400 (Reference 2), four types of initiating events of dominant significance have consistently been identified: 1) shutdowns; 2) transients (scrams); 3) losses of offsite power; and 4) loss of coolant accidents (LOCAs). These are the primary initiating events investigated in this analysis.

19D.3.1.1 Manual Shutdowns

Planned and unplanned manual shutdowns are controlled activities which present very mild challenges to the plant and seldom place demands on any standby safety equipment. These events are included in the analysis because of their frequency and since they do represent changes in operating states which could result in the initiation of accident sequences. On the basis of information provided in a 1985 analysis of operating plant data (Reference 3), a value of one event per year was judged appropriate to present an average of all annual scheduled and unscheduled controlled shutdowns in a mature ABWR plant.

19D.3.1.2 Transients

The transient initiating events included in this analysis encompass all types of unplanned scrams that have been encountered at operating BWRs. The factors in the initiating event that would significantly affect the ensuing sequence of events are the following:

(1) Whether or not the reactor has been isolated due to MSIV closures or failures of turbine bypass valves;

(2) Whether or not feedwater has been lost;

(3) Whether or not offsite power has been lost; and

(4) Whether or not an automatic scram signal is generated.

These four conditions are encompassed by the four transient initiating events used in this analysis:

(1) Non-isolation event;

(2) Isolation/loss of feedwater event;

(3) Inadvertent (stuck) open relief valve; and

(4) Loss of offsite power.

19D.3.1.2.1 Non-Isolation Events

All events that do not result in an isolation of the reactor due to closure of MSIVs or failure of all turbine bypass valves are included in this group. A turbine trip with bypass is selected to represent the plant response for this group due to its relative severity and frequency.

19D.3.1.2.2 Isolation/Loss of Feedwater Event

All events that result in closure of the MSIVs except LOCAs and losses of offsite power are included in this group, as are turbine trip and load rejection transients with assumed failure of all turbine bypass valves. In addition, since the challenge to ECCS for loss of feedwater events is very similar to that for MSIV closure events, these events are also included in this group.

19D.3.1.2.3 Inadvertent (Stuck) Open Relief Valve (IORV)

This event begins with one or more relief valves opening and remaining open while the reactor is under otherwise normal operating conditions. It is the only initiating event considered where there is no immediate automatic scram signal.

19D.3.1.2.4 Loss of Offsite Power

The loss of offsite power event is defined as a complete loss of power from the grid. It requires the startup and use of emergency diesel generators to provide power to plant systems until external power to the plant is recovered. Offsite power loss could be due either to grid failure or failure of plant connec

tions to the grid.

The expected loss of offsite power frequency and outage time distribution were derived from data presented in Reference 4. The total frequency of 0.1 per year for this event represents the upper 90 percent confidence bound. The data, documenting 57 loss of offsite power events, are based upon 732.3 site years of operation.

The remainder of the 1.0 transients per year ABWR design goal (0.9/year) was prorated between the remaining initiators (Non-Isolation, Isolation, and IORV) on the basis of information in Reference 3. Data chosen were the averages presented for the eighth year of BWR operation. These values were selected as being representative of a typical mature BWR since they are the average of data for the eighth year of operation of twenty plants, include 97 scrams, and are typical of other mid-life years.

19D.3.1.3 Loss of Coolant Accidents

Three types of coolant accidents (LOCAs) were considered for the ABWR: LOCAs within containment; LOCAs external to containment; and LOCAs in low pressure piping systems interfacing with high pressure piping systems connected to the reactor pressure vessel (RPV) - - also known as interfacing LOCAs.

19D.3.1.3.1 LOCAs Within Containment

As in a number of previous PRAs, the ABWR evaluation considers three sizes of inside containment LOCAs (small, medium, and large) which are established on the basis of reactor core cooling success criteria. The small break LOCA category represents break sizes characterized by very slow or no vessel depressurization and small gradual inventory loss similar to a transient. The medium break category differs from the small in that reactor coolant inventory is lost at a substantially greater rate such that RCIC has insufficient capacity to maintain water level. For the medium break category, the initial depressurization rate is sufficiently slow to require manual safety relief valve actuation (either ADS or non-ADS) for rapid depressurization. The large break LOCA is of sufficient size to provide rapid depressurization without the use of safety relief valves.

LOCA initiation frequencies corresponding to ABWR success criteria and presented in Table 19D.3-1 are the same data used in the GESSAR II PRA (Reference 5) and are based upon the Reactor Safety Study (Reference 2). Expected frequencies for the three break size categories are a matter of considerable uncertainty. The GESSAR values for the medium and large break categories correspond closely to the WASH-1400 values. Because of the different definitions for the small break LOCA, however, the GESSAR value for the small break category is much smaller than the WASH-1400 value.

For the ABWR analysis, breaks or leaks that are so small that they do not result in reactor trip on high drywell pressure are not considered as LOCAs. By this definition, the expected frequency of small LOCAs is much smaller than the WASH-1400 value.

After reviewing available information in other PRAs, it was judged appropriate to use the GESSAR values in ABWR event tree evaluations.

19D.3.1.3.2 LOCAs External to Containment

The LOCAs external to containment were studied and the results are discussed in Subsection 19E.2.3.3. The risk associated with external LOCAs is calculated to be a very small fraction of the total risk of severe accidents evaluated in the PRA. Therefore, the external LOCAs are not analyzed separately as initiating events for the PRA.

19D.3.1.3.3 LOCAs in Interfacing Systems

Piping and components in the interfacing systems are designed for pressures which are much lower than the normal RPV pressure. However, certain equipment failures or operator errors can subject these pipes to the normal RPV pressure and cause a potential for LOCA in the interfacing system. To prevent such LOCAs, piping in interfacing systems for accepted practical regions has been redesigned on an ultimate rupture strength basis to withstand normal RPV pressure. Therefore, these systems are not expected to rupture and the interfacing system LOCAs, which are discussed in Subsection 19B.2.15, are not a concern for the ABWR plant. So interfacing LOCAs are not analyzed separately as initiating events in the PRA.

19D.3.1.4 Other Potential Initiators

Several other potential initiating events were evaluated for inclusion in the PRA. These events, and the reasons for not including them in the

baseline PRA calculation, are discussed below.

19D.3.1.4.1 Reactor Pressure Vessel Failure

Disruptive failure of the RPV is an event that can be considered as being nearly incredible. There is no basis for estimating any specific probability or expected frequency of occurrence of such an event. Upper 99% confidence limits in order of 10^{-6} to 10^{-7} per year have been postulated but without firm statistical basis, since there has never been a disruptive failure of a reactor pressure vessel.

A survey of existing literature, data and reactor pressure vessel expert opinion yields the following pertinent points:

(1) Section 8 of Appendix XI of the Reactor Safety Study (Reference 6) cites the conclusion of WASH-1318 (Reference 9), that the upper limit (99% confidence) probability of a disruptive RPV failure event in any one nuclear reactor during any service year falls within the range of 10^{-7} to 10^{-6} and the mean value of this probability would be expected to even smaller. This conclusion was based on 725,000 vessel-years of service in U.S. fossil-fueled power plants without a disruptive failure. The authors of WASH-1318 further concluded that "...the estimated failure probability (for nuclear reactor vessels) may be reduced by an additional factor of 10 to 100 based on the detailed investigation of the influence and scheduling of the periodic inspections..." This conclusion was also noted in the Reactor Safety Study.

(2) The "leak before break" phenomenon is a key consideration which justifies a failure rate less than 10^{-7} per vessel year (Reference 10). WASH-1318 further states that the "pressure vessel will have a considerable margin to failure by (a) brittle fracture, even with large postulated initial flaws and (b) that leak-before-break capability is maintained even after a LOCA." This means that long before a crack could propagate to the point that a disruptive failure could occur the crack would propagate through the vessel wall and be detected due to significant leakage. The leak detection system would detect the existence of leaks and allow shutdown of the reactor to avoid propagation of the crack and vessel failure.

(3) Reactor vessels are subjected to periodic inspections, in accordance with Section XI of the ASME Code. This inspection is generally more intensive than that for non-nuclear vessels, and consists of an ultrasonic inspection of weld joints before the vessel goes into service and every 10 years thereafter, supplemented by surface inspections (visual, liquid penetrant test and magnetic particle test).

(4) In resent years, significant amount of research has been conducted in the area of pressure vessel integrity, and the factors relating to material specifications which play a key role in material embrittlement have been identified and well understood. The RPV material specifications and the RPV irradiation levels for the ABWR produce nil ductility temperature shifts that make the potential for nil ductility failures negligible.

(5) Recent GE work on the ABWR design evaluated large RPV bottom head breaks. Structural evaluations showed that loads on equipment and structures were insufficient to cause loss of structural integrity. These results show that the severe accident response for RPV failures would be no worse than for a large break LOCA severe accident.

(6) Reactor vessels are designed with a higher degree of protection from pressure transients and temperature events than are non-nuclear vessels. This higher degree of protection is assured by virtue of design measures, including over-pressure relief devices and operational control procedures.

(7) Reactor vessels are designed and constructed in accordance with Section III of the ASME Code. These rules are more restrictive than the rules of Sections I and VIII, which are used for non-nuclear vessels.

(8) Reactor vessels are operated in accordance with the limitations specified in NRC license technical specifications and no such requirements are imposed on non-nuclear vessels.

Based on the above considerations it is concluded that, while it is not possible to quantify the probability of RPV failure with great precision, the failure

probability of an RPV rupture for the ABWR plant is so low that its explicit inclusion in this analysis would not significantly impact the results. Furthermore, the RPV failure modes that are mechanistically plausible would produce consequences similar to the higher probability LOCA events because of the leak-before-break phenomenon.

19D.3.1.4.2 Loss of Control Room HVAC

The HVAC emergency cooling water (HECW) system delivers chilled water to the control building essential electrical equipment room coolers, the diesel generator zone coolers, and the main control room coolers during shutdown of the reactor, normal operating modes, and abnormal reactor conditions. The HECW system consists of three mechanically separated divisions, A, B and C. Each HECW division provides cooling to the control building essential electrical equipment room and the diesel generator zone in its division. Also, either division "B" or "C" can independently cool the main control room. Power is supplied to each division from independent Class 1E sources.

Division "A" of HECW consists of one pump, one refrigeration unit, instrumentation, and distribution piping and valves to the cooling coils. Divisions "B" and "C" are similar except that two parallel pumps and refrigeration units are used. System configurations for each division are illustrated in Figures 19D.3-1, 2 and 3.

The HECW system is capable of removing all heat loads with four of the five units running and one of the four pump and refrigerator units from divisions "B" and "C" in standby. At any given time the division with two pumps and refrigeration units in operation provides cooling to the main control room. The design philosophy is that if one of the refrigerators or pumps fails in this division, the standby refrigerator will automatically start and provide control room cooling while essential electrical equipment room and diesel generator zone cooling requirements will continue to be met by the remaining refrigerator in the affected division.

Cooling water for the HECW refrigerators is provided by the corresponding division of the Reactor Building Cooling Water (RCW) System which in turn rejects heat through the Reactor Service Water (RSW) System to the ultimate heat sink. Each division of RCW and RSW consists of two parallel trains interfacing through three heat exchangers. RCW and RSW system design capacities are such that one RCW train, one RSW train, and two of the three RSW/RCW heat exchangers in operation are sufficient to successfully meet equipment and control room heat removal requirements.

It is conservatively assumed that loss of an HVAC division results in the loss of essential electrical equipment room and diesel generator zone cooling, which leads to failure of the ECCS equipment of that division. Accordingly, system fault trees modeling the loss of the HVAC function in each division have been incorporated directly into the ABWR functional fault tree and sequence evaluations.

The potential for failure of both control room HVAC divisions as an initiating event and the expected consequences have also been investigated. Frequency of control room HVAC failure was estimated by calculating the product of the operating HVAC division random failure frequency, assuming 80 percent plant capacity factor, and the conditional probability that the standby division will fail to start and run, given failure of the operating division. The operator is certain to notice the loss of control room HVAC and is expected to initiate a manual shutdown, which then becomes the initiating event. The scenario which then follows can be conservatively approximated to that for the control room fire risk screening analysis presented in the Appendix 19M Fire Protection Probabilistic Risk Assessment. Conservatism is introduced by the fact that, following HVAC failure, the operator has time to take many recovery actions, and control systems continue to operate for a significant length of time, whereas in case of a fire, control room functions can be lost in a relatively short time.

The sequence of events is illustrated in the event tree of Figure 19D.3-4. The assumption is made that the only ECCS equipment available to respond to this transient event is that which can be controlled from the remote shutdown panel. This capability includes operation of HPCFB, RHRA, RHRB, and four SRVs for reactor depressurization. In addition, RCIC and the Fire Protection System are available to provide core cooling, since each is capable of remote operation independent of the control room, and thus is unaffected by the HVAC failure.

No information is currently available regarding qualification temperature limits of essential equipment in the control room, the rate of

temperature increase in the control room following HVAC loss, or the nature of and times available to initiate other possible recovery actions. The time available for recovering HVAC is increased by continued operation of the feedwater pumps (Event tree node Q). Consequently, the probability of failing to recover HVAC or to initiate alternate means of achieving adequate control room cooling was conservatively estimated to be 0.1. If feedwater is not available, no credit has been taken for recovery of HVAC.

On the above basis, core damage frequency resulting from the loss of control room HVAC initiating event was calculated to be less than 1.0E-08 per reactor-year. This result represents a conservative estimate within the context of the assumptions and constraints presented, and a more realistic analysis should produce an even lower CDF. Therefore this potential initiator was dismissed from further consideration.

19D.3.1.4.3 Loss of a Single AC or DC Bus

Loss of a single ac or dc bus will not cause a reactor scram. Direct current power is used to supply channel sensors for each division of the reactor protection system. Scram signals are 2-out-of-4 channels of logic, with each of four dc divisional power supplies providing power to one of the four channels. Loss of one division of dc power will result in a trip in only 1-out-of-4 of the channels, therefore the 2-out-of-4 criterion for scram would not be met. Loss of a single dc power division will not result in an RPS scram signal. Therefore, loss of a single dc bus is not analyzed as an initiator in the PRA.

Alternating current power is supplied to plant equipment at and above 480 volts through three divisions. There are four divisions of 120 volt ac power. Division 1 of 480 Volt ac powers the battery charger for Division 4 of 120 Volt ac power, which obtains an uninterruptible power supply from the batteries.

Scram solenoid groups A and B are powered by divisions 2 and 3, respectively, of 120 volt vital ac power. Loss of division 2 or 3 of ac power will not result in scram, since both the A and the B scram solenoids must be deenergized to cause scram insertion of control rods. Loss of power to one set of solenoids would result in a half scram signal but no control rod insertion.

Power from ac buses is also supplied to reactor internal pumps (RIPs), to control rod drive (CRD) pumps, to feedwater, condensate and circulating water pumps, to service water and cooling water pumps, to HVAC system pumps, to cleanup water pumps, to drain pumps, to vacuum pumps, to battery chargers, to building fans, to gas compressors, and to many other electrical panels and components during plant operation. Loss of a single ac power bus could result in power loss to as many as three RIPs. This would result in reactor power reduction to approximately 90%, at which point power operation could continue without scram.

Loss of a single bus of ac power to any of the other equipment noted above will result in local impact on the system but little, if any, impact on plant operation. Systems impacted are designed with redundancy in that major components, such as feedwater pumps and CRD pumps, are each powered by different ac power divisions, load groups and/or buses. For example, each of three feedwater pumps is on a separate 6.9kV bus, each of three circulating water pumps is on a separate 6.9kV bus, each of four condensate pumps is on a separate 6.9kV bus, each of two CRD pumps is on a separate 6.9kV bus. The five normal cooling water refrigerators are distributed among these separate 6.9kV buses, each of three turbine building service water pumps is on a separate 6.9kV bus, and each of three turbine building cooling water pumps is on a separate 6.9kV bus. Similarly, reactor building cooling water pumps 1A and 1D (loop A) are on one 6.9kV bus, pumps 1B and 1E (loop B) are on a different 6.9kV bus, and pumps 1C and 1F (loop C) are on a third 6.9kV bus. The four HVAC emergency cooling water (HECW) system refrigerators associated with control room cooling are on two different 480V buses so that loss of one bus will not disable all control room cooling.

Loss of a single component by loss of power will not cause scram and usually will not result in power reduction. Loss of a single feedwater pump or condensate pump may result in a temporary power reduction, but full power can be achieved with the remaining two pumps. In some cases (such as loss of one circulating water pump on a hot day), a reduction in power might result.

Standby equipment that requires ac power, such as pumps for the residual heat removal (RHR) system and the high pressure core flooder (HPCF) system, is not normally operating and would not

react directly to a loss of ac power. Thus, loss of a single bus of ac power to such equipment would not result in scram.

Other equipment supplied by ac or dc power either has adequate redundancy to avoid scram or is not related to scram signals. No single bus failure, ac or dc, will generate a scram signal or cause control rods to scram.

19D.3.1.4.4 Loss of One Division of the Reactor Service Water System

Loss of a single division of the reactor service water (RSW) system would not cause scram.

The RSW system is divided into three divisions which cool the three divisions of the reactor building cooling water (RCW) system. The RCW system cools a number of plant components. The RSW does not directly cool any components other than the RCW system.

Each of two RCW divisions cools five of the 10 reactor internal pumps (RIPs). Loss of either of these RCW divisions will cause its five RIPs to runback and trip from high RIP motor cooling water temperature. The ABWR is not to be licensed to operate with fewer than seven RIPs operating, so loss of five pumps would require shutdown. Thus, loss of one RSW division that provides cooling to five RIPs through the RCW division will lead to reactor shutdown, but not to a scram. Manual reactor shutdown has been analyzed (Subsection 19D.3.1.1).

The RCW system and the heating, venting and air conditioning (HVAC) normal chilled water system are used to cool the drywell. Drywell cooling is accomplished by three coolers which are in turn cooled by two RSW and two RCW divisions. In the most severe loss of a single RSW division two of the drywell coolers would be uncooled.

A study was made of single failure of equipment (loss of a division of both the RSW and RCW systems) during a loss of off-site power (LOOP). Drywell temperature increased from 57 to 75 C eight hours after LOOP. For components in the drywell the upper limit for long term exposure is 80 C, so there would be no component damage requiring or causing scram for loss of a single RSW division. There would also be ample time for operator action to restore drywell cooling. The pressure increase resulting from the calculated drywell heating would not be high enough to cause scram. It is concluded that a single RSW division failure would not result in drywell conditions leading to scram.

Some of the equipment cooled by the RCW system does not require cooling during normal operation. This includes the residual heat removal (RHR) system, the reactor core isolation cooling (RCIC) system, the emergency diesel generators, the flammability control system (FCS), the high pressure core flooder (HPCF) system, and the standby gas treatment system (SGTS). For these items the loss of one division of RCW would not cause or require scram.

None of the other equipment cooled by the RCW system has functional requirements such that loss of one RCW division (or one RSW division) would result in reactor scram. Such equipment includes the heating, ventilating and air conditioning (HVAC) system, the cleanup water (CUW) system, the fuel pool cooling (FPC) system, the containment air monitoring (CAM) system, the high conductivity waste (HCW) system, and the instrument air and service air systems. These systems operate during plant operation, but they have adequate redundancy to assure that plant operation can continue without scram in event of loss of cooling from one RCW division.

Therefore, loss of a single RSW system division is not analyzed as an initiator in the PRA.

19D.3.1.4.5 Reactor Vessel Water Level Instrumentation Failure

Failure of a single water level instrument or instrument channel would not cause scram.

The reactor pressure vessel (RPV) water level instruments are provided with four independent and separate divisions, electrically and mechanically. Water level signals are combined in 2-out-of-4 logic for reactor scram, so a failure in or of a single channel can only result in a half scram signal, with no control rod insertion.

Partial loss of drywell cooling as result of a failure of one of the three plant service water systems could result in an increase of drywell temperature from 57 to 75 C. This temperature increase would affect reading on all four water level instruments,because of piping inside the drywell, but not so much that a low water level trip would result.

Therefore, failure of a single water level instrument is not analyzed as an initiator in the PRA.

19D.3.1.4.6 Turbine Building Closed Cooling Water System

Total loss of the turbine building closed cooling water (TCW) system can result in scram. The TCW system provides cooling for the generator stator cooling water, and loss of this TCW cooling will lead to higher temperatures of the stator cooling water. As the stator temperature increases it will first give a high temperature alarm. The operator will act to reduce reactor power and will try to reestablish TCW system operation. If corrective action is not prompt and effective, the stator temperature will soon reach its high temperature trip and cause the generator to trip. This load rejection event will result in turbine trip and, at high power, the turbine trip will cause scram. Turbine trip is included in the basic initiating event frequency.

The TCW system has multiple pumps. If only one of these pumps is lost, the standby pump will start and there will not be transient temperature increases sufficient to result in equipment trips or scram. As noted in Subsection 19D.3.1.4.3, loss of a single ac bus will result in loss of no more than one TCW pump. Because of this redundancy of active TCW system equipment, the probability that the entire system could be disabled is low.

A review of BWR scram data shows that over a period of 150 plant years of commercial operation, beginning in 1984, U.S. BWRs suffered 425 automatic scrams. None resulted from failures in the turbine cooling water system. This further emphasizes the low probability of total loss of the ABWR TCW system.

Total loss of TCW system cooling would impact cooling for other systems that are required for continued plant operation. This includes the feedwater pump adjustable speed drives, condensate pump drives, circulating water pump drives, transformers, and the isolated phase bus duct. Such cooling losses would require prompt operator action to either shut the plant down or restore TCW system operation. Otherwise, reactor scram could soon occur from high or low RPV water level, from turbine trip caused by low vacuum, from electrical system troubles, or from gradual loss of instrument air pressure.

Loss of TCW system cooling would have no impact on any plant safety equipment. Such equipment is cooled by the reactor building cooling

water (RCW) system. Thus, loss of the TCW system would result in no reduction in plant safety.

Even if a scram were to occur, the net effect would be a negligible increase in the frequency of events already included in the PRA. Therefore, total loss of the TCW system is not analyzed as an initiator in the PRA.

19D.3.1.4.7 Trip of Circulating Water Pumps

As part of internal flood protection in the turbine building, instrumentation has been added to trip the circulating water pumps when a flood is detected in this building. Reliable instrumentation built with redundancy is provided to assure a high probability of tripping the pumps on demand while assuring a low frequency of inadvertent trip of the pump. The probability that this instrumentation will fail to trip pumps is 1.7×10^{-4}/year, including common mode failure.

An inadvertent trip of all circulating water pumps could cause scram because of loss of vacuum in the main condenser. It is estimated that the frequency of such trips is 1.8×10^{-4}/year, which is smaller than similar initiating events already included in the internal event PRA. It is therefore concluded that no separate analysis is needed for including the reactor trip initiated by the inadvertent trip of a circulating water pump.

19D.3.1.4.8 Loss of Instrument Air

Total loss of instrument air for a prolonged duration will result in reactor scram because air pressure is required to keep scram valves closed on the control rod drives. Loss of pressure allows scram valves to open, and rods will be driven into the core by hydraulic pressure. Instrument air pressure is also required to keep MSIVs open, so loss of air pressure would cause MSIV closure which would also result in scram.

Instrument air is supplied by the lead air compressor which operates as needed to maintain pressure in a large accumulator. The lead compressor is normally off, and it starts and takes up load when accumulator pressure drops. If the lead compressor fails to start or to continue operation, the second, standby compressor will start automatically and operate to supply system needs. The operators are alerted by alarm if the lead compressor fails to start on demand so they can take corrective action.

If both compressors fail, pressure in the system will decay through air leakage and eventually cause a low pressure alarm to signal need for operator corrective action. Operators can manually connect the service air system (which has two compressors) to replace any failed instrument air compressors. If leakage continues, eventually the scram valves will open and scram will occur. Also, MSIVs would close because of low air pressure and give a scram signal. If operators recognize that they will not be able to restore air pressure before scram, they may manually shut the reactor down without scram.

In event of a loss of off-site power, the instrument air compressors are automatically switched to the combustion turbine generator bus for power. They can be manually transferred to the diesel generator bus, if necessary.

Nitrogen is used to open the safety/relief valves (SRVs) of the automatic depressurization system (ADS), so complete loss of air would not impact SRVs or the ADS.

Because of the backup compressor and the large accumulator, the loss of ABWR instrument air is expected to be a low probability event. Experience from operating BWRs shows that loss of instrument air caused, from 1983 to 1987, fewer than 0.05 scrams per plant year ("A Risk-Based Review of Instrument Air Systems at Nuclear Power Plants", NUREG/CR-5472). Average U.S. BWR scram frequency during that period was greater than three scrams per year. Since 1987, efforts at scram frequency reduction have significantly reduced scram frequency from all causes. Scram frequency for U.S. BWRs in 1992 was below 1.3 scrams per year. For the ABWR the frequency of scrams resulting from the loss of instrument air is expected to be very small (< 0.02 per year).

The loss of instrument air event is very similar to an isolation event (which is analyzed in the PRA with an event frequency of 0.20 per year). Because of this, and since no other safety system needed for mitigation (such as ADS) is degraded significantly by loss of air, this event is judged to be already included in the isolation event analysis. It is therefore

concluded that no separate analysis is needed for including in the PRA the reactor trip initiated by the loss of instrument air.

19D.3.2 Generic Component Data

Applicable component failure rate data accompany each system fault tree presented in Section 19D.6. These data are primarily the values recommended for use by the General Electric Failure Rate Data Manual (Reference 6). They have been collected from a number of primary sources and have been screened and modified appropriately for application to the ABWR analyses. All of the values represent mean values and have been used in the analyses as single-point best estimate values.

19D.3.3 Human Error Probabilities

Human error probabilities used in this analysis are presented in the applicable component failure rate data tables which accompany each system fault tree presented in Section 19D.6. They were taken predominately from the GESSAR II PRA for which they were collected from various other sources and modified, as appropriate, for the GESSAR application. Most of these values were derived from the Swain and Guttman handbook (Reference 7). More recent studies suggest that these values may be somewhat conservative.

19D.3.4 Maintenance and Test Unavailabilities

Equipment maintenance or test unavailabilities

used in the initial ABWR PRA submittal were taken from the GESSAR PRA and were based upon BWR experience. In subsequent discussions with NRC regarding applicability of the GESSAR 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 two percent in the PRA model as shown in Table 19D.3-2. The final calaulated CDF of 1.56E-07 reflects inclusion of these values. Sensitivity of CDF to ECCS T&M outage times is summerized in Section 19D.9.

19D.3.5 Recovery of Offsite Power and Diesel Generator Restoration

Table 19D.3-3 represents the conditional probability of failing to recover offsite power and restoring one diesel generator as a function of time, given LOOP and DG failure to start and run on demand. The offsite power recovery probabilities were developed from data presented in Reference 4. Diesel generator values were taken from Reference 8.

19D.3.6 References

1. *Advanced Light Water Reactor Requirements Document, Appendix A: PRA Key Assumptions and Groundrules,* August, 1988 Draft, Page D4, Electric Power Research Institute.

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

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

4. *Losses of Offsite Power at U.S. Nuclear Power Plants - All Years Through 1986,* NSAC-111, May 1987, EPRI Nuclear Safety Analysis Center.

5. Amendment No. 11 to *GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant Probabilistic Risk Assessment,* Docket No. STN 50-447, December 3, 1982, General Electric Company.

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

7. *Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications,* Final Report, NUREG/CR-1278, A.D. Swain and H.E. Guttmann, August, 1983.

8. *Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants,* NUREG/CR-1362, March, 1980, EG&G Idaho, Inc.

9. *Analyses of Pressure Vessel Statistics From Fossil-Fueled Power Plant Service and Assessment of Reactor Vessel Reliability in Nuclear Power Plant Service*, USAEC Regulatory Staff; May 1974, WASH-1318.

10. *Status of Boiling Water Reactor Structural Integrity Programs at the General Electric Company,* E. Kiss and T. L. Gerber, *Nuclear Engineering and Design*, Vol. 59, No. 1, August 14, 1979.

Table 19D.3-1

Initiating Event Frequencies

Initiating Event		Frequency Per Reactor Year
Manual Shutdown		1.0
Isolation/Loss of Feedwater		0.18
MSIV Closure	0.04	
Loss of Condenser Vacuum	0.05	
Press. Reg./Bypass Valves Closed	0.01	
Loss of Feedwater	0.08	
Non-Isolation Event (Trip with bypass)		0.62
Inadvertent (Stuck) Open Relief Valve		0.10
Loss of Offsite Power		0.10
Less than 30 Minutes	0.0579	
30 Minutes to 2 Hours	0.0246	
2 to 8 Hours	0.0158	
Greater than 8 Hours	0.0017	
Small LOCA		0.0012
(Liquid Break <0.00545 ft^2)		
(Steam Break <0.3 ft^2)		
Medium LOCA		0.00067
(Liquid Break 0.00545 ft^2 to 0.3 ft^2)		
Large LOCA		0.00021
(Liquid Break 0.3 ft^2 or greater)		
(Steam Break 0.3 ft^2 or greater)		

Table 19D.3-2

ABWR MAINTENANCE AND TEST UNAVAILABILITIES

RCIC	0.02
HPCFB	0.02
HPCFC	0.02
RHRA	0.02
RHRB	0.02
RHRC	0.02

Table 19D.3-3

RECOVERY OF ELECTRIC POWER

Conditional Probability of Not Recovering Offsite Power or/and Not Restoring One Diesel Generator for ABWR

T (Hr)	Time Interval (Hours)	Failure to Recover Offsite Power Before Time T: Complementary Cumulative	Failure to Recover Offsite Power Before Time T: Conditional	Failure to Recover One Diesel Generator Before Time T: Complementary Cumulative	Failure to Recover One Diesel Generator Before Time T: Conditional
0.5	0-0.5	0.421	0.421	0	0
2.0	0.5-2.0	0.175	0.416	0.66	0.66
8.0	2.0-8.0	0.0175	0.100	0.23	0.35

T (Hr)	Time Interval (Hours)	Failure to Recover Either Offsite Power or One Diesel Generator Before Time T: Complementary Cumulative	Failure to Recover Either Offsite Power or One Diesel Generator Before Time T: Conditional
0.5	0-0.5	0.421	0.421
2.0	0.5-2.0	0.116	0.276
8.0	2.0-8.0	0.004	0.034

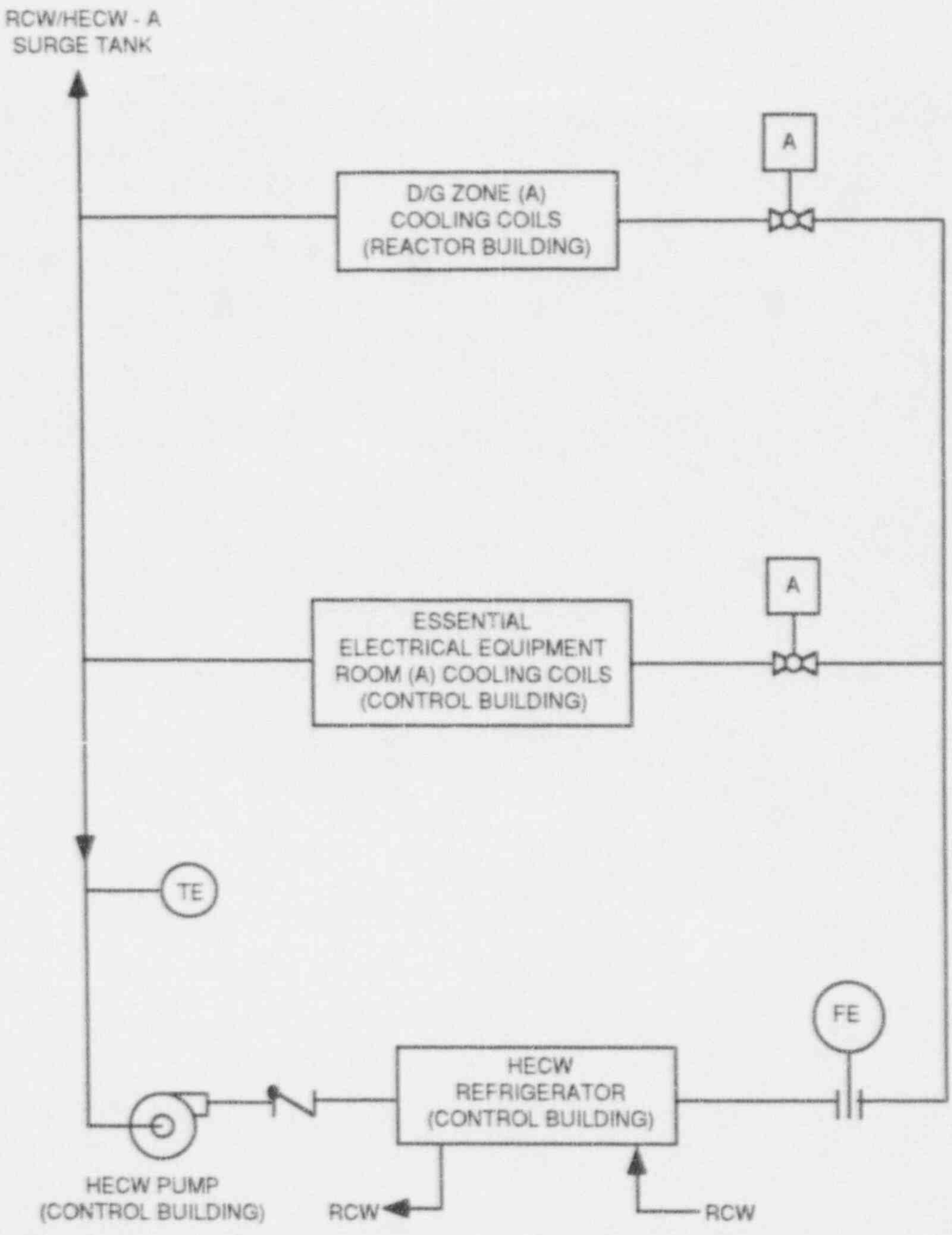


Figure 19D.3-1 HECW DIVISION A

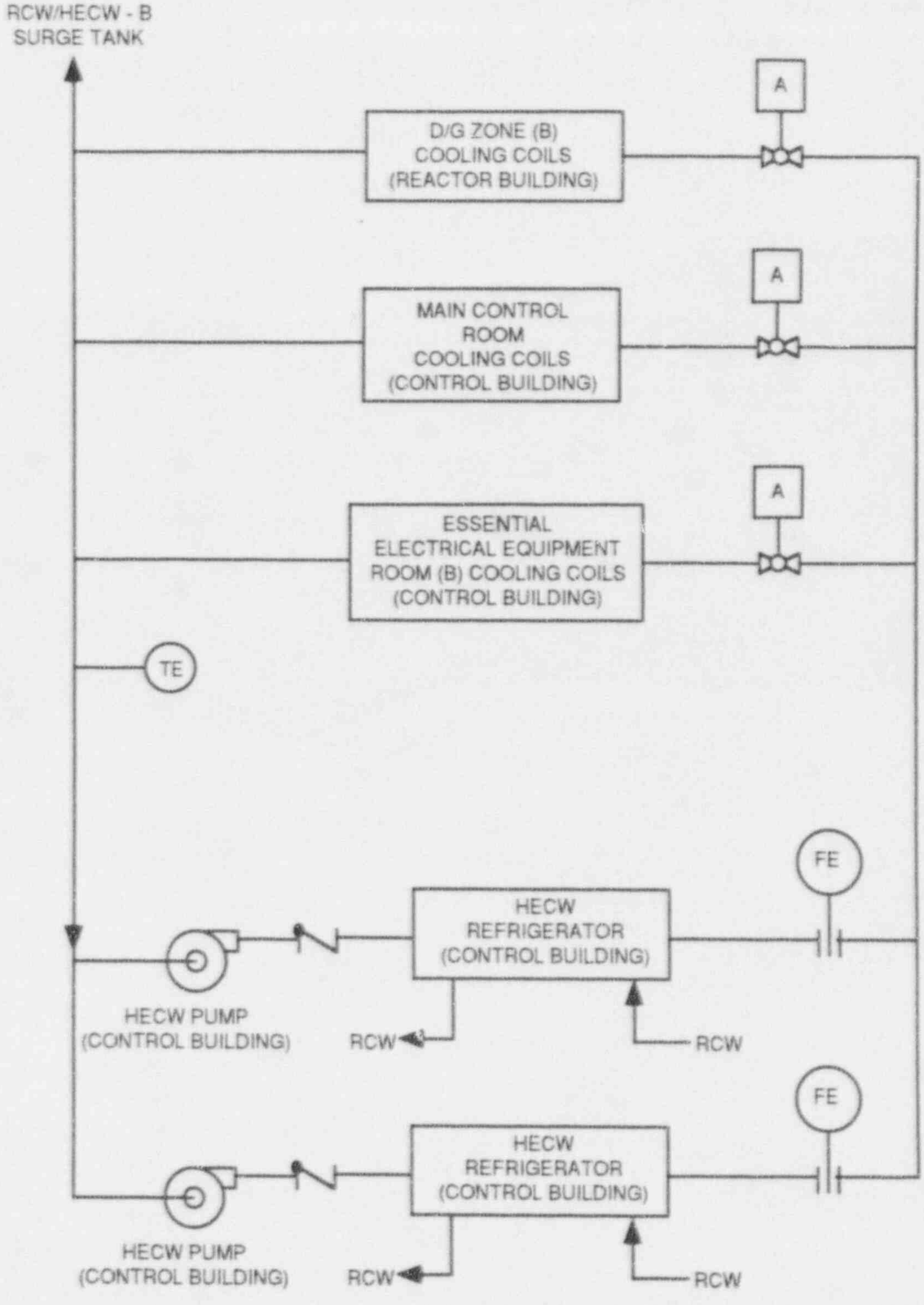


Figure 19D.3-2 HECW DIVISION B

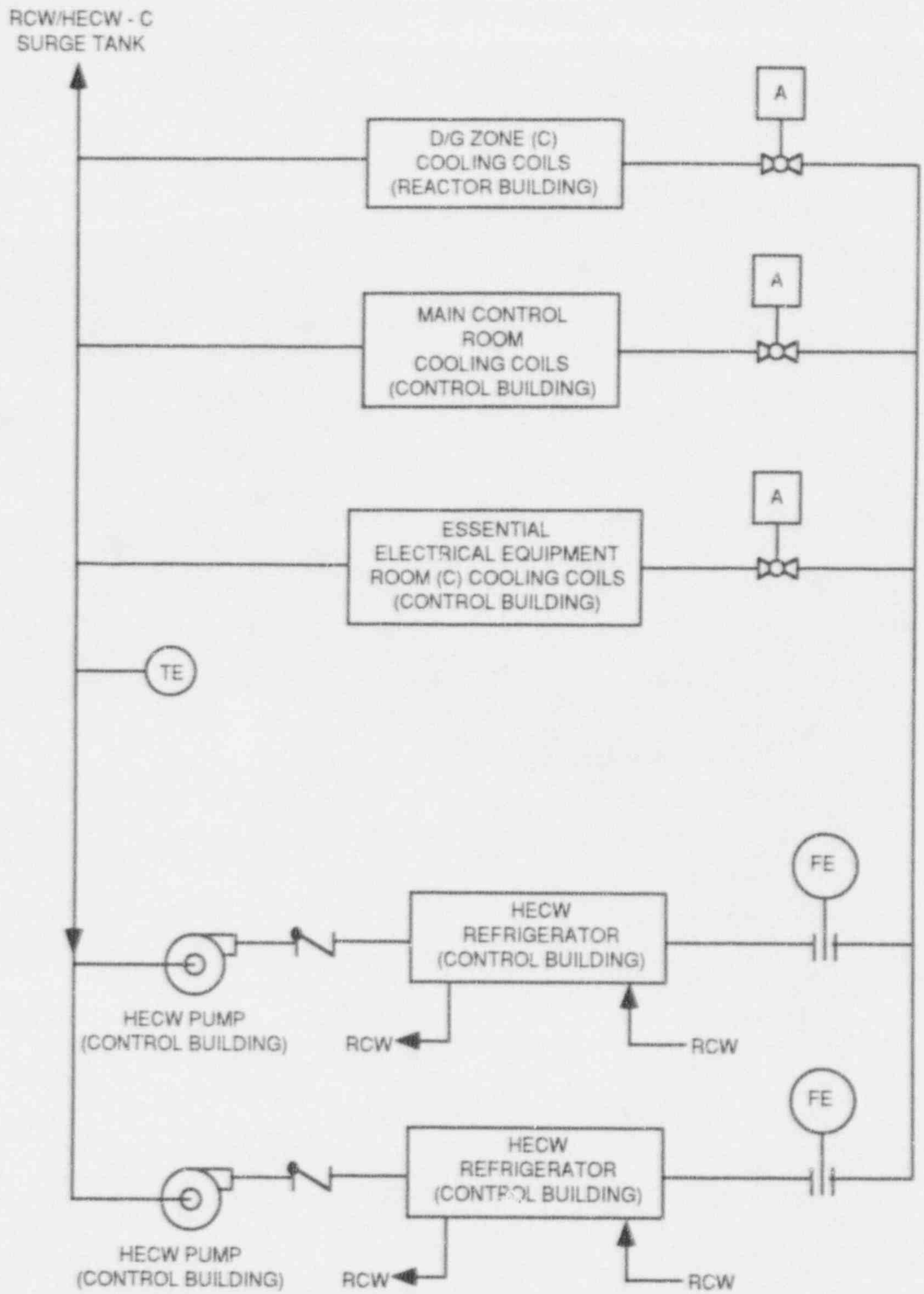


Figure 19D.3-3 HECW DIVISION C

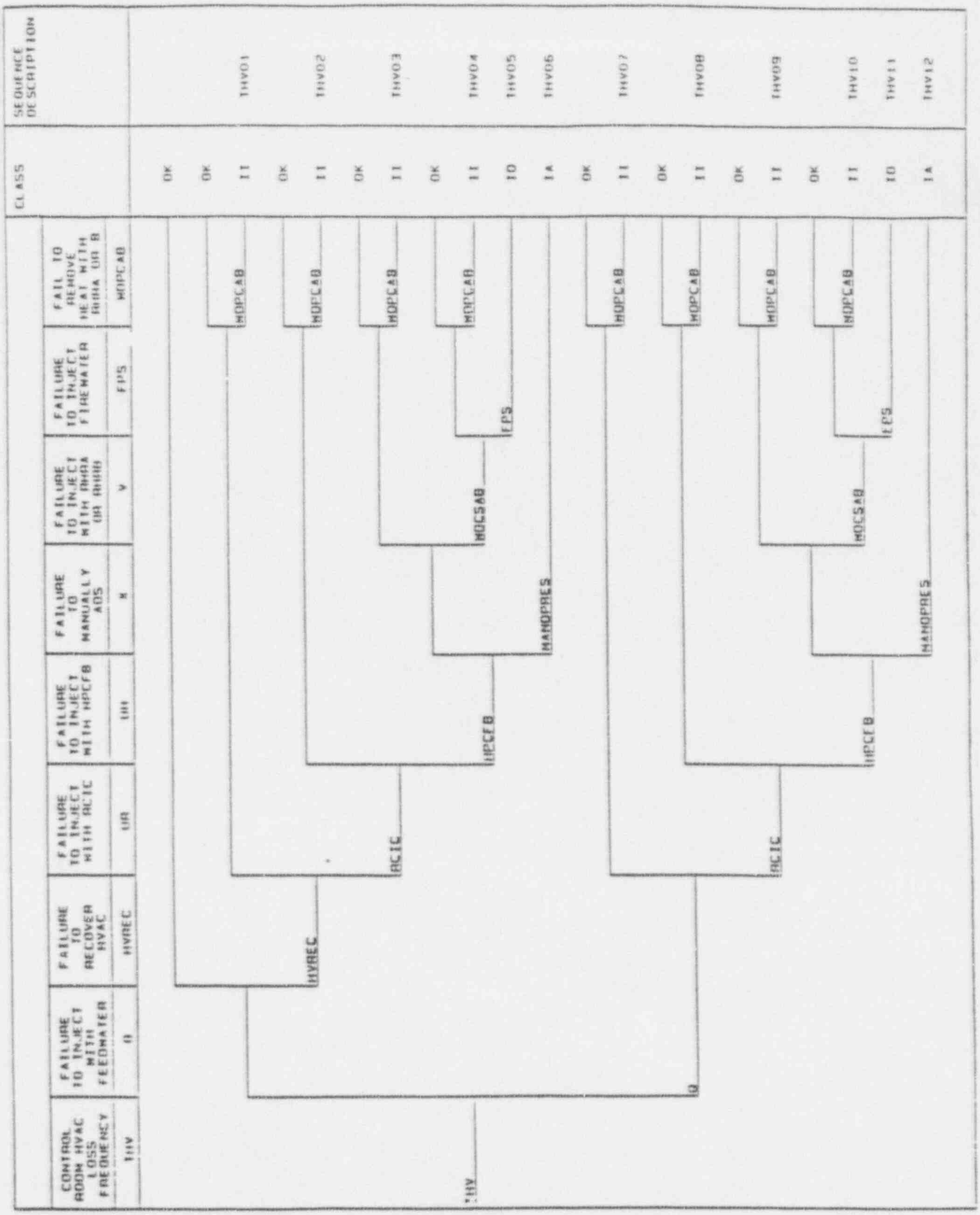


Figure 19.3-4 LOSS OF CONTROL ROOM HVAC

SECTION 19D.4

CONTENTS

SECTION 19D.4

CONTENTS (Continued)

TABLES

SECTION 19D.4

TABLES (Continued)

ILLUSTRATIONS

SECTION 19D.4

ILLUSTRATIONS (continued)

19D.4 ACCIDENT EVENT TREES

19D.4.1 Accident Event Tree Analysis

19D.4.1.1 Introduction

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 Section 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. Tabulations of failure probabilities and references for system or functional requirements defined by event tree branches but not derived from Section 19D.6 fault trees are presented in tabulations accompanying each tree. Accident event tree sequences which lead to core damage or loss of heat removal are further treated in the containment event trees of Section 19D.5 to determine frequencies of radioactive releases to the environment.

19D.4.1.2 Accident Event Tree General Description

For purposes of illustration, consider Figure 19D.4-1, the event tree for the reactor shutdown initiating event. The initiating event frequency is given as the first branch in 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 final two columns labeled "CLASS" and "PROB" document the outcome of each accident sequence. The first column contains the classification of each sequence; either successful termination (OK), core damage (Classes I and III), or a sequence which is developed further in another accident tree or transferred to the appropriate containment event tree (such as Classes II and IV). The final column contains the yearly frequency of occurrence of each sequence. Each value is the product of the initiating event frequency and the linked solution of Section 19D.6 fault trees and other estimated performance probabilities of the system functions identified in the sequence.

19D.4.1.3 Safety Functions and Success Criteria

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. The necessary combinations of minimum system requirements were established on the basis of best estimate predictions. Success criteria are provided in Subsection 19.3.1.3.1.

19D.4.1.4 Branch Point Probabilities

Event tree branch point values were defined in most cases by the system fault trees documented in Section 19D.6. In the remaining cases, values were estimated on the basis of available operating plant performance data or engineering judgement. Tables 19D.4-1 and Tables 19D.4-3 through 19D.4-17, which accompany specific accident event trees, provide bases for core cooling and heat removal branch point values other than those derived by evaluating the fault trees of Section 19D.6.

These trees, including support systems, and other estimates were linked with initiating event frequencies as defined by each sequence, and each sequence equation solved directly to obtain contribution to core damage frequency. These sequence outcomes were then summed to obtain total core damage frequency.

19D.4.1.5 Accident Sequence Classification

As stated previously, the consequence of each accident sequence may be either successful termination (i.e., the achievement of adequate core cooling and containment heat removal) or core damage. Each sequence that results in core damage or inadequate containment heat removal is assigned to an accident class for further consequence evaluation in the containment event trees of Section 19D.5. The bases for sequence classification are discussed in detail in that section.

19D.4.1.6 ATWS and LOCA Sequence Treatment

Sequences leading to either ATWS or LOCA events are not processed to their final dispositions in the primary transient event trees, and consequently are transferred to other event trees for further development. For example, in the non-isolation event tree, Figure 19D.4-2, the bottom sequence representing a failure to scram (ATWS) is routed to the ATWS event tree, Figure 19D.4-15, for additional treatment. In these sequences, the event tree name (ATWS in this example) is indicated in the accident class column.

19D.4.1.7 Accident Sequence Evaluation

The frequency of each accident sequence is developed by initiating event in the event trees of Figures 19D.4-1 through 19D.4-15. Sequence outcomes are summed by assigned accident class and these totals are then routed to the appropriate containment event trees for further analysis of containment and related systems response. Table 19D.4-2 provides a summary of accident event tree results by initiating event and accident class.

19D.4.2 Event Tree Descriptions

This subsection provides a description of each of the event trees developed and illustrated in Figures 19D.4-1 through 19D.4-15. Each tree is accompanied by a table which provides event tree branch point probablities and references where values were not derived from Section 19D.6 fault trees.

19D.4.2.1 Reactor Shutdown

The event tree for reactor shutdown is presented in Figure 19D.4-1. Reactor shutdown includes any event in which the reactor is shut down under normal operating conditions. Not all of the primary safety functions are required for this event. By definition, reactivity control is not a required function for mitigating this event. Neither is the reactor pressure control function required since there is no rapid pressure increase associated with this event. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-3.

19D.4.2.2 Non-Isolation (Turbine Trip)

The event tree for a non-isolation trip is shown in Figure 19D.4-2. Non-isolation events include any event in which the turbine is tripped and removed from the steam loop, but the condenser and feedwater remain available. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-4.

19D.4.2.3 Isolation/Loss of Feedwater

The event tree for isolation of loss of feedwater events is shown in Figure 19D.4-3. Isolation events are those events in which the reactor is isolated from the power conversion system, resulting in loss of the feedwater and the condenser. Loss of feedwater events are absorbed into this event category, as MSIV closure represents the bounding case for loss of feedwater. Although the feedwater and condenser are initially lost, there is a probability that they will be recovered. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-5.

19D.4.2.4 Loss of Offsite Power and Station Blackout Event Tree

Loss of offsite power event trees describes the progression of events in which all sources of offsite electric power are lost. Station blackout event trees describe events in which all station diesel generators and the combustion turbine generator, in addition to all sources of offsite electric power, are lost. Figure 19D.4-4 provides the basis for subdividing the overall loss of offsite power initiator between loss of offsite power only and station blackout scenarios and into time intervals of interest. Sequence outcomes are transferred to other event trees as indicated for additional processing. If offsite power is recovered within one half hour, no core damage will have occurred, and the event is very

similar to a reactor shutdown event. Therefore, this outcome is transferred to Figure 19D.4-1. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-6.

19D.4.2.5 Loss of Offsite Power for 30 Minutes to Two Hours

The event tree for loss of offsite power for 30 minutes to two hours is presented in Figure 19D.4-3. This event includes any scenario for which no external power is available to the plant for two hours following its loss. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-7.

19D.4.2.6 Loss of Offsite Power for Two to Eight Hours

The event tree for loss of offsite power from two to eight hours is shown in Figure 19D.4-6. This event tree includes any scenario for which no external power is available to the plant for two to eight hours following its loss. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-8.

19D.4.2.7 Loss of Offsite Power for More Than Eight Hours

The event tree for loss of offsite power for more than eight hours is shown in Figure 19D.4-7. This event tree includes any scenario for which no external power is available to the plant for more than eight hours. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-9.

19D.4.2.8 Station Blackout for Less Than Two Hours

Figure 19D.4-8 presents the event tree for station blackout for less than two hours. This event tree includes any scenario for which neither external power nor station diesel or combustion turbine generator power are available to the plant for two hours following the loss of offsite power. For this situation, RCIC is the only injection system available for core cooling. The heat removal function is not impaired, since its operation is not required prior power to restoration. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-10.

19D.4.2.9 Station Blackout for Two to Eight Hours

The event tree for station blackout from two to eight hours is presented in Figure 19D.4-9. This event tree includes any scenario for which neither external power nor station diesel or combustion turbine generator power are available to the plant for two to eight hours following the loss of offsite power. Initially RCIC is the only injection system available for core cooling. DC power for control will be available, since battery life is expected to be at least eight hours without any AC power to the battery chargers. In addition, the RCIC pump and turbine are expected to operate for at least eight hours without room coolers. Given successful RCIC operation for eight hours, the remaining injection systems become available upon the recovery of power. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-11.

19D.4.2.10 Station Blackout for More Than Eight Hours

The event tree for station blackout greater than eight hours is shown in Figure 19D.4-10. This event tree includes any scenario for which neither external power nor station diesel or combustion turbine generator power are available to the plant for more than eight hours. All sequences are conservatively assumed to lead to core damage but are discriminated by accident class since timing and consequences differ. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-12.

19D.4.2.11 Inadvertent Open Relief Valve (IORV) Event Tree

The event tree for IORV is shown in Figure 19D.4-11. This event includes those scenarios which begin with one or more relief valves opening and remaining open while the reactor is under otherwise normal operating conditions. Since a stuck open relief valve will eventually result in depressurization of the reactor, the reactor pressure control function is not required. This initiating event is the only one considered where there is no immediate automatic scram signal. Eventually, an automatic scram signal will be initiated by high drywell pressure should the operator have failed to initiate scram based on high suppression pool temperature. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-13.

19D.4.2.12 Small Break LOCA Event Tree

The small break LOCA event tree is shown in Figure 19D.4-12. A small LOCA as defined under the success criteria of Section 19.3.1.2 is a liquid break of area less than 0.005 square feet or a steam break of area less than 0.3 square feet. Similar to an IORV sequence, this event does not require the reactor pressure control function for successful mitigation. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-14.

19D.4.2.13 Medium Break LOCA Event Tree

The event tree for a medium break LOCA is shown in Figure 19D.4-13. This accident is defined as a liquid break between 0.005 and 0.3 square feet in area. This sequence, also similar to an IORV, does not require the reactor pressure control safety function. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-15.

19D.4.2.14 Large Break LOCA Event Tree

The event tree for a large LOCA is shown in Figure 19D.4-14. A large LOCA is defined as a liquid or steam break having an area greater than 0.3 square feet. The initiating event frequency for this event includes transfers from those sequences in other event trees where the SRVs do not open on demand thus causing large water or steam breaks. The event does not require the reactor pressure control safety function. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-16.

19D.4.2.15 ATWS Accident Sequence Event Tree

The event tree for ATWS is shown in Figure 19D.4-15. The initiating event frequency for this tree is the sum of those sequences in other event trees in which the control rods are not inserted by either the RPS or the alternate rod insertion (ARI) system. Based upon the ATWS success criteria defined in Section 19.3.1.2 and in light of the low frequency of ATWS initiators, a single ATWS event tree for an isolation event is adequate to conservatively assess ATWS. Event tree input not derived from Section 19D.6 fault trees is documented in Table 19D.4-17.

Table 19D.4-1

BASES FOR CORE COOLING AND HEAT REMOVAL FUNCTION EVENT TREE BRANCH INPUTS NOT DERIVED FROM APPENDIX 19D.6 FAULT TREES

	Symbol	Description	Value
1.	Q	Feedwater Unavailability Following a Transient (1 FW Pump + 1 Condensate Pump + 1 Cond. Transfer Pump) It is estimated that 50% of the time feedwater pumps will trip on high water level. In the event of loss of feedwater, failure to manually recover at least one pump train is estimated to be 0.1.	5.0E-02
2.	V2	Failure to Recover 1 Condensate and 1 Cond. Transfer Pump Similar to the preceeding estimate, but at low pressure, failure to manually recover at least one pump train is estimated to be 0.1. For loss of offsite power events, a diesel generator bus transfer is required and assumed.	1.0E-01
3.	W1	Normal Heat Removal (NHR) The NHR failure probability of 1.0E-02 is taken from Secton D.1.5 of GESSAR and is a conservative application due to the improved reliability expected from the use of motor-driven feedwater pumps.	1.0E-02
4.	W2	Reactor Water Cleanup System (RWCU) The RWCU system is capable of removing decay heat at high RPV pressures if return water bypasses the regenerative heat exchanger. Failure to manually activate this alternate heat removal system is taken to be 0.1.	1.0E-01

Table 19D.4-2

ABWR INTERNAL EVENT PRA CORE DAMAGE FREQUENCY SUMMARY

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INITIATING EVENT	ACCIDENT CLASS											
	IA	IB-1	IB-2	IB-3	IC	ID	II	IIIA	IIID	IV	TOT.CDF	PERCENT
TM	1.10E-08	---	---	---	---	5.02E-10	9.79E-13	---	---	---	1.15E-08	7.37E+00
TT	6.45E-09	---	---	---	---	3.45E-10	3.67E-11	---	---	---	6.83E-09	4.38E+00
TIS	1.61E-08	---	---	---	---	8.61E-10	1.57E-11	---	---	---	1.70E-08	1.09E+01
TE2	4.44E-09	---	---	---	---	2.65E-11	4.55E-13	---	---	---	4.47E-09	2.86E+00
TE8	2.86E-09	---	---	---	---	1.45E-11	9.95E-13	---	---	---	2.88E-09	1.84E+00
TEO	5.75E-10	---	---	---	---	1.06E-09	5.46E-11	---	---	---	1.69E-09	1.08E+00
BE2	1.56E-12	---	---	---	---	6.67E-08	---	---	---	---	6.67E-08	4.27E+01
BE8	---	2.57E-08	---	---	---	---	---	---	---	---	2.57E-08	1.65E+01
BEO	---	---	1.62E-08	8.86E-10	---	---	---	---	---	---	1.71E-08	1.10E+01
TIO	1.09E-09	---	---	---	---	1.54E-10	9.25E-13	---	---	---	1.24E-09	7.98E-01
S2	---	---	---	---	---	---	---	2.45E-10	1.02E-11	---	2.55E-10	1.64E-01
S1	---	---	---	---	---	---	---	1.42E-10	2.00E-10	---	3.42E-10	2.19E-01
SO	---	---	---	---	---	---	---	---	9.02E-11	---	9.02E-11	5.78E-02
ATWS	---	---	---	---	2.79E-13	---	2.24E-16	---	---	2.70E-10	2.70E-10	1.73E-01
TOT. CDF	4.25E-08	2.57E-08	1.62E-08	8.86E-10	2.79E-13	6.97E-08	1.10E-10	3.87E-10	3.00E-10	2.70E-10	1.56E-07	1.00E+02
PERCENT	2.72E+01	1.65E+01	1.04E+01	5.68E-01	1.79E-04	4.46E+01	7.07E-02	2.48E-01	1.93E-01	1.73E-01		1.00E+02

Table 19D.4-3

BRANCH POINT VALUES FOR REACTOR SHUTDOWN EVENT TREE NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
Tm	Event Frequency	Table 19D.3-1	1.0579 (1.0 +Transfer from Fig. 19D.4-4)
Q	Failure to inject with feedwater	Table 19D.4-1	5.0E-02
V2	Failure to inject with condensate	Table 19D.4-1	1.0E-01
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02
W2	Failure to actuate RWCU	Table 19D.4-1	1.0E-01

Table 19D.4-4

BRANCH POINT VALUES FOR NON-ISOLATION EVENT TREE

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
Tt	Event Frequency	Table 19D.3-1	0.62
PO	Failure of SRVs to open	GESSAR, Table D.1.2-1	1.0E-06
PC	Failure of SRVs to reclose	GESSAR, Table D.1.2-1	3.0E-03
Q	Failure to inject with feedwater	Table 19D.4-1	5.0E-02
V2	Failure to inject with condensate	Table 19D.4-1	1.0E-01
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02
W2	Failure to actuate RWCU	Table 19D.4-1	1.0E-01

Table 19D.4-5

BRANCH POINT VALUES FOR ISOLATION/LOSS OF FEEDWATER EVENT TREE

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
Tis	Event Frequency	Table 19D.3-1	0.18
PO	Failure of SRVs to open	GESSAR, Table D.1.2-1	1.0E-06
PC	Failure of SRVs to reclose	GESSAR, Table D.1.2-1	3.0E-03
Q(1)	Failure to inject with feedwater	Table 19D.4-1	0.43
V	Failure to inject with condensate	Table 19D.4-1	1.0E-01
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02
W2	Failure to actuate RWCU	Table 19D.4-1	1.0E-01

Note

1. 40% of the initiating event frequency (Tis) represents loss of feedwater events. By definition, Q = 1.0 for such events and thus:

$$Q = 0.4(1.0) + 0.6(0.05) = 0.43$$

Table 19D.4-6

BRANCH POINT VALUES FOR LOSS OF OFFSITE POWER AND STATION BLACKOUT EVENT TREE NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
Te	Event Frequency	Table 19D.3-1	0.10
E1	Failure to recover offsite power within 30 minutes	Table 19D.3-2	0.421
EDG-CTG	Failure of three diesel generators and combustion turbine generator	NEDE-22056, GE Failure Rate Man.	4.00E-05
PO1	Failure of SRVs to open (scram)	GESSAR, Table D.1.2-1	1.0E-06
PO2	Failure of SRVs to open (no scram)	GESSAR, Table D.1.2-1	1.0E-04
E20	Failure to recover offsite power within 2 hours	Table 19D.3-2	0.416
E2D	Failure to recover one diesel generator within 2 hours	Table 19D.3-2	0.66
E80	Failure to recover offsite power within eight hours	Table 19D.3-2	0.10
E8D	Failure to recover one diesel generator within 8 hours	Table 19D.3-2	0.35

Table 19D.4-7

BRANCH POINT VALUES FOR LOSS OF OFFSITE POWER EVENT TREE

(RECOVERY TIME 30 MINUTES TO 2 HOURS)

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
TE2	Event Frequency	Figure 19D.4-4	2.46E-02
V2	Failure to inject with condensate	Table 19D.4-1	1.0E-01
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02
W2	Failure to actuate RWCU	Table 19D.4-1	1.0E-01

Table 19D.4-8

BRANCH POINT VALUES FOR LOSS OF OFFSITE POWER EVENT TREE

(RECOVERY TIME TWO TO EIGHT HOURS)

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
TE8	Event Frequency	Figure 19D.4-4	1.58E-02
PC	Failure of SRVs to reclose	GESSAR, Table D.1.2-1	3.0E-03
V2	Failure to inject with condensate	Table 19D.4-1	1.0E-01
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02
W2	Failure to actuate RWCU	Table 19D.4-1	1.0E-01

Table 19D.4-9

BRANCH POINT VALUES FOR LOSS OF OFFSITE POWER EVENT TREE

(RECOVERY TIME GREATER THAN EIGHT HOURS)

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
TE0	Event Frequency	Figure 19D.4-4	1.75E-03
PC	Failure of SRVs to reclose	GESSAR, Table D.1.2-1	3.0E-03
V2	Failure to inject with condensate	Table 19D.4-1	1.0E-01

Table 19D.4-10

BRANCH POINT VALUES FOR STATION BLACKOUT EVENT TREE

(RECOVERY TIME 30 MNUTES TO 2 HOURS)

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
BE2	Event Frequency	Figure 19D 4-4	1.22E-06

Table 19D.4-11

BRANCH POINT VALUES FOR STATION BLACKOUT EVENT TREE

(RECOVERY TIME TWO TO EIGHT HOURS)

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
BE8	Event Frequency	Figure 19D.4-4	4.46E-07
PC	Failure of SRVs to reclose	GESSAR, Table D.1.2-1	3.0E-03

Table 19D.4-12

BRANCH POINT VALUES FOR STATION BLACKOUT EVENT TREE

(RECOVERY TIME GREATER THAN EIGHT HOURS)

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
BE0	Event Frequency	Figure 19D.4-4	1.62E-08
PC	Failure of SRVs to reclose	GESSAR, Table D.1.2-1	3.0E-03

Table 19D.4-13

BRANCH POINT VALUES FOR INADVERTENTLY OPEN RELIEF VALVE EVENT TREE

NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
Tio	Event Frequency	Table 19D.3-1	1.0E-01
Q	Failure to inject with feedwater	Table 19D.4-1	5.0E-02
V2	Failure to inject with condensate	Table 19D.4-1	1.0E-01
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02

Table 19D.4-14

BRANCH POINT VALUES FOR SMALL BREAK LOCA EVENT TREE
NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
S2	Event Frequency	Table 19D.3-1	1.2E-03
V2	Failure to inject with condensate	Table 19D.4-1	1.0E-01
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02
W2	Failure to actuate RWCU	Table 19D.4-1	1.0E-01

Table 19D.4-15

BRANCH POINT VALUES FOR MEDIUM BREAK LOCA EVENT TREE NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
S1	Event Frequency	Table 19D.3-1	6.7E-04
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02

Table 19D.4-16

BRANCH POINT VALUES FOR LARGE BREAK LOCA EVENT TREE NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
S0	Event Frequency	Table 19D.3-1	2.1E-04
W1	Failure to restore normal heat removal	Table 19D.4-1	1.0E-02

Table 19D.4-17

BRANCH POINT VALUES FOR ATWS ACCIDENT SEQUENCE EVENT TREE NOT DERIVED FROM SECTION 19D.6 FAULT TREES

Symbol	Description	Reference	Value
ATWS	Event Frequency	Transfer from Fig. 19D.4-2,-3,-4,-11, -12,-13,-14	1.0E-08
LPL	Level and pressure control failure	GESSAR, Table D.1.1-1	1.0E-02
PC	SRVs fail to close	GESSAR, Table D.1.2-1	1.0E-01
Q	Feedwater unavailable	Table 19D.4-1	5.0E-02

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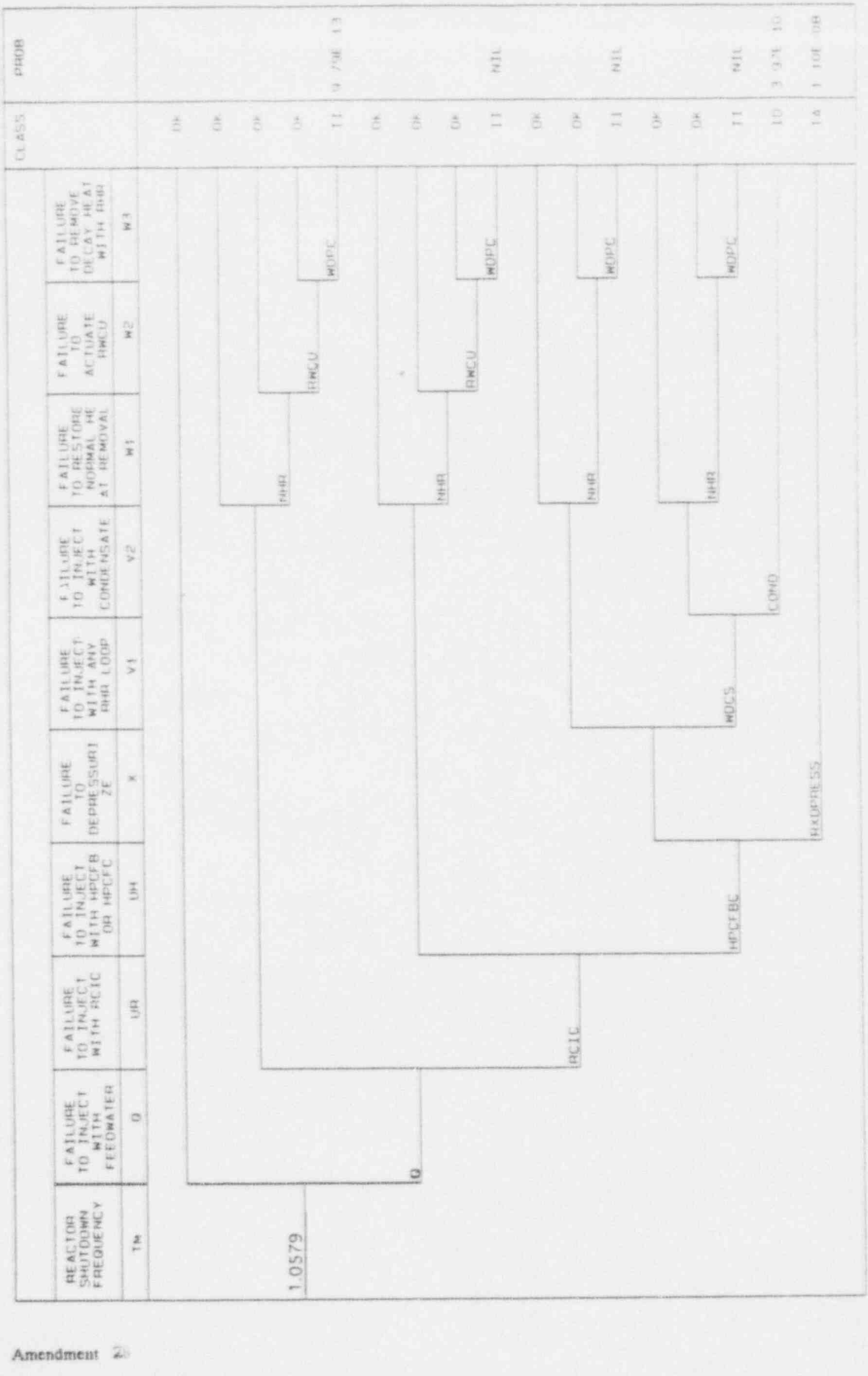


Figure 19D.4-1 Reactor Shutdown Event Tree

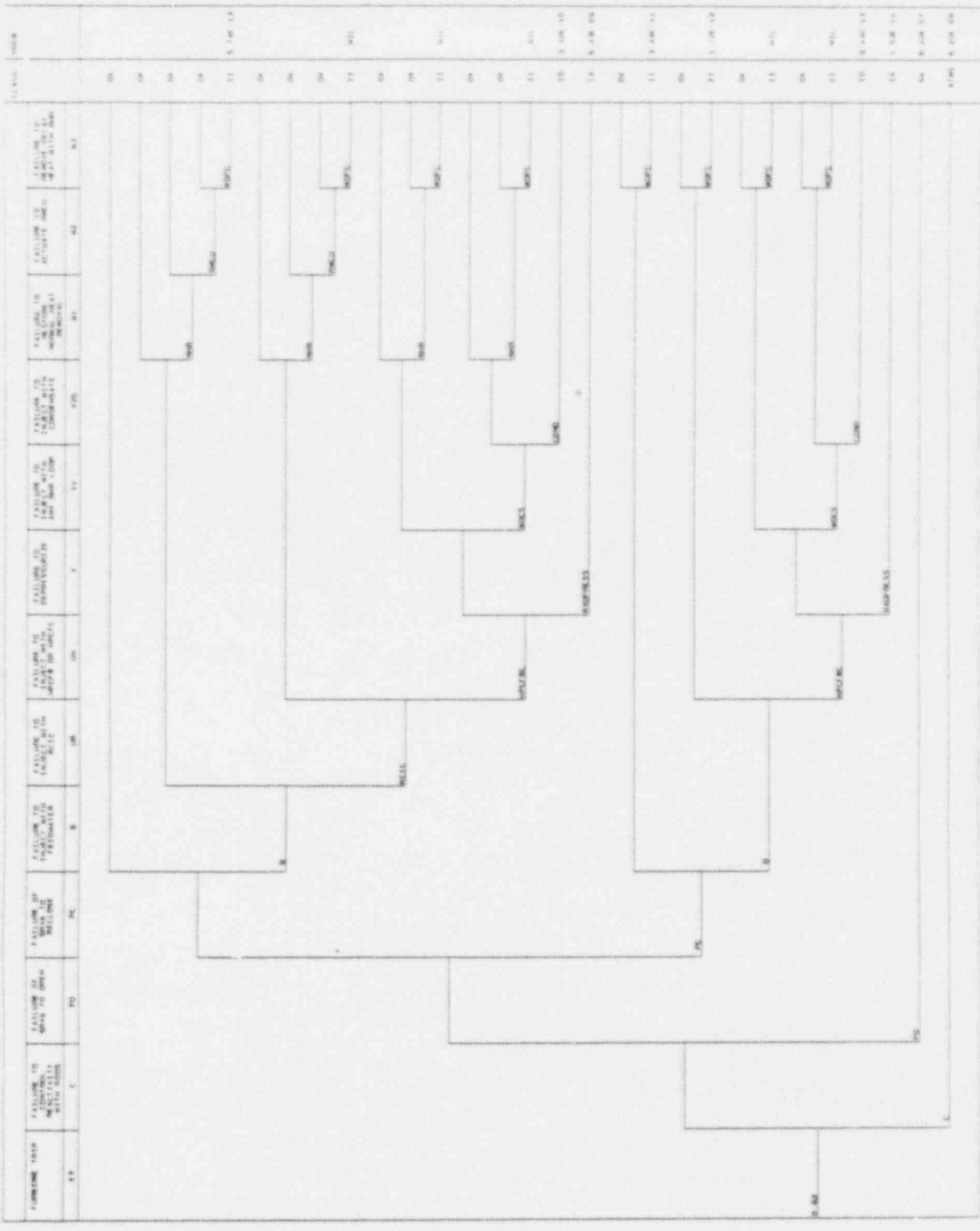


Figure 19D.4-2 Non-Isolation Event Tree

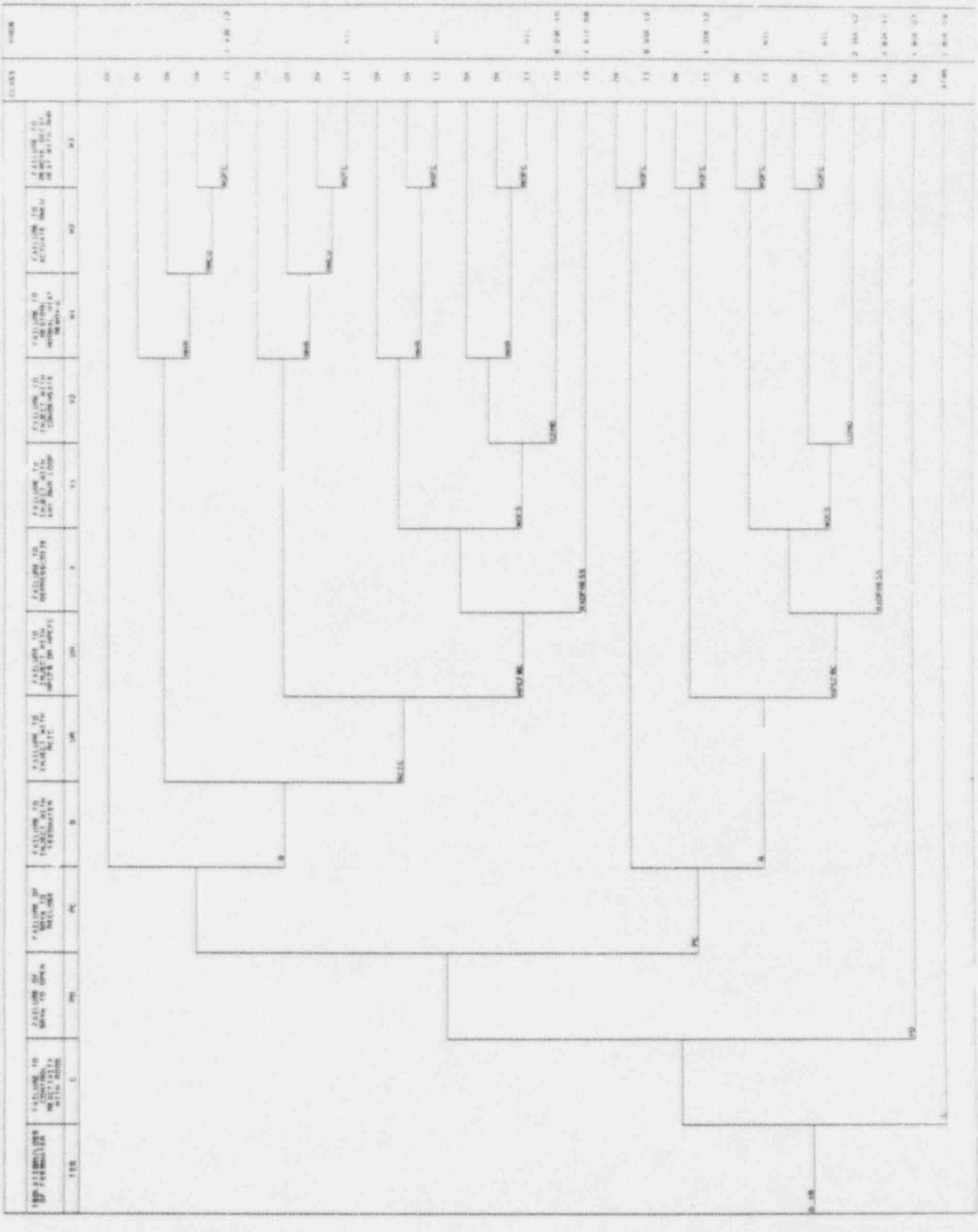
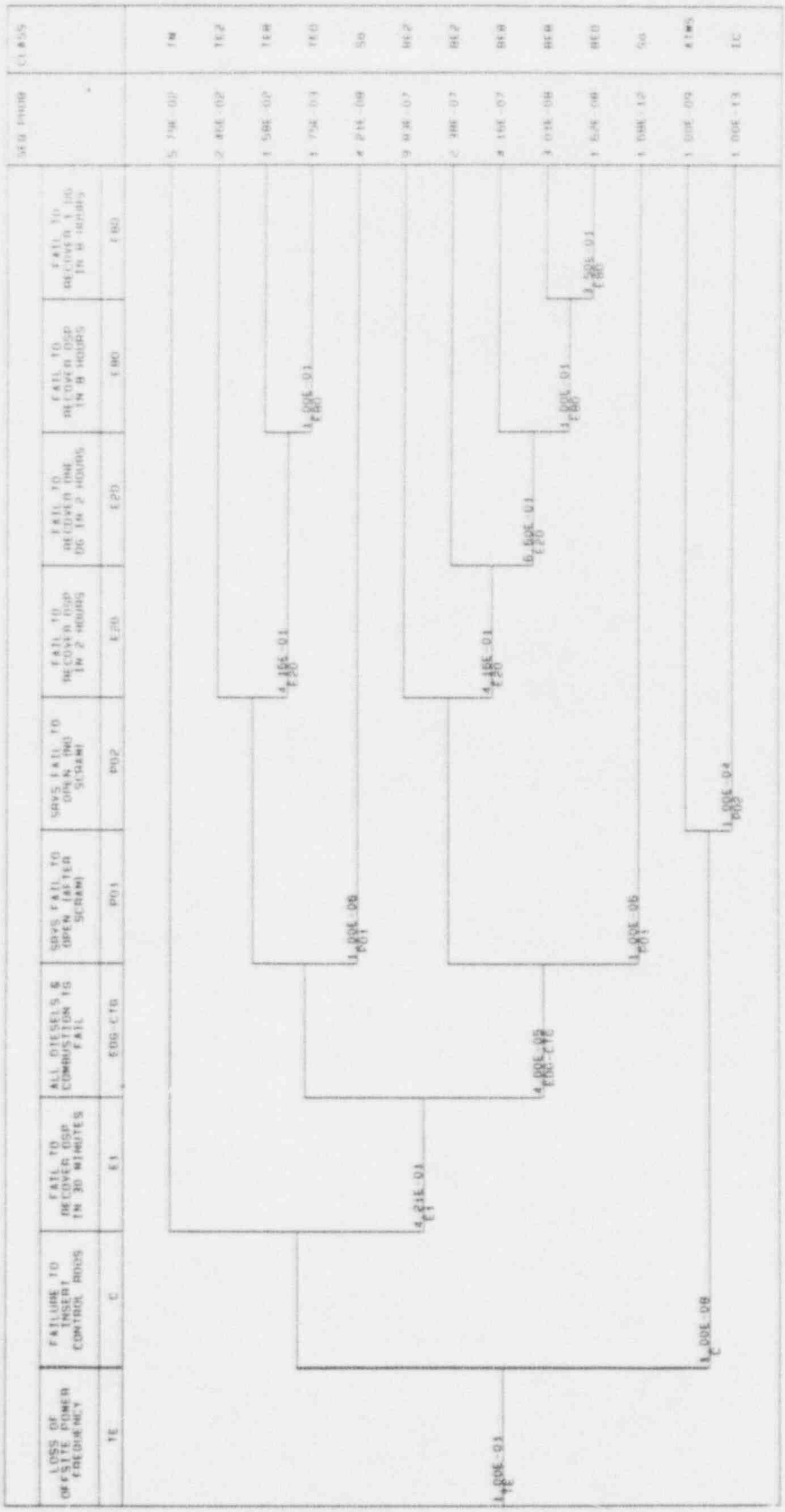


Figure 19D.4-3 Isolation/Loss of Feedwater Event Tree



19D.4-4 Loop and SBO Event Tree

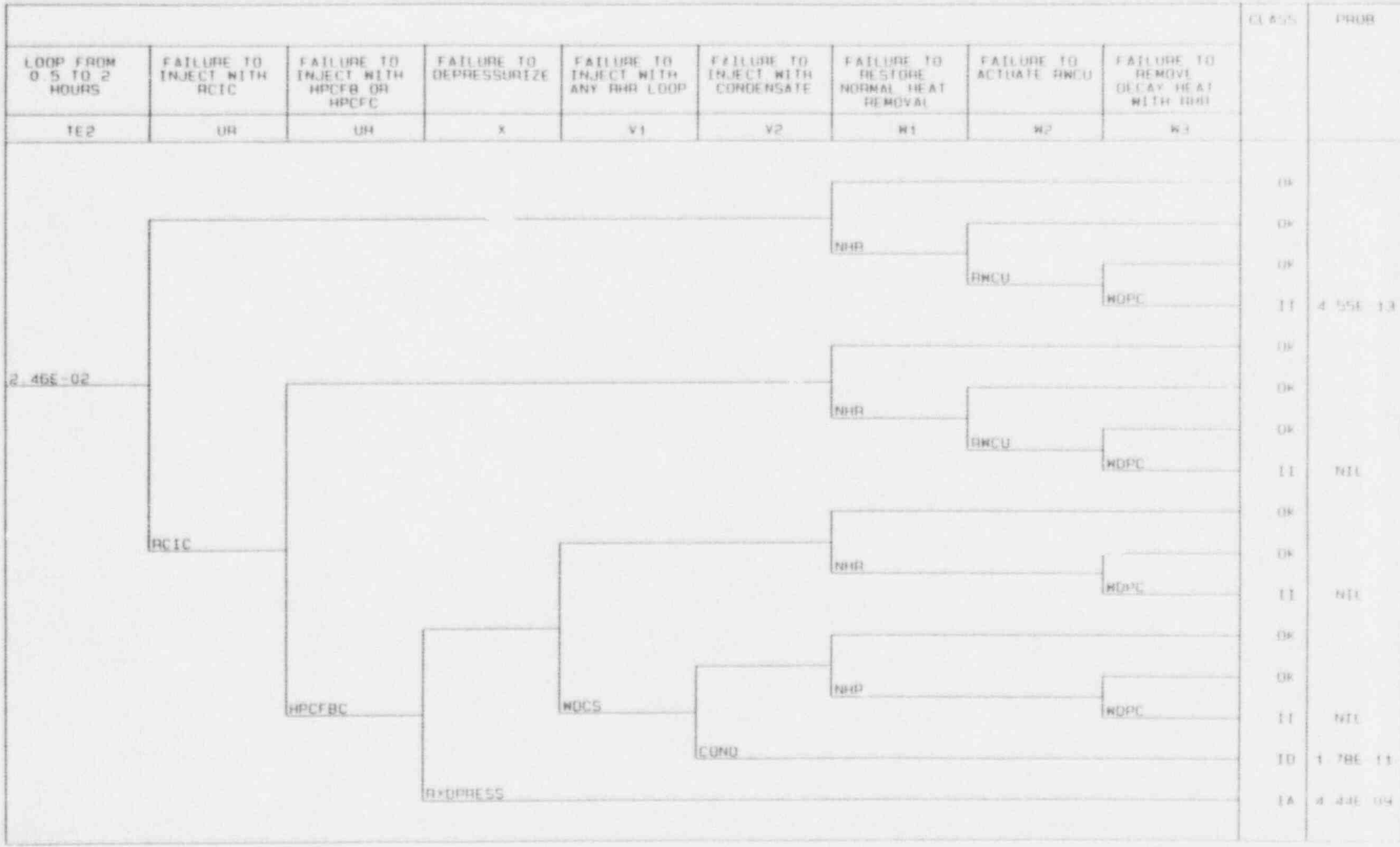


Figure 19D.4-5 Loss of Offsite Power Event Tree
(Recovery time: 30 min < t < 2 hrs)

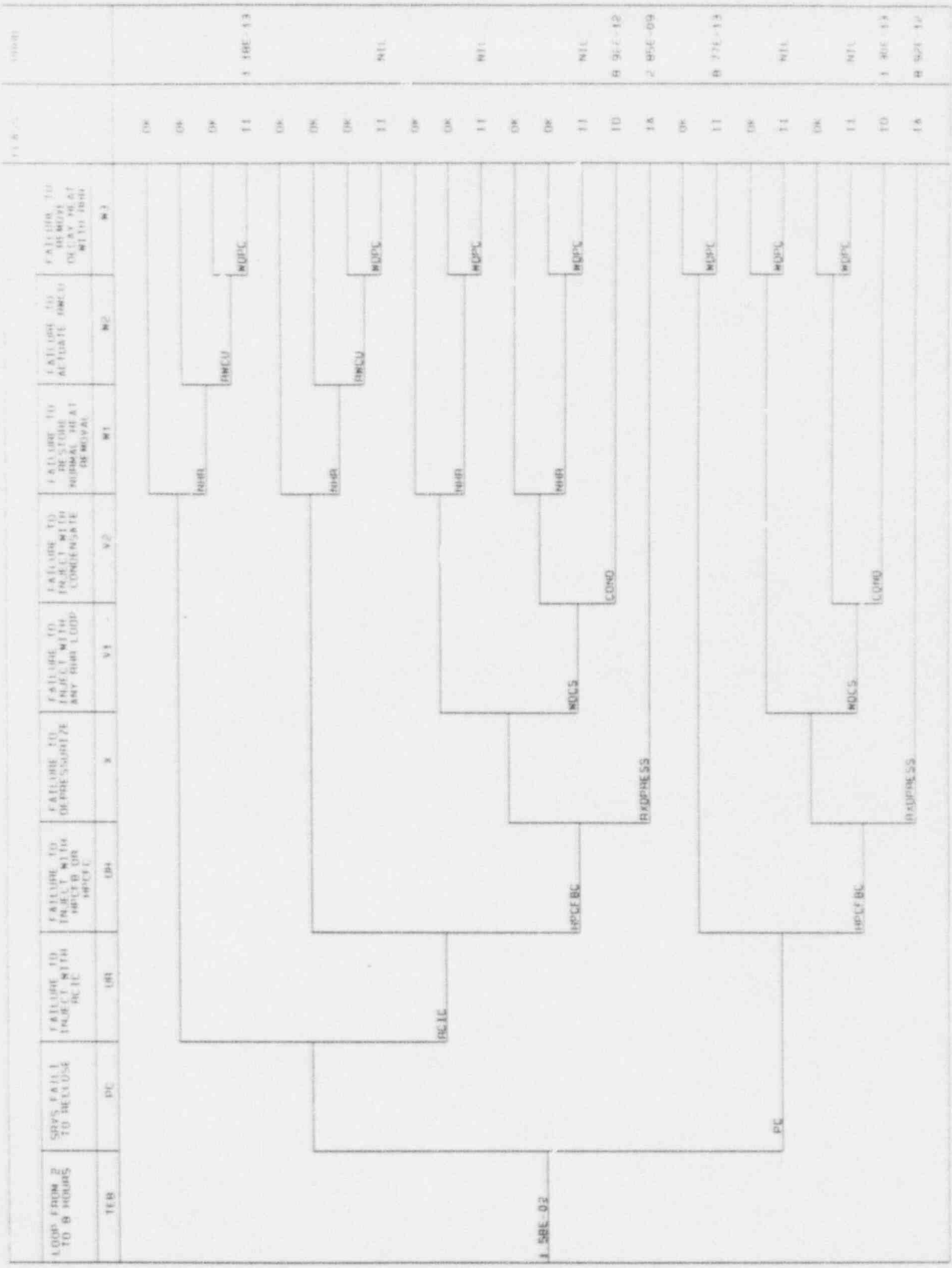


Figure 19D.4-6 Loss of Offsite Power Event Tree
(Recovery time: 2 hrs < t < 8 hrs)

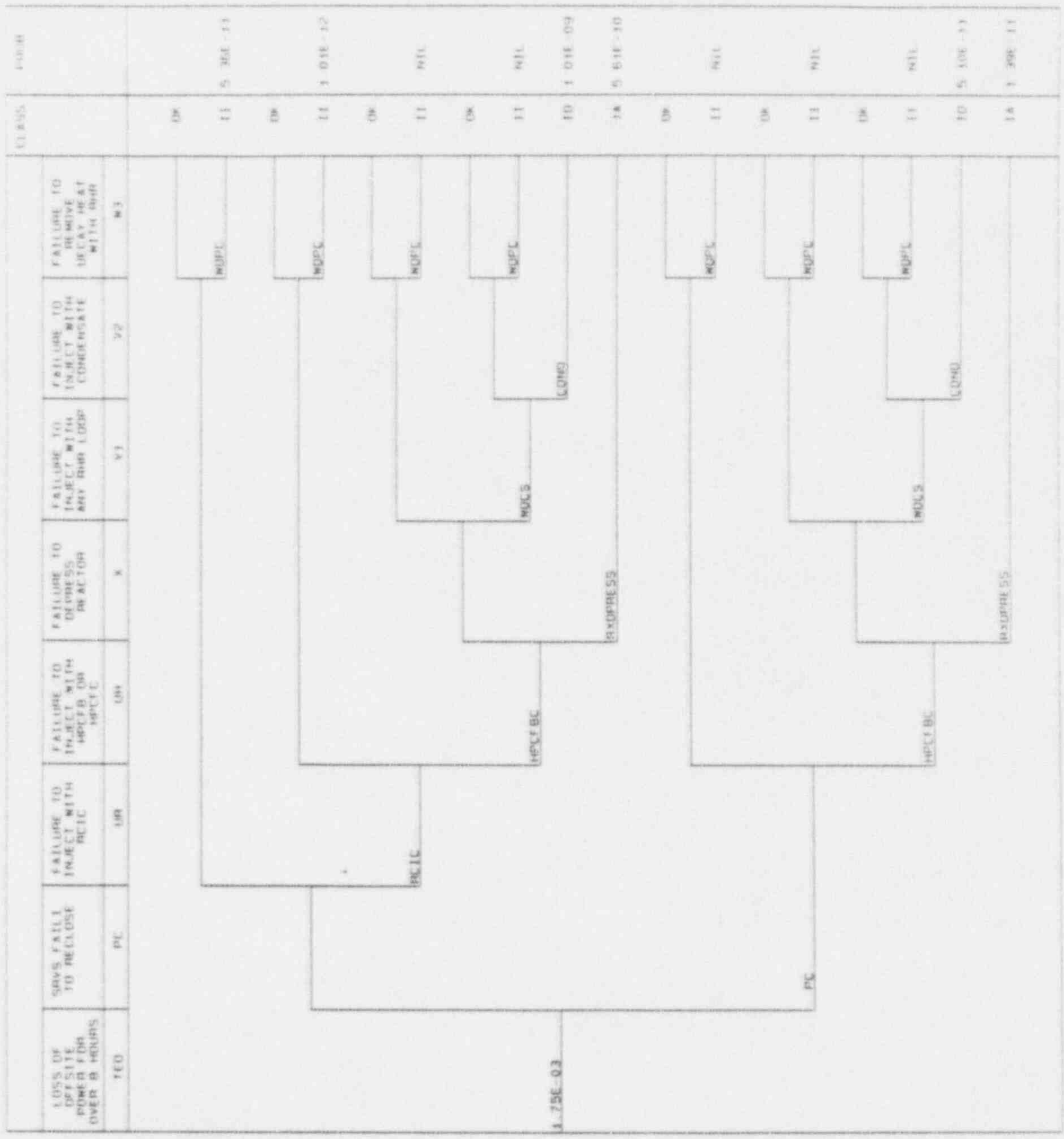


Figure 19D.4-7 Loss of Offsite Power Event Tree (Recovery time: t > 8 hrs)

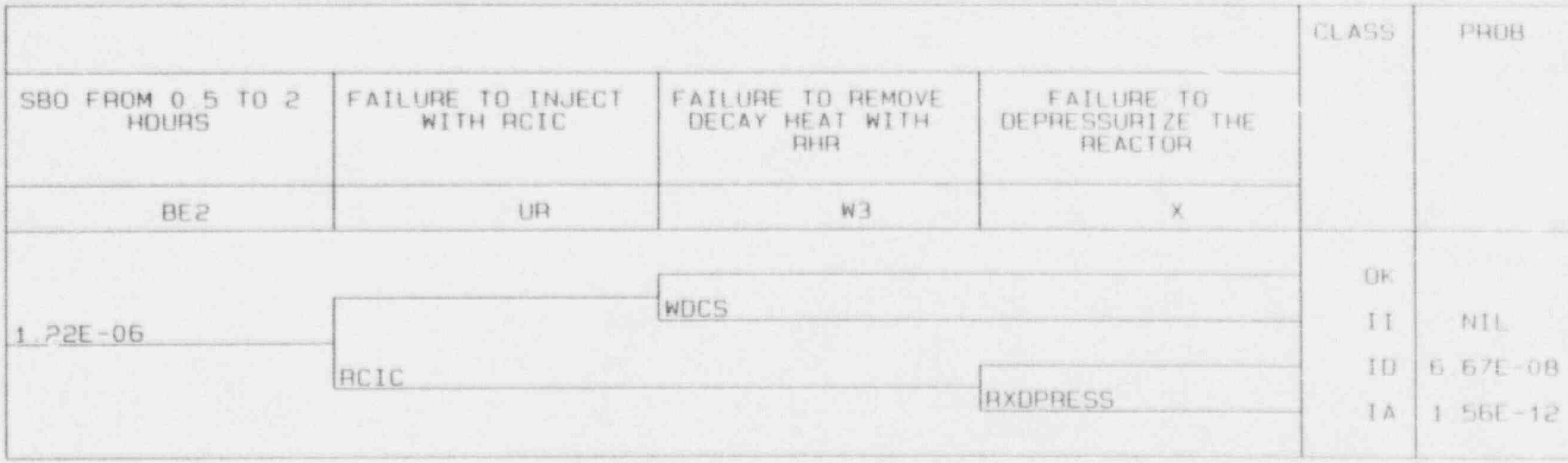


Figure 19D.4-8 Station Blackout Event Tree
(Recovery time: 30 min < t < 2 hrs)

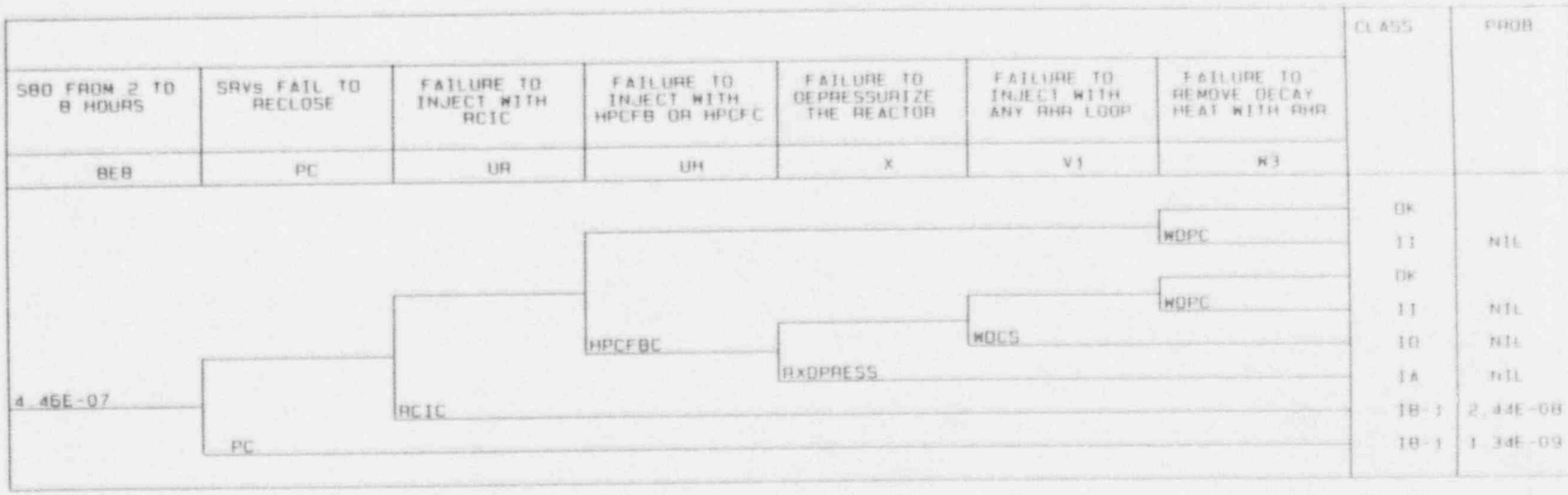


Figure 19D.4-9 Station Blackout Event Tree
(Recovery time: 2 hrs< t <8 hrs)

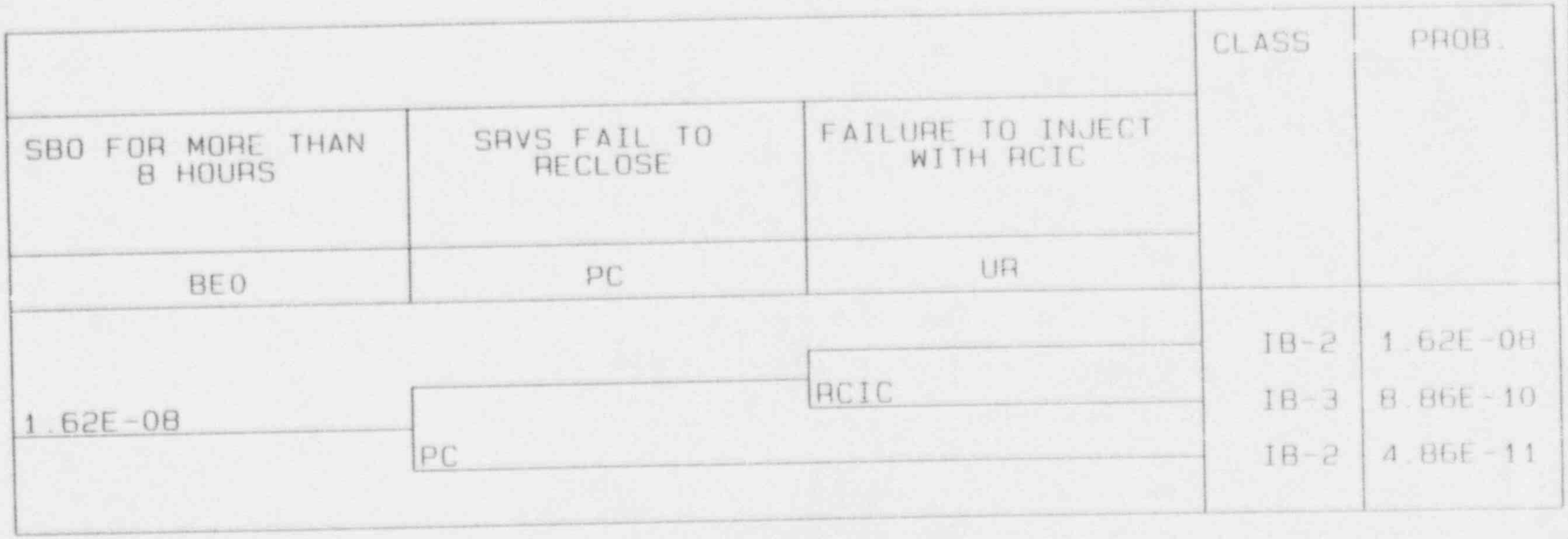


Figure 19D.4-10 Station Blackout Event Tree (Recovery Time t > 8hrs)

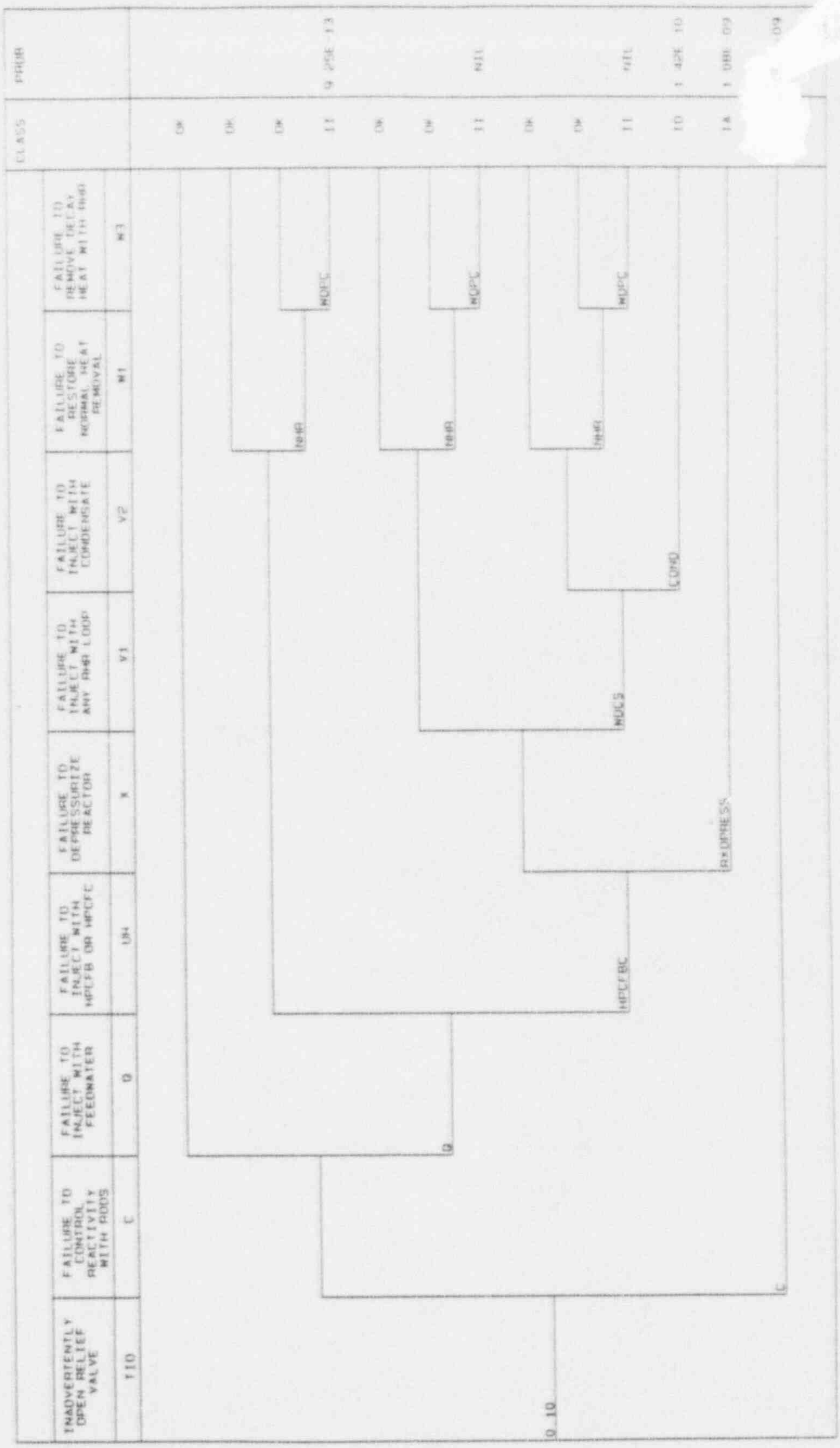


Figure 19D.4-11 Inadvertently Open Relief Valve (IORV) Event Tree

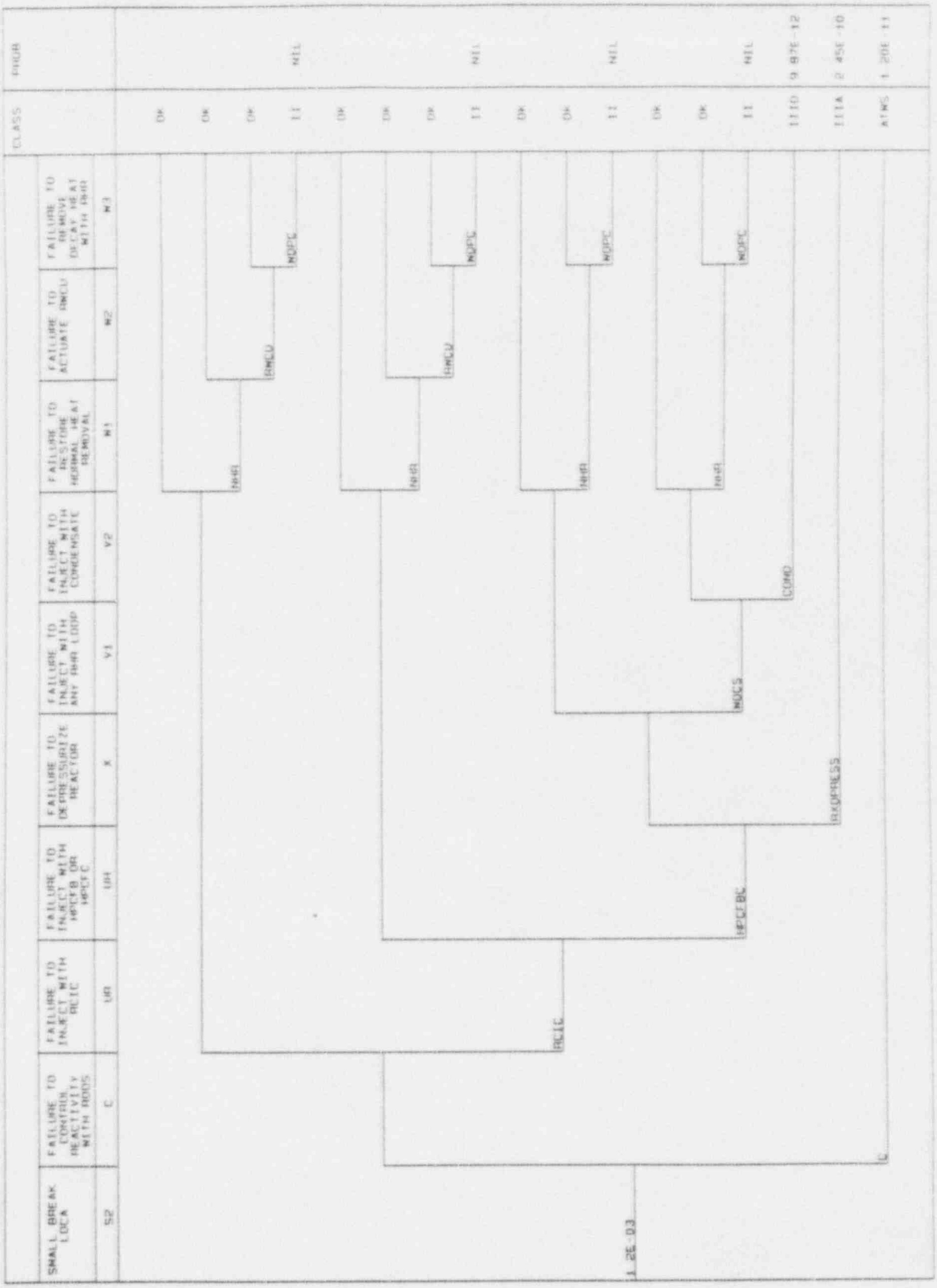


Figure 19D.4-12 Small Break LOCA Event Tree

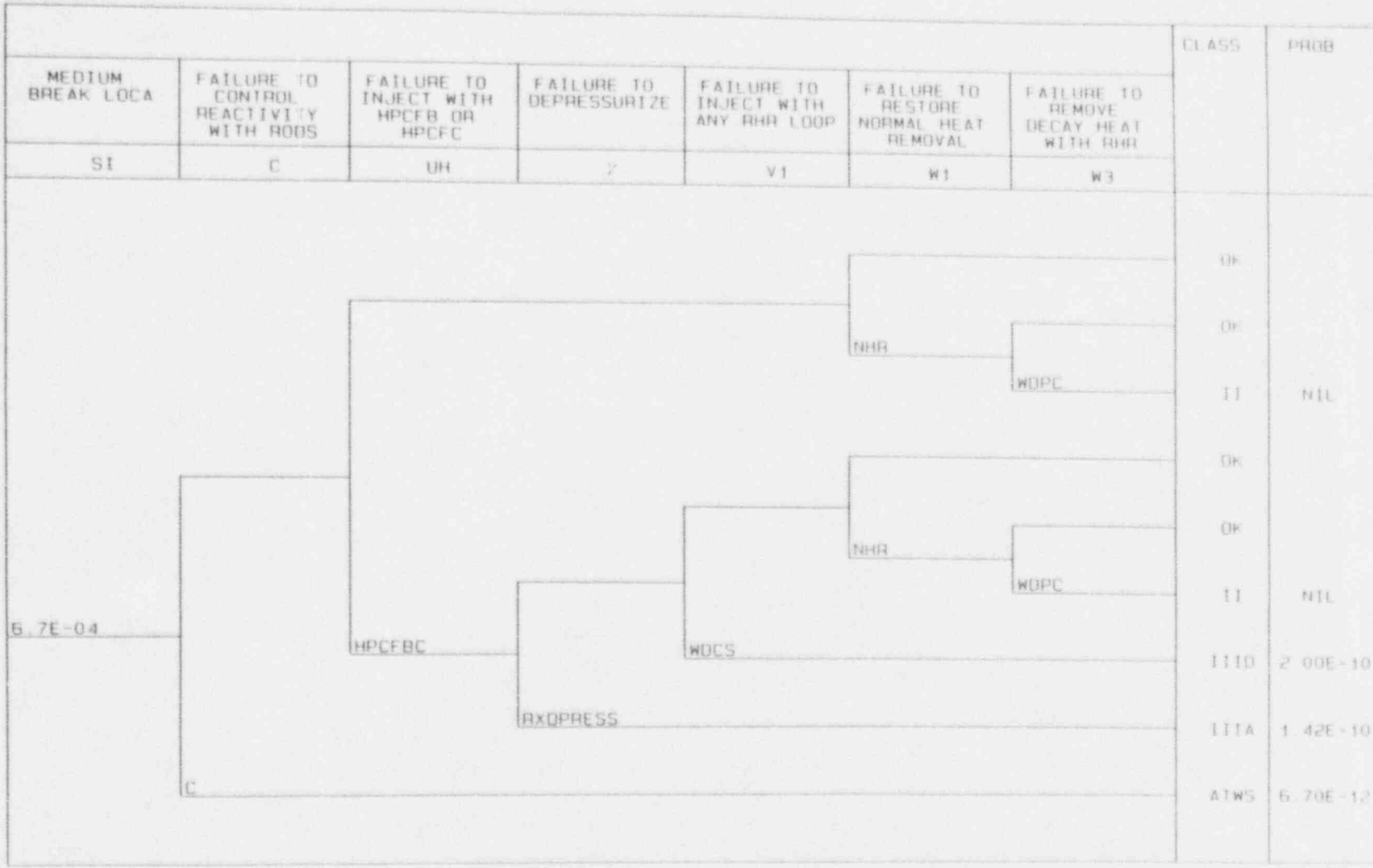


Figure 19D.4-13 Medium Break LOCA Event Tree

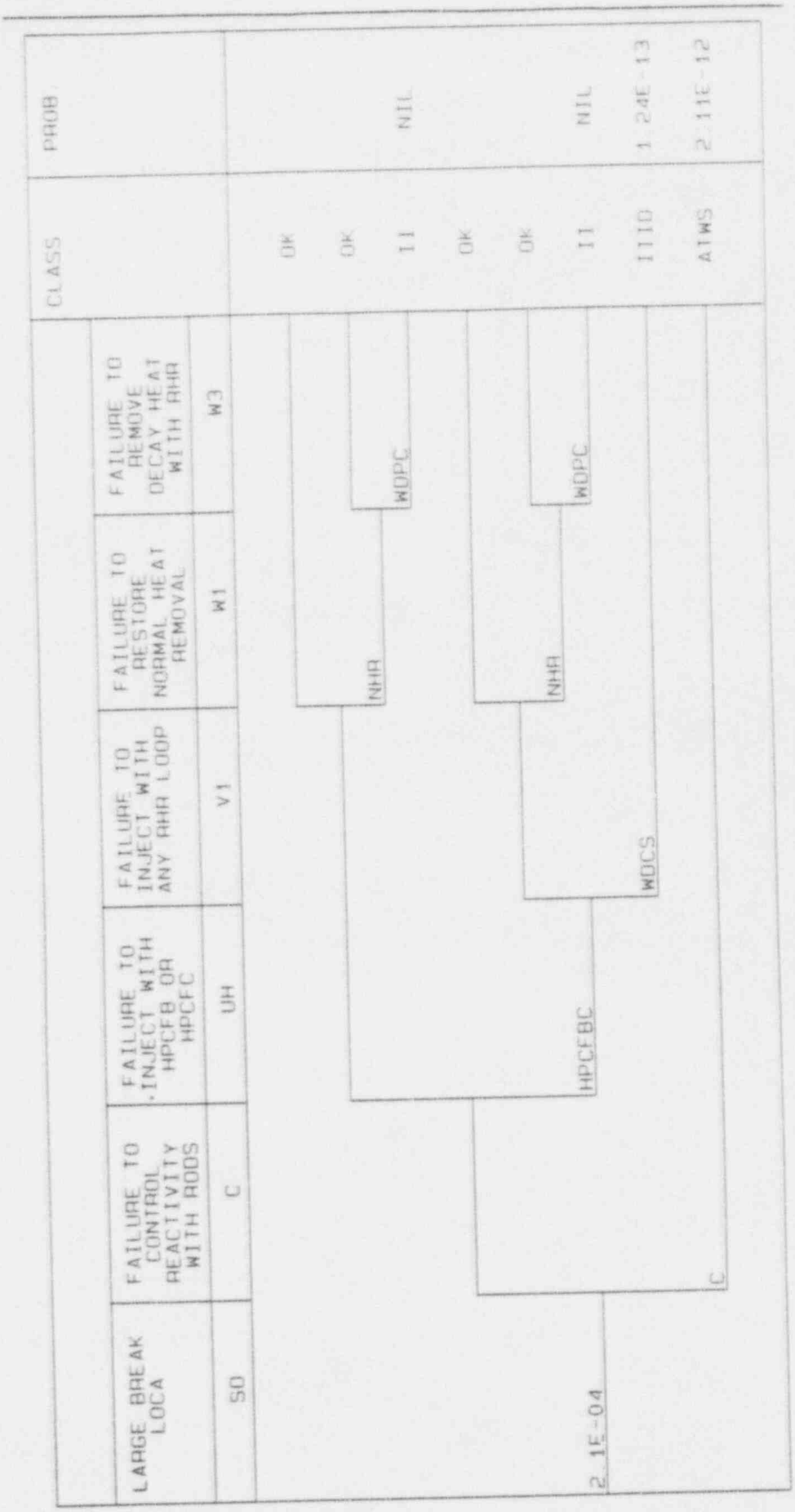


Figure 19D.4-14 Large Break LOCA Event Tree

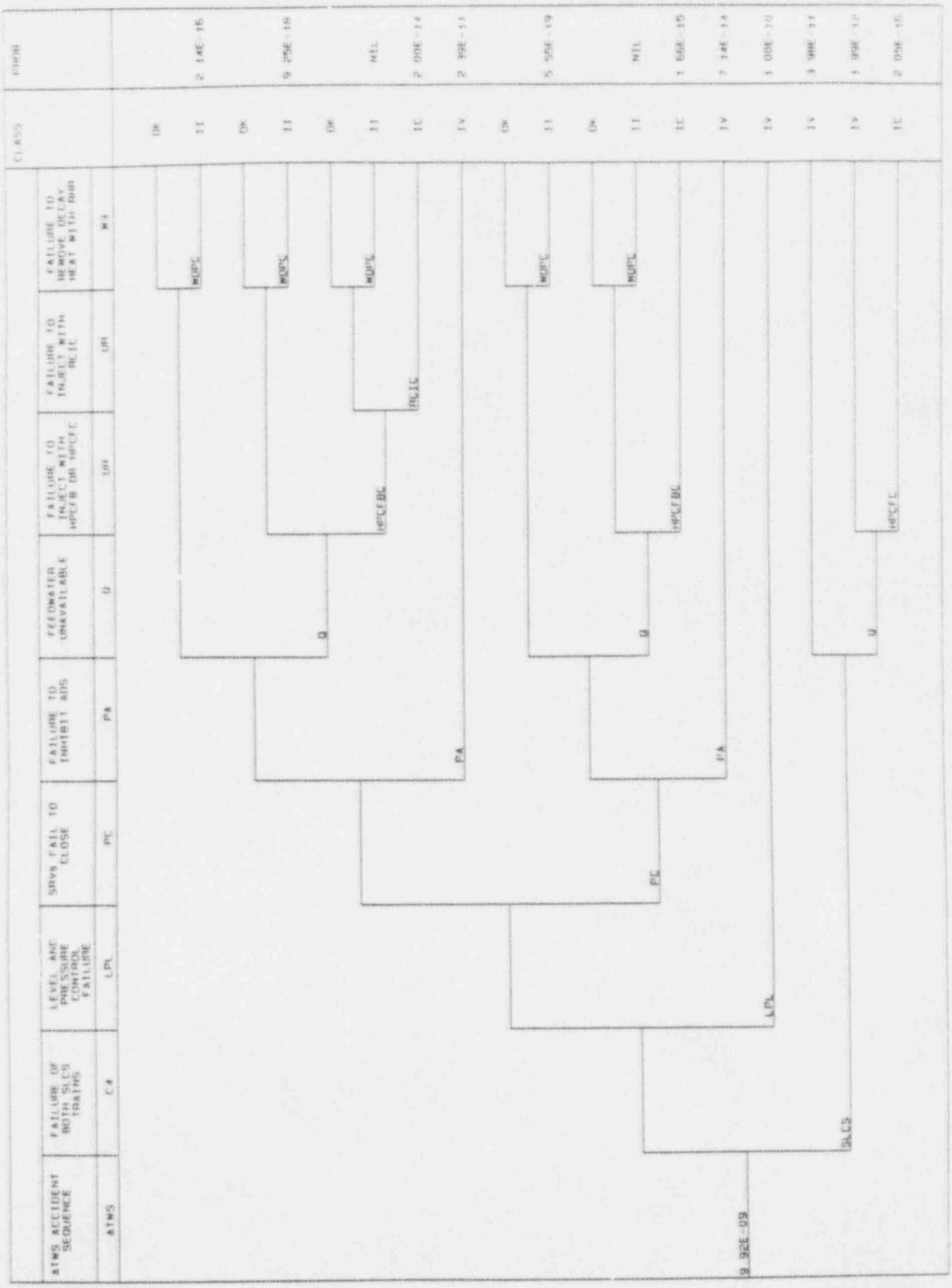


Figure 19D.4-15 ATWS Accident Sequence Event Tree

SECTION 19D.5

CONTENTS

SECTION 19D.5 (Continued)

CONTENTS

SECTION 19D.5 (Continued)

CONTENTS

SECTION 19D.5

TABLES

SECTION 19D.5

ILLUSTRATIONS

SECTION 19D.5

ILLUSTRATIONS (Continued)

19D.5 ABWR CONTAINMENT EVENT TREES

19D.5.1 Overview

The accident sequence event trees described in Section 19D.4 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 structural failure and the amount of fission product release to the environment.

There are five basic classes (I through V), and a total of eleven 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.

Generally, the event progressions for each of these classes of event are modeled in the containment event trees (CET). 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 Section 19E.2. The mode and location of containment structural failure is modeled based on a study of the containment capability discussed in Appendix 19F.

There is one CET for each of the 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 structural 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 Section 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 Section 19E.3.

19D.5.2 Accident Classes

In Section 19D.4 accident event trees are developed for each of the initiators. 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:

(1) Type of initiating event (transient, LOCA, etc.).
(2) Relative times of core melt and containment structural failure.
(3) 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 failure is available for decay 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 containment 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 description of these five classes is provided below and is summarized in Table 19D.5-1.

Class I: Most Class I events are transients with failure of core cooling systems. In these cases, core melt starts about one hour after event initiation and the RPV fails about one hour later. Following RPV failure, a mixture of molten core material and other metals, called corium, leaves the RPV and comes in contact with the concrete on the drywell floor. The corium-concrete interaction produces steam and non-condensable gases (such as hydrogen, CO and CO_2) and the containment eventually fails by slow over-pressurization.

Event progressions for ATWS events with failure of core cooling systems are similar and are also considered as Class I events.

Class II: Most Class II events are transients with successful core cooling, but with failure of the containment heat removal systems. The suppression pool heats up and the containment pressure builds up slowly until the containment overpressure protection system (COPS) set point is reached in about 24 hours. This period is available to the operator to try to recover the failed systems. If the COPS fails to actuate, structural failure of the containment could occur affecting the core cooling function. Consideration of this possibility is included in the containment event trees.

Loss of core cooling leads to core melt and RPV failure. Class II core damage sequences are thus characterized by containment structural failure followed by core melt. At the time of core melt, the containment is in a failed state and the fission products are released to the atmosphere without the benefit of residence time in the containment.

Event progressions for LOCAs with successful core cooling and ATWS events with successful boron injection and successful core cooling but with failure of the containment heat removal system are similar to the event progression described for transient events and these events are also considered Class II events.

Class III: LOCAs with loss of core cooling are Class III events. As in the case of Class I events, Class III events are also characterized by core melt followed by containment failure. However, because of the loss-of-coolant accident, core uncovery, core melt and RPV failure occurs faster than for Class I events.

Class IV: Class IV events are ATWS events without boron injection but with core cooling available. Under these conditions the reactor continues to produce up to 20% power. The steam produced in the reactor is routed to the suppression pool through the safety relief valves. If this situation continues unmitigated, the containment is overpressurized leading to rupture disk opening or structural failure. As discussed under Class II events, structural failure could lead to loss of core cooling function. This in turn could result in core melt. In summary, Class IV core damage events are characterized by fast containment structural failure followed by core melt.

Following core melt, fission products are released to the environment without radioactive decay due to holdup within the containment.

Class V: Class V events are events in which the suppression pool is bypassed. There are two types of Class V events. In the first, the pool is bypassed at the beginning of the event. An example of this type of event is a LOCA outside the containment. If the break is not isolated and if core cooling is unavailable, core melt will result, and the fission products will be released directly to the atmosphere without going through the suppression pool. The second type of Class V event consists of accidents in which the suppression pool is bypassed during the course of the accident. An example of this is the drywell rupture following core melt and RPV failure.

19D.5.3 Accident Subclasses

19D.5.3.1 Class I Events

The accident Class I is further divided into four subclasses, IA through ID, as discussed below. A summary of the differences is provided in Table 19D.5-2.

Class IA events are characterized by high RPV pressure when the core melts. These are transient events followed by failure of high-pressure water makeup systems coupled with failure to depressurize the reactor (ADS failure, for example). The subsequent core melt event is called the high-pressure core melt. The core melt and RPV failure could result in ejection of molten corium at high pressure into the drywell, which could increase the potential for drywell failure. On the other hand, RPV failure would depressurize the reactor making the low-pressure systems available for flooding the molten core.

Class IB events are broken into three categories.

Class IB-1 events are station blackout events with RCIC failure. Neither core cooling nor containment heat removal is available in the beginning and the core melt starts. However, on site power is recovered in eight hours which increases the likelihood of a core melt arrest and recovery of containment heat removal system. If core melt is not arrested and containment heat removal is not recovered then the containment structure fails on over pressure after 20 hours. Core melt arrest is discussed in Subsection 19D.5.8.

Class IB-2 events are a special class of Station Blackout events. The RCIC is available for core cooling for about eight hours, after which it is assumed to be unavailable.

The suppression pool continues to heat up when RCIC is in operation. This impacts the time of containment structural failure and the time available for decay of fission products released during the accident.

Class IB-3 events are similar to Class IB-1 events except that on-site power is not recovered in eight hours. This leads to core melt and increased likelihood of containment structural failure.

Class IC events are ATWS events without boron injection coupled with loss of core cooling. Core melt occurs faster than it does for other Class I events.

Conservatively, ATWS events with successful boron injection but with loss of core cooling are also included in this subclass.

Class ID includes low-pressure core melt events. These are transients followed by loss of high pressure core cooling, successful reactor vessel depressurization, and loss of low pressure core cooling. Following core melt and RPV failure, the molten core falls on the drywell floor. Unlike the Class IA event, low-pressure systems are not readily available to flood the molten core.

19D.5.3.2 Class II Events

Past analyses have shown that, as long as the core is kept covered with water, the containment response (especially the time required for containment structural failure) is relatively independent of the type of initiating event. Therefore Class II events have not been divided into sub-classes.

19D.5.3.3 Class III Events

Theoretically, Class III events could be sub-divided like the Class I event with four classes - A, B, C and D. However, Sub-Classes B and C which would represent LOCA coincident with loss-of-offsite power and LOCA coincident with ATWS are events with negligible frequencies of occurrence and negligible contribution to risk. These are therefore grouped as part of Class IIIA events.

Class IIIA events are small or medium LOCAs with failure of high pressure coolant makeup systems followed by failure to depressurize the reactor. The low pressure coolant systems may be available but cannot inject water into the reactor because of the high reactor pressure. Core melt occurs with the reactor at high pressure. The core melt and subsequent RPV failure could result in ejection of molten corium at high pressure into the drywell, which could increase the potential for drywell failure. On the other hand, RPV failure would depressurize the reactor making the low-pressure systems avail-

able for flooding the molten core. A large LOCA is not a Class IIIA event because the break depressurizes the reactor.

Class IIID events are LOCAs (small, medium or large) followed by failure of both the high pressure and low pressure coolant makeup systems. The reactor vessel is depressurized by the large LOCA or by the depressurization function for the small and medium LOCA. Following core melt and RPV failure, the molten core falls on the drywell floor. Unlike Class IIIA events, low pressure coolant makeup systems are not readily available to flood the molten core.

19D.5.3.4 Class IV Events

Class IV events are low probability events characterized by fast containment overpressurization (and failure) and it is judged that further sub-classification of this event is not necessary.

19D.5.3.5 Class V Events

Theoretically, there could be pool bypass events associated with each of the four accident Classes I through IV. However, past PRAs have shown that frequencies of pool bypass events and their contribution to plant risk are low, and it is reasonable to group them all under one class without dividing them into subclasses.

19D.5.4 Equipment Recovery

Recovery of the following systems or functions has been modeled in the containment event trees:

Core Cooling
Containment Heat Removal
On-site Power (includes diesel generators)
Off-site Power

Equipment recovery is achieved through component repair. Typical repairs are fuse replacement, valve operator replacement, pump or motor replacement, etc.

System recovery probabilities are 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

A mean time to repair of 19 hours based on the WASH-1400 data, was assumed for the repair of most system components as long as the core and the RPV are intact.

For events involving loss of offsite power or station black out, the MTTR was based on recovery of on-site or off-site power.

For systems involving (multiple) redundant divisions of equipment, there is a potential for recovery of each of the failed divisions. For instance, if all three RHR loops failed, there is a potential that any one of them can be recovered (P_f), and the probability of failing to recover can be modeled as $P_f \times P_f \times P_f$. However, because of potential for common cause failure and limitations on the number of available operators etc, it was judged that the probability of failure to recover the failed function would be taken as half the value calculated for a single system (ie $0.5 \times P_f$) and not $P_f \times P_f \times P_f$.

In the accident fault and event trees, the successful operation of the RHR was treated very conservatively. If the system failed to operate as designed then the branch of the tree was designated as failure. However, in the ABWR the most common failures of the RHR result in degraded performance or delayed initiation. For these cases, no repairs of the system are required to allow successful core cooling.

As an example, the presence of a LOCA signal is deemed a failure of supression pool cooling because this signal causes flow to be directed to the vessel. However, in the ABWR the valves are normally aligned such that the heat exchanger removes heat from the containment even when the RHR is in LPFL mode. Therefore, containment heat removal is successful.

A similar failure mode used in the fault and

accident event trees was common mode failure of a flow transmitter which could result in the minimum flow bypass line remaining open. Were this to happen about 10% of the RHR flow would bypass the heat exchanger. However, because the decay heat falls to a fraction of the RHR capacity before containment structural failure occurs, the degraded system is still capable of adequate containment heat removal.

A third type of failure considered in the fault trees and accident trees was failure of the operator to initiate the system. Failure to initiate could occur either due to human error or a failure of the suppression pool temperature alarm. However, because there is a very long time to containment structural failure the probability of recovering from human error is very high. Furthermore, in the long term, high drywell pressure will provide a diverse signal indicating the need for containment heat removal.

In order to determine the effects of these types of conservatisms, the fault trees were reevaluated with the appropriate nodes deleted. It was found that a recovery factor of 0.5 was appropiate.

When the remaining failure modes were examined it was found that the vast majority involved failures in the pump or valve rooms. If the pumps did not run after core damage began then the radiation levels would be less than 10 Rem/hr. Although this value somewhat high, pump and valve rooms are still accesible. Therefore, the time available for RHR recovery is the time to containment structural failure. In order to represent the affects of radiation in the pump room the failure to recover heat removal was multiplied by 2.

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 melt, containment overpressure, etc.). The available repair times were obtained based on the core melt progression analysis discussed in Section 19E.2.

19D.5.5 Containment Capability

The ABWR containment design pressure is 45 psig. Past stress analyses performed for other PRAs have shown that the containments are capable of withstanding much higher pressure (typically 2-3 times the design pressure). A discussion of the ABWR containment capability is provided in Appendix 19F. The ultimate pressure capability of the ABWR containment is limited by that of the drywell head. The drywell pressure capability depends upon the temperature in the containment. At 500°F, the containment ultimate strength is evaluated to be 134 psig.

19D.5.6 Containment Structural Failure Modes And Locations

In recent years, many PRAs have focused on the issue of containment performance following a severe accident. Of special interest are events with early loss of containment structural integrity or suppression pool bypass, and events involving large releases of radioactivity. In the case of non-inerted contain-

ments, hydrogen generation and potential for subsequent hydrogen detonation are also of special interest. The ABWR containment is inerted.

19D.5.6.1 Containment Structural Failure Modes

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). A certain amount of leakage is considered acceptable during plant operation since the risk associated with such leakage is judged to be acceptable. These acceptable limits are specified in the plant Technical Specifications. The amount of leakage increases as the containment is pressurized, but is not judged to be significant. (See Subsection 19E.2.3.4 for additional discussion.) However, if the temperature exceeds 500°F, there is a potential for degradation of seals in the large operable penetrations such as the drywell head, equipment hatches and personnel airlocks. Even with this high temperature degradation of these seals, a conservative evaluation shows that leakage is expected to occur only when the containment pressure exceeds 52 psig (Appendix 19F).

Conservatively, in this PRA, containment structural integrity is judged to be breached if the containment pressure exceeds 134 psig, or the containment pressure exceeds 52 psig and the containment temperature exceeds 500°F. In addition, under some conditions, exposure to very high temperatures could reduce the structural capability of the containment [see Subsection 19D.5.6.3(b)].

19D.5.6.2 Containment Failure Location & Probabilities

Based on a review of results of tests performed on sealed containment models, it is concluded in Appendix 19F that the most likely failure location is the drywell head. The ABWR containment event trees are developed such that the overpressurization events result in either drywell failure or opening of the COPS. The probability of drywell head failure is based on the uncertainty distributions for the head ultimate stength and the COPS as described in Subsection 19E.2.8.1.1. The sequences in which a combination of high temperature and pressure results in a containment leakage, the leakage is assumed to occur in the large drywell penetrations (e.g. equipment hatch) with a failure probability to 1.0.

19D.5.6.3 Failure Modes Explicitly Modeled in Containment Event Trees

The following containment failure modes are explicitly modeled in the CETs:

(1) Containment Overpressurization

Containment fails in the drywell when subjected to high pressure resulting from steam and non-condensable gases.

(2) Containment Leakage

Containment seals (such as the drywell head seal) fail when subjected to a combination of high temperature and pressure (500°F and 52 psig).

(3) High Temperature Failure

When subjected to a very high temperature (e.g. greater than 700°F) the drywell structural capacity is reduced due to reduction of material strength.

(4) Containment Failure at the Time of RPV Failure

Containment fails when the RPV fails due to factors such as vapor suppression failure, missile generation, etc.

19D.5.6.4 Failures Modes Not Explicitly Modeled in Containment Event Trees

(1) Steam Explosion

In-vessel and ex-vessel steam explosion leading to containment failure are not credible events as discussed in Subsection 19E.2.3.1 and Attachment 19EB. Therefore, they are not explicitly modeled in the CETs.

(2) Hydrogen Detonation

The ABWR containment is inerted during plant operation and therefore, failure modes

relating to hydrogen burning and detonation have been ruled out as having a negligible probability of occurrence. The risks associated with the small fraction of time (<1%) of ABWR plant operation when the containment is not inerted is negligible, since these are associated only with the plant startup or shutdown process, and inerting can be restarted if an accident is initiated. There is a potential for hydrogen combustion in the reactor building, following the release of gases after the containment fails. Since the containment structural failure directly results in suppression pool bypass in the ABWR CETs, special modeling of hydrogen combustion was considered not necessary.

(3) RPV Rupture

RPV rupture, an initiating event which could potentially cause a structural failure of the ABWR containment, is judged to be a negligible contributor to risk.

(4) Basemat Penetration

Basemat penetration following core melt is not expected to result in the release of radioactive materials to the environment (Section 19E.2).

19D.5.7 Suppression Pool Bypass

19D.5.7.1 Introduction

10

The magnitude of radioactive release to the environment for the severe accidents in which the suppression pool is bypassed is much higher than the severe accidents in which the release occurs through the suppression pool. Thus, suppression pool bypass paths are of special interest in BWR PRAs. This subsection discusses the various types of suppression pool bypass paths and describes how they are treated in the ABWR PRA. Some of these bypass paths are explicitly modeled in the CETs. Others have been studied separately in Subsection 19E.2.3.3 and found to be negligible contributors to ABWR plant risk. A summary of the various suppression pool bypass mechanisms and how they are treated in the ABWR PRA is provided in Table 19D.5-3.

19D.5.7.2 Ex-Containment LOCA

A LOCA can occur in the high pressure system piping which runs outside the containment. Rupture of high pressure piping in any of the following systems with a failure to isolate the break could result in this type of LOCA:

(1) Main steam lines
(2) Feedwater lines
(3) RCIC steam lines
(4) RWCU system lines

Another type of ex-containment LOCA is the interfacing systems LOCA where a low pressure system piping connected to a high pressure system is accidentally overpressurized resulting in the rupture of the low pressure piping. This, coupled with a failure to isolate, results in the interfacing systems LOCA. In ABWR, potential for interfacing LOCA has been essentially eliminated by increasing the design pressure of low pressure piping connected to higher pressure piping.

In each of the above cases, if the ex-containment LOCA is not mitigated, core melt could result and the radioactive material would be discharged to the environment directly through the break, bypassing the suppression pool.

This type of suppression pool bypass path has been studied as part of the suppression pool bypass study in Subsection 19E.2.3.3 and it was determined that this is not a significant contributor to ABWR plant risk. Therefore, this bypass path is not modeled in the CETs.

19D.5.7.3 Failure of Isolation Valves and Pipe Ruptures

There are other combinations of failure of isolation valves and rupture of pipes which could lead to suppression pool bypass. But unlike the ex-containment LOCA discussed above, these are by themselves not accident initiators but provide a path for bypassing the suppression pool for severe accidents which are initiated through other initiators. An example of this type of suppression pool bypass path is the one resulting from the rupture of RCIC piping, failure of RCIC discharge valve and operator inability to isolate the break.

This type of bypass paths is analyzed in detail in Subsection 19E.2.3.3 and it was determined that this is not a significant contributor to ABWR plant risk. Therefore, this bypass path is not modeled in the CETs.

19D.5.7.4 Failure of Drywell Vacuum Breaker

If the containment drywell vacuum breaker is in a failed state during a core melt event that leads to RPV failure, the suppression pool is bypassed. This bypass path has been studied as part of the suppression pool bypass study in Subsection 19E.2.3.3 and it was determined that could be a significant contributor to ABWR plant risk. Therefore, this bypass path is modeled in the CETs.

19D.5.7.5 Containment Structural Failure

The most likely structural failure of the containment occurs in the drywell. This failure mode bypasses the suppression pool and is modeled in the CETs. As discussed in Subsection 19D.5.6.3, two additional containment failure modes are modeled in the CETs. These also result in suppression pool bypass.

19D.5.7.6 Uncovery of Horizontal Vents

If after the RPV failure, the horizontal vents are uncovered due to low water level in the suppression pool, the pool will be bypassed. Calculations show that for all events other than ATWS, initial suppression pool inventory is sufficient to compensate for the evaporation loss for over 24 hours without uncovering the horizontal vents. It is assumed that suppression pool make up will be initiated within 24 hours (by using fire trucks if necessary) for each of these accidents. Therefore this type of suppression pool bypass is not explicitly modeled in the CETs.

19D.5.7.7 Low Probability Bypass Events

Some events such as RPV rupture, and in-vessel steam explosion, with extremely low probability of occurrence have the potential for causing suppression pool bypass. These are not specifically treated in the CETs because past PRAs have shown negligible contribution to risk attributable to these events. For references which provide additional details see Table 19D.5-3.

19D.5.8 Core Melt Arrest Success Criteria

19D.5.8.1 Introduction

After core melt has been initiated, the process can still be arrested if the core debris is cooled with sufficient water. The success criteria for arresting core melt is described in this subsection. The analytical basis for the success criteria is developed in Subsection 19E.2.1.4.3.

There are two ways to arrest core melt: 1) during the early stages of an accident core melt can be arrested prior to RPV failure, 2) if the RPV has been breached, core melt can be arrested prior to loss of structural integrity of the containment.

In each case, a means of getting water to the corium and a means of removing heat from the containment are required.

The core melt arrest success criteria is summarized in Table 19D.5-4.

19D.5.8.2 Core Melt Arrest Prior to RPV Failure

For arresting core melt within the RPV, one of the core cooling systems must be recovered within about an hour of the core melt initiation.

If the reactor is at high pressure, operation of one of the high pressure systems (HPCF B or C, RCIC, feedwater system) must be restored.

If the reactor is at low pressure, in addition to the high pressure system, operation of one of the low pressure systems (LPFL, condensate injection) may be recovered. Alternately, arrangement could be made to get the diesel driven fire water into the reactor.

CRD water supply is assumed to be not sufficient to arrest core melt.

For removing the heat from the containment, one of the RHR loops must be available.

19D.5.8.3 Core Melt Arrest Prior to Loss of Containment Structural Integrity

Following RPV failure, the molten core drops on the drywell floor. Core melt can be arrested by operating one of the two HPCF or any one of the low pressure systems (LPFL or condensate injection to the reactor vessel or diesel driven fire water system).

CRD water supply is assumed to be not sufficient for arresting core melt.

If none of the low pressure systems can be recovered in time to quench the corium on the lower drywell floor, the corium continues to heat up the lower drywell area. This melts the fuse at the ends of pipes in the passive flooder system resulting in the transfer of the suppression pool water to the lower drywell area. This passive flooder system is described in Subsection 9.5.12. The suppression pool water quenches the molten corium and the core melt process is arrested. This process is modeled in the CETs by the node P, representing "passive mitigation". It should be noted that even after passive flooder operation, the suppression pool water level stays high enough to cover the horizontal vents.

After the core melt is arrested, it is still necessary to remove heat from the containment. Containment heat can be removed by operation of one of the RHR systems. If the core melt is arrested but the RHR is not available, radioactivity is eventually released to the environment.

19D.5.9 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.

Detailed discussion of the binning process is provided in Subsection 19D.12.5.

19D.5.10 Containment Overpressure Protection

Subsection 6.2.5 describes a mitigation system called the overpressure protection subsystem of the atmospheric control system. This system protects the containment structural integrity and provides for controlled fission product release. If the containment exceeds its service level C limit, a rupture disk opens providing containment pressure relief. Since the system originates in the wetwell airspace, any fission product release will be scrubbed. The operation of COPS is indicated by "RD open" in the LCS node of the CETs.

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19D.5.11 Description of Containment Event Trees

19D.5.11.1 Introduction

A description of the CETs is provided in this section.

There is one CET for each of the accident classes. 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 structural failure of the containment. As discussed in Subsection 19D.5.9 the end states are assigned a release category under the column titled CET Bin/ Transfer. Some of the end states in the CETs require further CET development before release categories can be assigned. In such cases, under this column CET Bin/Transfer, the CET figure number, in which this sequence is developed in detail, is identified. IA.1, IB-1.1, etc. are examples of CET transfers. Each CET row is identified by a row number for purpose of identification in the text. Each CET consists of two figures. The first figure consists of the tree structure and the second figure consists of description of the accident sequence for each CET row.

Eleven accident classes were developed in the accident event trees. However, two of the accident classes (IC and IV) had negligibly low occurrence frequencies and CETs were not developed for those accident classes. Class IC event frequencies were added to the class IA frequencies Class IV events, like Class II events, are characterized by containment structural failure which might, in turn, cause failure of core cooling. The mechanisms for core cooling failure are discussed in Subsection 19D.5.11.3. No credit is taken for the firewater system to prevent core damage in this class because the stability of the reactor during an ATWS at low power has not been examined. The fission product release associated with this event has been binned with SBRCPFDH, as this bin resulted in the most severe consequences. Some accident classes, on the other hand, were analyzed in more than one CET. For example, class IA is analyzed in two event trees, IA and IA.1. Class V sequences were treated in the suppression pool bypass study in Subsection 19E.2.3.3 and no CETs were developed. A summary of how each of the accident classes is treated in the CETs is provided in Table 19D.5-7.

19D.5.11.2 CET for Class IA Event

CET for Class IA event is shown in Figures 19D.5-4 and -5. Class IA events are transients followed by failure of high-pressure water makeup systems coupled with failure to depressurize the reactor.

The first CET node, OP, checks if the operator depressurized the reactor. In this particular case this action has been included in the ADS fault tree (and therefore the accident event trees) and hence this node has been assigned a failure probability value of 1.0 (i.e. the reactor is not depressurized).

The second CET node, CHR, checks if containment heat removal was available at the time of the accident. If RHR is available, the accident sequence continues as discussed below. If RHR is unavailable, the event is tracked in a separate CET, IA.1 in Figure 19D.5-5.

The next CET node, ARV, checks if the core melt is arrested within the reactor pressure vessel, through recovery of one of the high pressure core cooling systems. If the core melt is arrested, the outcome will have negligible fission product release and is classified as LCHP IVNN (Row #2). A detailed description of the release categories is provided in Subsection 19E.2.2. If core melt is not arrested within the RPV, the core melt process continues leading eventually to RPV failure. The next node, CI, checks if the containment fails when the RPV fails as a direct and immediate consequence of vessel failure. If the containment fails, it is conservatively assumed to fail at the drywell and high fission product releases are expected (see Row #7).

The next node HTF checks if the containment failed due to high temperature resulting from the presence of uncooled corium in the upper drywell. The next CET node, ARC, checks if the core melt process is arrested prior to reaching the pressure at which the containment structure fails - which happens in about 20 hours.. If the core melt is arrested, no significant fission

product release is expected. However, to account for leakage through the containment, this release has been classified as LCHPFSNN release (Row #3). If the core melt process is not arrested by the time the containment ultimate pressure is reached, the sequence is developed in CET IA.1 in Figure 19D.5.11-5. The next node, P, checks for passive mitigation which occurs when the passive flooder system initiates and the suppression pool water quenches the molten core on the drywell floor. The next node, RCH, examines if the containment heat

removal is recovered in time to prevent containment structural failure. The last node containment vent, VT, allows containment structural failure to be prevented by opening the vent. However, in this analysis no credit is taken for containment venting. Thus this node has been assigned a failure probability of 1.0 for all CETs.

19D.5.11.3 CET for Class II Event

The CET for Class II Event is significantly different from those for the Class I events.

The first node, RCH, examines the probability of recovery of the RHR prior to containment structural failure. If the RHR system is successfully recovered, containment structural failure is averted. The second node VT models the containment vent. Since a vent is not used in the base analysis, this node is assigned a failure probability of 1.0. Therefore, if the RHR is not recovered at node RCH, containment pressure increases, resulting eventually in structural failure.

The node CC models the probability that core cooling will be impacted following structural failure of the containment. The quatification of this node is described be w.

For cases in whi the core is successfully cooled, but the containment is not, the containment will pressurize. The containment boundary will eventually be breached, but if core cooling is maintained, the offsite consequences of the breach will be negligible. If the containment boundary failure causes core cooling failure, the consequences would be more severe. Therefore, this potential was reviewed. Several general areas were reviewed; 1) drywell head failure, 2) high temperatures in the suppression pool, and 3) high drywell temperatures. Each is briefly discussed in the following paragraghs.

Drywell head failure would pressurize the relatively small volume between the head and concrete shield plugs. This could levitate some of the plugs which would then fall, potentially causing equipment damage. There is no potential for plugs falling between the reactor vessel and drywell wall - the annular space is too small. The vessel vent could be damaged, but the consequences would be no worse than a small LOCA. Although unlikely, plugs could fall through the vertical equipment hatch and damage electrical equipment and/or an RHR heat exchanger. It is extermly unlikely that more than one division of core cooling would be lost as a result. There is a remote possibility for plugs which were very energetically elevated to fall on the control room, possibly causing temporary loss of core cooling, until cooling could reestablished from the remote shutdown panel.

High temperatures in the suppression pool would result in increased suction temperature for core cooling pumps. However, pump performance should not be impaired because the pumps are designed for water temperatures as high as 360^{o}F. Further, condensate storage tank water and fire tank water temperatures would not be affected.

High drywell temperatures were considered for their potential effects on SRV performance, electrical equipment and water level instrumentation. SRV performance should not be degraded because the expected temperature/time history is less severe than the LOCA condition for which the SRVs will be qualified. There is no electrical equipment in the drywell which is required to operate to establish or maintain core cooling. Effects on water level instrument accuracy should be small since the reference and variable legs experience the same elevation drop in the drywell.

After reviewing these potential causes of core cooling loss resulting from high temperature conditions/containment failure, it was judged that the probability of core cooling loss ranged between 0.01 and 0.001. A value of 0.01 was used in the analyses for loss of conventional core cooling. Since the firewater addition system is much less vulnerable, it was assumed available 90% of the time, even if conventional cooling was lost.

19D.5.11.4 CET for Other Classes

The structure of other CETs is generally similar to the class IA CET. The trees reflect the accident progression as analyzed in Section 19E.2. The CET node values are calculated based on a number of factors, such as the recovery probability or fault tree evaluation, which in turn depend upon the accident sequence progression.

19D.5.12 Discussion of Results

19D.5.12.1 Introduction

The results of the containment event tree analyses are discussed in this subsection. To recapitulate, the accident sequence event trees described in Section 19D.4 identified nine accident classes. The total accident sequence frequency was calculated to be 1.3E-6 per reactor-year. The CETs were used to identify the radioactive release categories and calculate the associated frequencies for use in the consequence analysis described in Section 19E.3. In addition the CETs helped answer the following questions:

a) What fraction of these accident sequences resulted in core damage?

b) In what fraction of these cases was core melt arrested?

c) What fraction of these accident sequences resulted in COPS actuation?

d) What fraction of these accident sequences resulted in containment failure?

e) Are the core damage and containment-related goals met?

The answers to these questions follow:

19D.5.12.2 Core Damage Frequency

The total internal event core damage frequency (CDF) calculated from the sum of all releases except OK on Table 19D.5-8 is 1.6E-7 per reactor-year. Thus ~12% of the total accident sequence frequency (1.3E-6 per reactor-year) results in core damage. In other words, 88% of the accident sequences do not result in core damage.

Classes I, III and IV result in core damage. However, only 1E-6 of all class II events (i.e., the frequency of STC #52 in Figure 19D.5-3 divided by the frequency of all Class II) results in core damage. This low CDF value is attributable to the recovery of failed systems (or AC power), the ability of the ABWR RHR pumps to pump saturated water without cavitating and the ability of core cooling systems to continue to inject water to the reactor following operation of the COPS system.

19D.5.12.3 Core Melt Arrest

In addition to the 94% of accident sequences which do not result in core damage, in another 11% (Release Node N frequency of 1.6E-7 per reactor-year) the core melt is arrested either in the RPV or in the containment, without ever breaching the containment structural integrity. This means that in virtually all of the accident sequences, either radioactive material remains in the reactor vessel or is contained within the containment boundaries and not released to the environment (except through normal containment leakage). This is attributable to equipment and power recovery prior to containment failure and to "passive mitigation," i.e., flooding of the molten core from the suppression pool water when passive flooder system actuates.

Considering only those accident sequences in which core melt starts, (i.e., exclude certain class II events where core melt was not initiated), then the core melt arrest constitutes approximately 86% of all such sequences. The frequency of core damage with significant fission product release, which includes all categories except NCL and OK, is 2.2E-8 per reactor year.

The containment design incorporates a containment overpressure protection system which is designed to ensure that any sequence which is not arrested in the containment will have low consequences. This system consists of a line originating in the wetwell which exhausts to the plant stack. If the containment pressure rises to a level where containment integrity could be challenged, a rupture disk opens relieving the containment pressure. If there is no suppression pool bypass, the containment does not reach the rupture disk setpoint for about 24 hours. This ensures a late release with low magnitude. The frequency of these events is 2.1E-8, or 13% of all core damage events. The frequency of all other events is only 9.8E-10. Thus, the upper bound for releases with the potential to be early or have high magnitude is 0.6%.

19D.5.12.4 Probability of Containment Structural Failure due to Loss of Heat Removal

One of the goals of the ABWR design is to assure that highly reliable heat removal systems be provided to reduce the probability of containment failure by loss of heat removal.

The frequency of containment structural failure resulting

from a loss of containment heat removal systems is evaluated to be 1.1E-9 per reactor-year. Core damage occurs in only 0.1% of these events. This low number demonstrates that the goal is met for the ABWR design. The ABWR features and other factors that contribute to this low value are:

(1) Three divisions of heat removal systems.

(2) Ability to re-establish the main condenser as a heat sink in certain accidents.

(3) Ability to remove heat using RWCU heat exchanger.

(4) Long times before containment pressure reaches a value which could threaten containment integrity, which enables recovery of power and failed heat removal systems.

(5) Presence of the containment overpressure protection system.

(6) Ability of the core cooling systems to continue to maintain the core cooling function following structural failure of the containment.

19D.5.12.5 Frequencies for Radioactive Release Categories

The important release characteristics for each of the severe accident sequences are summarized in Figure 19D.5-3. The first branch of the tree identifies the initiating event for each sequence. This information is used to specify the first four letters of the severe accident sequences used for the deterministic analyses performed in Section 19E.2.2. Later branches identify the potential impact of other important issues such as flooder operation and mode of fission product release. Table 19D.5-7 identifies the deterministic accident sequence associated with each of the end states in Figure 19D.5-3 with a frequency of at least 1E-11. Note that all sequences with an intact containment and no rupture disk opening are assigned to class NCL (Normal Containment Leakage). Sequences with a frequency of less than 1E-11 are neglected.

The deterministic sequences are then binned according to the characteristics of the fission product release. Table 19E.3-6 indicates combination of the deterministic sequences into release bins. This combination was done by considering the timing and magnitude of the releases. Column P(i) of Table 19E.3-6 gives the probabilities associated with each of the consequence bins with frequency above 1E-10. These values are simply the result of summing all of the sequences in a given consequence bin.

STC#53 in Figure 19D.5-3 was binned with Case 9, the worst of the consequence bins. This is a very conservative assumption since the frequency associated with this sequence is the initiating event frequency for ATWS events. The assumption is made only because there is a negligible effect on the consequence analysis. If this assumption impacts the risk, a containment event tree should be developed for ATWS events.

19D.5.13 Sensitivity of Containment Performance Analysis to RHR Recovery Assumptions

It was noted that late RHR recovery is considered prior to pool bypass in the CET. Hence, the effects of pool bypass on the probability of RHR recovery are not explicitly considered in the model.

The probabilities of late RHR (non-)recovery under various conditions are summarized in Table 19D.5-6. These probabilities assess whether RHR is recovered prior to operation of COPS or overpressure structural failure of the containment. The probability of RHR recovery will vary for different accident subclasses, for sequences with the core damage progression terminated in-vessel, and for sequences with active injection to the lower drywell after RPV failure because of differences in the availability of AC power, sequence timing and other factors.

19D.5.13.1 Minimum RHR Recovery Probability with Pool Bypass

MAAP analysis indicates that the time available for late recovery of RHR prior to COPS actuation would be 5 hours for sequences with pool bypass. Using the method described in Subsection 19D.5.4 the probability of failure to recover at least one division of RHR is 0.4.

This non-recovery probability is considered a bounding value which will be applied for sequences in all accident classes under all conditions for a conservative estimate of the impact of pool bypass and RHR recovery on COPS operation and containment structural failure. Note that late RHR

recovery is only an issue for those accident classes with RHR unavailable at core damage initiation. Accident classes IA_0 and IIIA_0 have RHR available and need not be considered. Furthermore, the impact of pool bypass on RHR recovery is not an issue for accident class II since these sequences are transients where steam discharge occurs directly to the suppression pool.

The corrected probability of RHR non-recovery can be approximated from the previously calculated probability (which neglected the effect of pool bypass) by adding an additional term to account for sequences which would otherwise have had successful recovery. The probability of these sequences is multiplied by the probability of bypass and the probability of recovery for sequences with pool bypass. The modified RHR non-recovery probabilities are shown on Table 19D.5-8 for comparison with the existing values.

19D.5.13.2 Impact on Sequences with In-vessel Core Damage Mitigation

For transient sequences with core melt arrest in-vessel steam discharges which occur will be directed into the suppression pool and the existence of a pool bypass pathway (open vacuum breaker) does not impact containment heatup and the time available to recover RHR. Only for LOCA sequences terminated in-vessel in accident classes IIIA_1 and IIID would pool bypass potentially impact containment pressurization rates and the RHR recovery probability. However, the probability of class IIIA_1 sequences with core melt arrest in-vessel is negligible.

The probability of class IIID sequences with core melt arrest in-vessel is 1.89E-10. The impact of adjusting the RHR recovery probability for this sequence is given in Table 19D.5-9.

19D.5.13.3 Sequences with RPV Failure

An inspection of the CETs shows that almost all of the RPV failure sequences have drywell sprays available. For these sequences pool bypass will not effect containment pressurization since the steam will be quenched by the spray. Therefore, only sequences with no drywell spray availability need to be considered further. The impact of pool bypass on the probability of COPS operation and containment structural failure for these sequences is summarized below.

Frequency of sequences without in-vessel core damage arrest or sprays (and without early containment failure or drywell overtemperature failure) may be calculated from the CETs by multiplying the accident class frequency times the branch point probabilities for all branches up to and including HTF for the pathway following "NO DW INJECT". Most of the classes have negligible impact on the risk (i.e. the frequency of sequence pathway is less than 1E-11). Only class IB1 and ID must be considered further. The impact of pool bypass for these events is then estimated by replacing the original RHR non-recovery probabilities with the values taken from Table 19D.5-8. The results are shown in Table 19D.5-9.

19D.5.13.4 Conclusions

The release category frequencies were modified to account for the impact of suppression pool bypass on the time available for recovery of containment heat removal. The overall frequency of COPS operation is 2.1E-8 and that for late drywell head structural failure is 5.3E-10. The impact of pool bypass on the late RHR recovery probability is to increase the frequency of COPS operation by 4.8E-12 (an increase of 0.02%) and of drywell head failure by 7.7E-14 (an increase of .02%). Thus, consideration of pool bypass in the calculation of RHR recovery has no impact on risk.

Table 19D.5-1

DESCRIPTION OF ACCIDENT EVENT CLASSES

Event	Boron Injected?	Core Cooling Available?	Containment Heat Removal Available?	Relative Time of Core Melt and Containment Structural Failure	Accident Class
Transient	Not Applicable (N/A)	No	Yes	Core Melts First	I
Transient	N/A	No	No	Core Melts First	I
Transient	N/A	Yes	Yes	Successful Mitigation	Plant OK
Transient	N/A	Yes	No	Containment Fails First	II
LOCA	N/A	No	Yes	Core Melts First	III
LOCA	N/A	No	No	Core Melts First	III
LOCA	N/A	Yes	Yes	Successful Mitigation	Plant OK
LOCA	N/A	Yes	No	Containment Fails First	II
ATWS	Yes	No	Yes	Core Melts First	I
ATWS	Yes	No	No	Core Melts First	I
ATWS	Yes	Yes	Yes	Successful Mitigation	Plant OK
ATWS	Yes	Yes	No	Containment Fails First	II
ATWS	No	No	Yes	Core Melts First	I
ATWS	No	No	No	Core Melts First	I

Table 19D.5-1 (Continued)

DESCRIPTION OF ACCIDENT EVENT CLASSES

Event	Boron Injected?	Core Cooling Available?	Containment Heat Removal Available?	Relative Time of Core Melt and Containment Structural Failure	Accident Class
ATWS	No	Yes	Yes	Containment Fails First	IV
ATWS	No	Yes	No	Containment Fails First	IV
Containment Bypass	N/A	No	Yes or No	Containment (i.e. suppression pool) is bypassed. Therefore, no suppression pool scrubbing of radioactive releases. Relative time of core melt and containment failure immaterial.	V
Containment Bypass	N/A	Yes	Yes or No	Successful mitigation.	Plant OK

Table 19D.5-2

DESCRIPTION OF ACCIDENT EVENT I SUB-CLASSES

Event	Boron Injected?	Reactor Pressure	Core Cooling Available?	Accident Class	Comments
All transients except certain station black-out (SBO) events	Not Applicable (N/A)	High	No	IA	Because of high reactor pressure, there is a potential for containment structural failure shortly after core melt.
Station blackout events	N/A	High	No	IB-1	No core cooling or containment heat removal at the beginning because of absence of on-site and off-site power and RCIC failure. However, on-site power recovered in eight hours increasing the likelihood of recovery of core cooling and containment heat removal system.
Station black-out events	N/A	High	RCIC Available for the first eight hours	IB-2	Sequence with core decay heat reduced due to RCIC operation. Also suppression pool heats up prior to core melt shortening the time to containment structural failure.
Station black-out events	N/A	High	No	1B-3	No core cooling or containment heat removal.
ATWS	Yes or No	High or Low	No	IC	
All transients	N/A	Low	No	ID	

Table 19D.5-3

TREATMENT OF SUPPRESSION POOL BYPASS MECHANISMS IN THE PRA

Suppression Pool Bypass Mechanism	How Treated in the PRA	Reference Section/ Subsection
1. Ex-Containment LOCA	Containment bypass study	19E.2.3.3
- High Pressure Systems		
- Interfacing Systems LOCA		
2. Failure of Isolation Valves, Pipe Rupture	Containment bypass study	19E.2.3.3
3. Normal Containment Leaks (containment temperature $<500^{O}F$ or pressure <52 psig)	Modeled in CETs	19E.2.4.3
4. Containment Leaks (due to high containment temperature ($>500^{O}F$) and high pressure (52 psig))	Modeled in CETs	
5. Containment structural failure due to overpressure (>100 psig)	Modeled in CETs	
6. High Temperature Failure of the containment ($>700^{O}F$)	Modeled in CETs	
7. Uncovery of Horizontal Vent	Not expected to occur in the first 24 hours and therefore not modeled in CETs	19D.5.7.4
8. Low Probability Events	Not modeled in CETs	
- RPV Rupture		
- In-Vessel Steam Explosion		19E.2.1.3.1
- Ex-Vessel Steam Explosion		19E.2.1.3.1
- Basemat Penetration Following Core Melt		19E.2.1.3.6
9. Vacuum breaker leakage of failure	Modeled in CETs	

Table 19D.5-4

SUCCESS CRITERIA FOR CORE MELT ARREST

CASE	REACTOR AT HIGH PRESSURE	REACTOR AT LOW PRESSURE
CORE MELT ARREST IN RPV	Core Cooling 1 of 2 HPCF or RCIC or FW Containment Heat Removal 1 of 3 RHR	Core Cooling 1 of 2 HPCF or 1 of 3 LPFL or Condensate Injection or Fire System Water Injection Containment Heat Removal 1 of 3 RHR
RPV FAILS BUT CORE MELT ARRESTED PRIOR TO FISSION PRODUCT RELEASE	Not Applicable	Core Cooling 1 of 2 HPCF or 1 of 3 LPFL or Condensate Injection or Fire System Water Injection or Passive Flooder System Containment Heat Removal 1 of 3 RHR

Table 19D.5-5

(This Table deleted)

Table 19D.5-6

(This Table deleted)

Table 19D.5-7

Summary of How Each Accident Class is Treated in the CETs

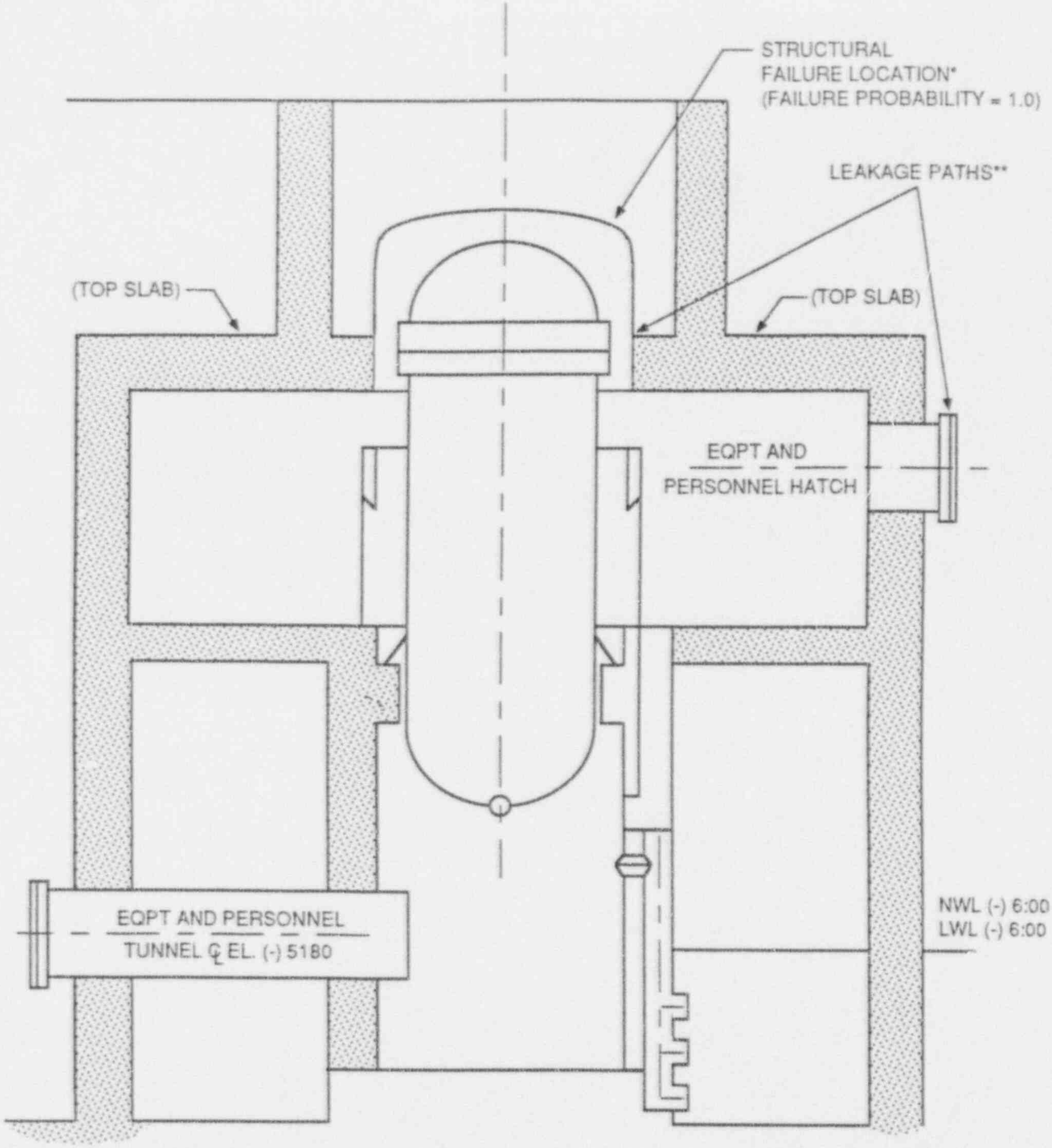
Accident Class	Containment Event Trees	Figure Number	Remarks
IA	IA	19D.5-4	Two CETs
	IA.1	19D.5-5	
IB-1	IB-1	19D.5-6	Two CETs
	IB-1.1	19D.5-7	
IB-2	IB-2	19D.5-8	
IB-3	IB-3	19D.5-9	Two CETs
	IB-3.1	19D.5-10	
IC	---	---	Low frequency event, included as part of IA CET
ID	ID	19D.5-11	
II	II	19D.5-12	
IIIA	IIIA	19D.5-13	Two CETs
	IIIA.1	19D.5-14	
IIID	IIID	19D.5-15	
IV	---	---	Low frequency event, assumed that containment structural failure leads to core melt 1% of the time.
V	---	---	Treated outside of CETs. See Subsection 19E.2.3.3

Table 19D.5-8

FREQUENCIES FOR RADIOACTIVE RELEASE CATEGORIES

(Per reactor year)

Accident Class:	IA	IA.1	IB-1	IB-1.1	IB-2	IB-3	IB-3.1	ID	II	IIIA	IIIA.1	IIID	Total
Release Categories													
NCL	4.2E-08	4.2E-10	1.9E-08	1.1E-10	1.7E-08	6.0E-10	3.6E-12	6.2E-08		4.3E-09	4.3E-11	1.2E-08	1.6E-07
LCHPIVDN		1.1E-12									1.1E-13		1.2E-12
LCHPFSDL		4.0E-12		1.1E-12			1.9E-13				4.1E-13		5.7E-12
LCHPPSDN		4.0E-13		1.1E-13			3.8E-15				4.1E-14		5.6E-13
LCHPPFPH	4.0E-10	4.1E-12		1.1E-12			3.8E-14			4.1E-11	4.2E-13		4.5E-10
LCHP00EH	4.0E-11	4.1E-13		1.1E-13			3.8E-15			4.1E-12	4.2E-14		4.5E-11
LCLPIVDN								6.2E-09					6.2E-09
LCLPFSDL			1.9E-10			3.1E-11		6.8E-10					6.8E-10
LCLPPFDH			1.9E-11			6.4E-13		1.4E-11					3.3E-11
LCLP00PH			0.0E+00			0.0E+00		0.0E+00					0.0E+00
LCLP00EH			0.0E+00			0.0E+00		0.0E+00					0.0E+00
SBRCIVD0					8.9E-10								8.9E-10
SBRCFSDH					1.0E-11								1.0E-11
SBRC00PH					0.0E+00								0.0E+00
SBRCPFDH					2.7E-11								2.7E-11
LBL[illegible]VDN												1.2E-09	1.2E-09
LBLCFSDL												1.3E-10	1.3E-10
LBLCPFDM												2.6E-12	2.6E-12
LBLC00PH												0.0E+00	0.0E+00
LBLC00DH												0.0E+00	0.0E+00
LHRC00DH									2.5E-10				2.5E-10
OK									2.5E-06				2.5E-06
Total	4.3E-08	4.3E-10	1.9E-08	1.1E-10	1.8E-08	6.4E-10	3.8E-12	6.9E-08	2.5E-06	4.4E-09	4.4E-11	1.3E-08	2.7E-06



* STRUCTURAL FAILURE OCCURS AT 100 psig WHEN TEMPERATURE IS 500°F

** LEAKAGE THROUGH DRYWELL HEAD EQUIPMENT HATCHES, PERSONAL AIR LOCKS OCCUR WHEN TEMPERATURE EXCEEDS 500°F AND PRESSURE EXCEEDS 52 psig.

Figure 19D.5-1 ABWR CONTAINMENT FAILURE LOCATION AND PROBABILITIES

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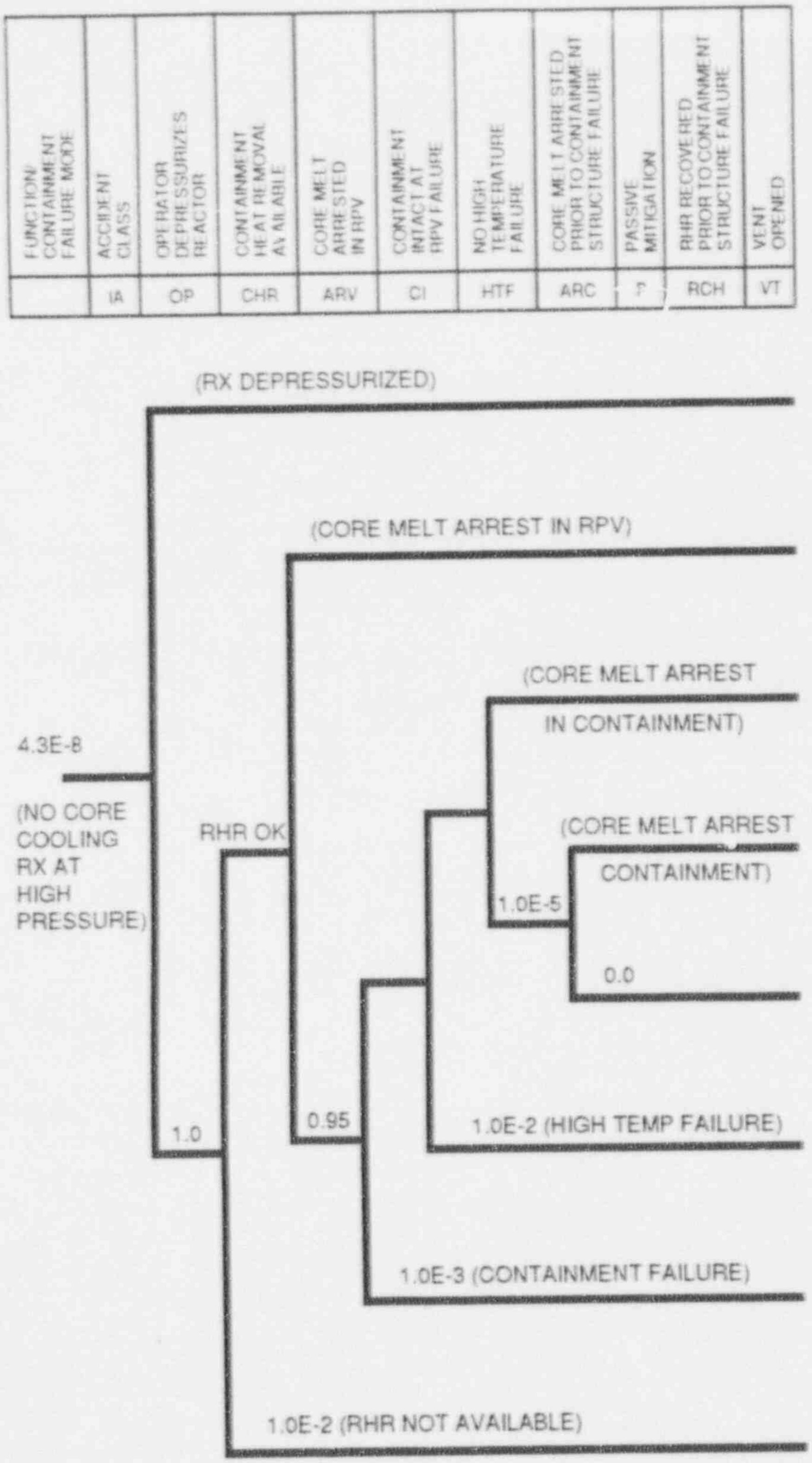
(Figure 19D.5-3 Deleted)

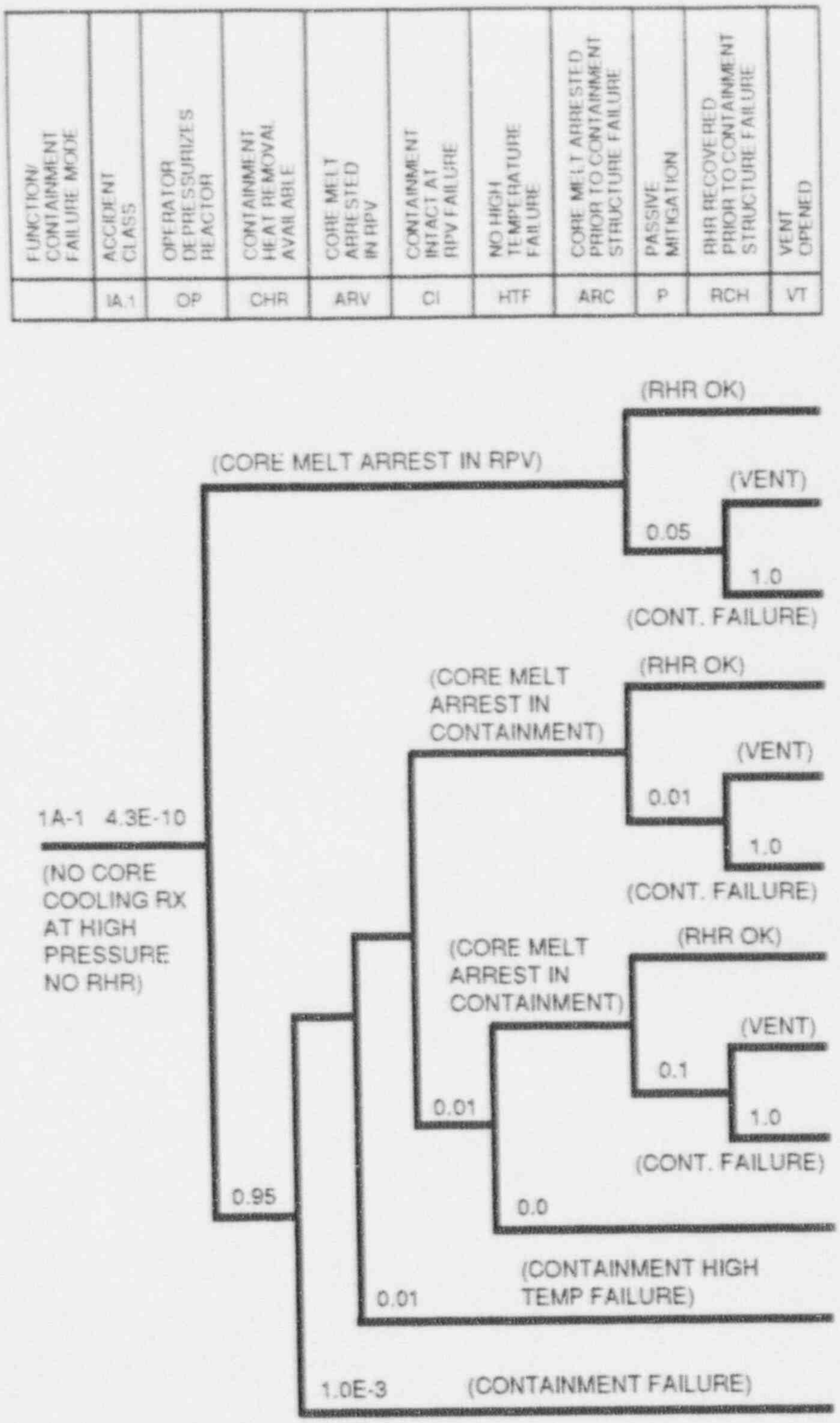


FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
0.0	-	-	-	N/A	1
2.1E-9	Y	N	N	LCHPIVNN	2
4.0E-8	Y	Y	N	LCHPFSNN	3
4.0E-13	Y	Y	N	LCHPPSNN	4
0.0	Y	Y	Y	N/A	5
4.0E-10	Y	Y	Y	LCHPPFPH	6
4.0E-11	Y	Y	Y	LCHPOOEH	7
4.3E-10	-	-	-	CET 1A.1	8

Figure 19D.5-4 TRANSIENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT HIGH PRESSURE (SHEET 1 0F 2)

CET SEQUENCE NUMBER	REMARKS
1	RX DEPRESSURIZATION BY OPERATOR IS MODELED IN FAULT TREES. THIS BRANCH IS NOT DEVELOPED IN THIS CET BUT IS INCLUDED IN CET FOR STATION BLACKOUT EVENTS.
2	CORE MELT ARRESTED IN RPV WHEN CORE COOLING IS RECOVERED IN ~1 HOUR. RHR IS AVAILABLE, THEREFORE CONTAINMENT STAYS INTACT.
3	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, CORE COOLING RECOVERED PRIOR TO REACHING VENT PRESSURE, CORE MELT ARRESTED, RHR IS AVAILABLE, THEREFORE CONTAINMENT STAYS INTACT.
4	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR IS AVAILABLE, THEREFORE CONTAINMENT STAYS INTACT.
5	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
6	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
7	CORE MELT, RPV FAILURE, CONTAINMENT STRUCTURE FAILS AT THE TIME OF RPV FAILURE, RESULTING IN EARLY AND HIGH RELEASE.
8	CONTINURED IN CET 1A.1 (FIGURE 19D.5-5).

Figure 19D.5-4 TRANSIENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT HIGH PRESSURE (SHEET 2 OF 2)

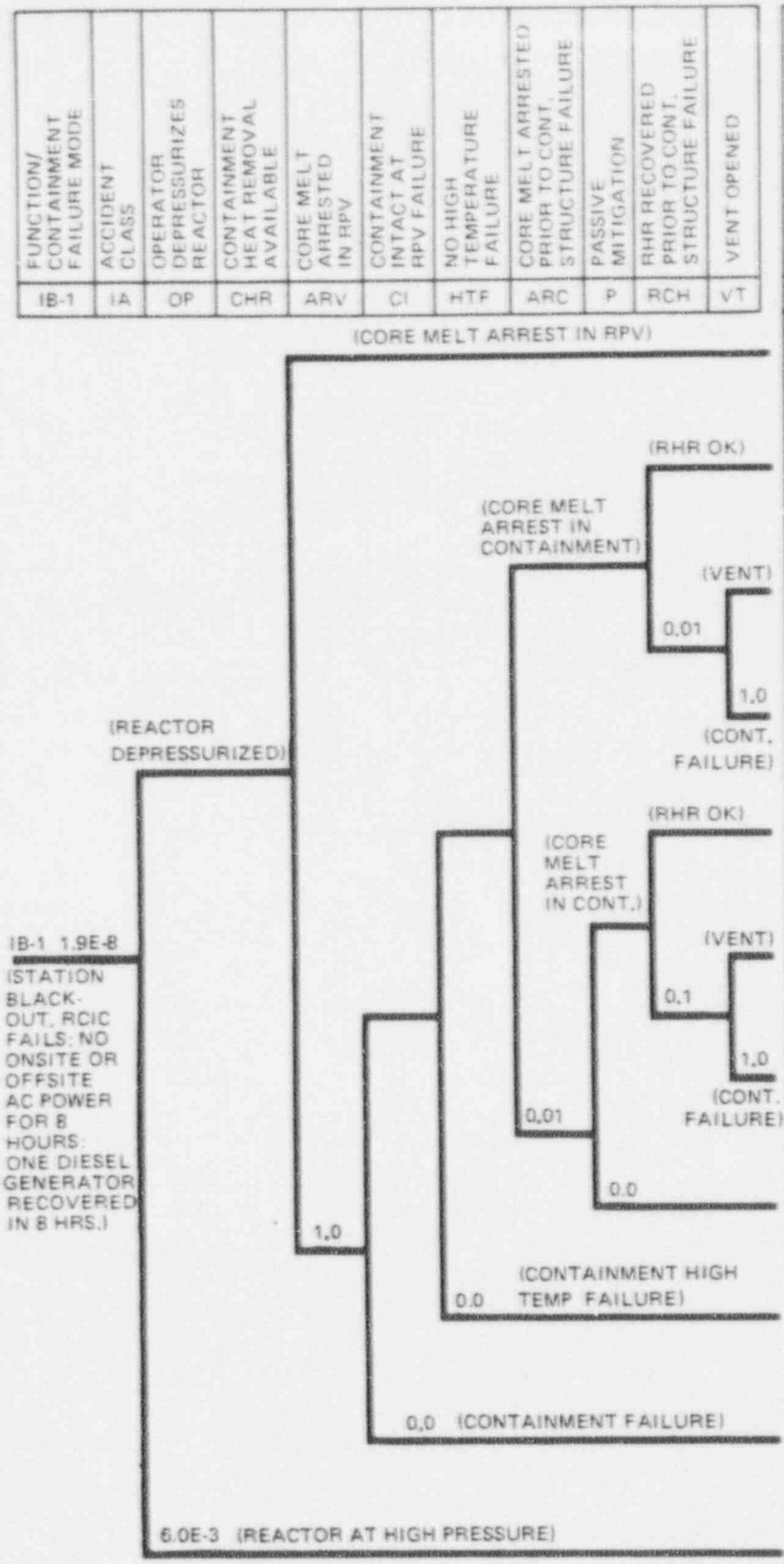


FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
2.0E-11	Y	N	N	LCHPIVNN	1
0.0	Y	N	N	LCHPIVVN	2
1.1E-12	Y	Y	Y	LCHPIVDN	3
4.0E-10	Y	Y	N	LCHPHRNN	4
0.0	Y	Y	N	LCHPFSVN	5
4.0E-12	Y	Y	Y	LCHPFSDL	6
3.6E-12	Y	Y	N	LCHPHRNN	7
0.0	Y	Y	N	LCHPPSVN	8
4.0E-13	Y	Y	Y	LCHPPSDN	9
0.0	Y	Y	Y	N/A	10
4.1E-12	Y	Y	Y	LCHPPFPH	11
4.1E-13	Y	Y	Y	LCHPOOEH	12

Figure 19D.5-5 LOSS OF CORE COOLING WITH REACTOR AT HIGH PRESSURE AND CONTAINMENT HEAT REMOVAL SYSTEMS UNAVAILABLE (SHEET 1 OF 2)

GET SEQUENCE NUMBER	REMARKS
1	LOSS OF HIGH PRESSURE CORE COOLING, RX NOT DEPRESSURIZED, RHR NOT AVAILABLE, CORE MELT ARRESTED IN RPV WHEN CORE COOLING IS RECOVERED IN ~1 HOUR, RHR RECOVERED PRIOR TO CONTAINMENT STRUCTURE FAILURE, CONTAINMENT INTACT.
2	CORE MELT SIMILAR TO SEQUENCE 1 EXCEPT RHR NOT RECOVERED BUT CONTAINMENT INTEGRITY MAINTAINED BY OPENING VENT.
3	CORE MELT SIMILAR TO SEQUENCE 1 AND 2 BUT NEITHER RHR IS RECOVERED, NOR IS VENT OPENED. CONTAINMENT STRUCTURE FAILS, CORE COOLING FAILS LONG AFTER CONTAINMENT FAILURE. MOST OF THE RELEASE OCCURS THROUGH THE SUPPRESSION POOL, THEREFORE LATE-LOW RELEASE EXPECTED.*
4	LOSS OF HIGH PRESSURE CORE COOLING, RX NOT DEPRESSURIZED, RHR NOT AVAILABLE, RPV FAILS, CORIUM ON DRYWELL FLOOR, CORE MELT ARRESTED PRIOR TO CONTAINMENT STRUCTURE FAILURE, RHR RECOVERED AND CONTAINMENT INTEGRITY IS MAINTAINED.
5	CORE MELT SIMILAR TO SEQUENCE 4, RHR NOT RECOVERED BUT VENT IS OPENEDTO MAINTAIN CONTAINMENT INTEGRITY.*
6	CORE MELT SIMILAR TO SEQUENCE 4, NO RHR, NO VENT, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
7	LOSS OF HIGH PRESSURE CORE COOLING, RX NOT DEPRESSURIZED, RHR NOT AVAILABLE, RPV FAILS, CORIUM ON DRYWELL FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR RECOVERED AND CONTAINMENT INTEGRITY IS MAINTAINED.
8	CORE MELT SEQUENCE SIMILAR TO SEQUENCE 7, RHR NOT RECOVERED BUT VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
9	CORE MELT SIMILAR TO SEQUENCE 7, NO RHR, NO VENT, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
10	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
11	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING FAILURE OF DRYWELL SPRAY FUNCTION.
12	CORE MELT, RPV FAILURE, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS, RESULTING IN EARLY AND HIGH RELEASE.
	*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0.

Figure 19D.5-5 LOSS OF CORE COOLING WITH REACTOR AT HIGH PRESSURE AND CONTAINMENT HEAT REMOVAL SYSTEMS UNAVAILABLE (SHEET 2 OF 2)



FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
0.0	–	–	–	LCLPIVNN	1
1.9E-8	Y	Y	N	LCLPHRNN	2
0.0	Y	Y	N	LCLPFSVN	3
1.9E-10	Y	Y	Y	LCLPFSDL	4
1.7E-10	Y	Y	N	LCLPHRNN	5
0.0	Y	Y	N	LCLPPFVN	6
1.9E-11	Y	Y	Y	LCLPPFDM	7
0.0	Y	Y	Y	N/A	8
0.0	Y	Y	Y	LCLPOOPH	9
0.0	Y	Y	Y	LCLPOOEH	10
1.1E-10	–	–	–	CET IB 1.1	11

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Figure 19D.5–6 LOSS OF CORE COOLING DURING A STATION BLACKOUT EVENT LASTING EIGHT HOURS, REACTOR AT LOW PRESSURE (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	STATION BLACKOUT (SBO), RCIC FAILS, CORE MELT ARREST IN RPV NOT CONSIDERED.
2	SBO, NO CORE COOLING, CORE MELT, RPV FAILURE, CORE MELT ARRESTED PRIOR TO CONTAINMENT STRUCTURE FAILURE AND RHR RECOVERED DUE TO RECOVERY OF ON-SITE AC POWER BEFORE 20 HOURS.
3	CORE MELT ARREST SIMILAR TO SEQUENCE 2, BUT RHR NOT RECOVERED. VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
4	CORE MELT ARREST SIMILAR TO SEQUENCE 2, BUT NEITHER RHR IS RECOVERED NOR VENT OPENED, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
5	SBO, NO CORE COOLING, CORE MELT, RPV FAILURE, CORIUM ON DRYWELL FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR RECOVERED WHEN ON-SITE AC POWER RECOVERS, CONTAINMENT INTEGRITY MAINTAINED.
6	CORE MELT ARREST SIMILAR TO SEQUENCE 5, RHR NOT RECOVERED BUT VENT OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
7	CORE MELT ARREST SIMILAR TO SEQUENCE 5, BUT NEITHER RHR IS RECOVERED NOR VENT IS OPENED, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT IMPACT OF CORE MELT ARREST.*
8	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
9	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
10	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS RESULTING IN EARLY AND HIGH RELEASE.
11	TRANSFER TO CET IB-11 (FIGURE 19D.5-7).

*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0

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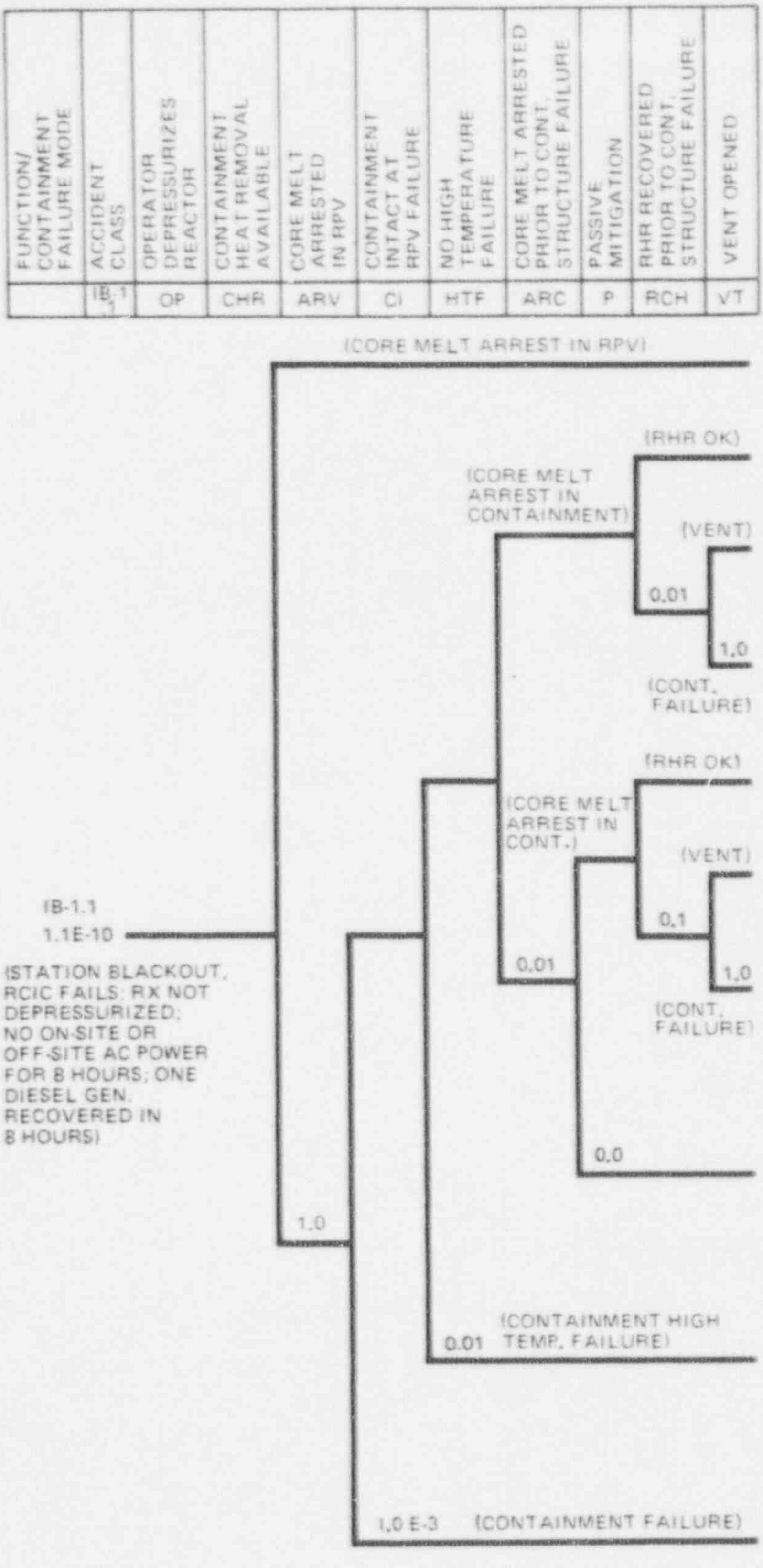
Figure 19D.5–6 LOSS OF CORE COOLING DURING A STATION BLACKOUT EVENT LASTING EIGHT HOURS, REACTOR AT LOW PRESSURE (SHEET 2 OF 2)



FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
0.0				LCHPIVNN	1
1.1E-10	Y	Y	N	LCHPHRNN	2
0.0	Y	Y	N	LCHPFSVN	3
1.1E-12	Y	Y	Y	LCHPFSDL	4
1.0E-12	Y	Y	N	LCHPHRNN	5
0.0	Y	Y	N	LCHPPSVN	6
1.1E-13	Y	Y	Y	LCHPPSDN	7
0.0	Y	Y	Y	N/A	8
1.1E-12	Y	Y	Y	LCHPPFPH	9
1.1E-13	Y	Y	Y	LCHPOOEH	10

89-326-25

Figure 19D.5–7 LOSS OF CORE COOLING DURING A STATION BLACKOUT EVENT LASTING EIGHT HOURS, REACTOR AT HIGH PRESSURE (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	STATION BLACKOUT (SBO), RCIC FAILS, CORE MELT ARREST IN RPV NOT CONSIDERED.
2	SBO, NO CORE COOLING, RPV AT HIGH PRESSURE CORE MELT, RPV FAILURE, CORE MELT ARRESTED BEFORE CONTAINMENT STRUCTURE FAILURE AND RHR RECOVERED DUE TO RECOVERY OF ON-SITE AC POWER BEFORE 20 HOURS.
3	CORE MELT ARREST SIMILAR TO SEQUENCE 2, BUT RHR NOT RECOVERED, VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
4	CORE MELT ARREST SIMILAR TO SEQUENCE 2, BUT NEITHER RHR IS RECOVERED NOR VENT OPENED, CONTAINMENT FAILS, CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
5	SBO, NO CORE COOLING, RPV AT HIGH PRESSURE, CORE MELT, RPV FAILURE, CORIUM ON DRYWELL FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR RECOVERED WHEN ON-SITE AC POWER RECOVERS, CONTAINMENT INTEGRITY MAINTAINED.
6	CORE MELT ARREST SIMILAR TO SEQUENCE 5, RHR NOT RECOVERED BUT VENT OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
7	CORE MELT ARREST SIMILAR TO SEQUENCE 5, BUT NEITHER RHR IS RECOVERED NOR IS VENT OPENED, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT IMPACT OF CORE MELT ARREST.*
8	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
9	CORE MELT, RPV AT HIGH PRESSURE, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
10	CORE MELT, RPV AT HIGH PRESSURE, RPV FAILURE, CORIUM ON FLOOR, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS RESULTING IN EARLY AND HIGH RELEASE.

*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0

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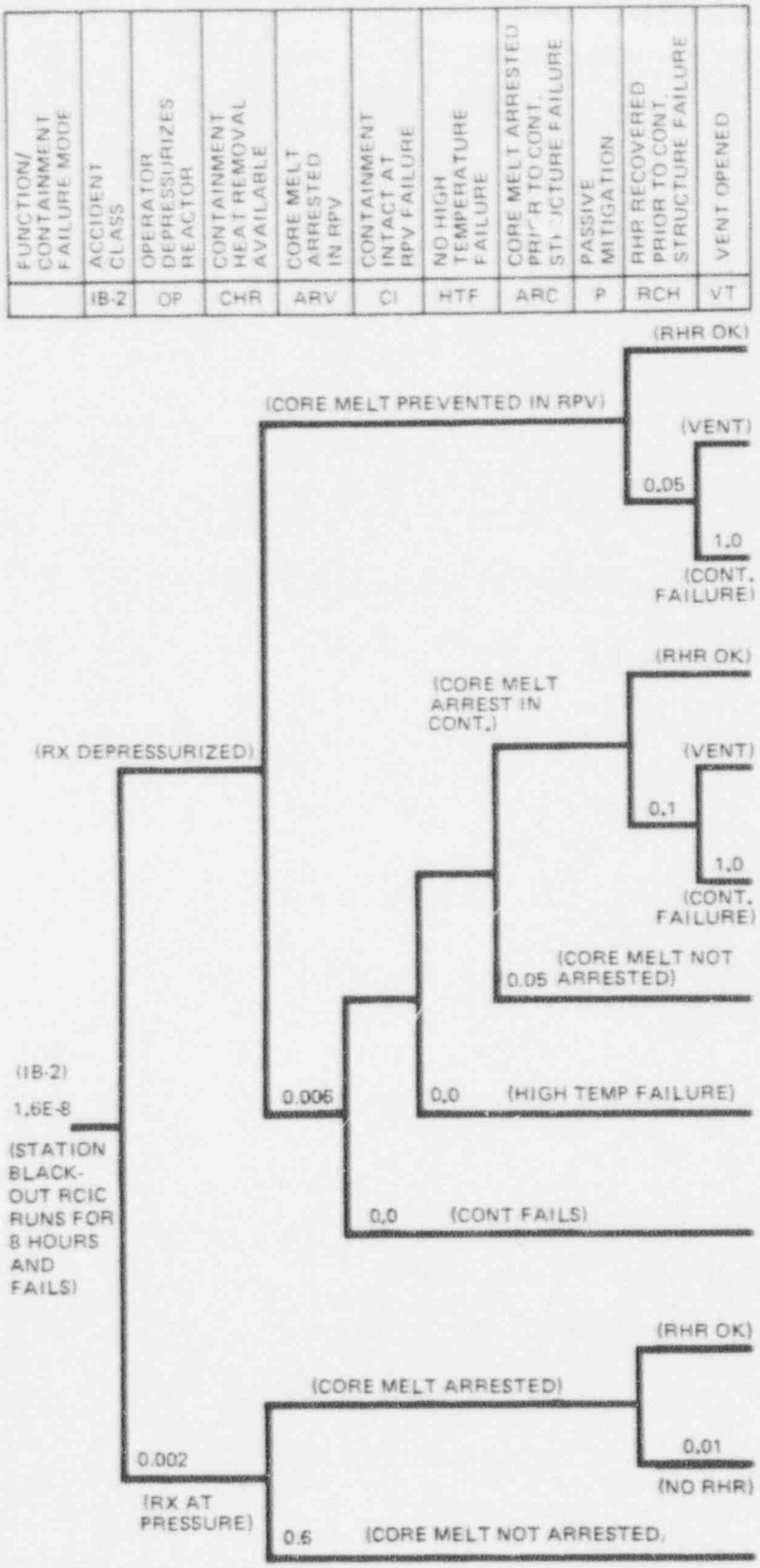
Figure 19D.5–7 LOSS OF CORE COOLING DURING A STATION BLACKOUT EVENT LASTING EIGHT HOURS, REACTOR AT HIGH PRESSURE (SHEET 2 OF 2)



FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
1.5E-8	Y	N	N	SBRCFANO	1
0.0	Y	N	N	SBRCFAVO	2
7.7E-10	Y	Y	Y	SBRCFADO	3
7.9E-11	Y	Y	N	SBRCHRNN	4
0.0	Y	Y	N	SBRCFSVN	5
8.8E12	Y	Y	Y	SBRCFSDL	6
8.8E-11	Y	Y	Y	SBRCPFDH	7
0.0	Y	Y	Y	SBRCOOPH	8
0.0	Y	Y	Y	SBRCOOEH	9
1.2E-11	Y	N	N	SBRCIVNN	10
1.2E-13	Y	Y	Y	SBRCIVDN	11
1.9E-11	Y	Y	Y	SBRCPFDH	12

89-326-27

Figure 19D.5–8 LOSS OF CORE COOLING AFTER EIGHT HOURS OF RCIC OPERATION DURING A STATION BLACKOUT EVENT (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	STATION BLACKOUT (SBO); RX DEPRESSURIZED; IF POWER IS RECOVERED WITHIN 2 HOURS OF RCIC FAILURE CORE MELT IS ARRESTED IN RPV AND RHR IS RECOVERED TO MAINTAIN CONTAINMENT INTEGRITY.
2	CORE MELT ARREST AS IN SEQUENCE 1, BUT RHR IS NOT RECOVERED, VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
3	CORE MELT ARREST AS IN SEQUENCE 1, BUT NEITHER RHR IS RECOVERED NOR VENT IS OPENED. CONTAINMENT FAILS, CORE COOLING FAILS LONG AFTER CONTAINMENT FAILURE. MOST OF THE RELEASE OCCURS THROUGH THE SUPPRESSION POOL, THEREFORE LATE-LOW RELEASE EXPECTED.*
4	SBO, RX DEPRESSURIZED. IF POWER IS RECOVERED PRIOR TO CONTAINMENT STRUCTURE FAILURE (~20 HOURS), CORE MELT IS ARRESTED IN CONTAINMENT, RHR RECOVERY MAINTAINS CONTAINMENT INTEGRITY.
5	CORE MELT ARREST AS IN SEQUENCE 4, BUT RHR IS NOT RECOVERED. VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
6	CORE MELT ARREST AS IN SEQUENCE 4, BUT NEITHER RHR IS RECOVERED NOR VENT IS OPENED. CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT THE EFFECT OF CORE MELT ARREST.*
7	SBO, RX DEPRESSURIZED, POWER NOT RECOVERED AND CORE MELT NOT ARRESTED, CORE MELT AND CONTAINMENT STRUCTURE FAILURE OCCURS, RESULTING IN LARGE RELEASE.
8	SBO, CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE OCCURS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
9	SBO, CORE MELT, RPV FAILURE, CORIUM ON FLOOR, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS RESULTING IN EARLY AND HIGH RELEASE.
10	SBO, RX NOT DEPRESSURIZED. IF POWER IS RECOVERED WITHIN 2 HOURS OF RCIC FAILURE, CORE MELT IS ARRESTED IN RPV AND RHR IS RECOVERED TO MAINTAIN CONTAINMENT INTEGRITY.
11	CORE MELT ARREST AS IN SEQUENCE 10, BUT RHR IS NOT RECOVERED, ASSUME VENT CANNOT HELP. CONTAINMENT FAILS RESULTING IN LARGE RELEASE.
12	SBO, RX AT PRESSURE, CORE MELT, NO RECOVERY, LEADS TO RPV FAILURE, CONTAINMENT STRUCTURE FAILURE AND HIGH RELEASE.

*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0.

89-326-28

Figure 19D.5–8 LOSS OF CORE COOLING AFTER EIGHT HOURS OF RCIC OPERATION DURING A STATION BLACKOUT EVENT (SHEET 2 OF 2)

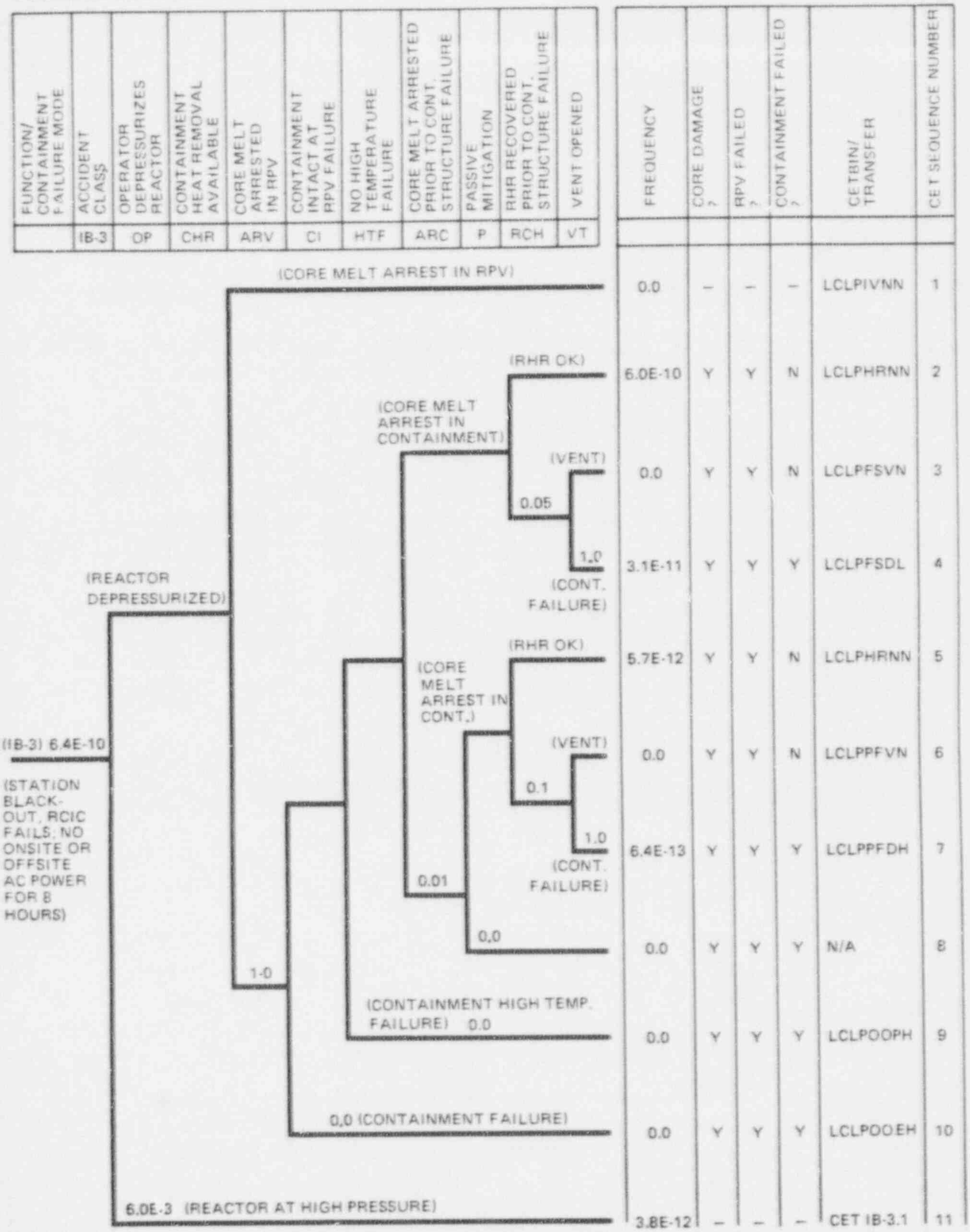


Figure 19D.5–9 LOSS OF CORE COOLING DURING A STATION BLACKOUT EVENT LASTING EIGHT HOURS, REACTOR AT LOW PRESSURE (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	STATION BLACKOUT (SBO), RCIC FAILS, CORE MELT ARREST IN RPV NOT CONSIDERED.
2	SBO, NO CORE COOLING, CORE MELT, RPV FAILURE, CORE MELT ARRESTED BEFORE CONTAINMENT STRUCTURE FAILURE AND RHR RECOVERED DUE TO RECOVERY OF ON-SITE AC POWER BEFORE 20 HOURS.
3	CORE MELT ARREST SIMILAR TO SEQUENCE 2, BUT RHR NOT RECOVERED, VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
4	CORE MELT ARREST SIMILAR TO SEQUENCE 2, BUT NEITHER RHR IS RECOVERED NOR VENT OPENED. CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
5	SBO, NO CORE COOLING, CORE MELT, RPV FAILURE, CORIUM ON DRYWELL FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR RECOVERED WHEN ON-SITE AC POWER RECOVERS, CONTAINMENT INTEGRITY MAINTAINED.
6	CORE MELT ARREST SIMILAR TO SEQUENCE 5, RHR NOT RECOVERED BUT VENT OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
7	CORE MELT ARREST SIMILAR TO SEQUENCE 5, BUT NEITHER RHR IS RECOVERED NOR VENT IS OPENED. CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT IMPACT OF CORE MELT ARREST.*
8	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
9	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
10	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS RESULTING IN EARLY AND HIGH RELEASE.
11	TRANSFER TO CET IB-31 (FIGURE 19D.5–10).

*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0.

89-326-30

Figure 19D.5–9 LOSS OF CORE COOLING DURING A STATION BLACKOUT EVENT LASTING EIGHT HOURS, REACTOR AT LOW PRESSURE (SHEET 2 OF 2)

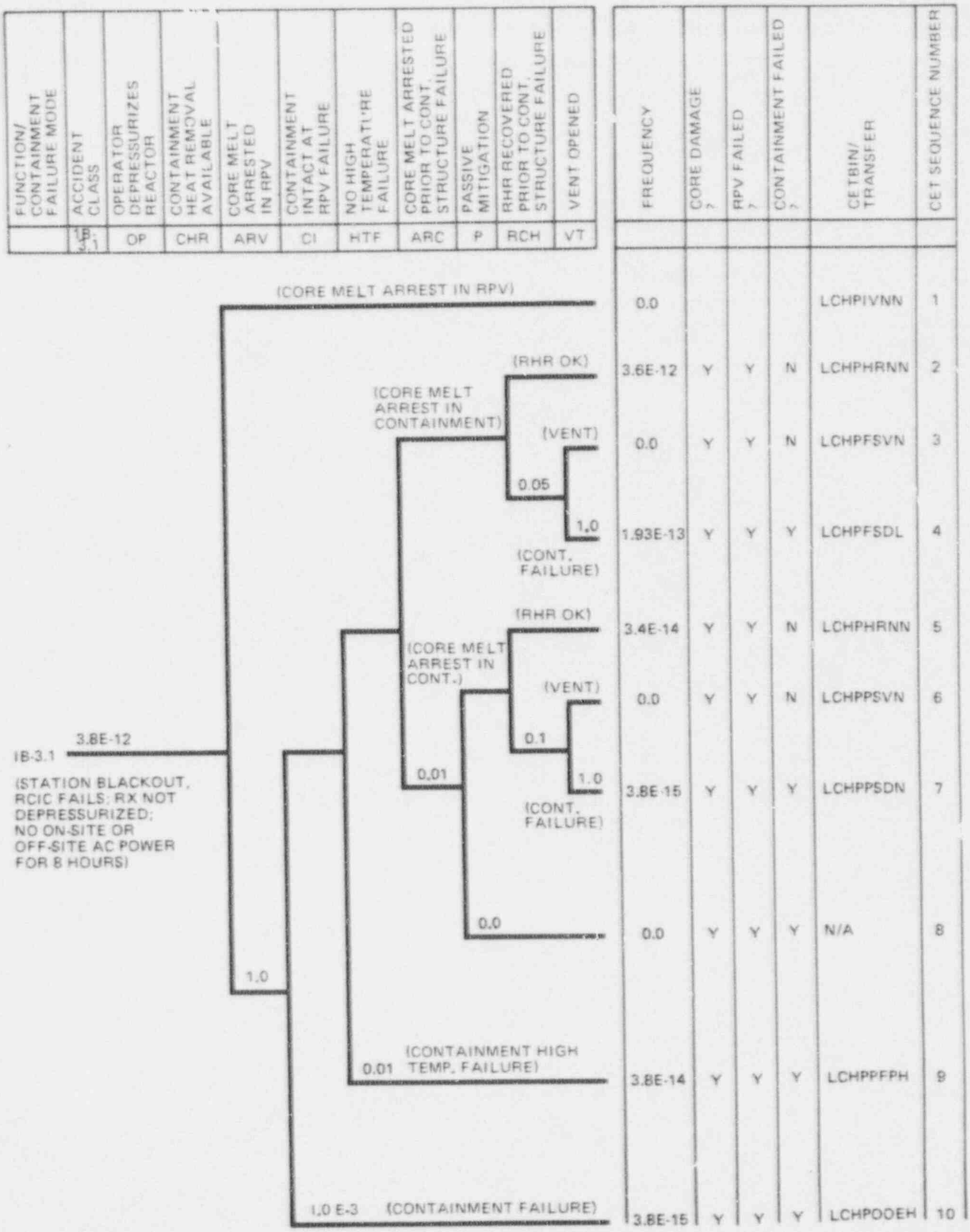


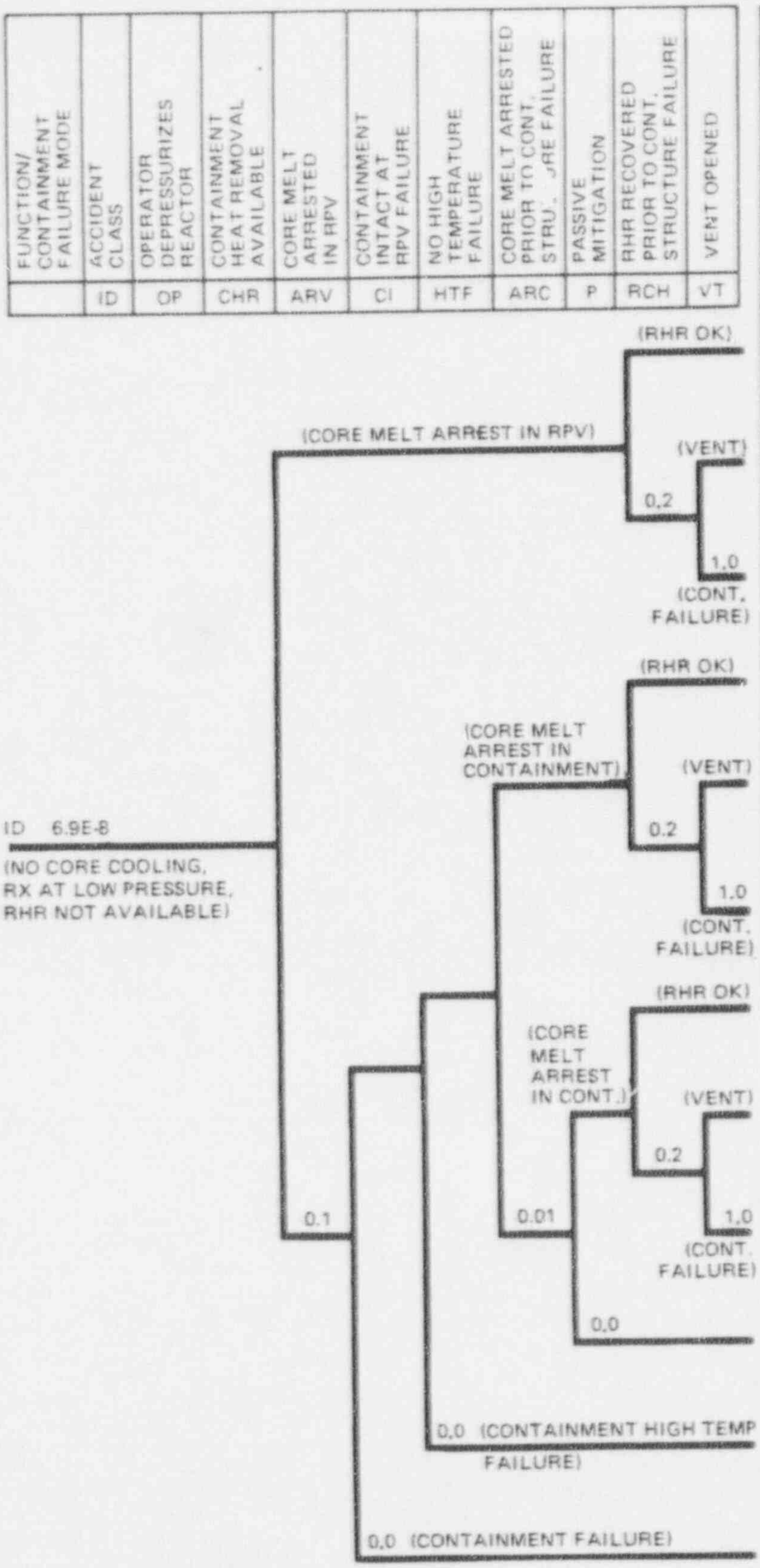
Figure 19D.5–10 LOSS OF CORE COOLING DURING A STATION BLACKOUT EVENT, REACTOR AT HIGH PRESSURE (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	STATION BLACKOUT (SBO), RCIC FAILS, CORE MELT ARREST IN RPV NOT CONSIDERED.
2	SBO, NO CORE COOLING, RPV AT HIGH PRESSURE CORE MELT, RPV FAILURE, CORE MELT ARRESTED PRIOR TO CONTAINMENT STRUCTURE FAILURE AND RHR RECOVERED DUE TO RECOVERY OF ON-SITE AC POWER BEFORE 20 HOURS.
3	CORE MELT ARREST SIMILAR TO SEQUENCE 2, BUT RHR NOT RECOVERED, VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
4	CORE MELT ARREST SIMILAR TO SEQUENCE 2, BUT NEITHER RHR IS RECOVERED NOR VENT OPENED. CONTAINMENT FAILS, CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
5	SBO, NO CORE COOLING, RPV AT HIGH PRESSURE, CORE MELT, RPV FAILURE, CORIUM ON DRYWELL FLOOR, PASSIVE FLOODER INITIATES, SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR RECOVERED WHEN ON-SITE AC POWER RECOVERS, CONTAINMENT INTEGRITY MAINTAINED.
6	CORE MELT ARREST SIMILAR TO SEQUENCE 5, RHR NOT RECOVERED BUT VE[illegible]ED TO MAINTAIN CONTAINMENT INTEGRITY.*
7	CORE MELT ARREST SIMILAR TO SEQUENCE 5, BUT NEITHER RHR IS RECOVERED NOR VENT IS OPENED, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT IMPACT OF CORE MELT ARREST.*
8	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
9	CORE MELT, RPV AT HIGH PRESSURE, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
10	CORE MELT, RPV AT HIGH PRESSURE, RPV FAILURE, CORIUM ON FLOOR, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS RESULTING IN EARLY AND HIGH RELEASE.

*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0.

89-326-32

Figure 19D.5–10 LOSS OF CORE COOLING DURING A STATION BLACKOUT EVENT, REACTOR AT HIGH PRESSURE (SHEET 2 OF 2)



FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
5.0E-8	Y	N	N	LCLPIVNN	1
0.0	Y	N	N	LCLPIVVN	2
1.2E-8	Y	Y	Y	LCLPIVDN	3
5.5E-9	Y	Y	N	LCLPHRNN	4
0.0	Y	Y	N	LCLPFSVN	5
1.4E-9	Y	Y	Y	LCLPFSDL	6
5.5E-11	Y	Y	N	LCLPHRNN	7
0.0	Y	Y	N	LCLPPFVN	8
1.4E-11	Y	Y	Y	LCLPPFDH	9
0.0	Y	Y	Y	N/A	10
0.0	Y	Y	Y	LCLPOOPH	11
0.0	Y	Y	Y	LCLPODEH	12

89-326-33

Figure 19D.5–11 TRANSIENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT LOW PRESSURE (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	LOSS OF LOW PRESSURE CORE COOLING, RX DEPRESSURIZED, RHR NOT AVAILABLE, CORE MELT ARRESTED IN RPV WHEN CORE COOLING IS RECOVERED IN ~ 1 HOUR, RHR RECOVERED BEFORE VENT PRESSURE IS REACHED, CONTAINMENT INTACT.
2	CORE MELT SIMILAR TO SEQUENCE 1 EXCEPT RHR NOT RECOVERED BUT CONTAINMENT INTEGRITY MAINTAINED BY OPENING VENT.*
3	CORE MELT SIMILAR TO SEQUENCE 1 AND 2 BUT NEITHER RHR IS RECOVERED, NOR VENT IS OPENED. CONTAINMENT STRUCTURE FAILS, CORE COOLING FAILS LONG AFTER CONTAINMENT FAILURE. MOST OF THE RELEASE OCCURS THROUGH THE SUPPRESSION POOL, THEREFORE LATE-LOW RELEASES EXPECTED.*
4	LOSS OF LOW PRESSURE CORE COOLING, RX DEPRESSURIZED, RHR NOT AVAILABLE, RPV FAILS, CORIUM ON DRYWELL FLOOR, CORE MELT ARRESTED PRIOR TO CONTAINMENT STRUCTURE FAILURE, RHR RECOVERED AND CONTAINMENT INTEGRITY IS MAINTAINED.
5	CORE MELT SIMILAR TO SEQUENCE 4, RHR NOT RECOVERED BUT VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
6	CORE MELT SIMILAR TO SEQUENCE 4, NO RHR, NO VENT, CONTAINMENT STRUCTURE FAILS. CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
7	LOSS OF LOW PRESSURE CORE COOLING, RX DEPRESSURIZED, RHR NOT AVAILABLE, RPV FAILS, CORIUM ON DRYWELL FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR RECOVERED AND CONTAINMENT INTEGRITY IS MAINTAINED.
8	CORE MELT SEQUENCE SIMILAR TO SEQUENCE 7, RHR NOT RECOVERED BUT VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
9	CORE MELT SIMILAR TO SEQUENCE 7, NO RHR, NO VENT, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT EFFECTS OF CORE MELT ARREST.*
10	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
11	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
12	CORE MELT, RPV FAILURE, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS RESULTING IN EARLY AND HIGH RELEASE.

*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0.

89-326-34

Figure 19D.5–11 TRANSIENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT LOW PRESSURE (SHEET 2 OF 2)

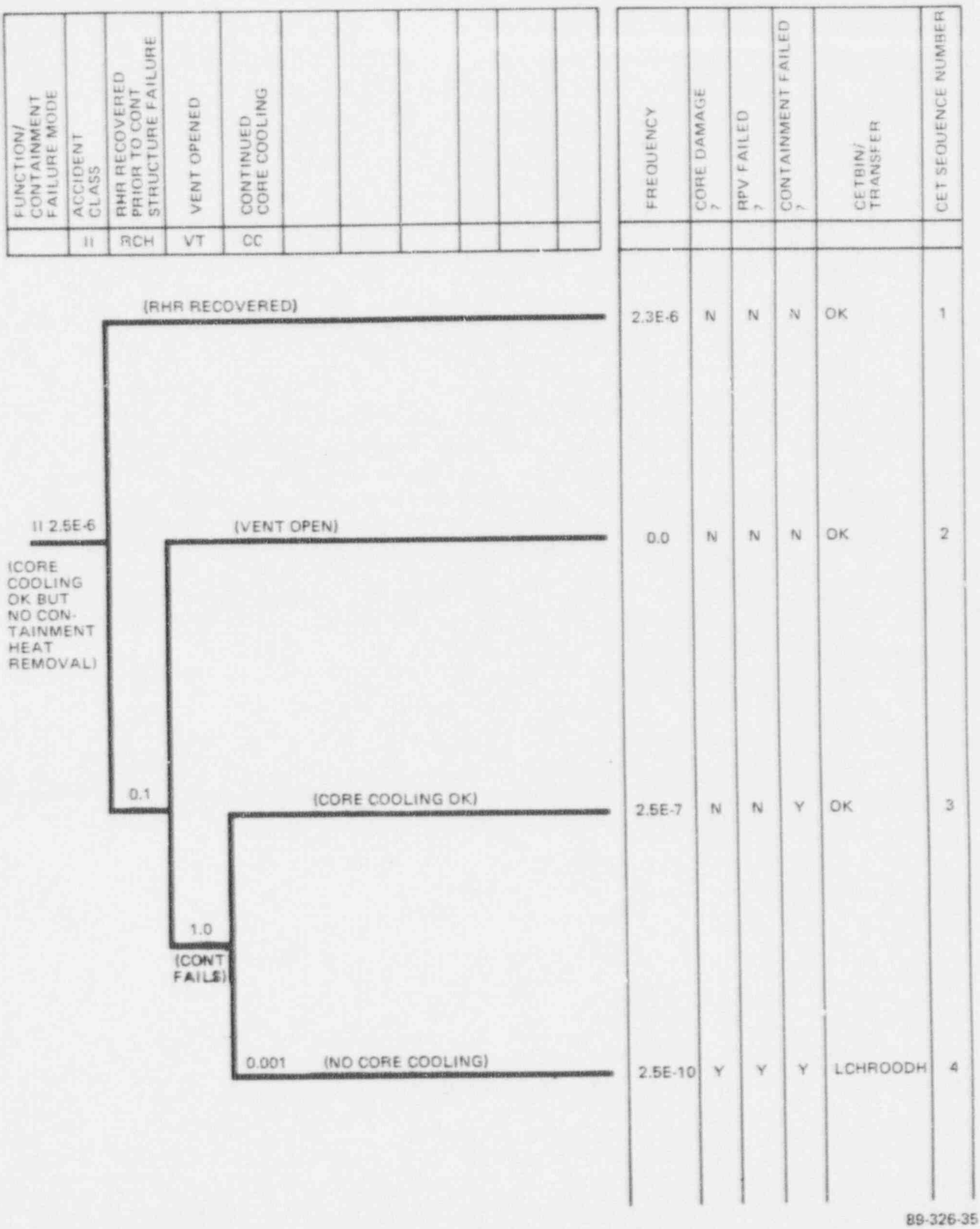
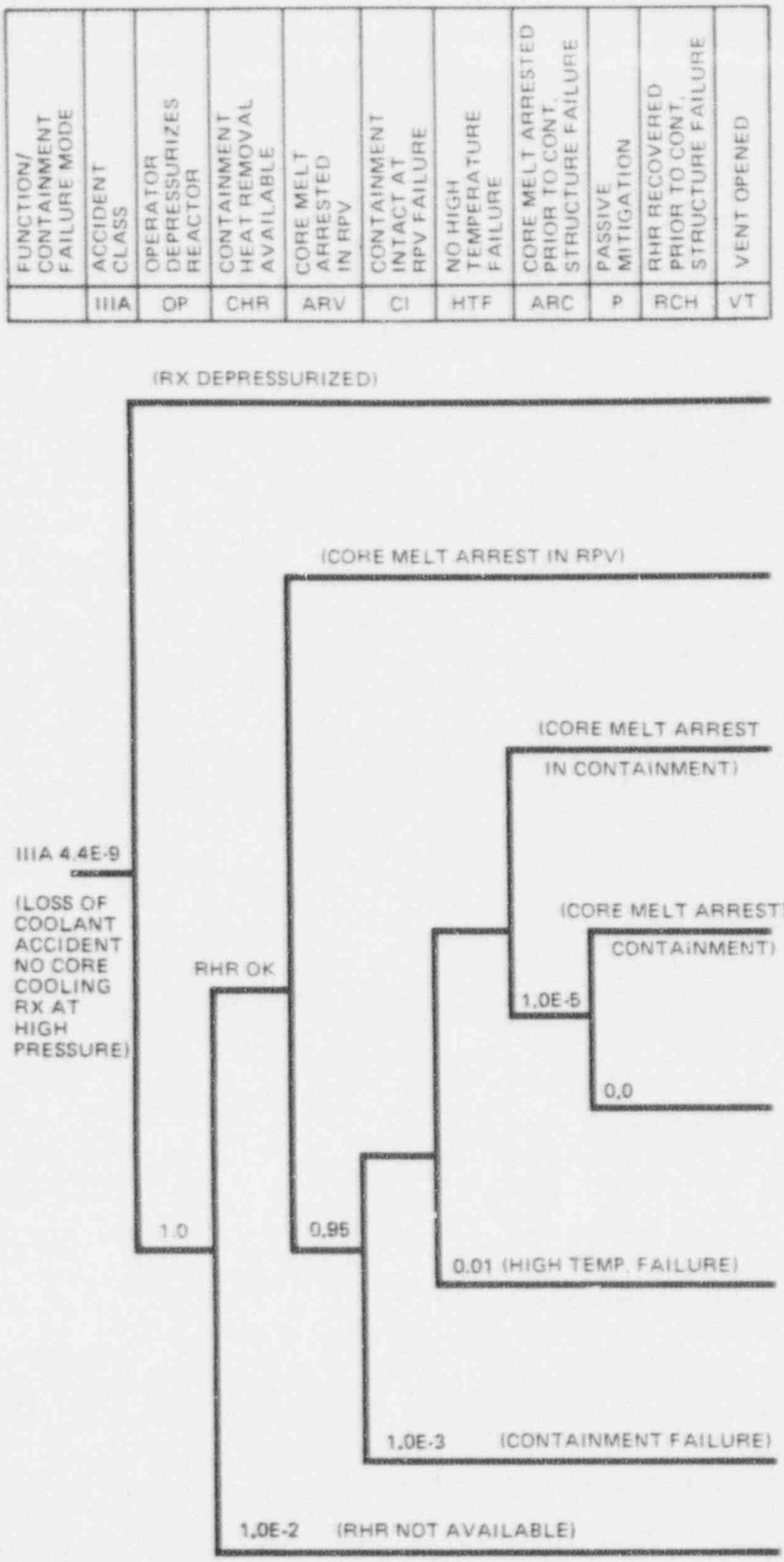


Figure 19D.5–12 TRANSIENT OR ACCIDENT WITH SUCCESSFUL CORE COOLING BUT FAILURE OF CONTAINMENT HEAT REMOVAL SYSTEMS (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	CORE COOLING SUCCESSFUL, CONTAINMENT HEAT REMOVAL NOT AVAILABLE INITIALLY, BUT IS RECOVERED PRIOR TO CONTAINMENT STRUCTURE FAILURE, CORE COOLING AND CONTAINMENT HEAT REMOVAL SUCCESSFUL.
2	CORE COOLING SUCCESSFUL, CONTAINMENT HEAT REMOVAL NOT AVAILABLE, OPERATOR OPENS VENT WHEN VENT PRESSURE IS REACHED. CORE COOLING AND CONTAINMENT HEAT REMOVAL SUCCESSFUL.*
3	CORE COOLING SUCCESSFUL, BUT CONTAINMENT HEAT REMOVAL IS NOT AVAILABLE, VENT NOT OPEN, CONTAINMENT FAILS, MOST LIKELY LOCATION OF FAILURE IS IN DRYWELL, CORE COOLING CONTINUES DESPITE CONTAINMENT STRUCTURE FAILURE.*
4	CORE COOLING INITIALLY SUCCESSFUL, WITHOUT CONTAINMENT HEAT REMOVAL CONTAINMENT STRUCTURE FAILS, CORE COOLING CONTINUES AFTER CONTAINMENT STRUCTURE FAILURE FOR A LONG TIME, CORE COOLING EVENTUALLY FAILS LEADING TO CORE MELT, HOWEVER, IN THIS SEQUENCE CORE COOLING IS ASSUMED TO FAIL EARLIER LEADING TO A LATE HIGH RELEASE.
	*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0.

89-326-36

Figure 19D.5–12 TRANSIENT OR ACCIDENT WITH SUCCESSFUL CORE COOLING BUT FAILURE OF CONTAINMENT HEAT REMOVAL SYSTEMS (SHEET 2 OF 2)



FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
0.0	–	–	–	N/A	1
2.2E-10	Y	N	N	LCHPIVNN	2
4.1E-9	Y	Y	N	LCHPFSNN	3
4.1E-14	Y	Y	N	LCHPPSNN	4
0.0	Y	Y	Y	N/A	5
4.1E-11	Y	Y	Y	LCHPPFPH	6
4.1E-12	Y	Y	Y	LCHPOOEH	7
4.4E-11		–	–	CET IIIA.1	8

89-326-37

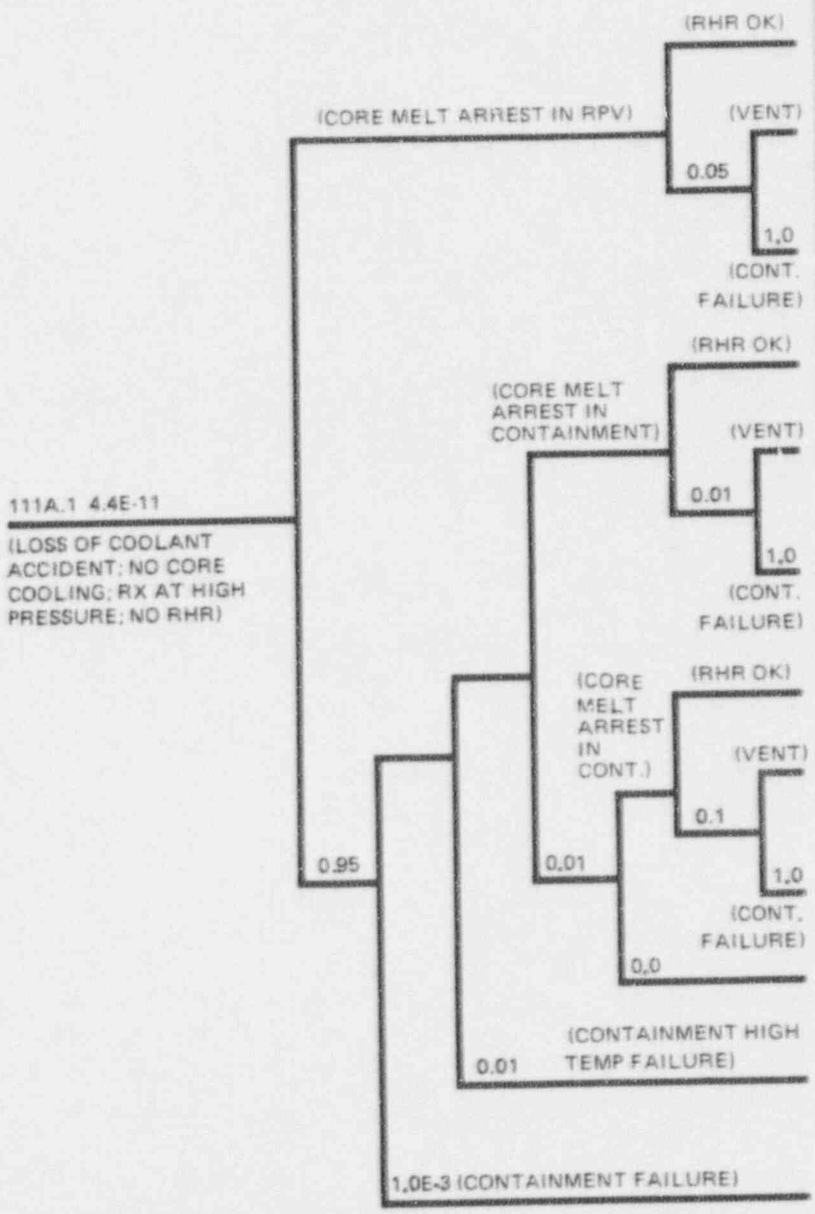
Figure 19D.5–13 LOSS OF COOLANT ACCIDENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT HIGH PRESSURE (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	RX DEPRESSURIZATION BY OPERATOR IS MODELED IN FAULT TREES. THIS BRANCH IS NOT DEVELOPED IN THIS CET.
2	CORE MELT ARRESTED IN RPV WHEN CORE COOLING IS RECOVERED IN ~1 HOUR. RHR IS AVAILABLE, THEREFORE CONTAINMENT STAYS INTACT.
3	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, CORE COOLING RECOVERED PRIOR TO REACHING VENT PRESSURE, CORE MELT ARRESTED, RHR IS AVAILABLE, THEREFORE CONTAINMENT STAYS INTACT.
4	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR AVAILABLE, THEREFORE CONTAINMENT STAYS INTACT.
5	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
6	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT STRUCTURE FAILURE RESULTS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
7	CORE MELT, RPV FAILURE, CONTAINMENT STRUCTURE FAILS AT THE TIME OF RPV FAILURE RESULTING IN EARLY AND HIGH RELEASE.
8	CONTINUED IN CET IIIA.1 (FIGURE 19D.5-14).

89-326-38

Figure 19D.5–13 LOSS OF COOLANT ACCIDENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT HIGH PRESSURE (SHEET 2 OF 2)

FUNCTION/ CONTAINMENT FAILURE MODE	ACCIDENT CLASS	OPERATOR DEPRESSURIZES REACTOR	CONTAINMENT HEAT REMOVAL AVAILABLE	CORE MELT ARRESTED IN RPV	CONTAINMENT INTACT AT RPV FAILURE	NO HIGH TEMPERATURE FAILURE	CORE MELT ARRESTED PRIOR TO CONT. STRUCTURE FAILURE	PASSIVE MITIGATION	RHR RECOVERED PRIOR TO CONT. STRUCTURE FAILURE	VENT OPENED
	IIIA 1	OP	CHR	ARV	CI	HTF	ARC	P	RCH	VT



FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
2.1E-12	Y	N	N	LCHPIVNN	1
0.0	Y	N	N	LCHPIVVN	2
1.1E-13	Y	Y	Y	LCHPIVDN	3
4.1E-11	Y	Y	N	LCHPHRNN	4
0.0	Y	Y	N	LCHPFSVN	5
4.1E-13	Y	Y	Y	LCHPFSDL	6
3.7E-13	Y	Y	N	LCHPHRNN	7
0.0	Y	Y	N	LCHPPSVN	8
4.1E-14	Y	Y	Y	LCHPPSDN	9
0.0	Y	Y	Y	N/A	10
4.2E-13	Y	Y	Y	LCHPPFPH	11
4.2E-14	Y	Y	Y	LCHPODEH	12

89-326-39

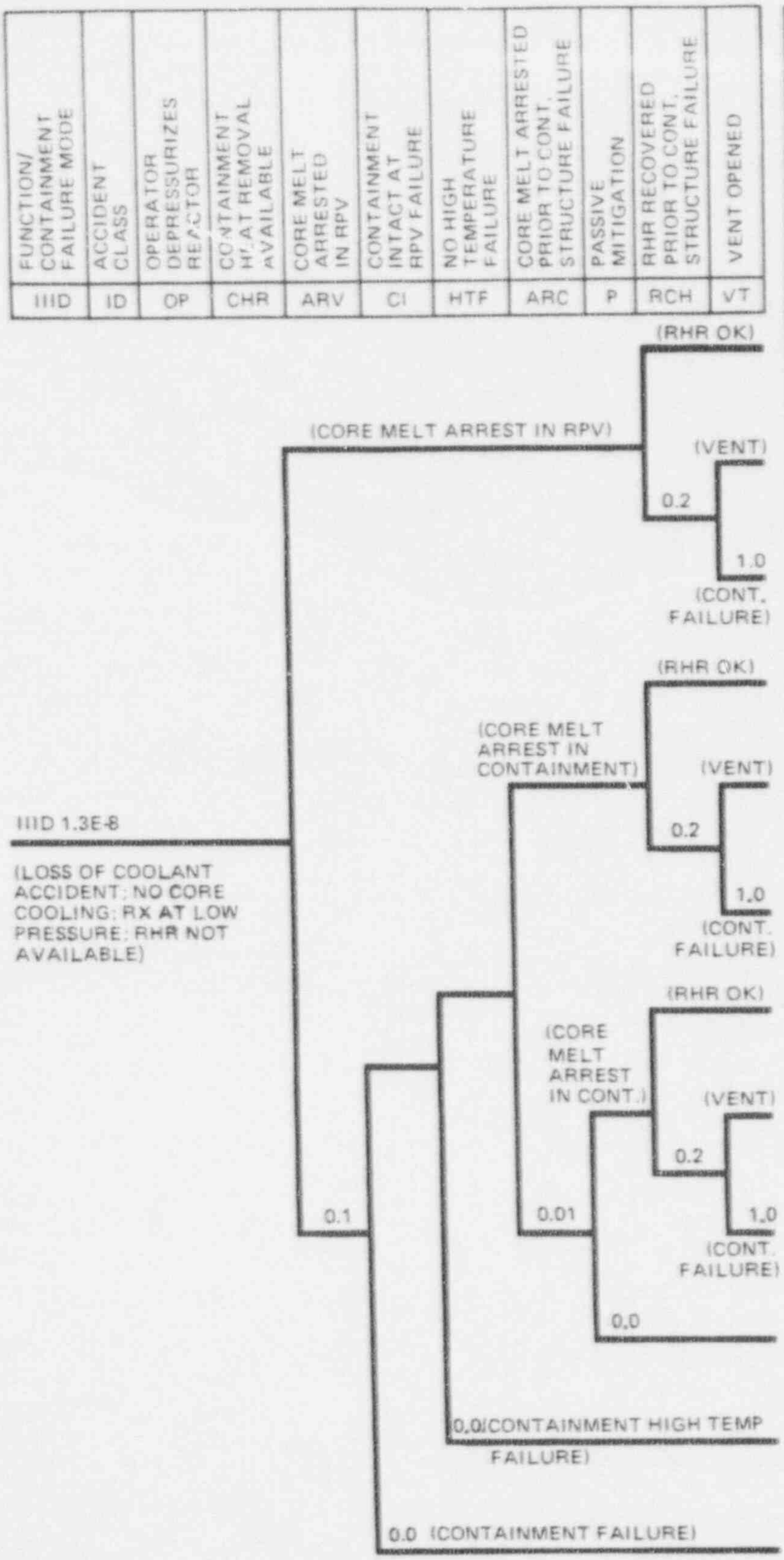
Figure 19D.5–14 LOSS OF COOLANT ACCIDENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT HIGH PRESSURE, CONTAINMENT HEAT REMOVAL SYSTEMS UNAVAILABLE (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	LOSS OF HIGH PRESSURE CORE COOLING, RX NOT DEPRESSURIZED, RHR NOT AVAILABLE, CORE MELT ARRESTED IN RPV WHEN CORE COOLING IS RECOVERED IN ~1 HOUR, RHR RECOVERED PRIOR TO CONTAINMENT STRUCTURE FAILURE, CONTAINMENT INTACT.
2	CORE MELT SIMILAR TO SEQUENCE 1 EXCEPT RHR NOT RECOVERED BUT CONTAINMENT INTEGRITY MAINTAINED BY OPENING VENT.
3	CORE MELT SIMILAR TO SEQUENCE 1 AND 2 BUT NEITHER RHR IS RECOVERED, NOR IS VENT OPENED. CONTAINMENT STRUCTURE FAILS, CORE COOLING FAILS LONG AFTER CONTAINMENT FAILURE. MOST OF THE RELEASE OCCURS THROUGH THE SUPPRESSION POOL, THEREFORE LATE-LOW RELEASE EXPECTED.*
4	LOSS OF HIGH PRESSURE CORE COOLING, RX NOT DEPRESSURIZED, RHR NOT AVAILABLE, RPV FAILS, CORIUM ON DRYWELL FLOOR, CORE MELT ARRESTED BEFORE CONTAINMENT STRUCTURE FAILS, RHR RECOVERED AND CONTAINMENT INTEGRITY IS MAINTAINED.
5	CORE MELT SIMILAR TO SEQUENCE 4, RHR NOT RECOVERED BUT VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
6	CORE MELT SIMILAR TO SEQUENCE 4, NO RHR, NO VENT, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
7	LOSS OF HIGH PRESSURE CORE COOLING, RX NOT DEPRESSURIZED, RHR NOT AVAILABLE, RPV FAILS, CORIUM ON DRYWELL FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR RECOVERED AND CONTAINMENT INTEGRITY IS MAINTAINED.
8	CORE MELT SEQUENCE SIMILAR TO SEQUENCE 7, RHR NOT RECOVERED BUT VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
9	CORE MELT SIMILAR TO SEQUENCE 7, NO RHR, NO VENT, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
10	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
11	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING FAILURE OF DRYWELL SPRAY FUNCTION.
12	CORE MELT, RPV FAILURE, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS, RESULTING IN EARLY AND HIGH RELEASE.

*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0.

89-326-40

Figure 19D.5–14 LOSS OF COOLANT ACCIDENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT HIGH PRESSURE, CONTAINMENT HEAT REMOVAL SYSTEMS UNAVAILABLE (SHEET 2 OF 2)



FREQUENCY	CORE DAMAGE ?	RPV FAILED ?	CONTAINMENT FAILED ?	CETBIN/ TRANSFER	CET SEQUENCE NUMBER
9.4E-9	Y	N	N	LBLCIVNN	1
0.0	Y	N	N	LBLCIVVN	2
2.3E-9	Y	Y	Y	LBLCIVDN	3
1-0E-9	Y	Y	N	LBLCHRNN	4
0.0	Y	Y	N	LBLCFSVN	5
2.6E-10	Y	Y	Y	LBLCFSDL	6
1.0E-11	Y	Y	N	LBLCHRNN	7
0.0	Y	Y	N	LBLCPFVN	8
2.6E-12	Y	Y	Y	LBLCPFDM	9
0.0	Y	Y	Y	N/A	10
0.0	Y	Y	Y	LBLCOOPH	11
0.0	Y	Y	Y	LBLCOOEH	12

89-326-41

Figure 19D.5–15 LOSS OF COOLANT ACCIDENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT LOW PRESSURE (SHEET 1 OF 2)

CET SEQUENCE NUMBER	REMARKS
1	LOSS OF LOW PRESSURE CORE COOLING, RX DEPRESSURIZED, RHR NOT AVAILABLE, CORE MELT ARRESTED IN RPV WHEN CORE COOLING IS RECOVERED IN ~ 1 HOUR, RHR RECOVERED BEFORE VENT PRESSURE IS REACHED, CONTAINMENT INTACT.
2	CORE MELT SIMILAR TO SEQUENCE 1 EXCEPT RHR NOT RECOVERED BUT CONTAINMENT INTEGRITY MAINTAINED BY OPENING VENT.*
3	CORE MELT SIMILAR TO SEQUENCE 1 AND 2 BUT NEITHER RHR IS RECOVERED, NOR VENT IS OPENED, CONTAINMENT STRUCTURE FAILS, CORE COOLING FAILS LONG AFTER CONTAINMENT FAILURE. MOST OF THE RELEASE OCCURS THROUGH THE SUPPRESSION POOL, THEREFORE LATE-LOW RELEASES EXPECTED.*
4	LOSS OF LOW PRESSURE CORE COOLING, RX DEPRESSURIZED, RHR NOT AVAILABLE, RPV FAILS, CORIUM ON DRYWELL FLOOR, CORE MELT ARRESTED PRIOR TO CONTAINMENT STRUCTURE FAILURE, RHR RECOVERED AND CONTAINMENT INTEGRITY IS MAINTAINED.
5	CORE MELT SIMILAR TO SEQUENCE 4, RHR NOT RECOVERED BUT VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
6	CORE MELT SIMILAR TO SEQUENCE 4, NO RHR, NO VENT, CONTAINMENT STRUCTURE FAILS. CONSERVATIVELY NEGLECT EFFECT OF CORE MELT ARREST.*
7	LOSS OF LOW PRESSURE CORE COOLING, RX DEPRESSURIZED, RHR NOT AVAILABLE, RPV FAILS, CORIUM ON DRYWELL FLOOR, PASSIVE FLOODER INITIATES SUBMERGING CORIUM UNDER WATER FROM SUPPRESSION POOL, RHR RECOVERED AND CONTAINMENT INTEGRITY IS MAINTAINED.
8	CORE MELT SEQUENCE SIMILAR TO SEQUENCE 7, RHR NOT RECOVERED BUT VENT IS OPENED TO MAINTAIN CONTAINMENT INTEGRITY.*
9	CORE MELT SIMILAR TO SEQUENCE 7, NO RHR, NO VENT, CONTAINMENT STRUCTURE FAILS, CONSERVATIVELY NEGLECT EFFECTS OF CORE MELT ARREST.*
10	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, NO WATER TO CORIUM, CONTAINMENT STRUCTURE FAILS.
11	CORE MELT, RPV FAILURE, CORIUM ON FLOOR, HIGH TEMPERATURE CONTAINMENT FAILURE RESULTS FOLLOWING THE FAILURE OF DRYWELL SPRAY FUNCTION.
12	CORE MELT, RPV FAILURE, CONTAINMENT STRUCTURE FAILS WHEN RPV FAILS RESULTING IN EARLY AND HIGH RELEASE.

*NOTE: VENT HAS BEEN ASSIGNED A FAILURE PROBABILITY OF 1.0.

89-326-42

Figure 19D.5–15 LOSS OF COOLANT ACCIDENT FOLLOWED BY LOSS OF CORE COOLING, REACTOR AT LOW PRESSURE (SHEET 2 OF 2)

SECTION 19D.8

CONTENTS

TABLES

19D.8 DEPENDENT FAILURE TREATMENT

19D.8.1 Summary

Dependent failures have been included as integral parts of the overall PRA system and functional analyses. Dependencies of frontline systems on support systems and interdependencies between frontline or support systems have been modeled explicitly in the fault tree and event tree analyses. Common-cause failures (CCFs) have also been explicitly included from the standpoint of multiple component failures within systems, and as a result of human error.

[illegible]

19D.8.2 General Considerations

Dependent failures, including those frequently categorized as common-mode or common-cause, must be realistically and adequately addressed in any comprehensive evaluation of risk. The term dependent failure refers to two or more elements failing as a result of the same cause or failure mode. A number of considerations are important in the analysis of common-cause failures, including the following:

(1) Common-cause effects generally are of greatest significance in redundant systems. The analysis was structured to insure that no important common-cause effects were overlooked, particularly for backup and support systems.

(2) Common-cause effects are limited by design, manufacture, and procedural diversity. Such factors were taken into consideration in the analysis.

(3) Isolation barriers and physical separation of redundant systems and components can reduce or eliminate common causes of failure, and these factors were considered and taken into account in the analysis as appropriate.

19D.8.3 Multiple Equipment Failures from a Common Cause

Causes of multiple equipment failures include common manufacturing errors, design errors, and extreme environmental conditions. Failures of this type generally have a low frequency of occurrence and would not significantly influence system failure probabilities, except in those cases where design redundancy is incorporated to achieve very low overall system or function failure probabilities. Consequently, systems for which common-cause effects of the above type are potentially most significant—and, therefore, addressed explicitly—included the following:

(1) emergency power supply system,

(2) automatic depressurization system,

(3) control rod drive system,

(4) reactor protection system, and

(5) instrumentation and control.

Design-related CCFs from the above systems are included in the analysis for the following special components:

- ESF logic,
- transmission network (MUX),
- sensor & transmitter miscalibration,
- output logic units,
- digital trip units,
- trip logic units,
- main scram load drivers,
- backup scram relays,
- pressure sensors,
- APRMs,
- diesel generators,
- batteries,
- offsite power sources,
- safety relief valves.

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.

19D.8.4 Multiple Failures Due to Human Error

Multiple failures can also occur due to human error, if an operator or maintenance technician repeatedly makes the same mistake. Multiple instrument miscalibration is judged the most likely event of this type and is explicitly included in the analysis. Human error as a common-cause failure mechanism is discussed further in Section 19D.7.

19D.8.5 Functional Interdependencies

Interdependencies within and between systems are treated rigorously in the fault tree and event tree analyses. Examples of interdependencies that are accounted for are electric power, service water, room cooling, and control instrumentation.

Interdependency factors that could affect automatic or manual initiation of systems were evaluated. In some cases, the same sensors, signal transmitters, reference fluid pressure columns or logic units are used in the initiation of more than one system. This type of dependency between systems was represented in detail to assess the probability of safety system operability. In the ABWR PRA, many sensors provide input to more than one system. Each of these sensors and their logic are designated by unique acronyms throughout the analysis to insure that dependencies among systems are rigorously accounted for in the functional fault tree evaluations.

19D.8.6 Generic Component CCFs

In response to a request from the Nuclear Regulatory Commission, additional common-cause failures were added to the basic analysis and the effects on system availability and core damage frequency were evaluated. The basic PRA analysis, as presented elsewhere in this document, was not modified. Component CCFs for the following systems were identified, evaluated and included in a CCF requantification (in addition to the CCFs included in the basic analysis):

HPCF	2/2 trains	14 components
RHR Core Flooding Mode	3/3 trains	24 components
RHR SP Cooling Mode	3/3 trains	25 components
RBCW	internal to each division	4 components
RBCW	between Divisions A & B	6 components
RBCW	between Divisions A & C	6 components
RBCW	between Divisions B & C	6 components
RBCW	between Divisions A, B & C	6 components

These are the systems where generic component common-cause failures might have a significant effect. Generic component CCFs that are included in the system analysis are for pumps, pump auxiliary equipment, manual valves, motor-operated valves, check valves, room air conditioners, spargers, strainers, circuit breakers, flow transmitters, heat exchangers, and temperature elements.

For the RBCW CCFs internal within each division of RBCW, the component CCFs were included at appropriate places within the fault tree structures. For all other cases (interdivisional or between trains), the individual component CCFs were summed and added-in at the top as a CCF module. The RBCW interdivisional CCFs were added-in at the top of the fault trees for the safety systems that use RBCW.

Component CCFs were identified wherever redundancy occurs in the fault trees of the above systems (generally, for every "and" gate). The component CCFs were quantified using the "multiple Greek letter" method and using the CCF factors given in the EPRI ALWR Utility Requirements Document (Reference 1). Where common-cause factors were not given for specific component types, the recommended "generic" factors were used. For those cases, the results should be considered as "bounding" and are probably conservative.

The numerical results of the component CCF analysis in terms of system CCF probabilities are given below:

HPCF CCFs	2/2 loops	2.46E-3
RHR Core Flooding CCFs	3/3 loops	1.00E-3
RHR SP Cooling CCFs	3/3 loops	9.78E-4
RBCW CCFs within a division	1/2 pumps	1.52E-5
RBCW CCFs between Div. A & B	2/2	6.48E-6
RBCW CCFs between Div. A & C	2/2	6.48E-6
RBCW CCFs between Div. B & C	2/2	6.48E-6
RBCW CCFs between Div. A, B and C	3/3	5.93E-6

The numerical results of this analysis also can be viewed from two additional perspectives: the effect on system unavailability (Table 19D.8-1), and the effect on core damage frequency (Table 19D.8-2).

The effects of the added generic component CCFs on system unavailability are significant. The most significant effect is on the core flooding mode of RHR, where the system unavailability with component CCFs is over 11 times the system unavailability without component CCFs. The largest contributors to RHR CCF are common-cause failure of the RHR pumps to start, and common-cause failure of the injection valves to open is al a significant contributor to RHR CCF.

For the HPCF system, the most significant CCF contributors are common-cause failure of the pumps to

start, and mispositioning (closed) of manual valve F005. The reason for the large CCF of the manual valve is because of a very high assigned random failure probability (1.0E-2) as taken from WASH 1400 (Reference 2). This is a very conservative value.

The individual divisions of the RBCW system were not significantly affected by component CCFs. However, the interdivisional CCFs have a measurable effect on core damage frequency, as discussed in Table 19.8-2.

The most significant effect on CDF is due to the CCFs between all three divisions of RBCW. This is primarily due to the failure of both HPCF and RHR Core Flooding, given loss of all RBCW divisions. All other CCFs have relatively little effect on CDF.

The common-cause failure of all three divisions of RBCW is balanced among common-cause failure of heat exchangers and common-cause failure of pumps (failing to run). Plugged strainers and temperature control valves failing closed also contribute to the RBCW interdivisional CCFs.

The total effect on CDF of the addition of these CCFs (~20% increase) is not insignificant, partially due to the low absolute value of the ABWR CDF. These CCFs will be added to the plant model in any future revised basic quantification of the ABWR PRA.

19D.8.7 References

1. *Advanced Light Water Reactor Utility Requirements Document*, Vol II, Chp. 1, App. A, Rev. 5, EPRI, Palo Alto, CA, December 1992.

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

Table 19D.8-1
EFFECT ON SYSTEM UNAVAILABILITY (A)

System	Base A	A with CCFS*	% Increase
HPCF	2.33E-3	4.79E-3	105
RHR (flood)	9.65E-5	1.10E-3	1040
RHR (cool)	2.72E-4	1.25E-3	360
RBCW Div. A	3.09E-4	3.24E-4	4.85
RBCW Div. B	3.09E-4	3.24E-4	4.85
RBCW Div. C	3.09E-4	3.24E-4	4.85

* CCFs within that system.

Table 19D.8-2
EFFECT ON CORE DAMAGE FREQUENCY

System	CDF Increase/yr	% Increase
HPCF	5.6E-9	3.6
RHR (flood)	4.9E-9	3.1
RHR (cool)	5.4E-10	0.3
RBCW A, B & C	1.90E-8	12.2
TOTAL	3.00E-8	19.2

SECTION 19D.9

CONTENTS

TABLES

19D.9 CDF SENSITIVITY TO OUTAGE TIMES AND SURVEILANCE INTERVALS

19D.9.1 Summary

As a consequence of 1992 GE-NRC discussions of ABWR DSER questions regarding applicability of GESSAR test and maintenance (T&M) unavailabilities to the ABWR PRA, it was agreed that ABWR T&M unavailabilities 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 two percent in the PRA model, and the calculated core damage frequency of 1.56E-07 reflects inclusion of these values.

19D.9.2 Sensitivity to Test and Maintenance Outage Times

CDF sensitivity to T&M outage times was assessed by varying system values individually as well as in combination. Results presented in attached Tables 19D.9-1 and 19D.9-2 illustrate the impact of increasing system T&M unavailabilities by a factor of five from two to ten percent. Ten percent was judged to be a reasonable upper bound for T&M unavailability for a single system. As can be seen, calculated CDF is most sensitive to the RCIC system T&M unavailability. This is due in large part to the fact that station blackout sequences dominate CDF, and in these sequences RCIC is essential for successful core cooling. Since no credit was taken in the Level 1 PRA for fire water injection, this calculated CDF sensitivity to RCIC T&M unavailability is actually somewhat conservative. In addition, ample time is available for maintenance of RCIC during refueling outages without CDF risk implications, since during shutdown the system is unable to perform its ECCS function.

Second in importance is the T&M unavailability of HPCFB. HPCFB includes a hard-wired manual initiation backup in the control room. This provides a diverse means of manually initiating HPCFB in the event of essential multiplexing system common mode failure, a feature which increases the importance of HPCFB relative to other ECCS systems.

CDF is very insensitive to the T&M unavailability of systems other than RCIC and HPCFB, either individually or in combination. Table 19D.9-3 provides a summary of bounding scenarios in which individual systems are removed completely from service. These results support the conclusions drawn from Tables 19D.9-1 and 19D.9-2. Further, even with RCIC the most sensitive system completely out of service, the core damage frequency goal of 1.0E-05 is still satisfied.

19D.9.3 Sensitivity to Surveillance Intervals

Since no changes were made to established BWR surveillance intervals (GESSAR values were applied in the ABWR PRA), sensitivity to changes in surveillance intervals was not investigated.

Table 19D.9-1

CDF SENSITIVITY TO T&M OUTAGE UNAVAILABILITIES

System	Single System Perturbations To Base Case T&M Value Of Two Percent						
RCIC	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2
HPCFB	2.00E-2	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2	2.00E-2
HPCFC	2.00E-2	2.00E-2	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2
RHRA	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1	2.00E-2	2.00E-2
RHRB	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1	2.00E-2
RHRC	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1
CDF	**1.56E-7**	**2.93E-7**	**1.77E-7**	**1.57E-7**	**1.57E-7**	**1.56E-7**	**1.56E-7**
%INCR	-	86	13	<1	<1	<1	<1

Table 19D.9-2

CDF SENSITIVITY TO T&M OUTAGE UNAVAILABILITIES

System	Multiple System Perturbations To Base Case T&M Value Of Two Percent						
RCIC	1.00E-1	2.00E-2	2.00E-2	1.00E-1	2.00E-2	1.00E-1	1.00E-1
HPCFB	2.00E-2	1.00E-1	2.00E-2	1.00E-1	1.00E-1	1.00E-1	1.00E-1
HPCFC	2.00E-2	2.00E-2	1.00E-1	2.00E-2	1.00E-1	1.00E-1	1.00E-1
RHRA	1.00E-1	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1
RHRB	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1
RHRC	2.00E-2	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2	1.00E-1
CDF	**2.92E-7**	**1.78E-7**	**1.57E-7**	**3.14E-7**	**1.77E-7**	**3.15E-7**	**3.18E-7**
& INCR	88	14	<1	101	13	102	104

Table 19D.9-3

CDF SENSITIVITY TO T&M OUTAGE UNAVAILABILITIES

System	Impact Of Single Systems Completely Removed From Service						
RCIC	2.00E-2	1.00	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2
HPCFB	2.00E-2	2.00E-2	1.00	2.00E-2	2.00E-2	2.00E-2	2.00E-2
HPCFC	2.00E-2	2.00E-2	2.00E-2	1.00	2.00E-2	2.00E-2	2.00E-2
RHRA	2.00E-2	2.000E-2	2.00E-2	2.00E-2	1.00	2.00E-2	2.00E-2
RHRB	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00	2.00E-2
RHRC	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00
CDF	**1.56E-7**	**1.83E-6**	**4.08E-7**	**1.62E-7**	**1.64E-7**	**1.59E-7**	**1.57E-7**
% INCR	-	1073	162	4	5	2	<1

SECTION 19E.2

CONTENTS

SECTION 19E.2 (Continued)

CONTENTS

SECTION 19E.2 (Continued)

CONTENTS

SECTION 19E.2 (Continued)

CONTENTS

SECTION 19E.2(Continued)

CONTENTS

SECTION 19E.2(Continued)

CONTENTS

SECTION 19E.2(Continued)

CONTENTS

SECTION 19E.2 (Continued)

CONTENTS

SECTION 19E.2(Continued)

TABLES

SECTION 19E.2(Continued)

TABLES (Continued)

SECTION 19E-2 (Continued

ILLUSTRATIONS

SECTION 19E-2 (Continued

ILLUSTRATIONS

SECTION 19E-2 (Continued)

ILLUSTRATIONS (Continued)

SECTION 19E-2 (Continued)

ILLUSTRATIONS (Continued)

pressure core melt.

19E.2.1.2.2.5 Recovery Following Restoration of AC Power

All equipment necessary for restoration of power is located external to the reactor building secondary containment. With the exception of the control building and the RCIC room, 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 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 operation of the reactor building closed 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. 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 Station Blackout would be successful even if the design basis values were exceeded, as long as the ultimate capability were not exceeded.

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 less than 10% with the exception of the wetwell/drywell vacuum breakers. This is judged to be within the uncertainty of the PRA. Therefore, only the failure of the 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 Radiation Heating of the Equipment Tunnel

A potential concern for the ABWR during severe accidents is radiation 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 reaches 533 K (500°F). Upon opening, water from the suppression pool would flood the lower drywell, covering the corium. This stops any radiation 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 reaches 8 inches, water from the suppression pool would flood the lower drywell.

The pedestal cavity design meets the 0.2 sq. meters/MWt specification of the EPRI Debris Coolability Requirements for Advanced Light Water reactors (Reference 2). A conservative analysis was performed following the methods of the ARSAP Debris Coolability Requirement (Reference 3) and utilizing the concrete ablation rate from CORCON (Reference 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. This 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 Section 19D.5.

19E.2.1.3.8 Mode of Vessel

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 pictorial 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 than 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 retraints 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.2.1.4 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 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.2.4.1. 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.5.12. 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 structural failure. If structural integrity is maintained, 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 the seismic event trees, 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 are 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 will be investigated in detail. Each parameter will be considered individually, although interactions between some key phenomena are 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 will not be carried out as done in NUREG-1150.

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, GE has surveyed the available literature as discussed in subsection 19E.2.5. Some of the severe accident issues are 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 are included in the sensitivity studies which follow.

19E.2.1.5.2 Sensitivity Studies

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.

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 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. 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 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.2.1.6.1.

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 0.3% 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.2.1.6.7, 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.2.6.2.

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. Section 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.2.1.6.3.

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 Section 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. A sensitivity studies were performed 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. 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 conditional probability of 0.04. 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.2.1.5.3 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 how Direct Containment Heating (DCH), pool bypass, and Core Concrete Interaction (CCI) each impact the 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 phenomena:
- 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 0.1% for 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

The conclusion from all of these uncertainty calculations were:

1. For the dominant core melt sequences that release core material into the containment, 90% 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 1.5% of the sequences with significant CCI will suffer pedestal failure. Combining this conclusion with the first, only 0.15% 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 approximately 0.4%.

2. There is a 2% probability that there is a small leakage path between the drywell and wetwell. Based on the Morowitz plugging model, 90% 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 with no drywell spray available is 7.4E-11; 0.05% 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. Classes of accidents with frequencies greater than 10^{-8} were considered in selecting the accident sequences to be studied.

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 less than 10^{-8} and if it was significantly smaller than the overall frequency of the class then the next most severe case was examined. Note that the sequences with frequencies of less than 10^{-8} 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-1.

For each base 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 533K (500°F). This serves to keep the corium temperature low, preventing core-concrete interaction, and prevents radiation 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 in 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, 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 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 radiation 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 SSAR 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 of the SSAR 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 check valves (F100 and F104) to insure that RCS pressurization does not result in a breach of the injection path. The required flow rate for the firewater addition system in specified in Section 2.15.6 of the ITAAC.

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, and it provides cooling of the upper drywell region. The operator is instructed to operate the spray system if the vessel failure has occurred, as determined by the temperature of the drywell and the inability to maintain water level in the vessel. 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 level of the bottom of the vessel the operator is instructed to turn the firewater system off, turning it on again only as necessary to prevent the upper drywell temperature from exceeding 533K (500°F). If drywell head failure occurs the firewater spray system is to be restarted. This causes any fission product aerosols to agglomerate on the spray droplets, reducing the fission product release to the environment.

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 533K (500°F) and the pressure exceeds 0.515 MPa (60 psig). 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 Subection 6.2.5 and in 19E.2.8.1.1.

D: The drywell head is assumed to fail before the rupture disk opens. The mean failure pressure of the drywell head is 1.025 MPa (134 psig) if the temperature in the upper drywell is below 533K (500°F). However, as discussed in Attachment A to Appendix 19F, there is a small probability the drywell fails at lower pressure. At higher temperatures, the drywell head is assumed

to fail at lower pressures 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 earthquake. 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. This eighth character indicates the magnitude of the release predicted by MAAP. 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 2 hours after initiation of the transient, the lower vessel head fails.

The corium falls into the lower drywell along with the water that had been retained 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.

(a) 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 533K (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, quenches it, and 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 1600K (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, thus, the temperature of the lower drywell gas decreases (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 (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 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 over-predicts its severity. MAAP 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, 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 (90 psig) 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 533K (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 approximately a 2% 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 0.72MPa (90 psig) 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.

(b) 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 injection lines. When trying to prevent vessel failure the operator is instructed to inject water to the vessel via the LPFL line. If this is not accomplished in time to prevent vessel failure, 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.2-3A) 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.2-3B), 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 suppression pool 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.

Side 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

19E.2 Deterministic Analyses 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 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. MAAP 3.0B 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 MAAP3.0B-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 MAAP3.0B-ABWR. Specific ABWR features which required code changes are listed below.

(1) Containment Configuration

The ABWR configuration is different than previous BWR configurations. MAAP3.0B-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 heat exchangers in all 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

MAAP3.0B-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, the MAAP3.0B-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. MAAP3.0B-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 pedestal penetration occurs 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 while the corium was attacking the concrete and pressurizing. 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 MAAP3.0B-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 outside the scope of this submittal: for example, the type of concrete to be used in the plant is not included in this submittal.

(1) Condensate Storage Tank. The configuration for the condensate storage tank is assumed to be consistent with the description in 19.9.9. This is sufficient to satisfy the station blackout performance requirements discussed in Subsection 19E.2.1.2.2.

(2) Deleted

(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 negligible impact on containment performance. Limestone Sand concrete is representative of the concrete which might be used in much of the United States.

(4) Deleted

(5) Battery loading profiles will be developed to define appropriate load shedding during Station Blackout (see Subsection 19E.2.1.2.2.2(3)). This item has been identified as a COL Action Item in Section 19.9.9.

(6) RCIC room temperature will not exceed equipment design temperature without room cooling for at least 8 hours (See Subsection 19E.2.1.2.2.2(5)). This item has been identified as a COL Action Item in Section 19.9.9.

(7) Control room temperature will not exceed equipment design temperature for at least 8 hours without room cooling (See Subsection 19E.2.1.2.2.2(6)). This item has been identified as a COL Action Item in Section 19.9.9.

(8) Operator action during station blackout is consistent with the EPGs as specified in Subsection 19E.2.1.2.2.4.

(9) Deleted

(10) Deleted

(11) Deleted

(12) Deleted

19E.2.1.2.2 Station Blackout Performance

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 analyses summarized in this subsection show that the ABWR can withstand a station blackout without core damage or loss of containment integrity for a period of at least 8 hours. If AC power is still unavailable beyond this period, the core cooling function is assumed to be lost. 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. The following areas are considered to assure RCIC functionality during station blackout:

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 (50 psig). However, since the operator will be maintaining vessel pressure near 945 psig 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 no practical limit on the availability of DC power for operating SRV solenoids.

The control power for six of the 18 SRVs is taken from the Division 1 battery. The valves have been considered as part of the load on the Division 1 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 1 battery is sufficient for 8 hours of coping during station blackout.

Of the remaining 12 SRVs, 6 have their control power supply on the Divisions 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, 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 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 During Station Blackout.

The SRV actuators can open the SRVs with a pressure differential of approximately 70 psi (nitrogen supply pressure above containment pressure) without assistance from internal steam pressure. The SRV accumulators used for the ADS function (see Figure 19E.2-12), which are charged to a pressure of 170 psig (12 kg/cm^2g), have insufficient pressure and capacity to fully open the SRVs at 100 psig pressure in the containment and RPV, so additional gas at 170 psig is needed from outside the containment to ensure the pressure control and depressurization function.

The normal supply of N_2 gas to the SRVs from the atmospheric control system outside the containment is 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 2130 to 850 psig (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 must 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 170 psig 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 were above 100 psig or if somewhat less than 170 psig were supplied), the operator could adjust the set point of the pressure regulating valve above 170 psig at the local station (the relief valves are set at 210 psig).

The capacity of a group of ten 45 liter high

pressure N_2 gas bottles at 850 psig 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 100 psig. 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 DC batteries will be sized to be capable of operating the RCIC system for a minimum of 8 hours assuming the expected loading profiles for station blackout. These loading profiles, including load shedding, will be defined in detail as the ABWR design progresses.

4) Water Source Inventory.

The primary water source for the RCIC System during station blackout is the condensate storage tank (CST) which has been sized to provide sufficient inventory for a minimum of 8 hours during this scenario. In the event the CST became depleted, the backup source is the suppression pool. The RCIC system must be manually overridden to assure that the condensate storage tank will be maintained as the primary water source for core cooling and makeup.

5) RCIC Room Temperature.

Failure of the AC cooling power supplies 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 151°F (66°C), starting at the normal room temperature 0f 104°F (20°C.), for at least 8 hours.

6) Control Room Temperatures.

The safety related equipment required to function during station blackout and located in the main, lower and computer control rooms will be designed for a maximum operating temperature of 122°F (50°C). The ABWR plant will be designed to prevent the room temperature from reaching this equipment design temperature for at least 8 hours, starting at the normal room temperature of 79°F (26°C).

19E.2.1.2.2.3. Primary Containment Vessel (PCV) Integrity

Containment pressure and temperature analysis were performed to determine the containment atmospheric conditions after 8 hours of station blackout conditions 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 45 psig and 340°F (drywell)/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 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 action are specified in the EPGs to control the RCIC system (to avoid repeated restarting of the RCIC turbine) 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 (below first SRV 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 at the steam cooling pressure and thereby avoid a high

initiation level. The operator will be instructed to leave SRVs open to keep the vessel at low pressure

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 (90 psig) 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 533K (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.

There is approximately a 5% 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 0.72MPa (90 psig) 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 suppresion pool temperature is below the suppression pool heat capacity temperature limit at the time of vessel failure, SPV 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.

(a) 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.2-4E).

After the drywell sprays are turned on the containment slowly repressurizes (Figure 19E.2-4A). 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 is assumed to open when the wetwell pressure reached 0.72 MPa (90 psig) 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. The volatile fission product release continues for about 25 hours. The release fraction of CsI at 72 hours is less than 1E-7.

There is approximately a 2% 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 0.72MPa (90 psig) 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.

(b) 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 bottom of the vessel, at about 22 hours, the operator is assumed to turn off the firewater system. The corium in the upper drywell then causes the temperature in the upper drywell to increase. When the temperature in the upper drywell again reaches 500 K (440°F) the operator restarts the drywell spray. This causes the pressure and the upper drywell temperature to decrease. After 15 minutes the operator turns the system off in order to minimize excess water addition to the containment. The cycle is repeated many times

Due to MAAP code limitations the firewater spray was switched to drywell spray from the RHR system in the LPCI mode after the water level in the suppression pool reached the bottom of the vessel. The effect of this change 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 0.72MPa (90 psig) at 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 less than a 5% 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 0.72MPa (90 psig) 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.

(c) 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), qeunching 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.2-5B). 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 0.1 MPa (14.7 psid). 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. 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 0.66MPa (81 psig) (Figure 19E.2-5A). The containment does not reach the wetwell rupture disk setpoint pressure of 0.72MPa (90 psig), nor does it reach the pressure necessary to fail the drywell head. The drywell head failure pressure at 830K is reduced to 0.75MPa (94 psig) 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 (SBRC)

This accident initiator, SBRC, represents a station blackout sequence. These are characterized by the unavailability of all AC Power. Therefore, the RCIC system and firewater are the only systems available for core cooling. This sequence assumes RCIC operates for 8 hours, providing core cooling (per Subsection 19E.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 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 8 hours. After the RCIC system is presumed to fail, the water in the vessel continues to boil off to the suppresion pool. The pool begins to overflow to the lower drywell at about 9 hours.

The data in Figures 19E.2-6, 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.2-7 can be substituted for Figures 19E.2-6 for the first 6 hours.

(a) 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 manually depressurize the reactor and align the valves to begin injecting with the firewater addition system.

The depressurization causes the water to flash to steam, lowering the water level in the vessel (Figure 19E.2-6D). MAAP predicts that the core heats up to about 1150 K (1610°F) during this time (Figure 19E.2-6C). Therefore, severe core damage will not

occur. When the pressure reaches the shutoff head of the firewater addition system, 1.96 MPa (270 psig), water injection begins and the core cools rapidly. The water level in the vessel then rises until it reaches level 8 (Figure 19E.2-6C). The operator then maintains water level between level 8 and level 2.

During this time, the containment pressure increases slowly while the RCIC operates (Figure

19E.2-6A). 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 suppresion pool mass.

When the wetwell pressure reaches 0.72 MPa (90 psig) after 32.3 hours, the rupture disk opens. However, because no core damage has occurred there is no release of fission products.

(b) 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). 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 wetwell pressure to increase rapidly to the rupture disk rupture pressure, 0.72 MPa (90 psig) 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 approximately a 2% 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 0.72MPa (90 psig) 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 taker 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 68°C (155°F) 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 (90 psig), (Figure 19E.2-8A) and the rupture disk opens. After 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.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 steam line 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 drywell head failure 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).

(a) 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 0.72MPa (90 psig), 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 approximately a 2% 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 0.72MPa (90 psig) 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.

(b) 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(b), 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 0.72MPa (90 psig) 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 approximately a 5% 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 0.72MPa (90 psig) 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).

(a) 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 approximately a 2% 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 3.5E-2,

which is much greater than the release for the corresponding rupture disk case.

(b) Firewater Spray

If the operator begins injection using the firewater additions 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 th esuppression 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 approximately a 5% 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 0.72MPa (90 psig) 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 a LCHP event, corium is entrained into the wetwell and upper drywell (Figure 19E.2-11C).

(a) 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 threshhold (Figure 19E.2-11B). However, since the pressure is only 0.25 MPa (22 psig), leakage does not occur at this time but is delayed until 17.8 hours when the drywell pressure reaches 0.46 MPa (52 psig) as shown in Figure 19E.2-11A.

At 47.7 hours, the pressure in the drywell dips sharply by about .1 MPa (14.7 psid). 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 850K (1070°F). At the same time the drywell pressure has reached a steady value of 0.67MPa (82.5 psig) (Figure 19E.2-11A). The containment does not reach the rupture disk setpoint pressure of 0.72MPa (90 psig), nor does it reach the pressure necessary to fail the drywell head. The drywell head failure pressure at 850K is reduced to 0.71MPa (88 psig) 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.

(b) 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(b). 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 (3 psi) 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(b). 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 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 19E2-12F.

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).

(a) 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 0.72MPa (90 psig) 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 approximately a 2% 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 0.72MPa (90 psig) 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.

(b) 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 0.72MPa (90 psig) 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 less than 1E-7.

There is approximately a 5% 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 0.72MPa (90 psig) at 26.4 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 85 hours. The CsI release fraction at 72 hours is 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 drywell head failure are given. Also shown are the duration of the release and the release fraction of 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 are 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 5, 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 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 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 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 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 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.

(a) 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 9) in the form

$$\frac{8\pi^3\sigma}{\lambda^3} + \frac{2\pi g(\rho_h-\rho_L)}{\lambda} - \frac{4\pi^2\rho_h\rho_L(v_h-v_L)^2}{\lambda^2(\rho_h+\rho_L)} < 0 \qquad (1)$$

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 $\lambda/2$.

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

(d) 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,

$$E = (4/3)\rho\pi c_v r^3 (T - T_\infty) \quad (11)$$

is diminished at a rate q, for which

$$T - T_\infty = (T_i - T_\infty) \exp(-t/\gamma_h)$$

where the time constant π_h gives the convective time response as

$$\gamma_h = \rho c_v r / 3h \quad (12)$$

The internal conduction of heat occurs with an approximate time constant (Reference 10),

$$\gamma_c = 2r^2/\alpha \quad (13)$$

Either γ_c or γ_h may control the heat transfer rate to surrounding water. Figure 19E.2-15 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 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 12). The convective response time is seen to be proportional to the droplet radius in Figure 19E.2-15. 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.

(e) 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 9) for a spherical bubble,

$$RR'' + (3/2)(R')^2 = (P_b - P_\infty)/\rho_l \quad (14)$$

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

$$h_{fg} m'_g = hA_p (T_i - T_\infty) \exp(-t/\gamma_h) \quad (15)$$

where γ_h is given by equation (12).

An energy balance for the bubble growth may also be written:

$$P_b V' - h_g m'_g + U' = 0 \quad (16)$$

Assuming an ideal gas

$$U = P_b V_b/(k-1) \quad (17)$$

If the bubble is further assume to be spherical, one may combine equations (16) and (17) to yield

$$12\pi k P_b R^2 R' + 4\pi P'_b R^3 = 3m'_g h_g (k-1) \quad (18)$$

Combining this equation with the mass rate equation (15) and the Rayleigh bubble equation (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

$$\gamma_L = R/R' \quad (19)$$

was obtained which is plotted in Figure 19E.2-15.

Figure 19E.2-15 shows that γ_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.2-15 is primarily determined by the droplet surface heat transfer rate.

(f) Conditions for Self Triggering

Self-triggering could occur if the mechanical energy DW 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

$$\Delta W_{\sigma} = 4\pi\sigma r^2 \tag{20}$$

where σ is 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

$$\Delta W = \Delta W_{\sigma} \tag{21}$$

An estimate of the expansion work done by a steam bubble which expands to volume V_{∞} is given by

$$\Delta W = (P - P_{\infty})V_{\infty} \tag{22}$$

here P is an average pressure during expansion. The term $(P - P_{\infty})$ can be approximated from the Rayleigh bubble equation, written as

$$P - P_{\infty} = \rho_L(RR'' + 3(R')^2 / 2) \tag{23}$$

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 13). If the term RR'' is neglected, Equation (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 (10) and the convection response time of Equation (12), that is, R_{∞}/γ_h. It follows that the debris droplet radius which could promote self-triggering can be estimated from

$$r_{\text{self-trig}} > \frac{2\sigma}{9\rho_L h^2}(\rho c_v)^{1/3} \frac{(h_{fg})^{5/3}}{[v_{fg}(T_i - T_{\infty})]^{5/3}} \tag{24}$$

(g) 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 γ_h of equation (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

$$(P - P_{\infty}) A_L = M_L y'' \tag{25}$$

where $M_L = \rho_L A_L L$. The solution for y, based on an average pressure, is $y = ((P - P_{\infty})/\rho_L L)t^2/2$, for which the approximate hydrodynamic response time for the overlying pool is

$$\gamma_p^2 = \left.\frac{2y\rho_L L}{(P - P_{\infty})}\right|_{y=L} \tag{26}$$

The average pressure is estimated from

$$P \approx P_{\infty}(2V_{total}/A_L L) \tag{27}$$

The total steam volume V_{total}, if formed at ambient pressure, can be obtained from Equation (9) with E' replaced by

$$E'_{total} = NE' \tag{28}$$

where N is the total number of debris droplets participating in the steam formation process. It is possible for a steam explosion occur if the condition

$$\gamma_p \gg \gamma_h \tag{29}$$

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:

$r_{air} = 24$ mm

$r_{water} = 0.03$ mm

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:

$\gamma_h = 9.2$ s	convection
$\gamma_c = 1.8$ s	internal conduction
$\gamma_L = 0.006$ s	bubble growth, single droplet

That is, steam bubble growth from debris particle energy is limited by the convective heat transfer rate.

Equation (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 (29).

It was assumed that a debris stream which extended throughout the pool depth was the participating mass, corresponding to

$$M_d = \rho p R_0^2 L = 213 \text{ kg}$$

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

$$W = P_\infty E \frac{v_{fg}(P_\infty)}{h_{fg}(P_\infty)} = 19.3 \text{ MJ}$$

This indicates that about 7% of the total thermal energy is converted into mechanical energy. This is far higher that the 1% to 3% range reported in experiments (Reference 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

$\gamma_p = 0.38$ s

The convection heat transfer response time, found previously is

$\gamma_h = 9.2$ s

It follows that the condition for a steam explosion given in Equation (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 it 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 75 psig. 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 97 psig 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, Subarcticle 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 Paths

19E.2.3.3.1 Introduction

This section reviews the potential risk of certain suppression pool bypass paths and demonstrates that, with the exception of the wetwell drywell vacuum breakers, and certain other lines, bypass paths present no significant risk following severe accidents. Because of this insignificance, only the vacuum breakers and the other lines 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 14) review.

The results of the evaluation is that bypass lines evaluated contribute no more than about 10% 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 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 sequences develop during the course of an event, for example, as a result of low suppression pool water level. Such cases are considered in Section 19D.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 10CFR50 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 event, may make the early isolation of some lines impossible. This section evaluates the significance of bypass paths in order to justify that no additional treatment in the PRA is neccessary.

(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:

$$\text{Risk} = \text{Total [Event Frequency x Consequence]} \quad (30)$$

$$= F_{nbp} \times C_{nbp} + F_{bp} \times C_{bp} \quad (31)$$

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 (31) must be much less than the first, or:

$$\frac{F_{bp}}{F_{nbp}} \ll \frac{C_{nbp}}{C_{bp}} \quad (32)$$

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:

$$F_{bp} = F_{cd} \times \sum_i P_{cbpi} \quad (33)$$

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:

$$P_{cbpi} = P_{bpi} \times f_i \qquad (34)$$

where: f_i = The fraction of fission products generated during a core damage event which pass through line i (subsection 19E2.3.3.3 (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.

and P_{bpi} = The conditional probability of bypass in line i (Section 19E2.3.3.3 (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:

$$F_{nbp} = F_{cd} - F_{bp} \qquad (35)$$

Inserting equations (33), (34) and (35) into equation (32) yields:

$$P_{bpi} \times f_i << C_{nbp}/C_{bp} \qquad (36)$$

Assuming F_{bp} is much less than F_{cd} which would be consistent with the basis for containment isdolation.

If equation (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 36) was evaluated using the MAAP 3B-ABWR and CRAC codes to establish the approximate order of magnitude for evaluation purposes. To establish a threshold for this frequency, the consequence ratio [right side of equation (36)] was evaluated using the MAAP 3B-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", described in Appendix 19P, is the consequence from normal containment leakage; "Case 7" may be used as an approximation of the full suppression pool bypass consequence.

The corresponding ratio based on values in Table 19P.2-1 is 8.4E-4 which 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 RWCU 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 extremly 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 (36) of Section 19E2.3.3.1 establishes the need for evaluation of the flow splits and failure probability for each line not excluded in Table 19E.2-1. This section 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: 1) a path from the RPV which passes to the reactor building with the remainder passing to the suppression pool through the SRVs and 2) 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:

$$f = W_j/W_j + nW_k \qquad (37)$$

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:

$$f = f' / 1+f' \qquad (38)$$

where $f' = W_j/nW_k$

From the formula for turbulent compressible fluid flow (Reference 15)

$$W = 1891\, Yd^2\, [(dP)/KV]^{1/2} \qquad (39)$$

where W = j or k (lb/hr)
Y = Expansion factor
d = Internal diameter (in)
(dP) = Differential pressure (psid)
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 (cf/lb)

Solving for f',

$$f' = 1891\, Y_j d_j^{\,2}[dp/KV]^{1/2} / 1891\, ny_k d_k^{\,2}[dp/KV]^{1/2}$$
$$= Y_j d_j^{\,2}[dp/K]^{1/2} / nY_k d_k^{\,2}[dp/K]^{1/2} \qquad (40)$$

Equation (40) may be rearranged to show:

$$f' = (1/n)\,[Y_j/Y_k]\,[d_j/d_k]^2 \times [dP_j/dP_k]^{1/2}\,[K_k/K_j]^{1/2} \qquad (41)$$

The expressions in equation (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:

1. Containment pressure following the core melt is assumed to be at an average of 45 psig 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 (see Figures 19E.2-2 through 19E.2-12).

2. Prior to RPV melt-through, the reactor pressure vessel (RPV) is maintained at a relatively low pressure (100 psig) 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 24 inch drywell vent paths are consistent with the ABWR design configuration. For conservatism the vents are assumed to be one quarter uncovered.

3. 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 100 psig internal RPV pressure is assumed during the core melt process. This is based on an average 45 psig drywell pressure and an assumed SRV design which closes the SRV when a differential pressure of about 50 psid exists between the main steamline an the SRV discharge line.

   Depressurization of the RPV or containment through the bypass path is not considered. The assumption is made that pressure is continuously generated during the sever accident in sufficient quantity to uncover the SRV discharge or drywell vents.

4. The pressure from in 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 20 foot (6.0M) elevation head over the SRVs during the core melt process. For drywell sources a 15 foot (4.5M) 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.

5. The length of lines discharging to the suppression pool and through the bypass paths affects the resistance coefficient Equation (39). Based on the ABWR arrangement drawings this length is estimated to be approximately 85 ft. (25 M). For the drywell sources, the path to the suppression pool is estimated to be 5 ft. (1.5 M)

Other values used in the calculation are listed below:

Parameter	**Assumed Value**	**Basis**
Resistance Coefficient (K=f'L/D)		
Friction Factor	.011 to .018	Ref 14 (pg A-25) Size dependent)
Line Diameter (D)	Various	Line size (see Table 19E.2-1)
Other Resistances (K)		Ref 14 (pg A-30)
Gate valve	13	
Check valve	135	
Globe valve	340	
Entrance effects	.5	
Exit effects	1.0	
Expansion Factor (Y)	.6 to .9	Ref 14 (pg A022) (dP, K dep.)

Table 19E.2-19 shows sample results (f ' from equation 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:

(a) 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.

(b) No credit is taken for additional retention of fission products in the reactor building, in piping or through radioactive decay.

(c) 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) Current operating plant MSIV failure to close probability is about 4E-3/demand with a common mode failure probability (P_i) of about 1E-4/demand. For this evaluation the common mode failure probability of 1E-4 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 11.5 SCFH per valve. About 50% of the valves typically fail this local leak rate test at this level and about 10% are believed to typically exceed the 640 SCFH level allowed by ABWR proposed technical specifications. The leakage probability (P2) used in this analysis was based on three leakage groups:

		Probability	
Group	Leakage	Per Valve	Per Line
G1	<11.5 scfh	.5	.5
G2	11.5 to 640 scfh	.4	.2
G3	>640 scfh	.1	.01

The MSIV leakage probability (P2) is assigned a value of .71 to correspond to the total line leakage probability. Flow split fractions were determined for each of the groups 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 at 4E-3 from Reference 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. The industry failure rates for motor operated valves is about 3.6E-3/demand and 4.1E-3 for air operated valves. These failure rates are not significantly affected by the valve environments. A common cause failure among air operated valves was considered for lines containimg redundant series valves. For these lines a Beta factor of .18 was used for the failure of the second valve.

(f) 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 of 1.0. For this analysis a failure probability of 1.0 was conservatively 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 industry failure rates for check valves allowing complete reverse flow (P9), based on 7000 hours of operation per operating cycle, is about 8.4E-3 per 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. For these lines a Beta factor of .18 was used for the failure of the second 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. A value of 0.5 is judged reasonable especialy considering the potential for room environment degradation. 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 vale. A 0.5 probability is judged reasonable for the operators failure to act during the core damage event.

(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. A value of 1E-3 is judged reasonable during a core damage event.

(k) Pipe rupture is extremely rare in stainless steel piping. However, carbon steel piping has been observed to fail under certain conditions. The frequency of these failures has been widely studied and shown to be in the range of 1E-7 events/year. The probabilities of line rupture as a function of line size (P13, P14, P15) are taken from Reference 14. Four line segments outside of the containment are assumed for each bypass line. The intermediate line size (3 to 6 inches) probability is assumed to be twice that of the large line size (greater than 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 7000 hours). This approach of estimating pipe failure probability is judged to be conservative.

The failure probabilities used in the evaluation should be considered conditional probabilities, given a core melt. In general the above probabilities are not affected by the core melt process itself and can therefore be considered independent of the event process.

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 RWCU suction in combination with failure of the isolation valves to close. The concern is that 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 RWCU 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 Station Blackout event. Furthermore, should the isolation valves fail to close, the system arrangement assures that the core is not uncovered and EPGs require depressurization 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 RWCU lines above the core to avoid a potential siphon of the core inventory. In the event of an unisolated RWCU 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. This action should be possible prior to any impact on other ECCS equipment. These actions are included in Section 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.2-19) which illustrates the potential pathways. The evaluation is based on the conservative assumption of a station blackout event since it is believed to be the dominant core damage sequence and gives the highest bypass fractions.

(4) Evaluation of Results

Section 19E2.3.3.1 (4) provides a conservative justification that bypass paths with a total bypass fraction less than 8.4E-4 do not substantially increase the offsite risk. As is shown in Table 19E.2-21, the bypass probability is about 3.0E-5 for all potential paths not addressed in the Containment event trees. This total is well within the goal.

Potential bypass through the Wetwell-Drywell Vacuum Breakers are included in the containment event trees. (Section 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. These trees show that the total core damage frequency due to LOCAs outside of containment is about 1.3E-8 per year. 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 Appendix 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 Table 19E.2-21.

P_1 Oper. 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 warmup lines the operator is assumed to act to close an open valve, if needed. The basis for the value chosen (P10 in Section 19E.2.3.3 (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.

Q_1 Second Division 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 a second electrical division. It is assumed that such failure results from environmental effects from flooding or pressurization 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 noted 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 RWCU break in subsection 19R.4.5 shows that no cooling systems will be damaged.

Compartment pressurization and environmental effects of high pressure LOCAs in secondary containment were considered in the development of Figures 19E.2-20a through c.

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 next adjacent division in a clockwise progression through the reactor building.

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 qualification of the equipment to 212 degrees F and 100% humidity makes the probability of further system unavailability unlikely. Where a LOCA could occur in an area adjacent to a separate division, a value of 1E-3 was assumed for Q_1, based on conservative engineering judgement, to represent the remote possibility for failure of these adjacent systems.

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_0 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 and are summarized below:

COOLANT MAKEUP FAILURE (Q_0)
BREAK SIZE

	Small	Medium	Large
Div. not Affected	2.2E-7	6.2E-7	6.1E-7
1 Div. affected	1.1E-6	8.6E-6	8.5E-6
2 Div. affected	3.6E-4	3.7E-3	3.7E-3

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.

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. Since condensate is effectively an unlimited supply 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/RWCU bottom head drain (a unique situation addressed separately in Section 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 firewater. 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 lowering RPV level to a level below the affected penetration, if necessary. Adequate instrumentation and long term makeup from firewater and condensate sources would normally be available.

(3) Conclusion

For each of the event trees shown in Figures 19E.2-10a through c the total non-bypass and bypass core damage frequencies are shown and are summarized below:

Core Damage Frequency (events/yr)

	Non-Bypass	Bypass	Total
Small LOCAs	1.2E-8	1.1E-9	1.3E-8
Intermediate LOCAs	2.3E-10	1.2E-10	3.5E-10
Large LOCAs	2.0E-10	4.8E-13	2.0E-10
TOTAL	1.2E-8	1.2E-9	1.3E-8

Ex-containment LOCA events without bypass represent a small fraction of the total core damage frequency (1.6E-7) are therefore 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 is less than 1% of the total evaluated core damage frequency. Large LOCAs can be excluded from further consideration on the basis of low probability. Exclusion of Medium 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 Section 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 Events

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 retained and that, in effect, the suppression pool scrubbing is transferred from the suppression pool to the RHR equipment rooms.

19E.2.3.4.1 RHR Equipment Room Flooding

The RHR equipment room drains to a sump room below. This sump room also receives drains from the HPCF equipment room (in two cases) and from the RCIC room (in one case). The design of these drains is assumed to have a device to prevent one sump from filling and backflowing up to the HPCF or RCIC rooms. The device is assumed to function without AC power. This prevents the loss of HPCF or RCIC resulting from RHR equipment room flooding.

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. This wall runs parallel with the discharge pipe at a distance of approximately 4 ft. The length of this wall is 43 ft, the height is 20 ft. The RHR pump discharge piping is assumed to run 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 (<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 100 psi. 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 4 psi, if the maximum bubble pressure of 100 psi 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 most critical wall in a typical room was chosen for this investigation. This wall runs along the 90°-270° direction of the reactor building and connects to the exterior wall and the containment at both sides. 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 wall was examined for its ability to withstand a 2g earthquake which is more severe than that which leads to the heat exchanger mounting failure 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 4 psig 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 wall was modeled as an equivalent single degree of freedom system subjected to the pressure transient described above. The results show that the maximum dynamic amplification factor is 0.26. The equivalent static chugging pressure is thus about 1 psig.

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 by treating the wall as a flat plate with fixed supports along the edges. The resulting maximum moment is found to be about 56% 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 seismic-induced flood. This ensures that fission products can be scrubbed by the entrapped water.

19E.2.3.5 Potential for 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.

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 carried into the piping to become entrained out of the COPS at the stack. Calculation of entrainment at the surface of the suppression pool is also considered using the work of Rozen, et. al. (Reference 17) and is found to have an insignificant impact on fission product release.

19E.2.3.5.1 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-condensible 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:

$$\frac{dP}{dt} = \frac{0.665ART\sqrt{P\rho_g}}{V_w M_{a,w}} \tag{42}$$

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.2.3.5.2 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 to the level of the bottom of the vessel (Elevation 0.0 meters). The 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.2.3.5.2.1 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. A uniform void generation rate is assumed at each point in the liquid. The average void fraction is then given by:

$$\bar{a}_P = \frac{j_g / U_\infty}{2 + C_0 j_g / U_\infty} \tag{43}$$

(Reference 1) where the mass flow rate, W_p, at the top of the pool determines the superficial gas velocity:

$$j_g = W_p / A\rho_g \tag{44}$$

and the drift velocity, U_∞, is given by:

$$U_\infty = 1.53\left[\sigma g\left(\frac{\rho_l - \rho_g}{\rho_l^2}\right)\right]^{1/4} \tag{45}$$

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:

$$h = \frac{h_0 \rho_l}{\bar{\alpha}_p \rho_g + (1 - \bar{\alpha}_p)\rho_l} \tag{46}$$

where: h_0 = initial pool height.

19E.2.3.5.2.2 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 torus. 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:

$$\alpha = \frac{j_g / U_\infty}{1 + C_0 j_g / U_\infty} \tag{47}$$

(Reference 1) where the terms are analogous to those defined for Equations (43) through (45), but now refer to drywell conditions. Comparison of Equation (47) to Equation (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 in principle, can be calculated using the relationship in Equation (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:

$$r = \sqrt{\frac{2A}{10\pi}} \tag{48}$$

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.2.3.5.2.3 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 the 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:

$$\frac{d}{dt}(m_p h_f) = W_p h_g \tag{49}$$

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:

$$W_P = \frac{m_P \frac{dh_f}{dP}}{h_{fg}} \dot{P} \qquad (50)$$

The ideal gas law is used in the drywell to derive the relationship:

$$W_D = \dot{P} V_D M_{a,D} / R T_D \qquad (51)$$

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 (50) and (51):

$$\frac{W_D}{W_P} = \frac{V_D M_{a,D} h_{fg}}{R T_D m_P \frac{dh_f}{dP}} \qquad (52)$$

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 (100 C) and that the suppression pool is hot (160 C). Under these conditions the flow rate ratio is 0.065. 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 (52). The bounding calculation shows that less than 10% 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.2.3.5.2.4 Application to ABWR

Pool swell is maximized at high pressure and temperature conditions. 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 this condition, the average void fraction due to pool flashing is about 4%. This results in a pool swell of about 0.59 meters. Since the bottom of the COPS penetration is at 3.87 meters, this mechanism alone will not lead to flooding of the COPS penetration.

If the pool level were to rise an additional 3.28 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 20% due to through flow from the drywell is required for this additional pool swell. Applying Equations (47), (48) and the upper bound value from Equation (52), one arrives at an radius of 0.78 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.

19E.2.3.5.3 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 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:

$$U_{threshhold} = 2.7\left[\sigma g\left(\frac{\rho_l - \rho_g}{\rho_g^2}\right)\right]^{1/4} \qquad (53)$$

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 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. 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.2.4 Analysis for Recovery Sequences

In order to determine the sensitivity of the PRA to various assumptions made in both the deterministic and probabilistic portions of the analysis a series of sensitivity studies were performed using MAAP. Additionally, some sequences with unusual characteristics, such as those having no containment structural failure are considered in this section.

19E.2.4.1. Time of Drywell Spray Initiation

The drywell 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 2 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 two 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 two 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 two hour period, more of the water initially in the lower drywell would boil off. Eventually, the debris in the lower drywell could 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 in 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 up to half and hour. 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. 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.

Fission product scrubbing is also provided if the release is from the wetwell airspace, as would occur if the wetwell airspace were provided with a rupture disk. The release fractions associated with this type of release is examined in Subsection 19E.2.4.8. The results of that study are applied to this case in the effects analysis of Section 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 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 there is no 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. A loss of all core cooling with vessel failure at low pressure case was chosen for the analysis. The transient was run for two days. The drywell head failure pressure was raised to prevent containment rupture. The containment leakage area was chosen such that the leak rate was equal to the

technical specification limit of 0.4% per day at rated pressure.

The average pressure during the transient is 0.55 MPa (65 psig). The first appreciable fission product release occurs at 2.3 hours. After 50 hours the release fraction of noble gasses is 0.54%. The release fractions of the volatile fission products are much lower, 5.2E-6 and 3.8E-6 for CsOH and CsI, respectively. The results of this analysis is used in the consequence analysis of section 19E.3 with the sequence name NCL.

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 section 19D.5 the probability 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 until the time of containment structural failure is identical to the LCHP events described in Subsection 19E.2.2.2 until the time of vessel failure. At this time it is assumed that the drywell head fails. 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 (15 psig). Just before the 2 hour mark MAAP 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 (29 psig) before the aerosols are blown out and the containment pressure falls to about 0.2 MPa (15 psig).

The fission product release for this sequence is much higher than that for the base case (LCHP-PF-D-H). Fission product release begins at the time of vessel failure (1.4 hours). The noble gas release is very slow since most of the noble gasses are trapped in the wetwell. After 96 hours only 85% of the noble gasses have been released to the environment. 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 both about 47%.

Since the fission product release is significantly higher than that for the base case, this information will be included in the consequence analysis of Section 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 following a seismic event 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 was nearly equal to the initial level of the water in the suppression pool. 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.

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 pool. 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.

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 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 these various sources (References 19 through 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 Analysis

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 analysis 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 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) Plant 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 RV 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-condensible 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 one meter of sacrificial concrete. 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 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.

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.

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 and failure location of the containment,

(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 Section 19E.2.2, except that the model parameter for blockage and hydrogen generation (FCRBLK, see Reference 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 (52 psig). 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 operator blows the reactor 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 Section 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 Section 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 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 remain 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 in the design. Therefore, no

further investigation of the impact of fission product release from the fuel is required.

19E.2.6.3 CsI Re-evaporation

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 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 re-evaporization 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 200,000 kg of water falls into the lower drywell at the time of vessel failure.

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 sequence 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 Section 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 plenum 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 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 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 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,

$$\frac{\dot{Q}_{blade}}{\dot{Q}_{decay}} = \frac{(mc_p)_{blade}}{(mc_p)_{blade} + (mc_p)_{fuel} + (mc_p)_{can}} \quad (1)$$

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.

$$\dot{Q}_{blade} = (FA)_{blade}\,\sigma\left(T^4_{fuel} - T^4_{blade}\right) \quad (2)$$

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 recov... a time window of about seven minutes. Based ... standard models for recovery of systems and operator ..., 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 Section 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 about 70% 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 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 only 0.3% 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. The following sections examine the available experimental data base 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-condensible 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, 90 psig (0.72 MPa), was selected such that there is a very small probability that the containment structure fails. As shown in Appendix 19F.3.1, the weakest portion of the ABWR containment is the drywell head. The median failure pressure of the drywell head is estimated to be 134 psig (1.03 MPa abs). The other portions of the containment have an estimated failure pressure of 180 psig. 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 Subsection 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, two cases must be considered. For sequences in which the firewater system is used and water is added to the containment, as described in Section 19E.2.2, there is approximately a 5% chance that the drywell head will fail. For sequences without water addition to the containment, the drywell head failure probability is about 2%. These probabilities are used in the quantification of the containment event trees in Section 19D.5.

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 3% 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 20 square inches (.0129 m^2) 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 less than 5% 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 Attachment 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 (90 psig). 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% 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.

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 Section 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.2.7 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 Appendix 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 an estimated CCFP of 0.1% for all core damage events. Additional sensitivity analyses were considered which indicate that an upper bound on the impact of DCH is 1.5%. 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 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. 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 Appendix 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.

The conclusion from all of these uncertainty calculations were:

(1) For the dominant core melt sequences that release core material into the containment, 90% 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 1.5% of the sequences with significant CCI will suffer pedestal failure. Combining this conclusion with the first, only 0.15% 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 (approximately 98 percent) of the accident demands. Small amounts of leakage have a probability of 1.8 percent per event, and can result in medium volatile fission product releases (one to ten percent of initial inventory). Volatile fission product releases on the order of 10 to 20 percent 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 only 0.39 percent.

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 section 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 only 16 percent of the hypothesized severe accident sequences.

The COPS is part of the atmospheric control system and consists of two 8-inch diameter overpressure relief rupture disks mounted in series on a 14-inch line which connects the wetwell airspace to the stack. 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 insuring 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 was used to simulate a series of severe accident sequences for both release mechanisms. These severe accident sequences are described in Section 19E.2.2. Failure pressure of the drywell head was assumed to be equal to its median ultimate strength, 1.025 MPa (134 psig). 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 the structural integrity of the containment. As shown in Appendix 19F, the service level C for the ABWR is 97 psig, 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 A to Appendix 19F, 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 MPa (134 psig). 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.

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 (14 psi). 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 (7 psi).

A rupture disk setpoint of 0.72 MPa (90 psig) 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 90 psig setpoint results in a 5% probability of drywell head failure prior to rupture disk opening for a 0.1 MPa (14 psi) drywell to wetwell pressure difference. For a drywell to wetwell pressure difference of 0.05 MPa (7 psi), the drywell head failure probability prior to rupture disk opening is 2%. 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 (90 psig) at 366 K (200 F). Two cases were examined using MAAP 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 (88 psig) which corresponds to a rupture disk temperature of 422 K (300 F); and, second, the pressure setpoint was increased to 0.735 MPa (92 psig) 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 from about 2% for the base case to about 3% 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 to about 1.5%. 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 from about 5% for the base case to about 6.5% for the case with the rupture disk temperature of 311 K (100 F) and decreases to about 4% 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.

An eight-inch rupture disk was selected. This is sufficient to allow 35 kg/sec of steam flow at the opening pressure of 90 psig (0.72 MPa-a) and corresponds to a energy flow of about 2.4% rated power. 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 less than 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 calculations for the various accident scenarios were investigated in Section 19E.2.2 and 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 15%. For all cases with release via the COPS, MAAP predicts release fractions of less than 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 (90 psig) and the rupture disk will open at 31.1 hours. Without the rupture disk, the drywell will reach 1.025 MPa (134 psig) 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 only 4% (see 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 (134 psig). For the range of severe accident sequences described in Section 19E.2.2, the probability of RHR recovery in a similarly defined time window is about 11%.

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 (.0054 to 2.19 ft^2). A fully open vacuum breaker has an effective 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 bypass effective areas less than 50 cm^2 (.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 (.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. One 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 Summary

A wetwell pressure setpoint of 0.72 MPa (90 psig) 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 5.1% maximum 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 4%. Therefore, no significant error is introduced into the calculation.

Table 19E.2-27 indicates that the probability of drywell head failure increases by a factor 50 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 factor of 100 without the COPS system, the core damage probability associated with Class II events also increases by a factor of 100. Figure 19E.2-22 shows the probability of exceedence versus whole body dose at 1/2 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 section provides the bases for sizing the lower drywell flooder system. The system is described in detail in Section 9.5.12 of the ABWR SSAR.

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 triggering a fusible plug at the line exit (lower drywell side). Since four inch diameter fusible disks may be commercially available, the flooder line diameter was chosen as four inches.

The teflon disk resides between the stainless steel disk and the fusible plug in the flooder valve. 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 400°C and a maximum continuous operating temperature of 288°C both of which are above the plug melting temperature of 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 Section 19E.2.8.2.2. The expected flow rate in the flooder system can be

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 Section 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 one percent 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 percent 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 two 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/sec (18 liters/sec). Assuming a four inch throat as discussed in Section 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/sec.

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

$$A_f = \frac{\pi}{4} d_f^2 n_f \qquad (1)$$

$$= 0.073 m^2$$

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, see 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

$$\dot{v}_{fl} = A_f \sqrt{\frac{2g\Delta z}{1+k}} \qquad (2)$$

$$= 0.099 m^3 / sec$$

where $\dot{v}_{fl}$ is the total volumetric flow rate through nine lines and g is the acceleration of gravity. For a liquid density of 980 kg/m^3, this corresponds to a system flow rate of 97 kg/sec and an individual line flow rate

of 10.8 kg/sec. 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

$$\dot{v}_{vap} = \frac{\dot{Q}}{h_{fg}\rho_{liq}} \qquad (3)$$

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

$$\dot{v}_{fill} = \dot{v}_{fl} - \dot{v}_{vap} \qquad (4)$$

The time to fill the lower drywell to the exit of the flooder is

$$t_{fill} = \frac{V_{fill}}{\dot{v}_{fill}} \qquad (5)$$

where V_{fill} is 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 to 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 one percent (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 39.26 MW. If all of this heat is transferred to the flooder water, the rate and time to fill the lower drywell are

$$\dot{v}_{fill,d} = 0.080 m^3 / sec$$

$$t_{fill,d} = 21 \text{ minutes}$$

The maximum heat flux from the surface of a debris bed that has been experimentally observed (see Section 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

$$\dot{v}_{fill,min} = 0.022 m^3 / sec$$

$$t_{fill,max} = 1.3 \text{ hours}$$

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 1.3 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
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 grove, 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 increased from its base case value of 0.001 to 0.01, 0.1, and 1.0.

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 one, two and three orders of magnitude, respectively for the three sensitivity cases. However, the base case results for Dry CCI are so small that a three order of 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 a two fold increase for a two orders of magnitude increase in the passive flooder failure probability, and a ten fold increase for a three order of magnitude increase. 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 probability associated with the Dry CCI is only 3.5E-10. 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 does not affect net risk for probabilities above 3E-10. Therefore, no chart of the impact on risk was created. The value of the COPS system is not in 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/sec which can be provided by two of the ten flooder lines. The expected flow rate is 0.099 m^3/sec (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 1.3 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/MW_{th} 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 Section 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.

19E.2.9 References

1. MAAP-3.0B Computer Code Manual, EPRI NP-7071-CCML, November 1990.

2. Advanced Light Water Reactor Utility Requirements Document, EPRI Report NP-6780-L.

3. Advanced Reactor Severe Accident Program, *Technical Support for the EPRI Debris Coolability Requirement for Advanced Light Water Reactors, Task 8.3.5.6*, Fauske and Associates, Inc., Burr Ridge, IL, August 1988.

4. NUREG/CR-3920, SAND84-1246, Rev. 3, *CORCON-MOD2 User's Manual*, August 1984.

5. R.E. Henry, *Key Phenomenolgical Models for Assessing Explosive Steam Generation Rates*, IDCOR Technical Report 14.1A, 1983.

6. R.E. Henry, *Key Phenomenological Models for Assessing Explosive Steam Generation Rates*, IDCOR Technical Report 14.1B, 1983.

7. NUREG-75/014, WASH 1400, *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, October 1975.

8. T.G. Theofanous, *Scaling Considerations in Steam Explosions*, Procedures 1987 National Heat Transfer Conference, Pittsburgh, August 9-12, 1987, pp. 58-67.

9. Sir Horace Lamb, Hydrodynamics, Dover.

10. F. Kreith, Principles of Heat Transfer, 3rd Edition, IEP - A Dun-Donnelley Publisher, N.Y. 1976.

11. P.J. Berenson, *Film Boiling Heat Transfer from a Horizontal Surface*, Transactions of the ASME, Series C, Volume 83, 1961.

12. G.A. Green, *Experiments on Melt Spreading and Bubbling Heat Transfer*, Severe Accident Research Partners Meeting, Bethesda, MD, October 17-21, 1981.

13. F.J. Moody, *Pressure Suppression Containment Thermal-Hydraulics State of the Art*, NUREG/CP-0014, Volume 1, 1980.

14. 22A7007 Revision 21, GESSAR II, 238 Nuclear Island, General Electric Company.

15. Crane, *Flow of Fluids Through Valves, Fitting, and Pipe*, Technical Paper 410, 1969.

16. NEDE 22056 Revision 2, *Reliability Analysis Data Manual*, General Electric Company, January 1986.

17. A.M. Rozen, S.I. Golub and T. I. Vitintseva, *Calculating the Transported Entrainment During Sparging*, Translated by Polyglot Language Service.

18. S. Kutateladze, *Elements of the Hydrodynamics of Gas-Liquid Systems*, Fluid Mechanics - Soviet Research, 1, 4, 1972.

19. Nuclear Regulatory Commission. *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG-1150, June 1989.

20. Nuclear Regulatory Commission. *Evaluation of Severe Accident Risks: Peach Bottom, Unit 2*, NUREG-4551 Volume 4, December 1990.

21. Nuclear Regulatory Commission. *Evaluation of Severe Accident Risks: Grand Gulf, Unit 1*, NUREG-4551 Volume 6, December 1990.

22. Nuclear Regulatory Commission. *Individual Plant Examination: Submittal Guidance*, NUREG-1335, August 1989.

23. Nuclear Regulatory Commission, Generic Letter No. 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities*, November 23, 1988.

24. Probabilistic Risk Assessment for Kuosheng Nuclear Power Station, Unit 1, Republic of China Atomic Energy Council, July 1985.

25. Electric Power Research Institute, *Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B*, Draft 1992.

26. Nuclear Regulatory Commission, *MAAP 3.0B Code Evaluation - Final Report*, Draft, December 1991.

27. Advanced Light Water Reactor Utility Requirements Document, Volume II, Chapter 1, Appendix A: PRA Key Assumptions and Guidelines, EPRI Report NP-6780-L.

28. R. O. Gauntt, R. D. Gasser, L. J. Ott, *The DF-4 Fuel Damage Experiment in ACRR with a BWR Control Blade and Channel Box*, NUREG/CR-4671, SAND 86-1443, November 1989.

29. R. Tokraz and R. Libby, *Recriticality in a Boiling Water Reactor Following a Core Damage Accident*, Proceedings of the 17th Water Reactor Safety Meeting, October 1989.

30. K.C. Wagner, *Analysis of a High Pressure ATWS with Very Low Makeup Flow*, DOE/ID 10211, October 1988.

TABLE 19E.2-1

POTENTIAL SUPPRESSION POOL BYPASS LINES

DESCRIPTION	NUMBER OF LINES	PATHWAY FROM	PATHWAY TO	PATHWAY SIZE (mm) (1 in. = 25.4 mm)	ISOLATION VALVES	BASIS FOR EXCLUSION (SEE NOTES)
Main Steam	4	RPV	ST	700	(AO, AO)	-
Main Steam Line Drain	1	RPV	ST	200	MO, MO	3
Feedwater	2	RPV	ST	550	CK, CK	-
Reactor Inst. Lines	30	RPV	RB	6	CK	-
CRD Insert/Withdraw	103	RPV	RB	<1	CK, MA	1
HPCF Discharge	2	RPV	RB	200	CK, MO	-
HPCF Warmup	2	RPV	RB	25	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 Warmup Lines	2	RPV	RB	25	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

TABLE 19E.2-1 (Continued)

POTENTIAL SUPPRESSION POOL BYPASS LINES

DESCRIPTION	NUMBER OF LINES	PATHWAY FROM	PATHWAY TO	PATHWAY SIZE (mm) (1 in. = 25.4 mm)	ISOLATION VALVES	BASIS FOR EXCLUSION (SEE NOTES)
RWCU Suction	1	RPV	RB	200	(MO, MO)	-
RWCU Return	1	RPV	RB	200	MO, MO	5
RWCU Head Spray Line	1	RPV	RB	150	CK, MO, MO	3
RWCU 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	50	MO, MO	1
LDS Instruments	9	RPV	RB	6	CK	-
SPCU Suction	1	SP	RB	200	MO, CK	2
SPCU Return	1	SP	RB	250	MO, MO	2
Cont. Atmosphere Monitor	6	DW	RB	20	MA	8
LDS Samples	2	DW	RB	30	(SO, SO)	-
Drywell Sump Drains	2	DW	RB	100	MO, MO	-
HVCW/RBCW Supply	4	DW	RB	100	CK, MO	1
HVCW/DWCW Return	4	DW	RB	100	MO, MO	1
DW Exhaust/SGTS	2	DW	RB	250	AO, AO	7
Wetwell Vent to SGTS	1	WW	RB	250	AO, AO	2
DW Purge	1	DW	RB	300	AO	-
WW Inerting/Purge	2	WW	RB	550	AO	2
Instrument Air	2	DW	RB	50	CK, MO	1
SRV Pneumatic Supply	3	DW	RB	50	CK, CK	1
Flamability Control	1	DW	RB	100	(MO, MO)	3
ADS/SRV Discharge	8	RPV	WW	300	RV	-
ACS Crosstie	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

TABLE 19E.2-1 (Continued)

POTENTIAL SUPPRESSION POOL BYPASS LINES

LEGEND AND ACRONYMS

PATHWAY

Source (From)		Termination (To)	
RPV	Reactor Pressure Vessel	WW	Wetwell
DW	Drywell	RB	Reactor Bldg
SP	Suppression Pool	WW	Wetwell
		ST	Steam Tunnel

Isolation Valve Types

AO Air Operated
MO Motor Operated
RV Relief Valve
CK Check Valve
MA Manually Actuated
SO Solenoid Operated
( ) Common Mode Failure Potential (See Section 19E2.3.3.3 (2))

Bases for Exclusion

1. Closed systems 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, extreme restrictions in CRD seals provides the basis for excluding those 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 100 psig) 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 LPCF loop A and RWCU) are excluded since they are bounded by the evaluation of feedwater.

6. Acceptable long term leakage from the drywell to the reactor building following a design basis accident is specified at .4% of drywell volume per 24 hours. During severe accident conditions this leakage could be somewhat greater due to higher than design basis containment pressure. However, the contribution of this leakage to overall risk is ignored because this leakage is through numerous tortuous passages of small diameter which provide ample opportunity for plateout and plugging effects (see subsection 19E2.1.3.4). A discussion of the drywell access tunnels is included in section 19F.

7. Drywell purge lines are normally closed and 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 RPV Pressure	Core covered 50 psig RCIC trip 150 psig RCIC rated flow	Core covered >150 psig
b) D.C. Battery Capacity	11,400 amp-hrs Div. 1, 2, 3 & 4	Sufficient with load shedding
c) RCIC Water Source	1) CST - 20 x 10^3 ft^3 2) Suppression pool - 126 x 10^3 ft^3	CST sufficient with RPV pressurized
d) RCIC Room Temperature	151 F	<151 F
e) Drywell Temperature	340 F	<340 F
f) Drywell Pressure	45 psig	<45 psig
g) Wetwell Temperature	219 F	<219 F
h) Wetwell Pressure	45 psig	<45 psig
i) Control Rooms		
- Main	122 F	<122 F
- Lower	122 F	<122 F
- Computer	122 F	<122 F

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

TABLE 19E.2-3 (Continued)

DEFINITION OF ACCIDENT SEQUENCE CODES

Character 7: Mode of Release

N	Normal Containment Leakage
P	First release via leakage through Moveable Penetrations
R	Overpressure Protection Relief Rupture Disk
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 Rupture Disk Opens

Time	Event
0.0	MSIV Closure
4.2 sec	Reactor Scrammed
0.4 hr	Indicated Water Level at 2/3 Core Height One SRV Opened by Operator
1.8 hr	Vessel Failed
2.7 hr	Water in Lower Drywell Boiled Off Corium Heatup Begins
5.4 hr	Passive Flooder Opens
20.2 hr	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 sec	Reactor Scrammed
0.4 hr	Indicated Water Level at 2/3 Core Height One SRV Opened by Operator
1.8 hr	Vessel Failed
2.7 hr	Water in Lower Drywell Boiled Off Corium Heatup Began
4.0 hr	Firewater Spray Started
7.0 hr	Suppression Pool Overflows to the Lower Drywell
23.6 hr	Firewater Spray Stopped
31.1 hr	Rupture Disk Opened
56.6 hr	Water in Lower Drywell Boiled Off
61.1 hr	Passive Flooder Opened

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 sec	Reactor Scrammed
0.4 hr	Indicated Water Level at 2/3 Core Height One SRV Opened by Operator
1.8 hr	Vessel Failed
2.7 hr	Water in Lower Drywell Boiled Off Corium Heatup Began
4.0 hr	Firewater Spray Started
7.0 hr	Suppression Pool Overflows to the Lower Drywell
23.6 hr	Firewater Spray Stopped
31.1 hr	Rupture Disk Opened
56.6 hr	Water in Lower Drywell Boiled Off
61.1 hr	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 sec	Reactor Scrammed
0.3 hr	Core Uncovered
2.0 hr	Vessel Fails Corium and Water Entrained into Upper Drywell
2.0 hr	Passive Flooder Opens
4.0 hr	Drywell Spray Initiated
25.0 hr	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 sec	Reactor Scrammed
0.3 hr	Core Uncovered
2.0 hr	Vessel Fails Corium and Water Entrained into Upper Drywell
2.0 hr	Passive Flooder Opens
2.1 hr	Seal Degradation Temperature Reached
18.1 hr	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 sec	Reactor Scrammed
52 sec	RCIC Injection, Suction from CSP
1.3 hr	RCIC Suction Switched to Suppression Pool
4.4 hr	RCIC Suction Switched to CSP
8.0 hr	RCIC Failure
9.0 hr	Suppression Pool began to overflow to Lower Drywell
9.8 hr	Manual ADS
9.9 hr	Collapsed Water Level Fell to 2/3 core height Firewater Addition System Injection Began
32.3 hr	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 sec	Reactor Scrammed
52 sec	RCIC Injection, Suction from CSP
1.3 hr	RCIC Suction Switched to Suppression Pool
4.4 hr	RCIC Suction Switched to CSP
8.0 hr	RCIC Failure
9.3 hr	Core Uncovered
9.7 hr	One SRV opened by operator
12.3 hr	Vessel Fails
21.1 hr	Lower Drywell Water Boils Away
23.5 hr	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 sec	Reactor Scrammed
1.1 min	RCIC Injection
2.9 hr	Manual Open 1 SRV
3.0 hr	HPCF Injection
3.1 hr	RCIC Trip on Low Turbine Pressure
21.7 hr	Rupture Disk Opens
>72 hr	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 sec	High Drywell Pressure Signal
4.4 sec	Reactor Scrammed
14.9 sec	MSIV Closed
2.8 min	Core Uncovered
1.4 hr	Vessel Failed
5.7 hr	Passive Flooder Opened
19.1 hr	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 min	Core Uncovered
0.5 hr	One SRV Opened by Operator
1.3 hr	Vessel Failed
1.9 hr	Water in Lower Drywell Boiled Off Corium Heatup Begins
4.4 hr	Passive Flooder Opens
18.7 hr	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 hr	Vessel Fails Corium and Water Entrained into Upper Drywell
1.4 hr	Passive Flooder Opens
1.4 hr	Seal Degradation Temperature Reached
17.8 hr	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
4.1 min	Core Uncovered
1.9 hr	Suppression Pool Began to Overflow to Lower Drywell
3.6 hr	RCIC Tripped
3.8 hr	SRV Opened
5.6 hr	Vessel Failed
8.6 hr	Rupture Disk Opened Fission Product Release Began

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 Opening	End of CsI Release	Release Fraction of CsI @ 72 hours
LCLP-PF-R-N	1.8 hr	20.2 hr	20.2 hr	100 hr	<1E-7
LCLP-FS-R-N	1.8 hr	31.1 hr	31.1 hr	76 hr	<1E-7
LCHP-PS-R-N	2.0 hr	25.0 hr	25.0 hr	50 hr	<1E-7
LCHP-FS-R-N	2.0 hr	50 hr*	50 hr*	125 hr*	<1E-7*
LCHP-PF-P-M	2.0 hr	18.1 hr	N/A	70 hr	8.8E-2
SBRC-PF-R-N	12.3 hr	23.5 hr	23.5 hr	100 hr	<1E-7
LBLC-PF-R-N	1.4 hr	19.1 hr	19.1 hr	125 hr	<1E-7
LBLC-FS-R-N	1.4 hr	29.5 hr	29.5 hr	67 hr	<1E-7
NSCL-PF-R-N	1.3 hr	18.7 hr	18.7 hr	105 hr	<1E-7
NSCL-FS-R-N	1.3 hr	30.7 hr	30.7 hr	69 hr	<1E-7
NSCH-PF-P-M	1.3 hr	17.8 hr	N/A	65 hr	7.3E-2
NSCH-FS-R-N	1.3 hr	50 hr*	50 hr*	125 hr*	<1E-7*
NSRC-PF-R-N	5.6 hr	8.6 hr	8.6 hr	110 hr	<1E-7
NSRC-FS-R-N	5.6 hr	26.4 hr	26.4 hr	120 hr	<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
a	7.E-6 m^2/s	Thermal Diffusivity of Corium
c_v	480 J/kg-K	Specific Heat of Corium
q	9000 kg/m^3	Density of Corium
r	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
q_{air}	1.1 kg/m^3	Density of Air
q_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

The above 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 mm	in	Flow Split Fraction 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	MSIV closure	1.0E-4	a
P2	MSIV leakage probability	7.1E-1	b
P3	Turbine Bypass Isolation	4.0E-3	c
P4	Main condenser failure	1.0	c
P5	MSL break outside containment	8.0E-6	d
P6	Air operated valve (NO)	4.1E-3	e
P7	DC Motor operated valve (NO)	3.6E-3	e
P8	AC Motor Operated valve (NO-SBO)	1.0	f
P9	Check Valve	8.4E-3	g
P10	Motor operated valves (NC)	5.0E-1	h
P11	Motor operated valves (NC)	2.8E-4	i
P12	Inadvertent opening	1.0E-3	j
P13	Small line break	2.4E-4	k
P14	Medium line break	1.6E-5	k
P15	Large line break	8.0E-6	k

Table 19E.2-21

Summary of Bypass Probabilities

Lines from the RPV

Pathway	Flow Split Fraction	Bypass Probability Equation	Bypass Probability	Bypass Fraction	Figure 19E.2-19
Main Steam	6.7E-1	4*P1*(P3*P4+P5)	1.6E-6	1.1E-6	A
Main Steam Leakage	2.2E-5	4*P2*(P3*P4+P5)	1.1E-2	2.5E-7	A
Feedwater	5.2E-1	2*P9*P15	2.4E-8	1.3E-8	B
Reactor Inst. Lines	3.1E-5	30*P13*P9	6.0E-5	1.9E-9	D
HPCF Discharge	1.1E-1	2*P9*P10*P14	1.3E-7	1.5E-8*	C
HPCF Warmup	1.0E-3	2*P10*P11*P13	6.7E-8	6.7E-11*	C
SLC Injection	3.0E-3	1*P9*P13	3.6E-7	1.1E-9	B
RCIC Steam Supply	6.9E-2	1*P8*P14	1.6E-5	1.1E-6	E
LPFL Discharge	1.7E-1	2*P9*P10*P15	6.7E-8	1.1E-8*	C
LPFL Warmup Line	1.0E-3	2*P10*P11*P13	6.7E-8	6.7E-11*	C
RWCU Suction	1.2E-1	1*P8*P14	1.6E-5	2.0E-6	E
RWCU Inst Lines	3.1E-5	4*P13*P9	8.1E-6	2.5E-10	D
Post Acc Sampling	1.0E-3	4*P8*P13	9.6E-4	9.9E-7	J
LDS Instruments	3.1E-5	9*P13*P9	1.8E-5	5.7E-10	D
SRV Discharge	6.9E-2	8*P14	1.3E-4	8.8E-6	K
			Total	1.4E-5	

*These lines may be excluded for station blackout events

Table 19E.2-21

Summary of Bypass Probabilities (Continued)

Lines from the Drywell

Pathway	Flow Split Fraction	Bypass Probability Equation	Bypass Probability	Bypass Fraction	Figure 19E.2-19
Cont Atmos Monitor	8.9E-4	6*P9*P13	1.2E-5	1.1E-9	D
LDS Samples	1.7E-3	2*P8*P13	4.8E-4	8.2E-7	E
Drywell Sump Drain	3.0E-2	2*P8*P13	4.8E-4	1.4E-5	J
DW Purge	4.6E-1	1*P6*P11	1.1E-6	5.3E-7	I
ACS Crosstie	1.1E-1	2*P12	1.5E-6	1.6E-7	H
*WW-DW Vac Bkr	2.6E-1	8*P9	6.7E-2	1.7E-2	G
		Total excluding vacuum breaker		1.6E-5	
		Grand Total excluding vacuum breaker		3.0E-5	
			Goal	8.4E-4	

* 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	HPCS 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
81	Containment Spray Status Following Vessel Breach
103	Containment Vented Following Vessel Breach
106	Containment Spray Status Late
119	Containment Vented Late

Table 19E.2-22
NUREG/CR-4551 GRAND GULF APET EVENTS BY CATEGORY (CONTINUED)

Event Number	Description
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
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

Table 19E.2-22
NUREG/CR-4551 GRAND GULF APET EVENTS BY CATEGORY (CONTINUED)

Event Number	Description
Hydrogen Related Phenomena/Issues (continued)	
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-condensibles or Steam
124	Late Containment Failure Due to Non-condensibles 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)
  - H2 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)
  - H2 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
  - 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 re-evaporation
- 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-condensible gas generation
- Containment failure location
- Containment failure area
- Pool bypass
- High temperature failure of drywell
- Suppression Pool DFs

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	< 1E-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	3.7%
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)	7 psig	7 psig
Time of fission product release	20.2 hr	27.5 hr
CsI release fraction @ 72 hours	<1.E-7	4.8%
Probability of RHR recovery in time window	N/A	11%
Probability of eventual DW head failure w/o CHR	2%	100%
With Water Addition to Containment		
ΔP (Drywell-Wetwell)	14 psig	14 psig
Time of fission product release	31.1 hr	35.0 hr
CsI release fraction @ 72 hours	<1.E-7	3.7E-7
Probability of RHR recovery in time window	N/A	4%
Probability of eventual DW head failure w/o CHR	5%	100%

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)	2.08E-8	5.25E-10	1.09E-7	1.10E-9	1.10E-12
Without COPS	0.0	2.13E-8	0.0	1.10E-7	1.10E-10

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	6.73E-8	6.73E-8	6.73E-8	6.70E-8
Wet CCI	7.11E-9	7.11E-9	7.10E-9	7.07E-9
Dry CCI	3.45E-13	3.45E-12	3.45E-11	3.45E-10
Pedestal Condition				
No Ped Failure	7.41E-8	7.41E-8	7.40E-8	7.37E-8
Ped Failure	1.06E-10	1.06E-10	1.06E-10	1.06E-10
FP Release Mode				
COPS	7.58E-9	7.58E-9	7.57E-9	7.51E-9
DW Head	3.91E-10	3.91E-10	3.91E-10	3.89E-10
Pen. Overtemperature	3.60E-11	3.91E-11	6.98E-11	3.77E-10

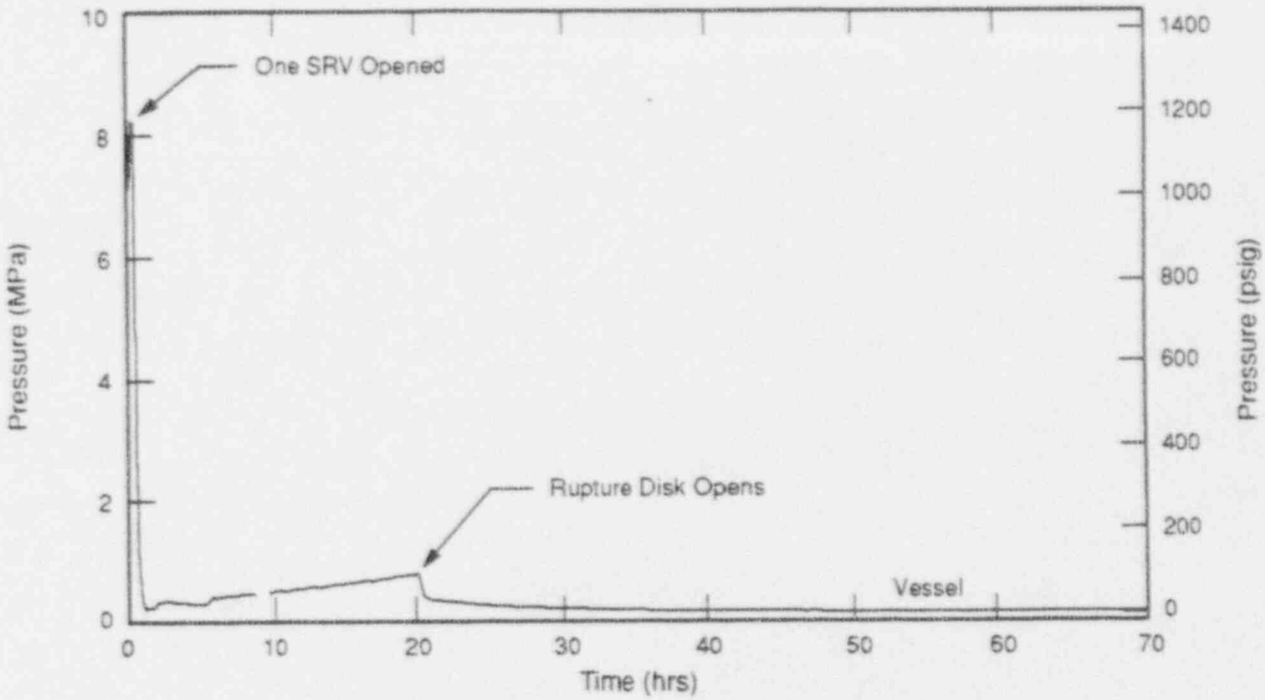


Figure 19E.2-2A LCLP-PF-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: VESSEL PRESSURE

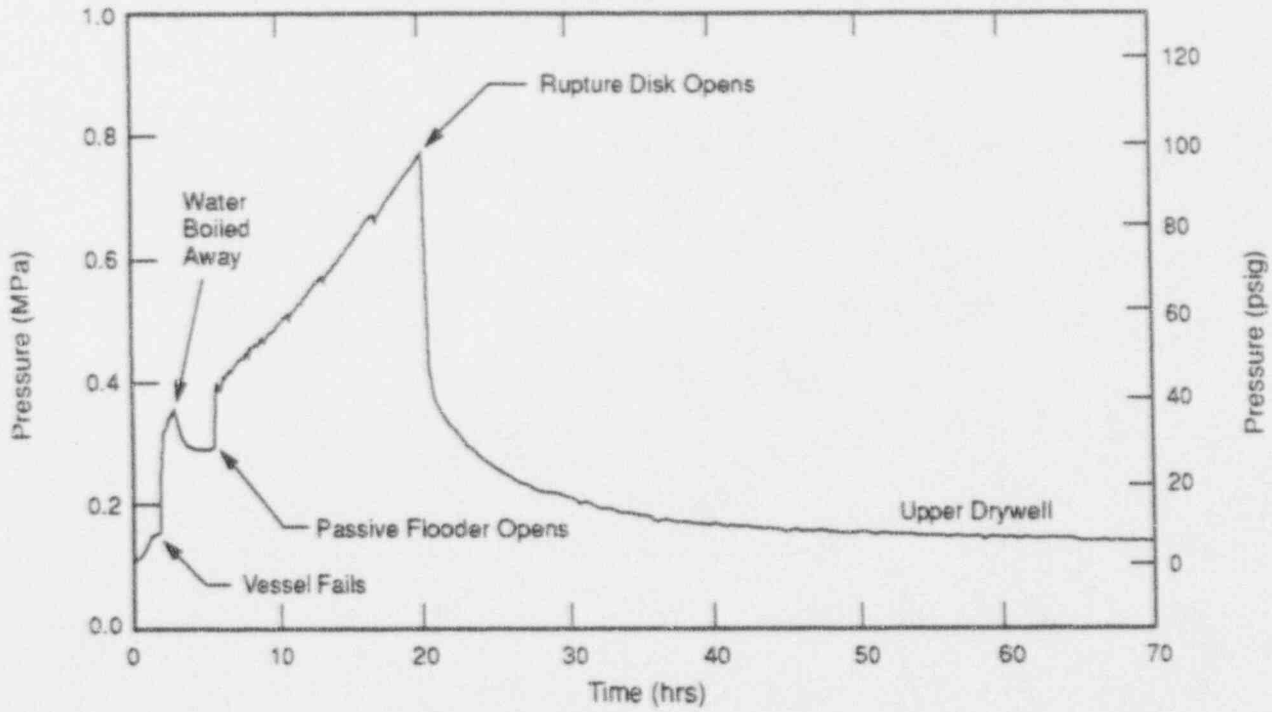


Figure 19E.2-2B LCLP-PF-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: UO_2 TEMPERATURE

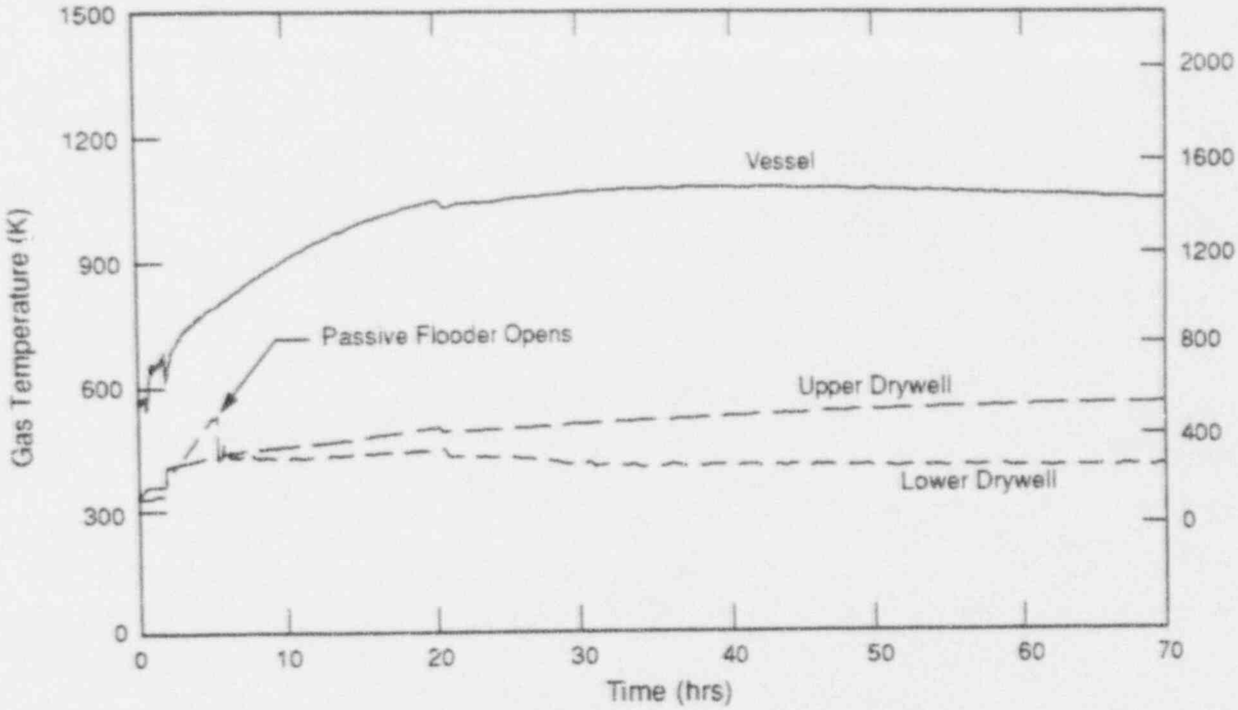


Figure 19E.2-2C LCLP-PF-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: GAS TEMPERATURE

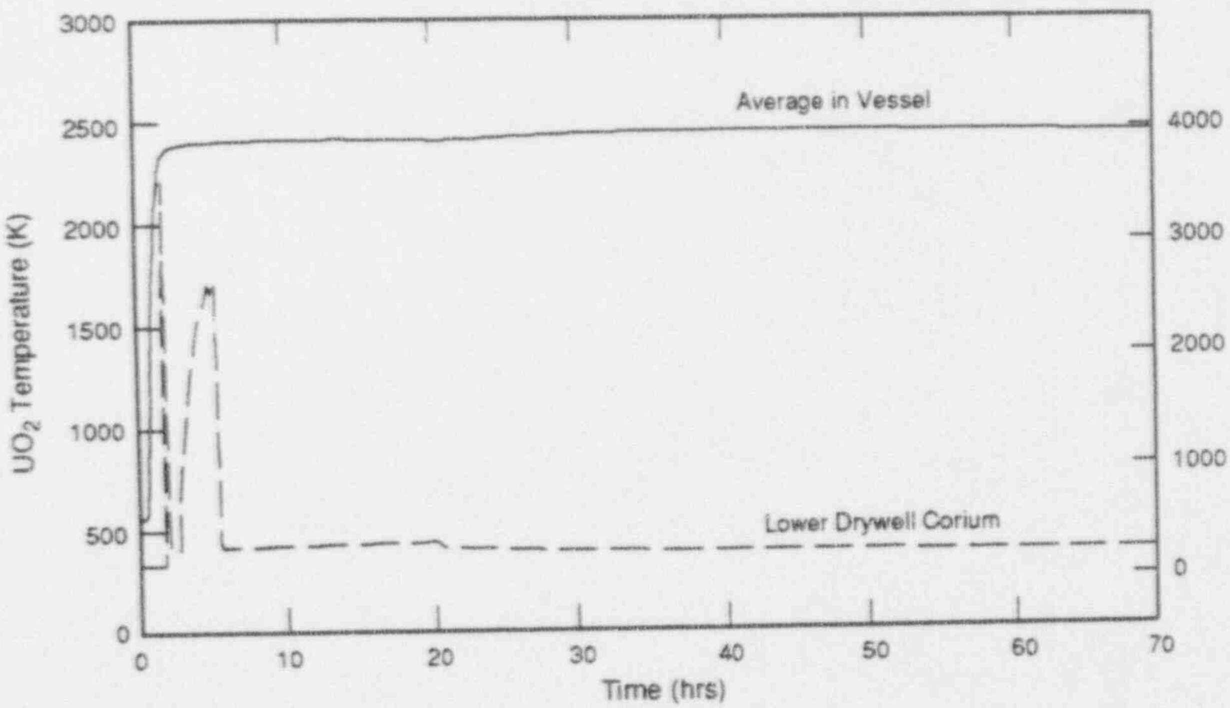


Figure 19E.2-2D LCLP-PF-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: UO_2 TEMPERATURE

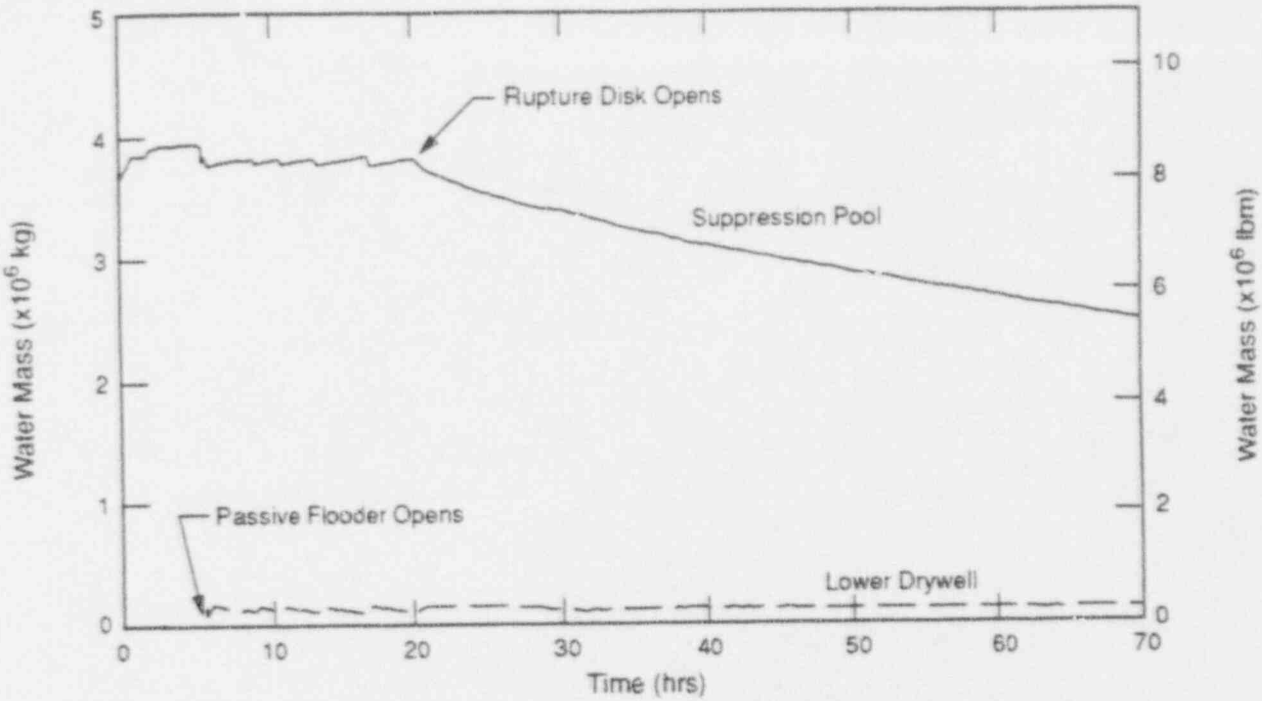


Figure 19E.2-2E LCLP-PF-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: VESSEL PRESSURE

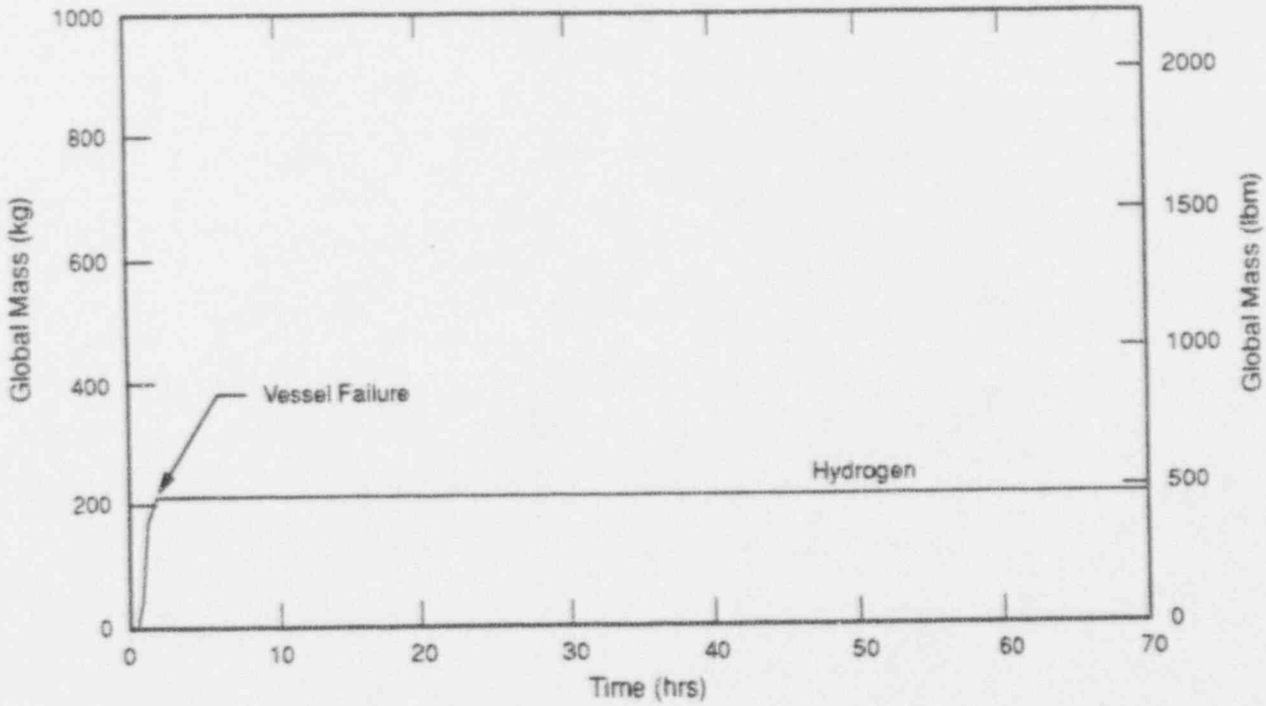


Figure 19E.2-2F LCLP-PF-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: MASS OF NON-CONDENSIBLES

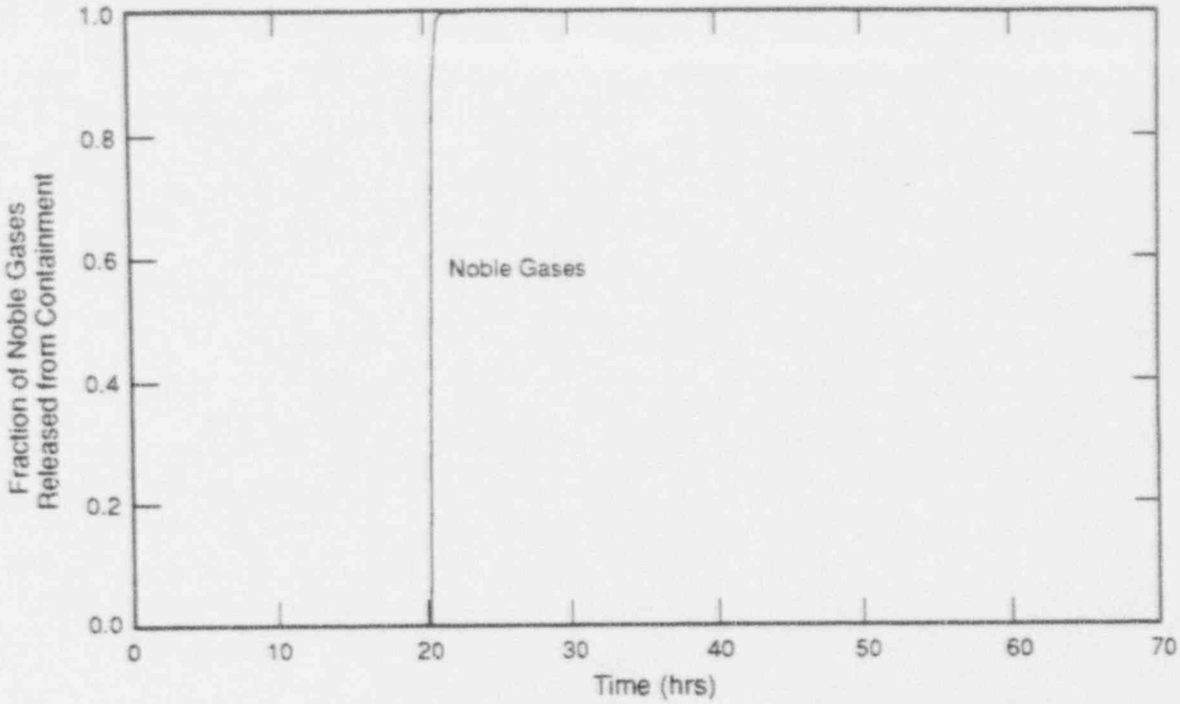


Figure 19E.2-2G LCLP-PF-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: NOBLE GASES

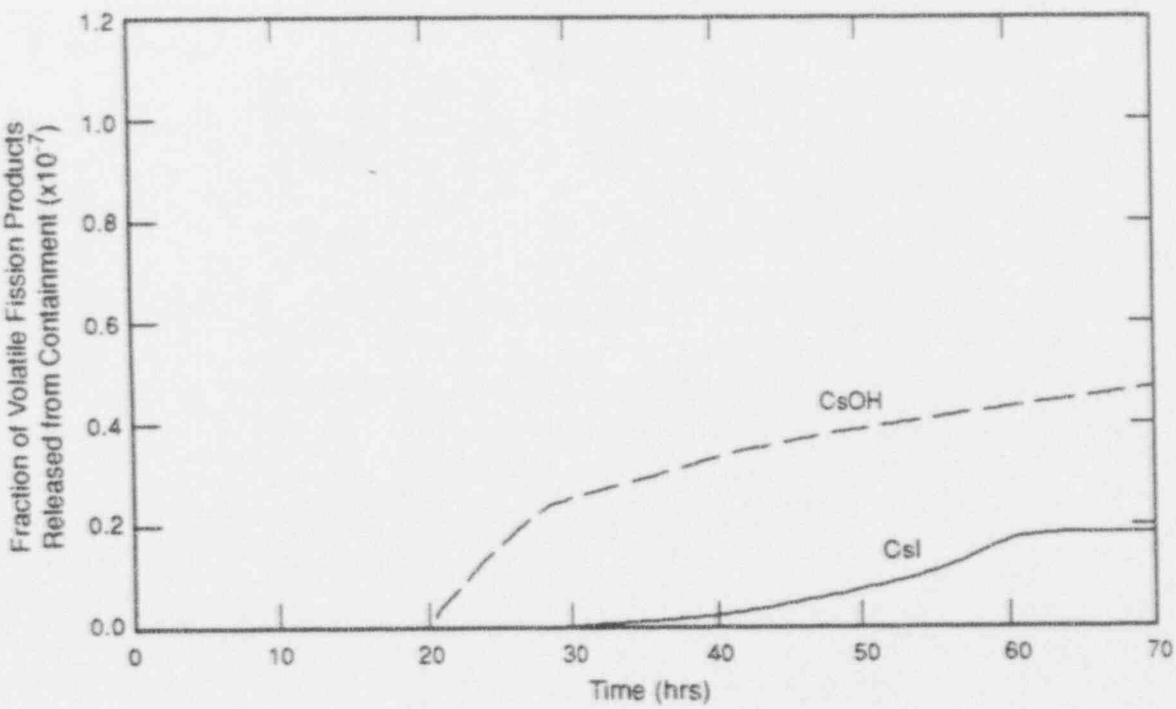


Figure 19E.2-2H LCLP-PF-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: VOLATILE FISSION PRODUCTS

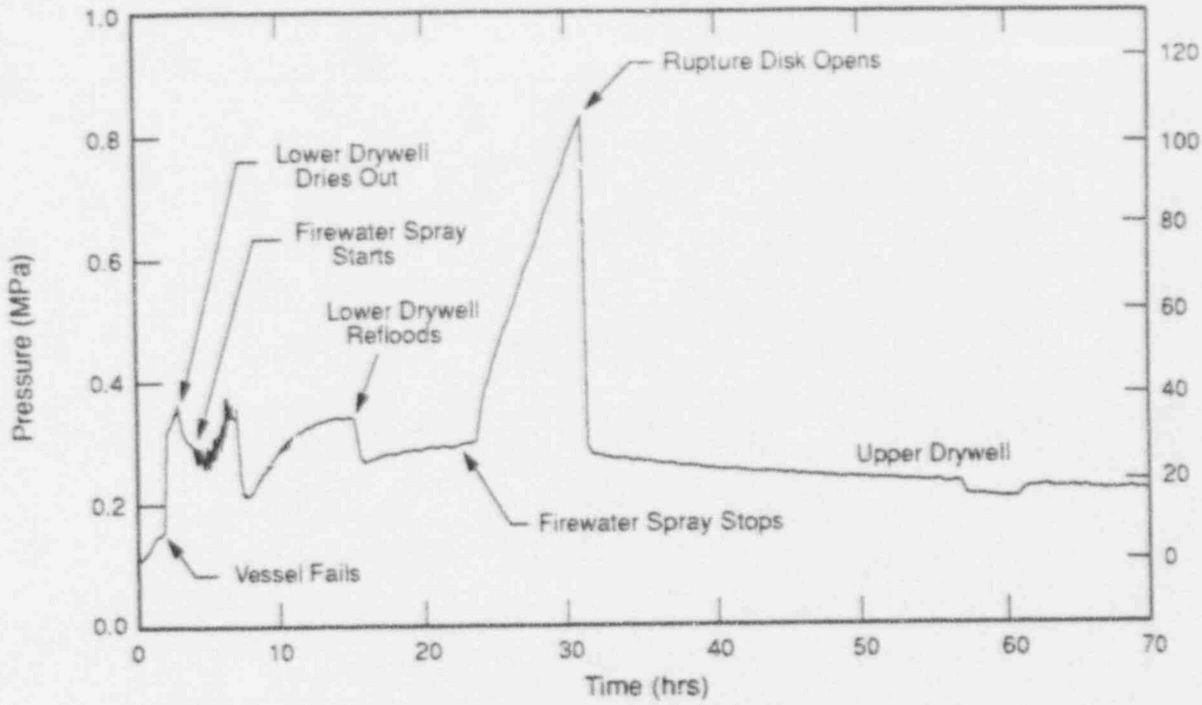


Figure 19E.2-3A LCLP-FS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, FIREWATER SPRAY OPERATES AND RUPTURE DISK OPENS: DRYWELL PRESSURE

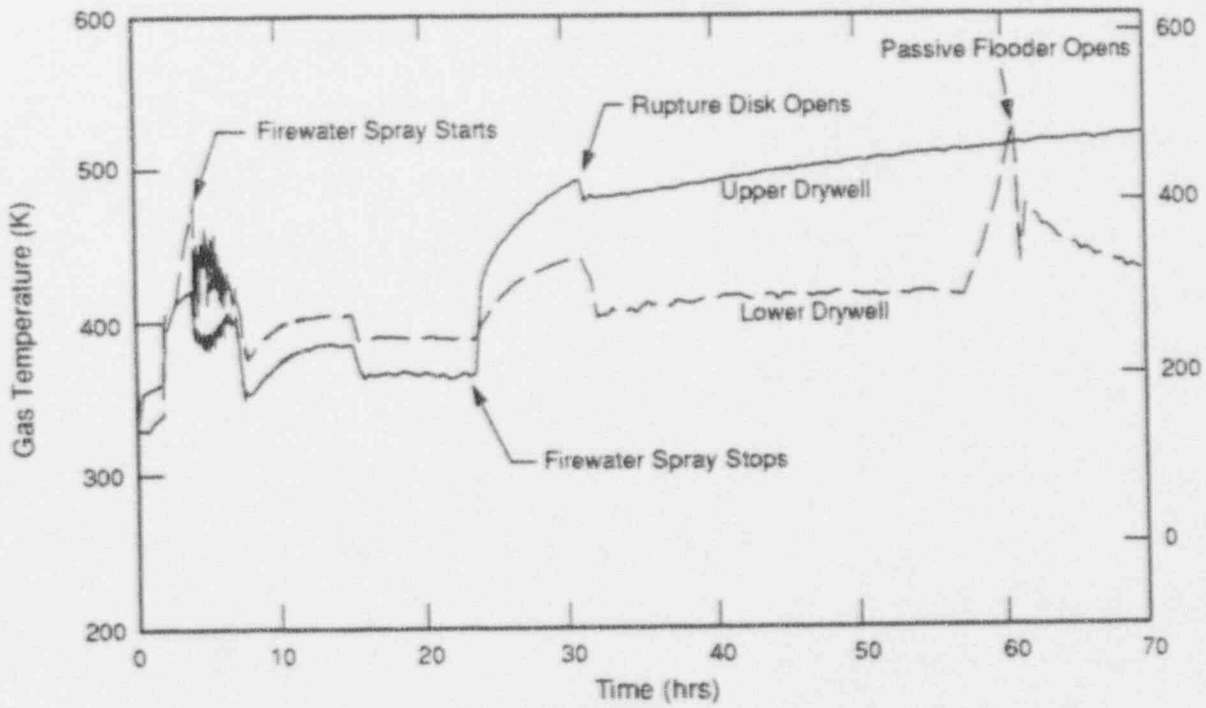


Figure 19E.2-3B LCLP-FS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, FIREWATER SPRAY OPERATES AND RUPTURE DISK OPENS: GAS TEMPERATURE

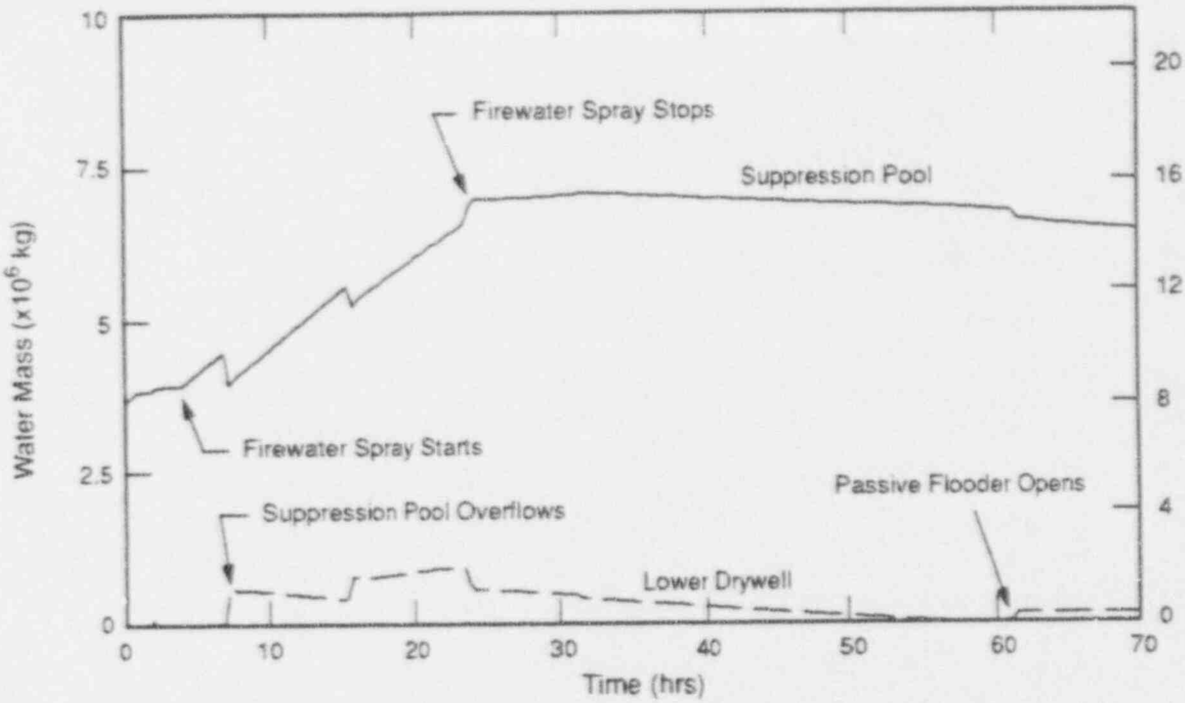


Figure 19E.2-3C LCLP-FS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, FIREWATER SPRAY OPERATES AND RUPTURE DISK OPENS: WATER MASS

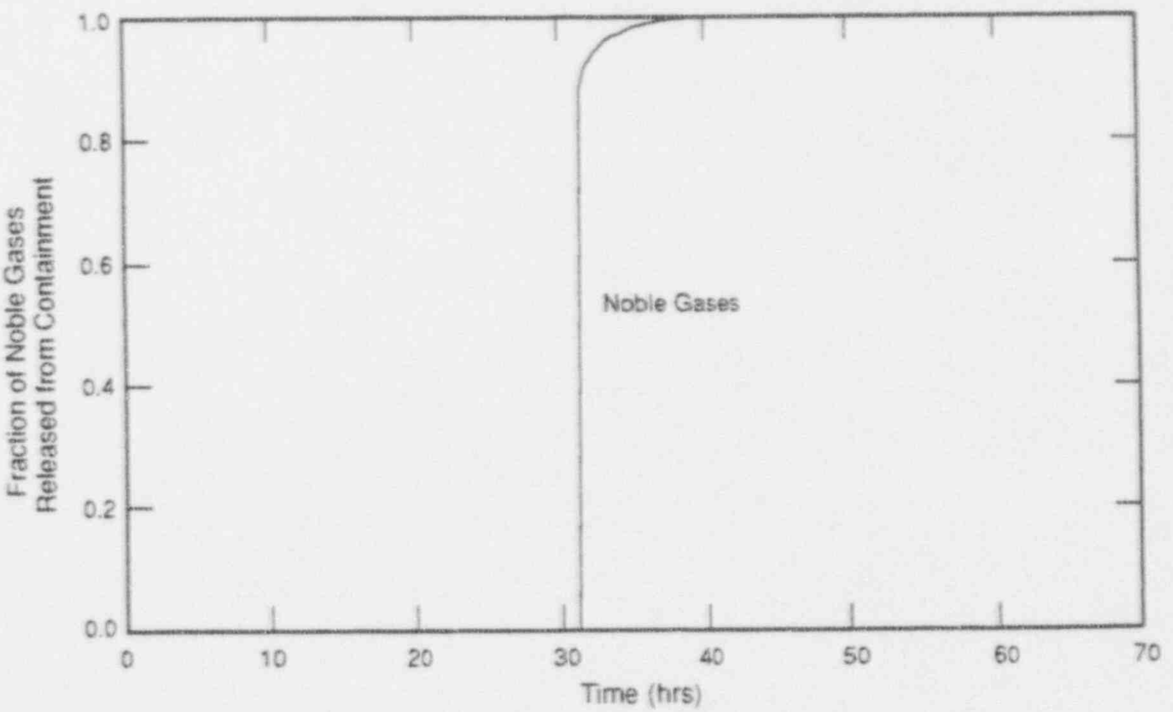


Figure 19E.2-3D LCLP-FS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, FIREWATER SPRAY OPERATES AND RUPTURE DISK OPENS: NOBLE GAS

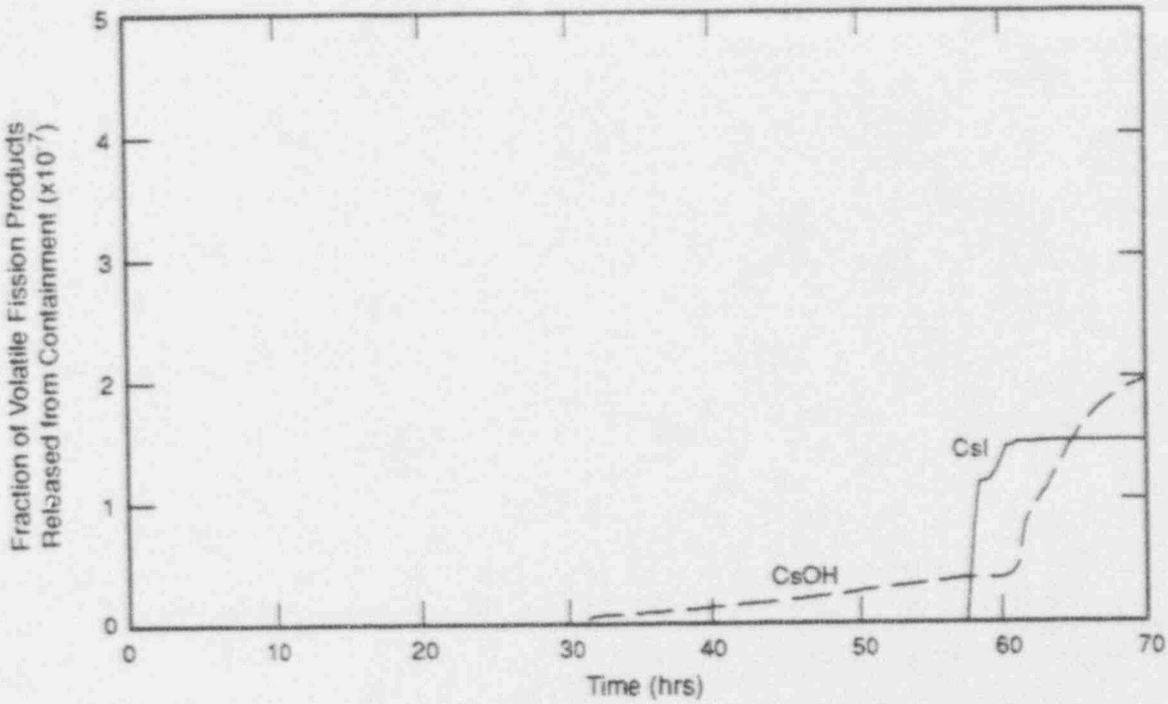


Figure 19E.2-3E LCLP-FS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT LOW PRESSURE, FIREWATER SPRAY OPERATES AND RUPTURE DISK OPENS: VOLATILE FISSION PRODUCTS

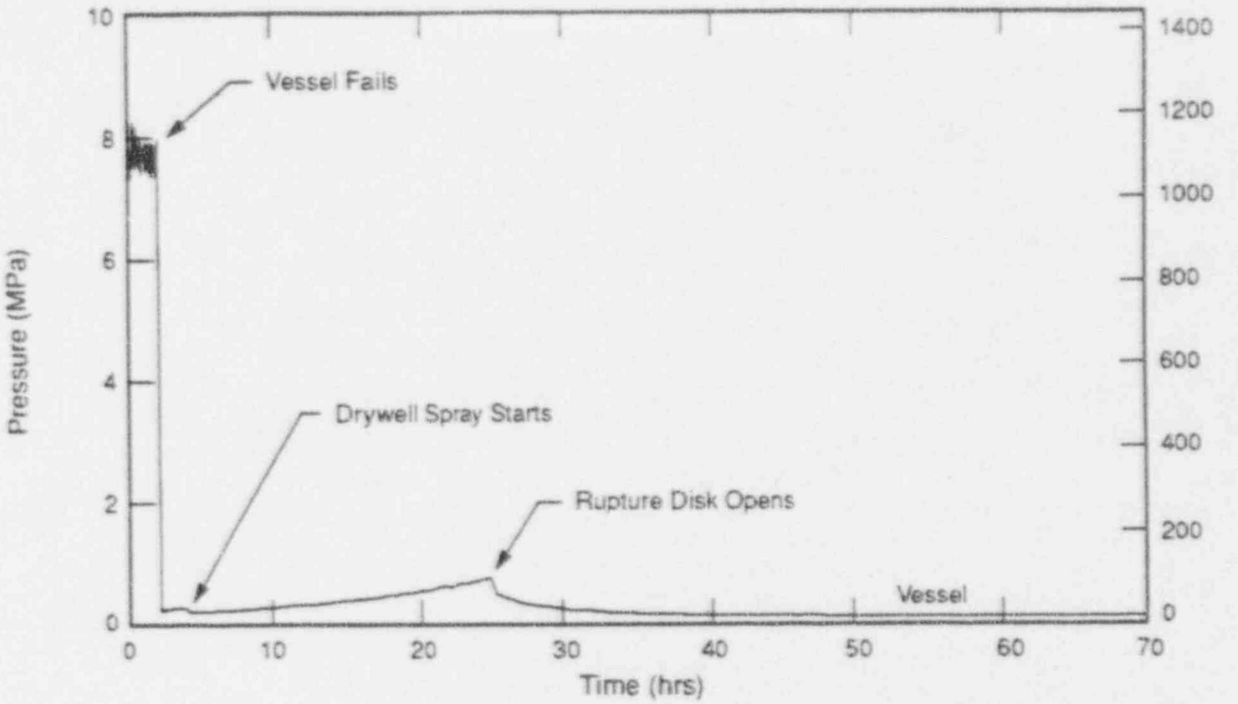


Figure 19E.2-4A LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRYWELL SPRAYS OPERATE, RUPTURE DISK OPENS: VESSEL PRESSURE

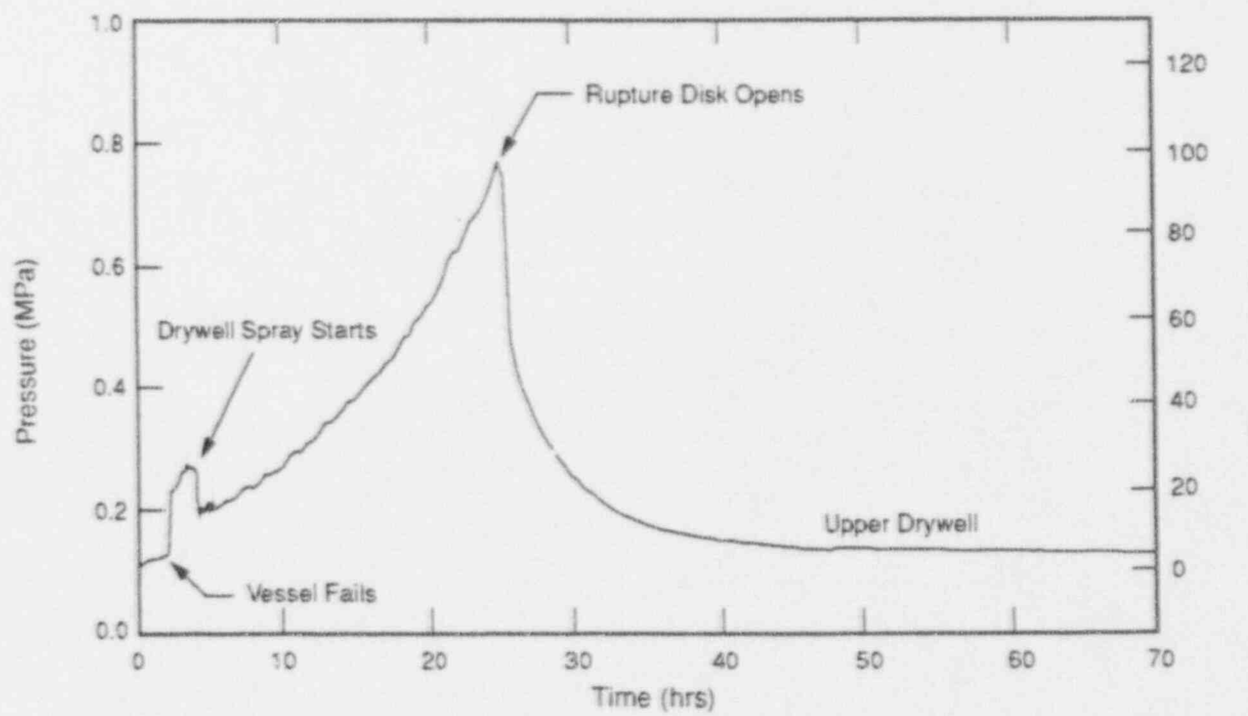


Figure 19E.2-4B LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRYWELL SPRAYS OPERATE, RUPTURE DISK OPENS: DRYWELL PRESSURE

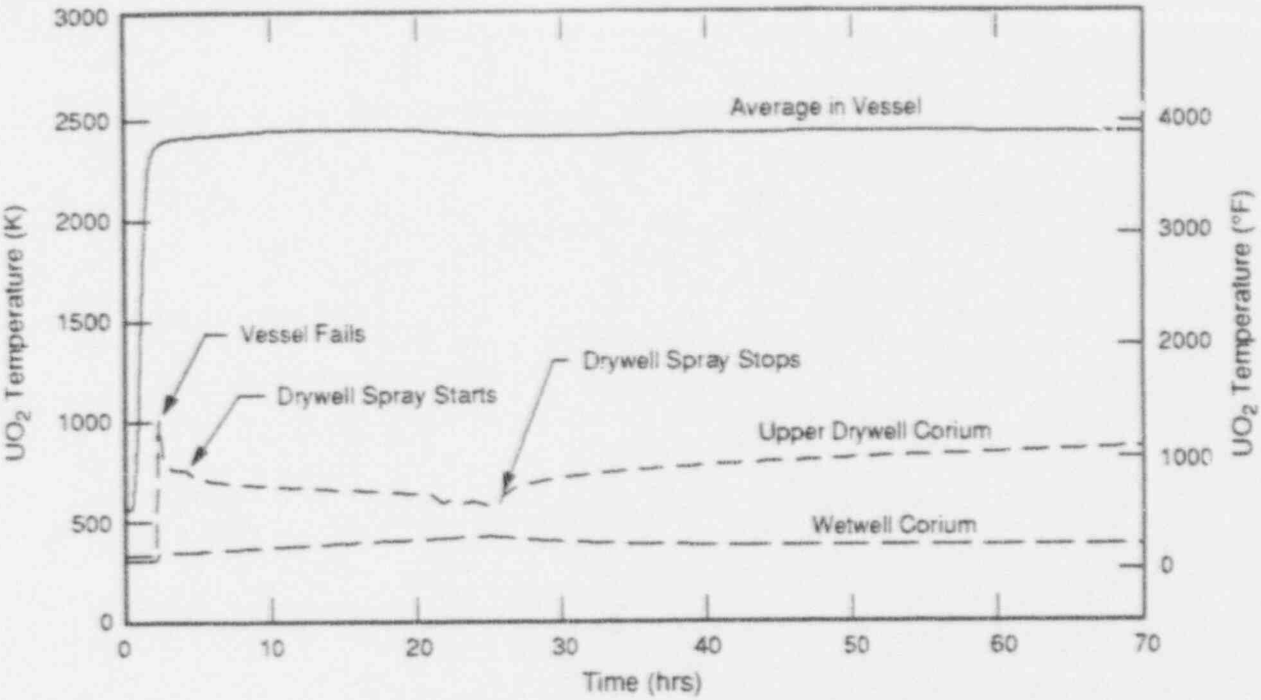


Figure 19E.2-4C LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRYWELL SPRAYS OPERATE, RUPTURE DISK OPENS: UO_2 TEMPERATURE

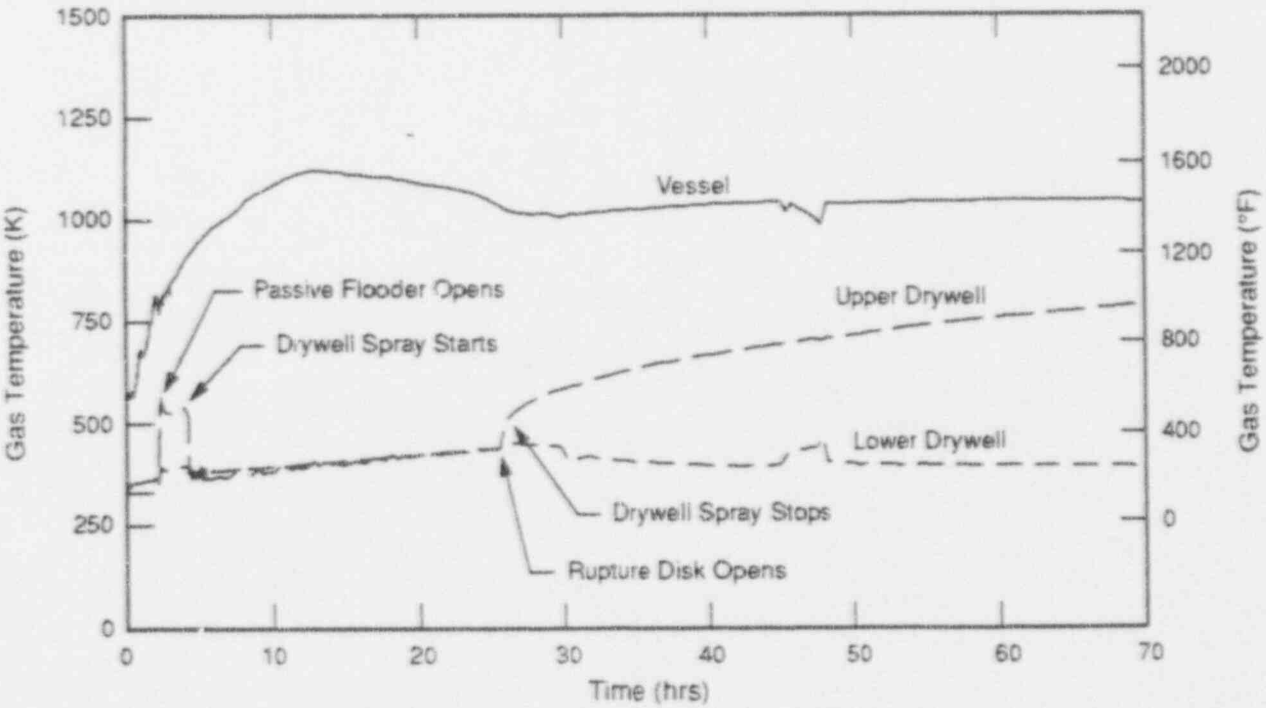


Figure 19E.2-4D LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRYWELL SPRAYS OPERATE, RUPTURE DISK OPENS: GAS TEMPERATURE

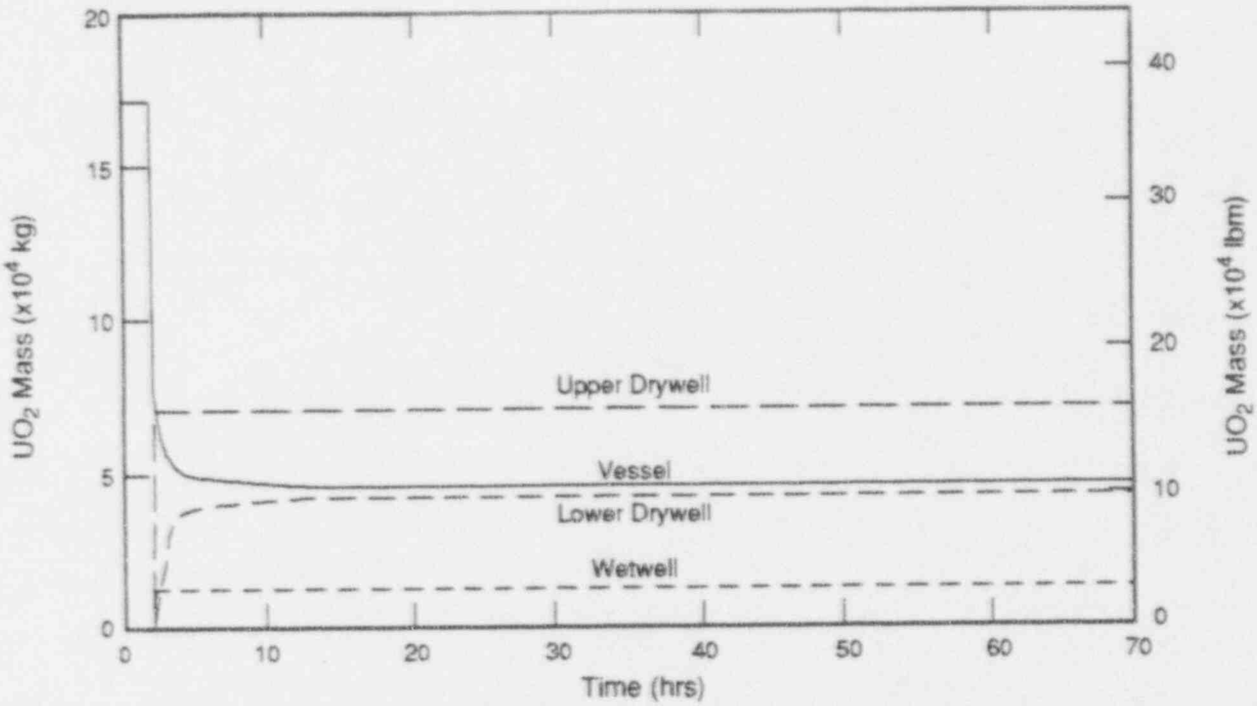


Figure 19E.2-4E LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRYWELL SPRAYS OPERATE, RUPTURE DISK OPENS: UO_2 MASS

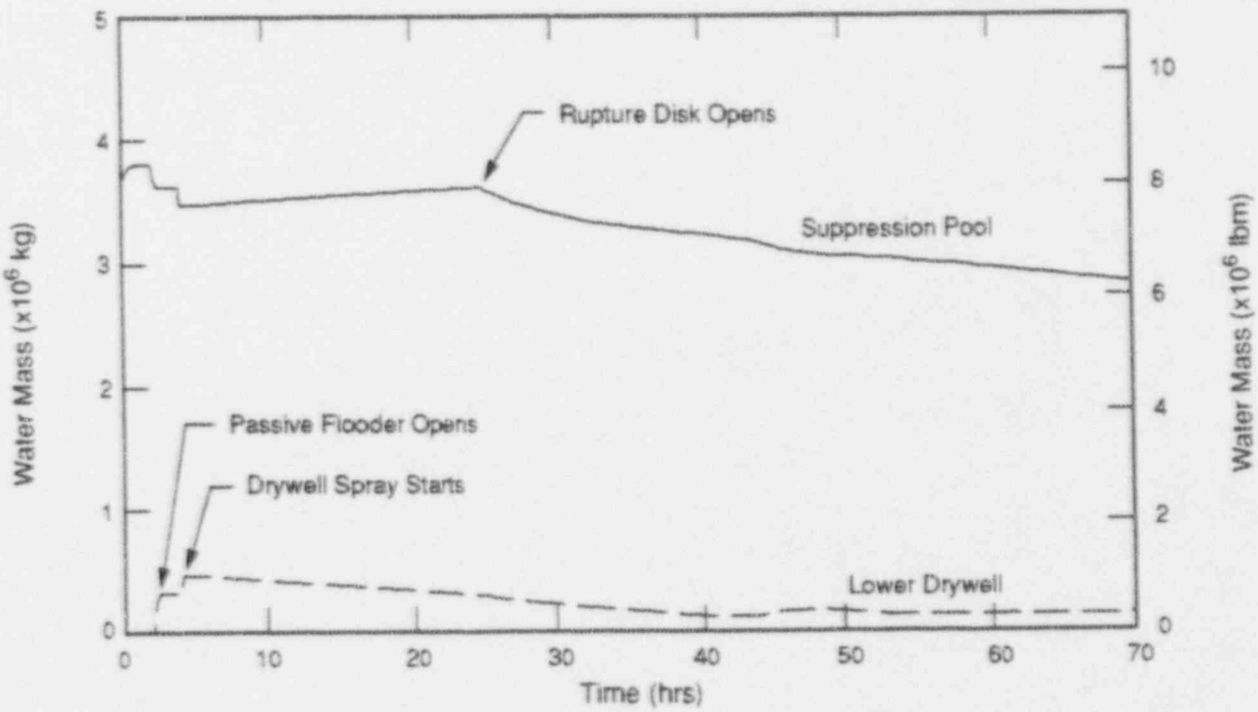


Figure 19E.2-4F LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRYWELL SPRAYS OPERATE, RUPTURE DISK OPENS: WATER MASS

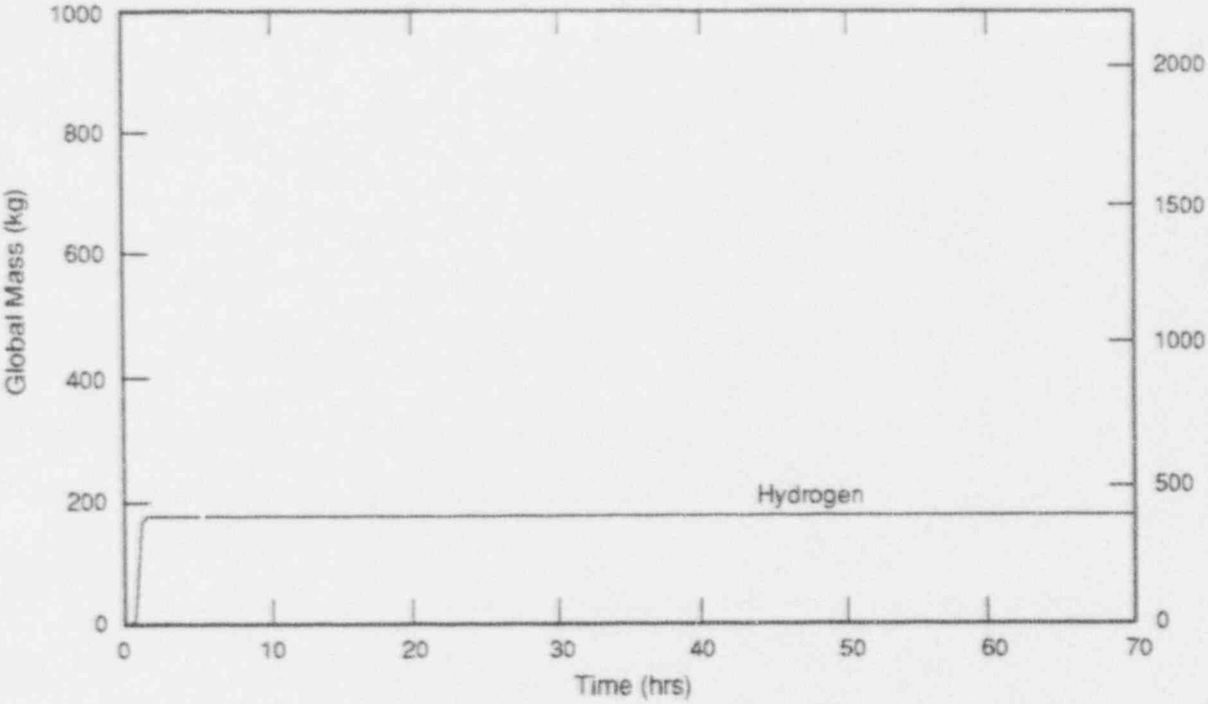


Figure 19E.2-4G LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRY-WELL SPRAYS OPERATE, RUPTURE DISK OPENS: GLOBAL MASS

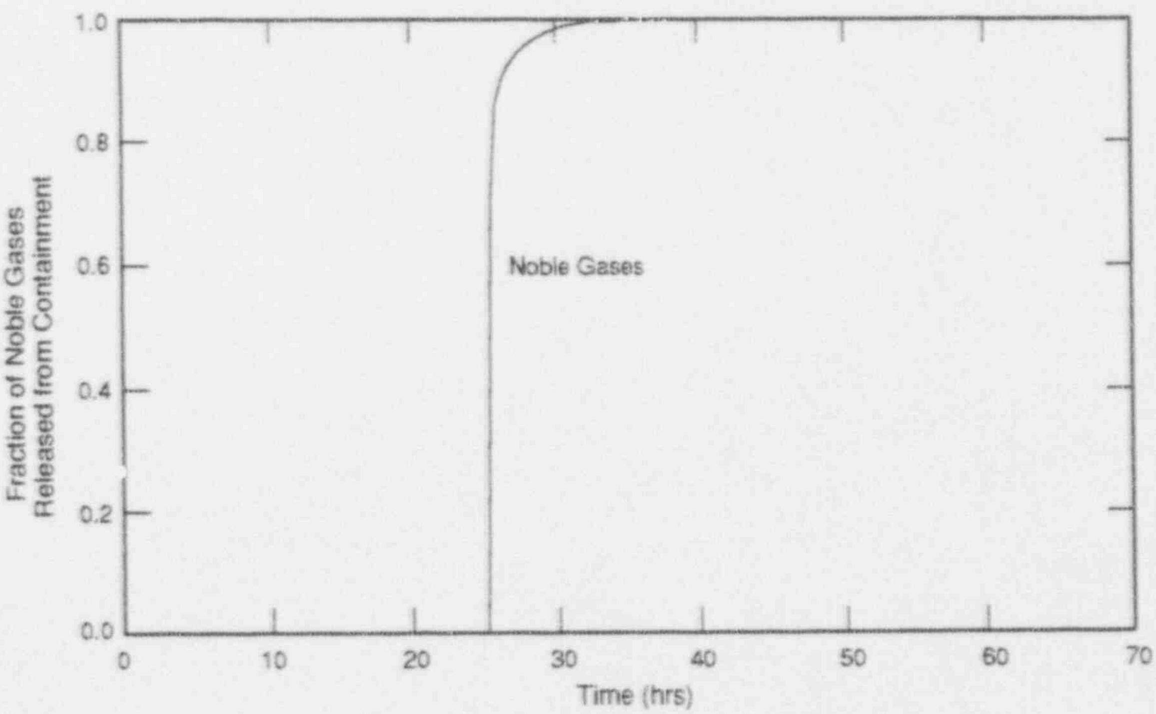


Figure 19E.2-4H LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRY-WELL SPRAYS OPERATE, RUPTURE DISK OPENS: NOBLE GASES

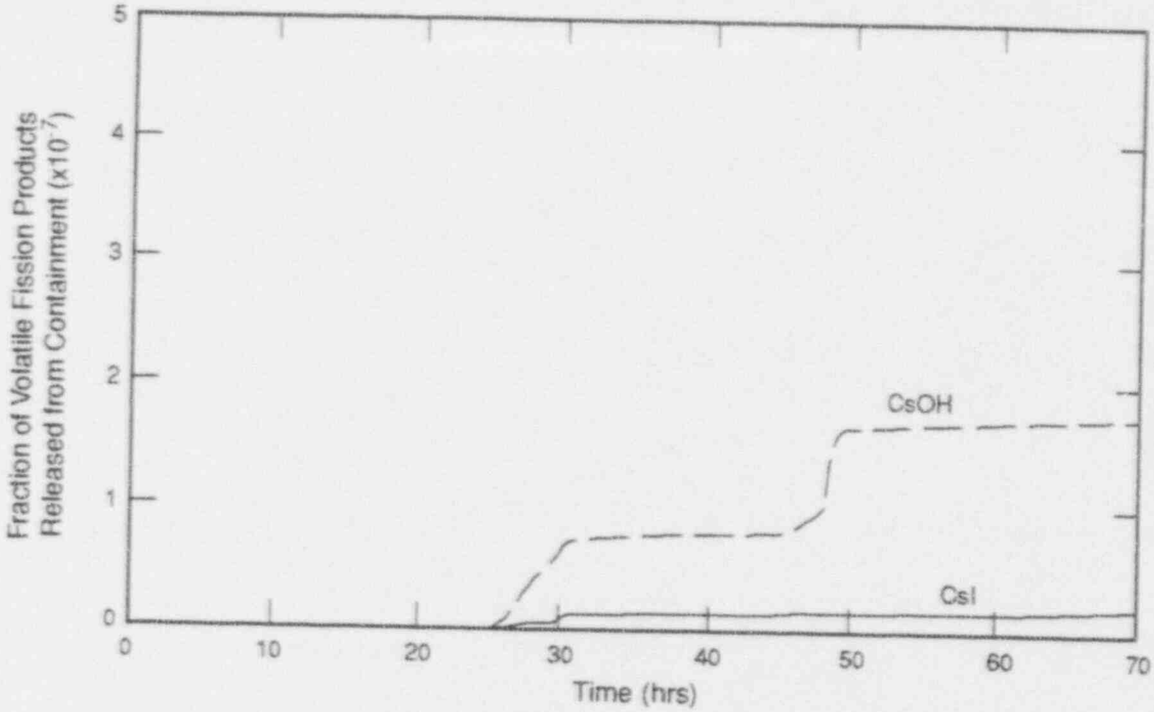


Figure 19E.2-4I LCHP-PS-R-N: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER AND DRY-WELL SPRAYS OPERATE, RUPTURE DISK OPENS: VOLATILES

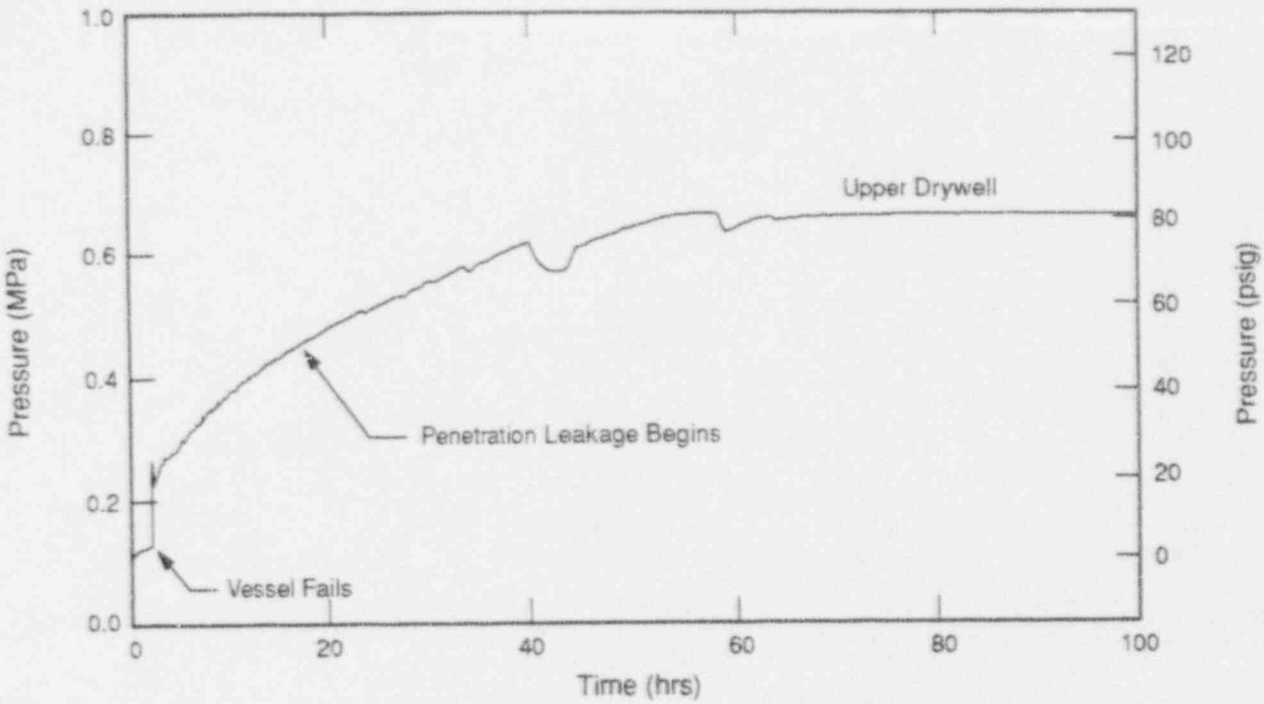


Figure 19E.2-5A LCHP-PF-P-M: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER OPERATES, PENETRATION LEAKAGE: DRYWELL PRESSURE

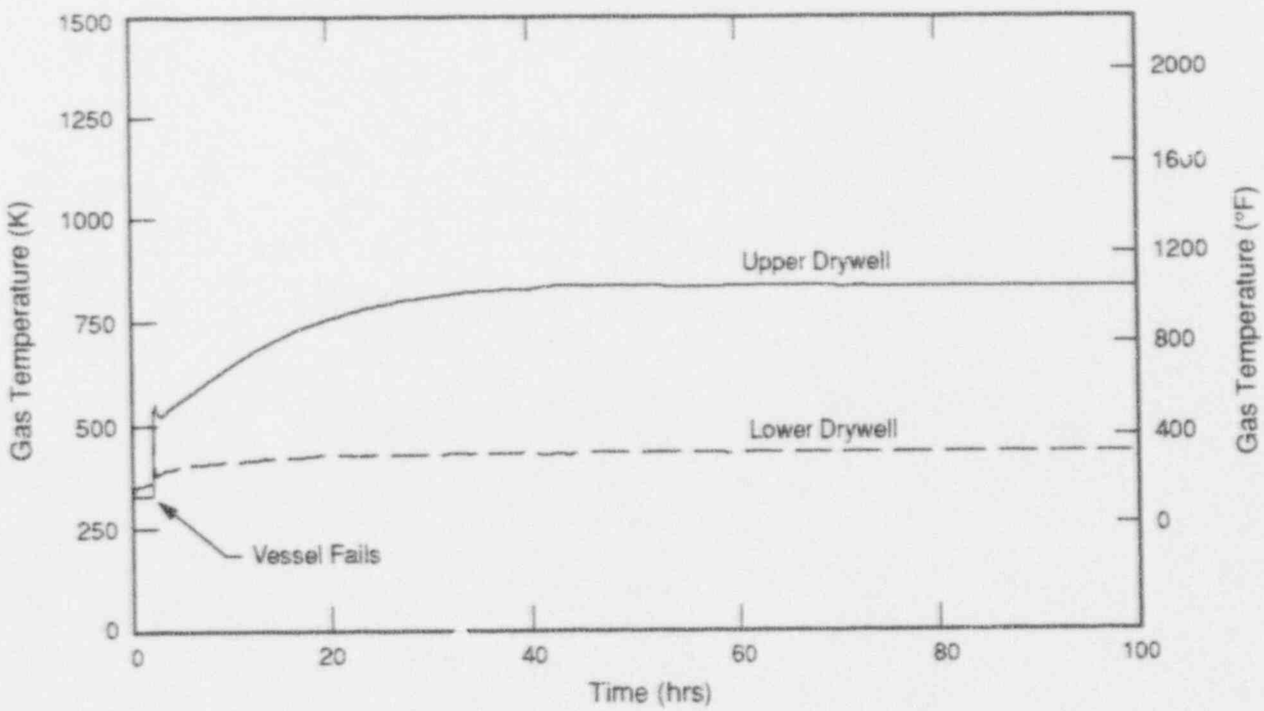


Figure 19E.2-5B LCHP-PF-P-M: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER OPERATES, PENETRATION LEAKAGE: GAS TEMPERATURE

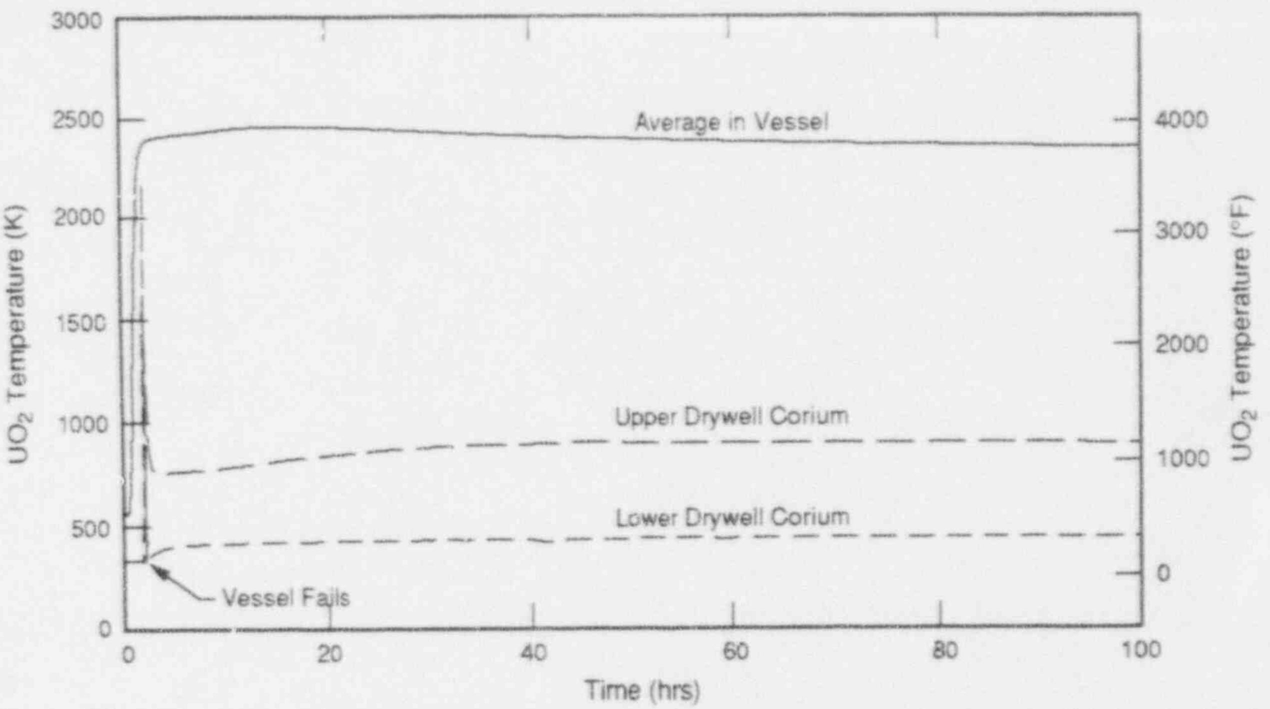


Figure 19E.2-5C LCHP-PF-P-M: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER OPERATES, PENETRATION LEAKAGE: UO_2 TEMPERATURE

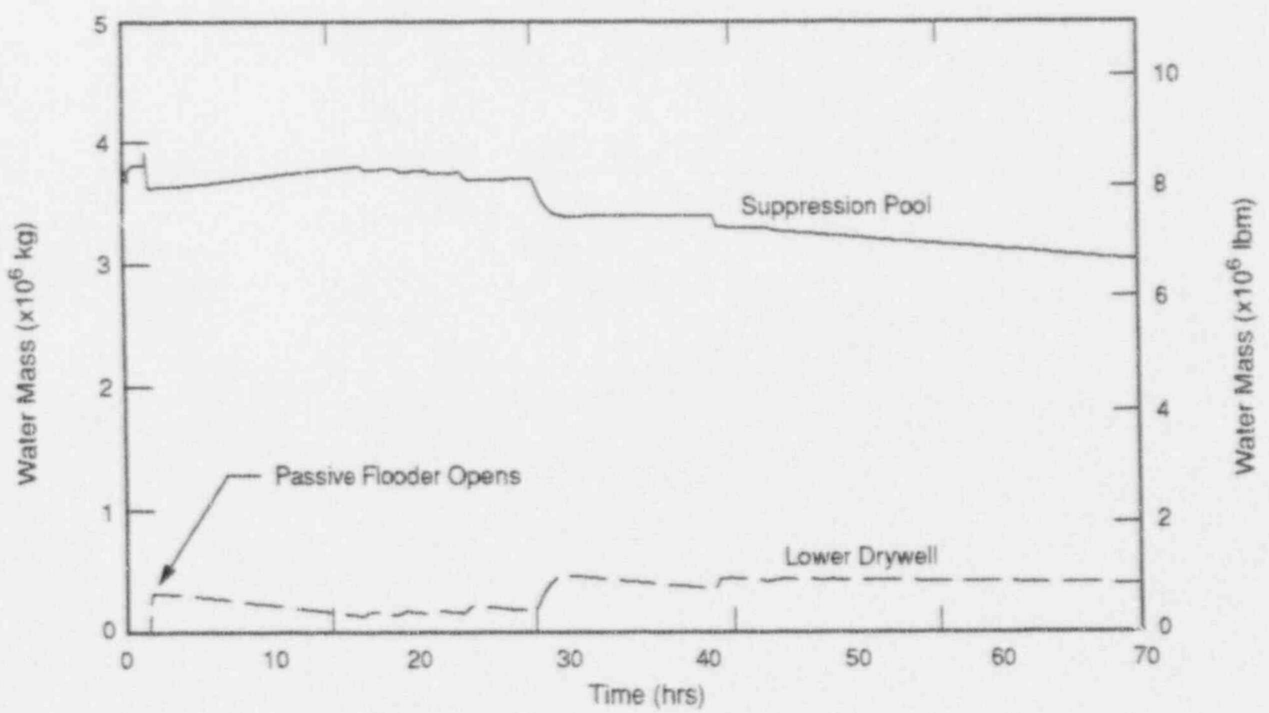


Figure 19E.2-5D LCHP-PF-P-M: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER OPERATES, PENETRATION LEAKAGE: WATER MASS

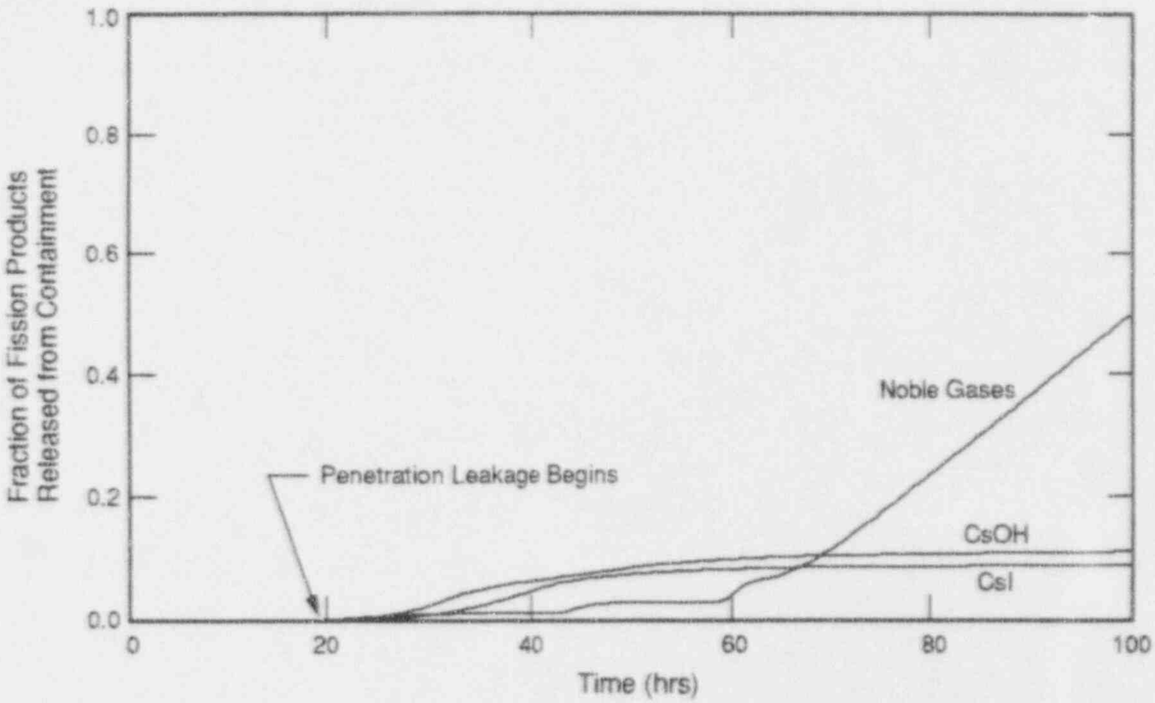


Figure 19E.2-5E **LCHP-PF-P-M: LOSS OF ALL CORE COOLING WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER OPERATES, PENETRATION LEAKAGE: FISSION PRODUCT RELEASE**

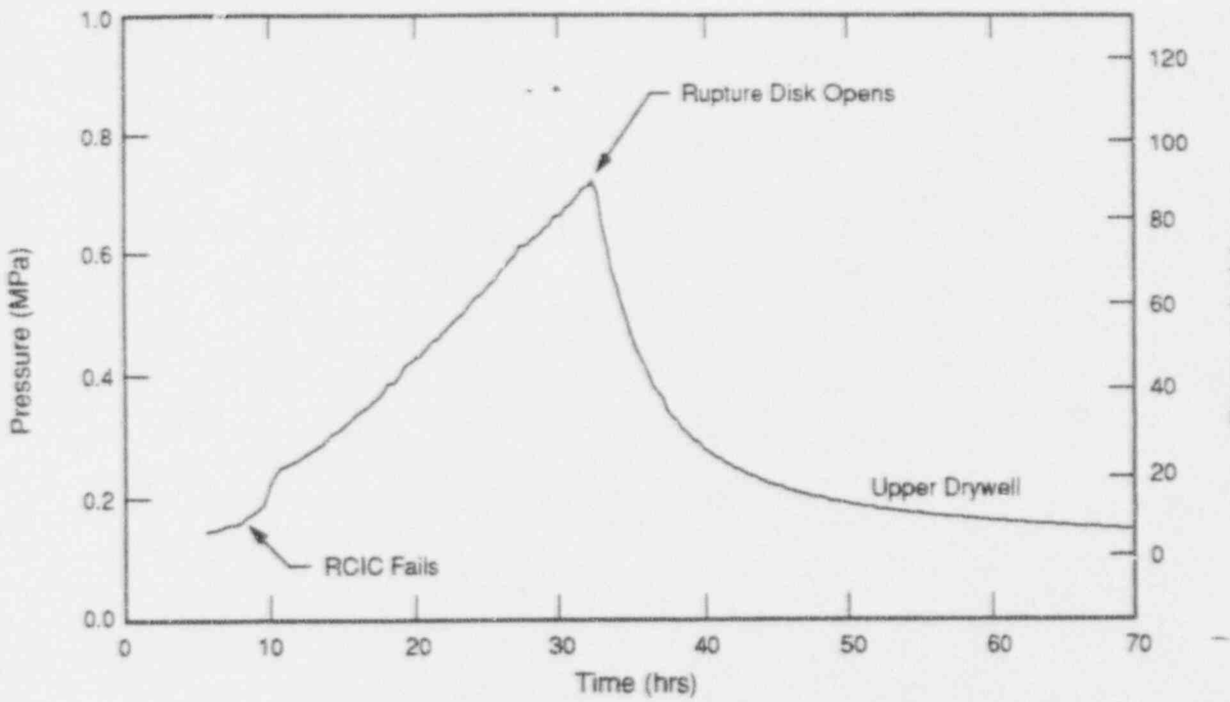


Figure 19E.2-6A SBRC-FA-R-0: STATION BLACKOUT, RCIC RUNS EIGHT HOURS, FIREWATER ADDITION PREVENTS CORE DAMAGE, RUPTURE DISK OPENS: DRYWELL PRESSURE

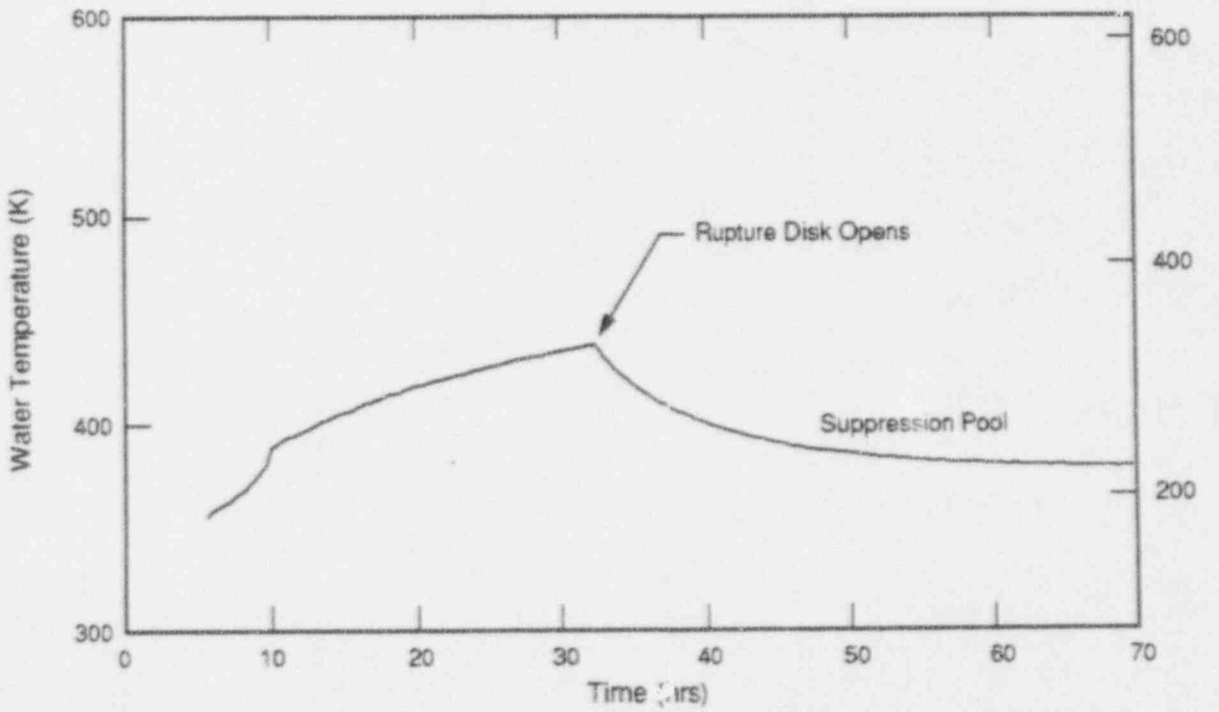


Figure 19E.2-6B SBRC-FA-R-0: STATION BLACKOUT, RCIC RUNS EIGHT HOURS, FIREWATER ADDITION PREVENTS CORE DAMAGE, RUPTURE DISK OPENS: WATER TEMPERATURE

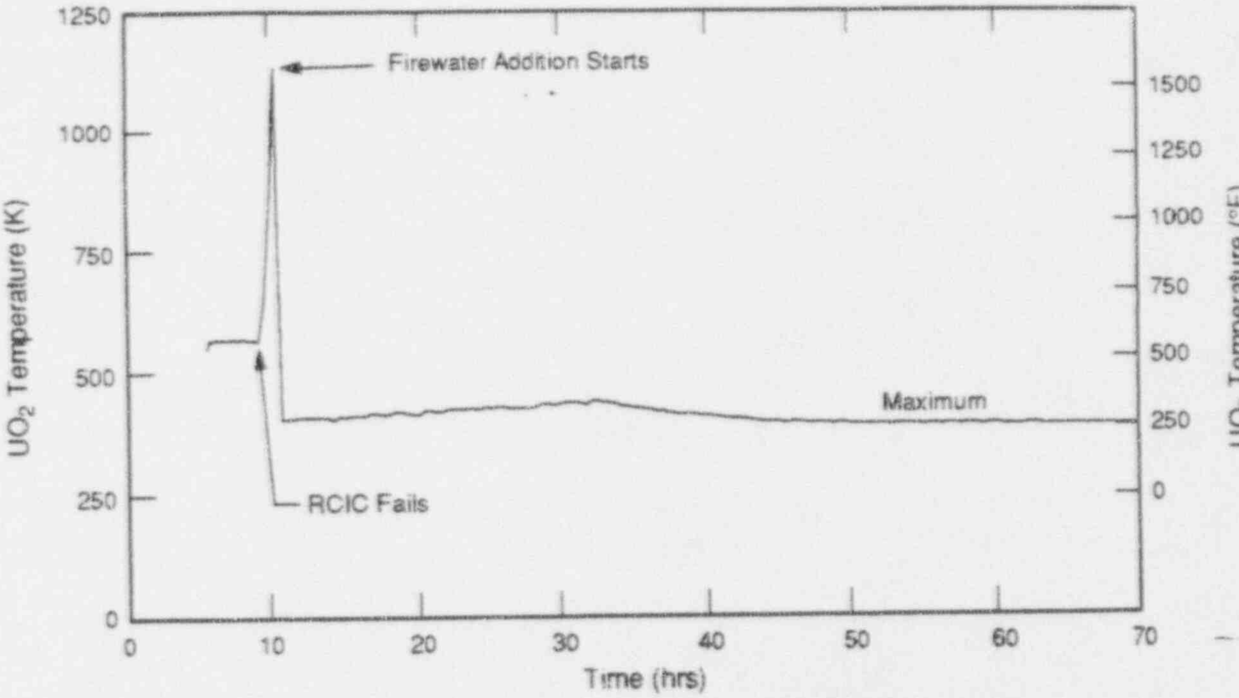


Figure 19E.2-6C SBRC-FA-R-0: STATION BLACKOUT, RCIC RUNS EIGHT HOURS, FIREWATER ADDITION PREVENTS CORE DAMAGE, RUPTURE DISK OPENS: UO_2 TEMPERATURE

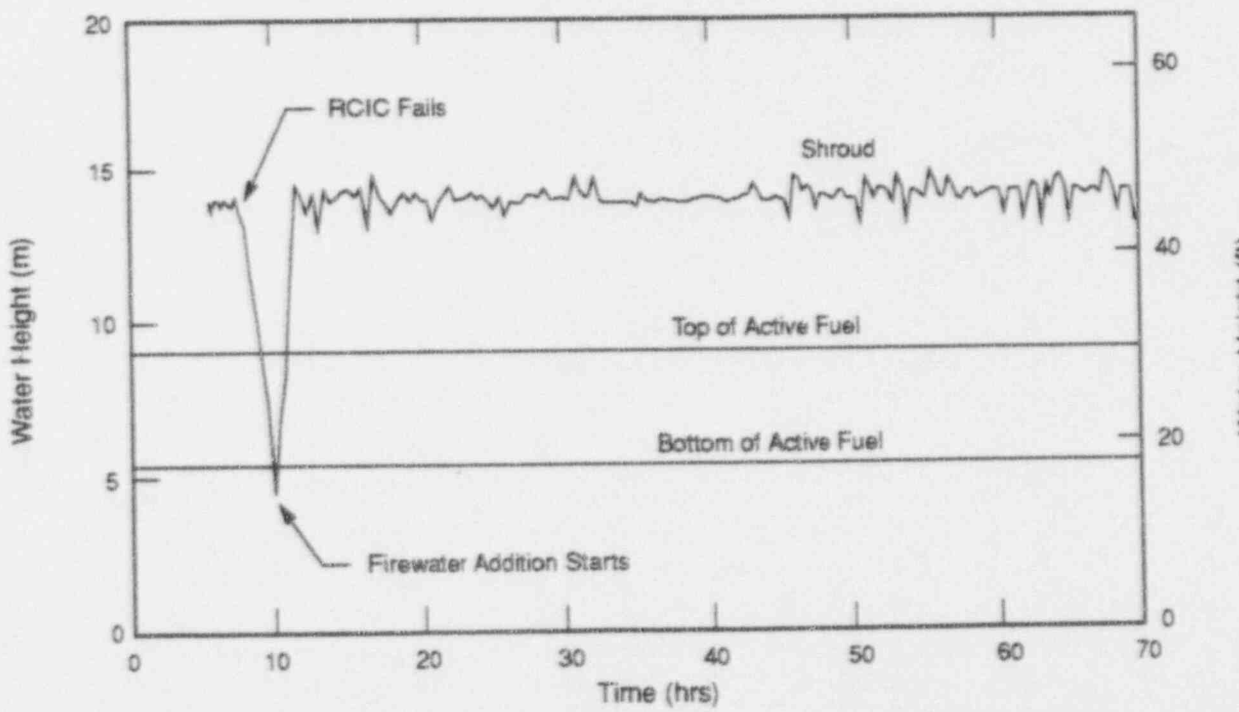


Figure 19E.2-6D SBRC-FA-R-0: STATION BLACKOUT, RCIC RUNS EIGHT HOURS, FIREWATER ADDITION PREVENTS CORE DAMAGE,

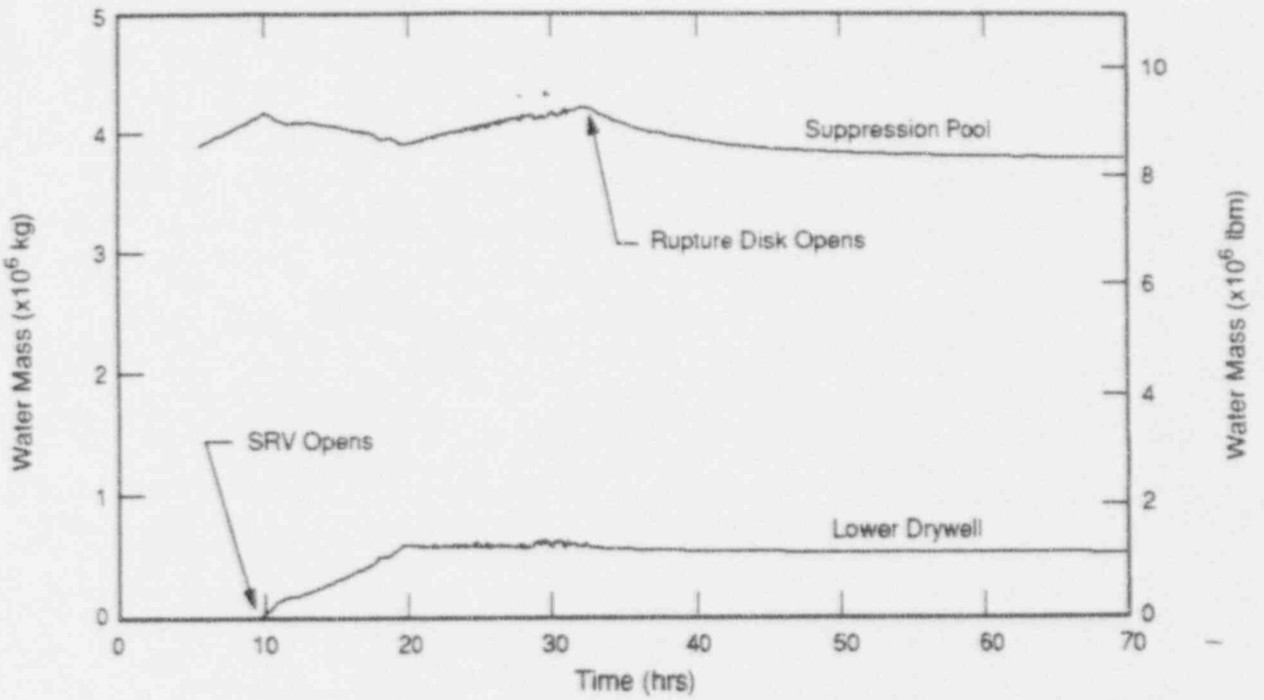


Figure 19E.2-6E SBRC-FA-R-0: STATION BLACKOUT, RCIC RUNS EIGHT HOURS, FIREWATER ADDITION PREVENTS CORE DAMAGE, RUPTURE DISK OPENS: WATER MASS

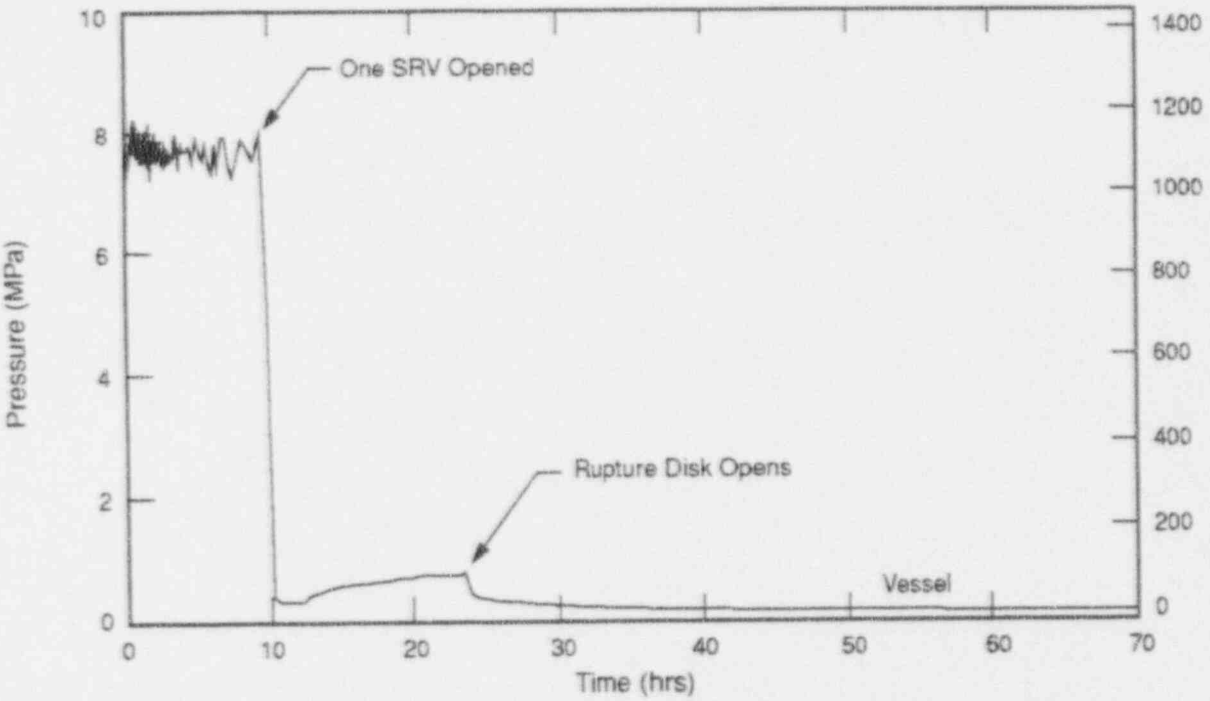


Figure 19E.2-7A SBRC-PF-R-N: STATION BLACKOUT WITH RCIC OPERATING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: VESSEL PRESSURE

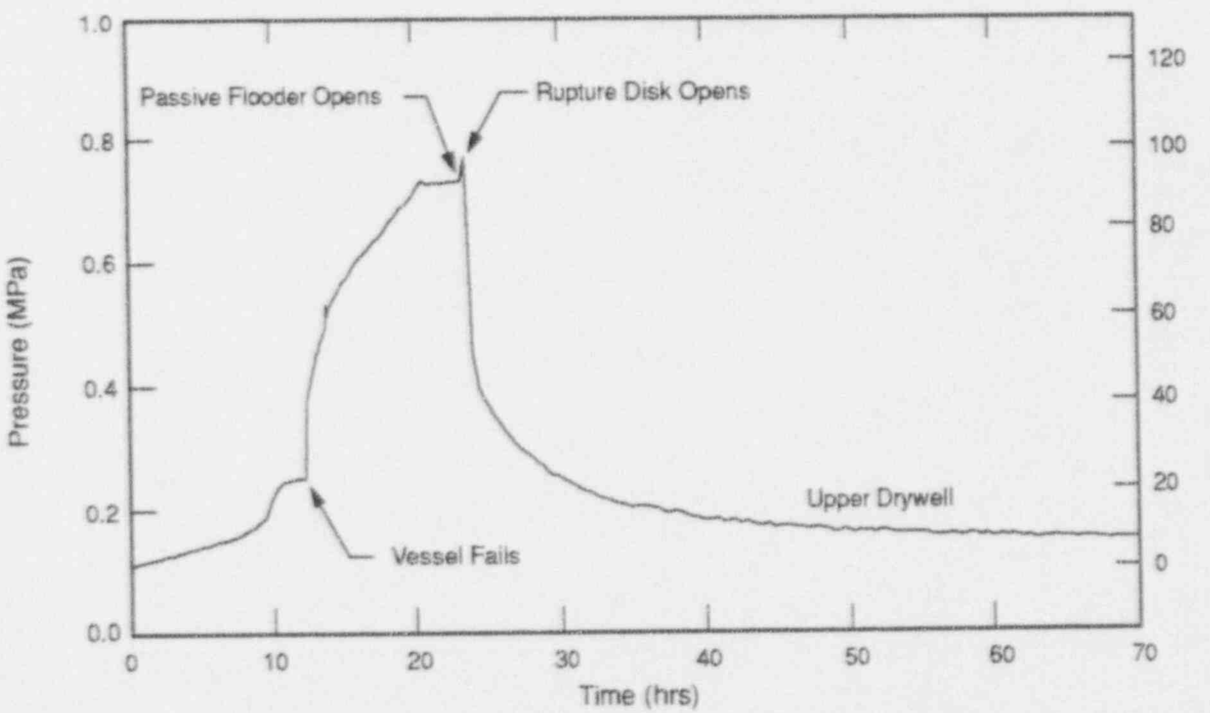


Figure 19E.2-7B SBRC-PF-R-N: STATION BLACKOUT WITH RCIC OPERATING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: DRYWELL PRESSURE

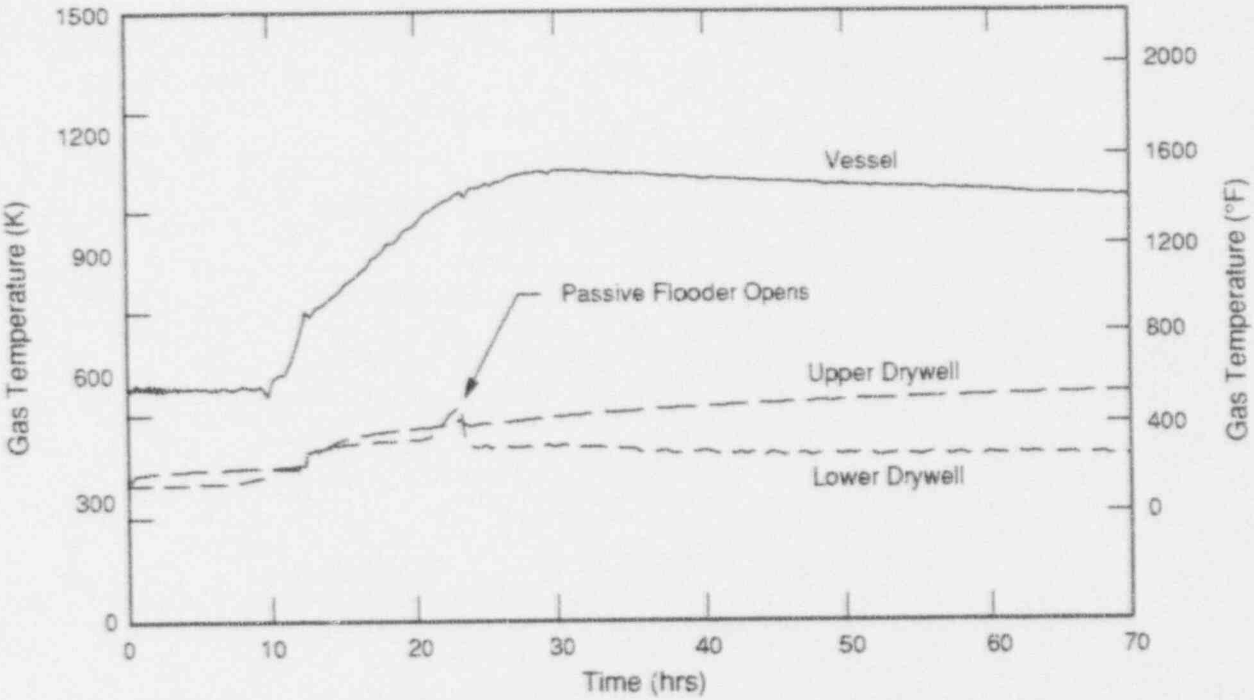


Figure 19E.2-7C SBRC-PF-R-N: STATION BLACKOUT WITH RCIC OPERATING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: GAS TEMPERATURE

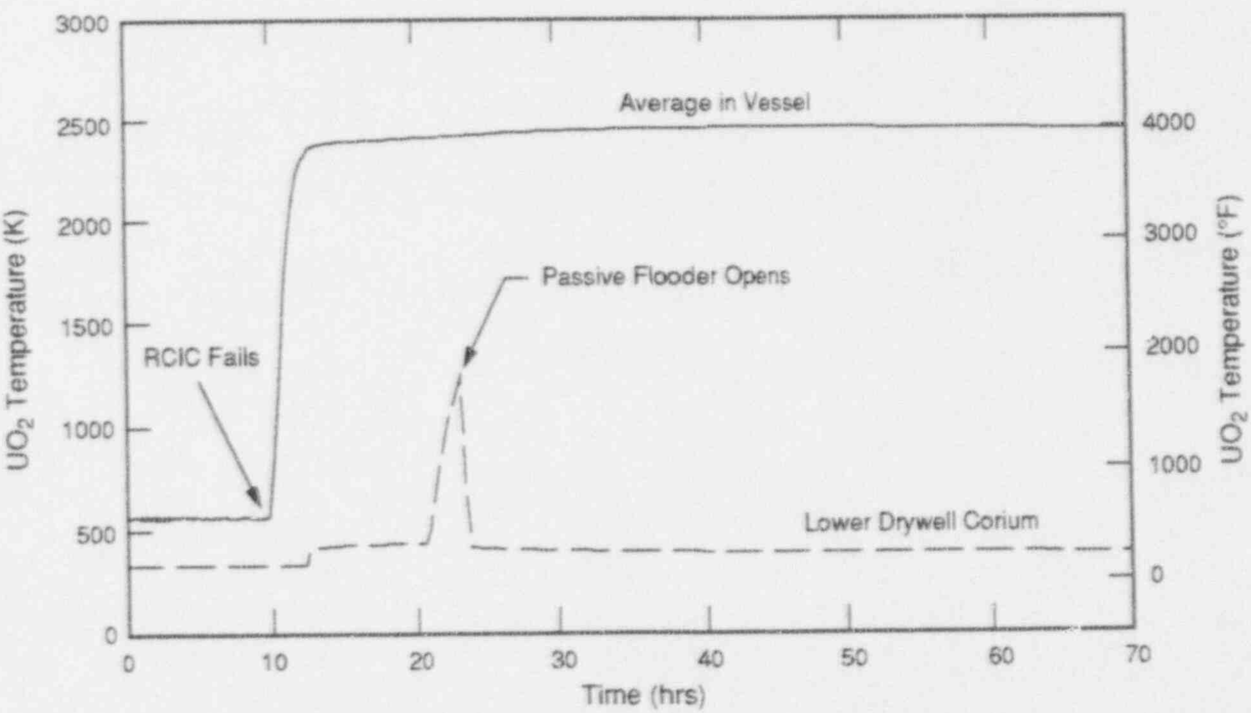


Figure 19E.2-7D SBRC-PF-R-N: STATION BLACKOUT WITH RCIC OPERATING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: UO_2 TEMPERATURE

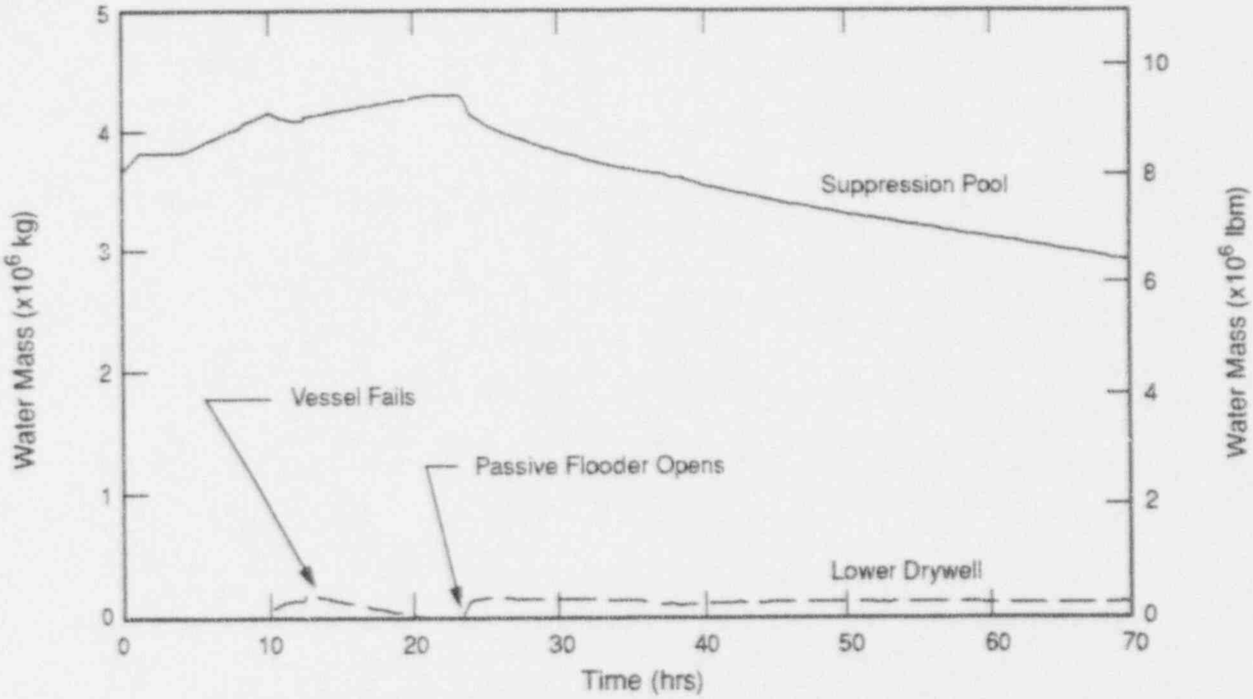


Figure 19E.2-7E SBRC-PF-R-N: STATION BLACKOUT WITH RCIC OPERATING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: WATER MASS

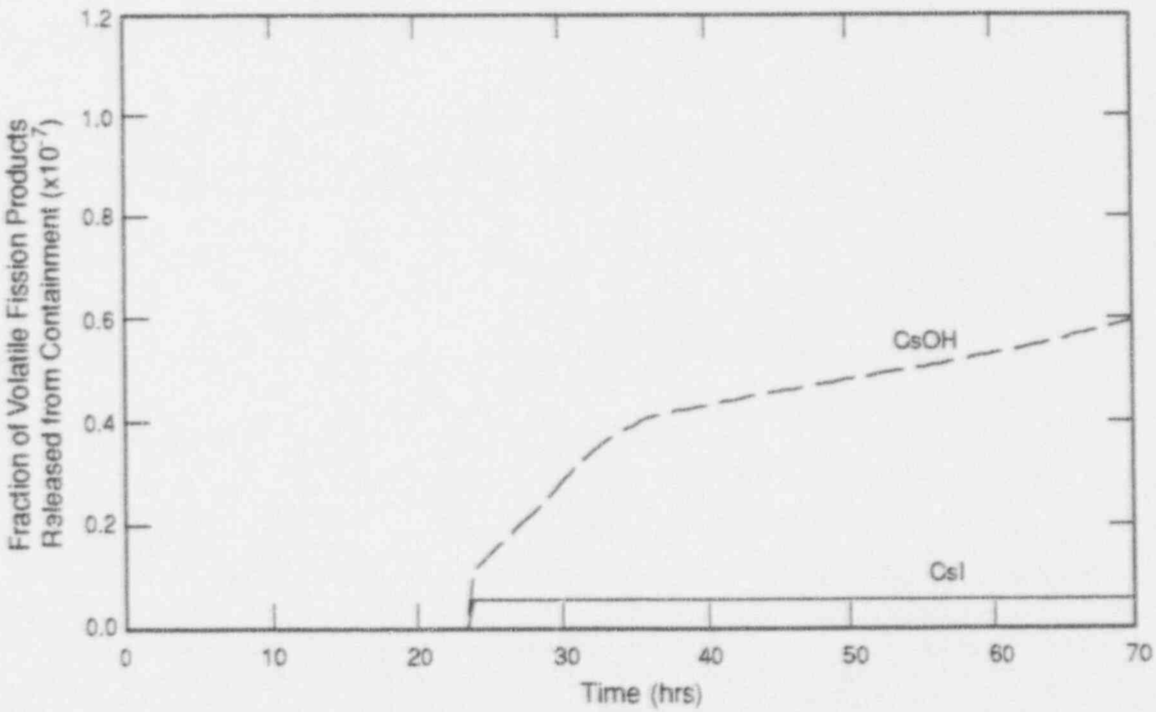


Figure 19E.2-7F SBRC-PF-R-N: STATION BLACKOUT WITH RCIC OPERATING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: VOLATILE FISSION PRODUCT RELEASE

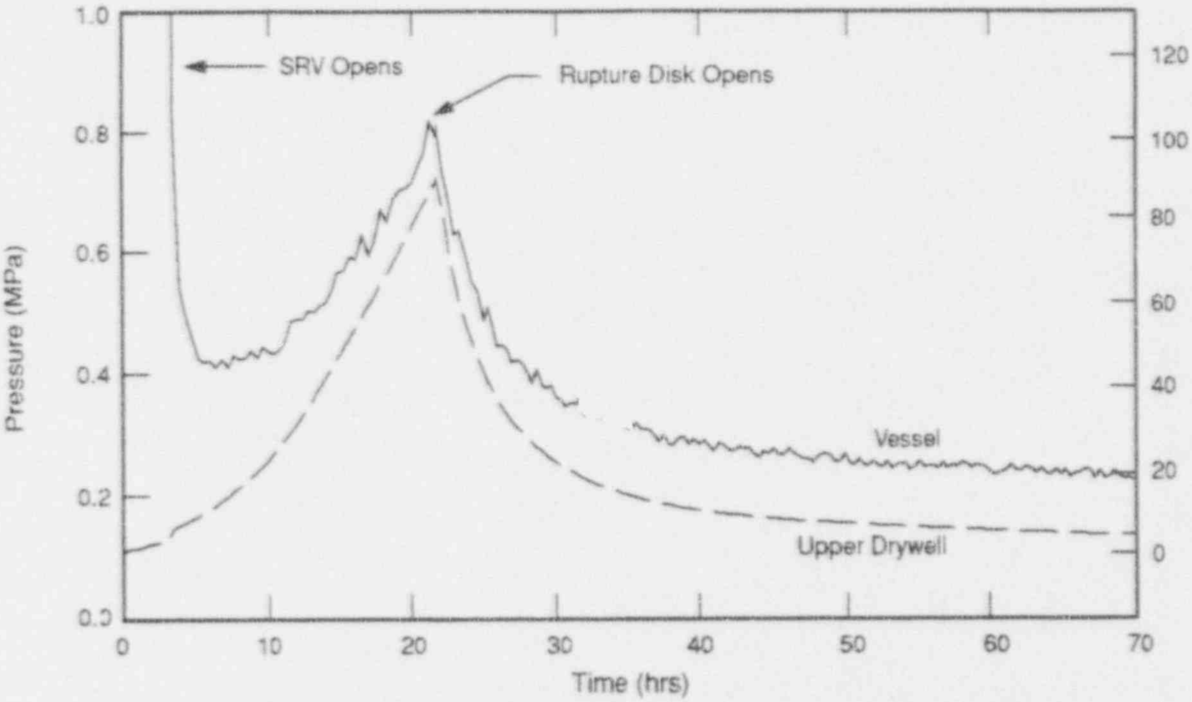


Figure 19E.2-8A **LHRC-00-R-0: ISOLATION WITH LOSS OF CONTAINMENT HEAT REMOVAL AND RUPTURE DISK OPENS: DRYWELL PRESSURE**

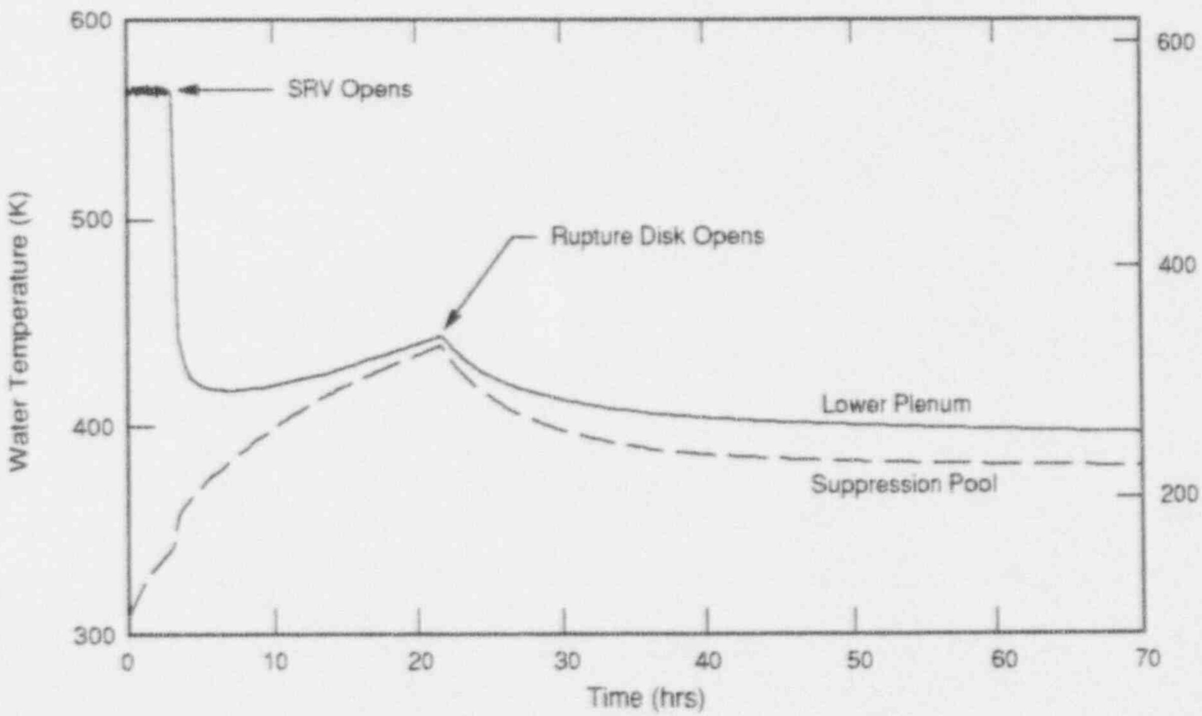


Figure 19E.2-8B **LHRC-00-R-0: ISOLATION WITH LOSS OF CONTAINMENT HEAT REMOVAL AND RUPTURE DISK OPENS: WATER TEMPERATURE**

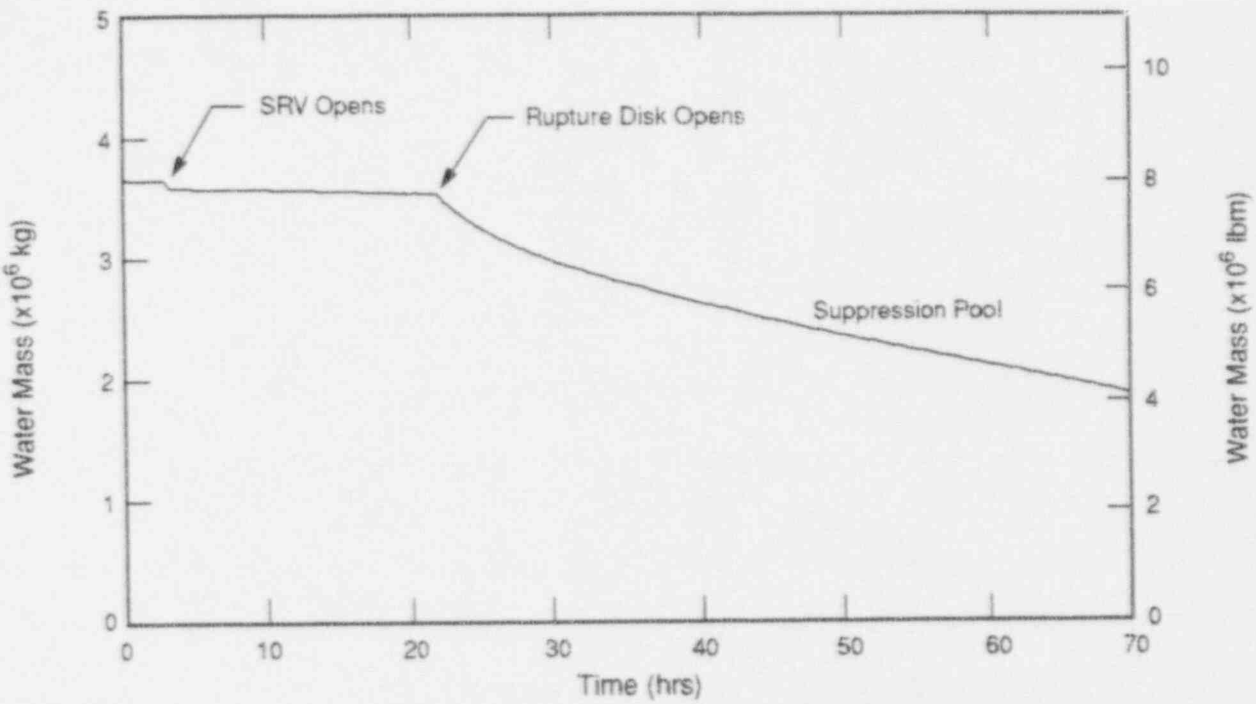


Figure 19E.2-8C LHRC-00-R-0: ISOLATION WITH LOSS OF CONTAINMENT HEAT REMOVAL AND RUPTURE DISK OPENS: WATER MASS

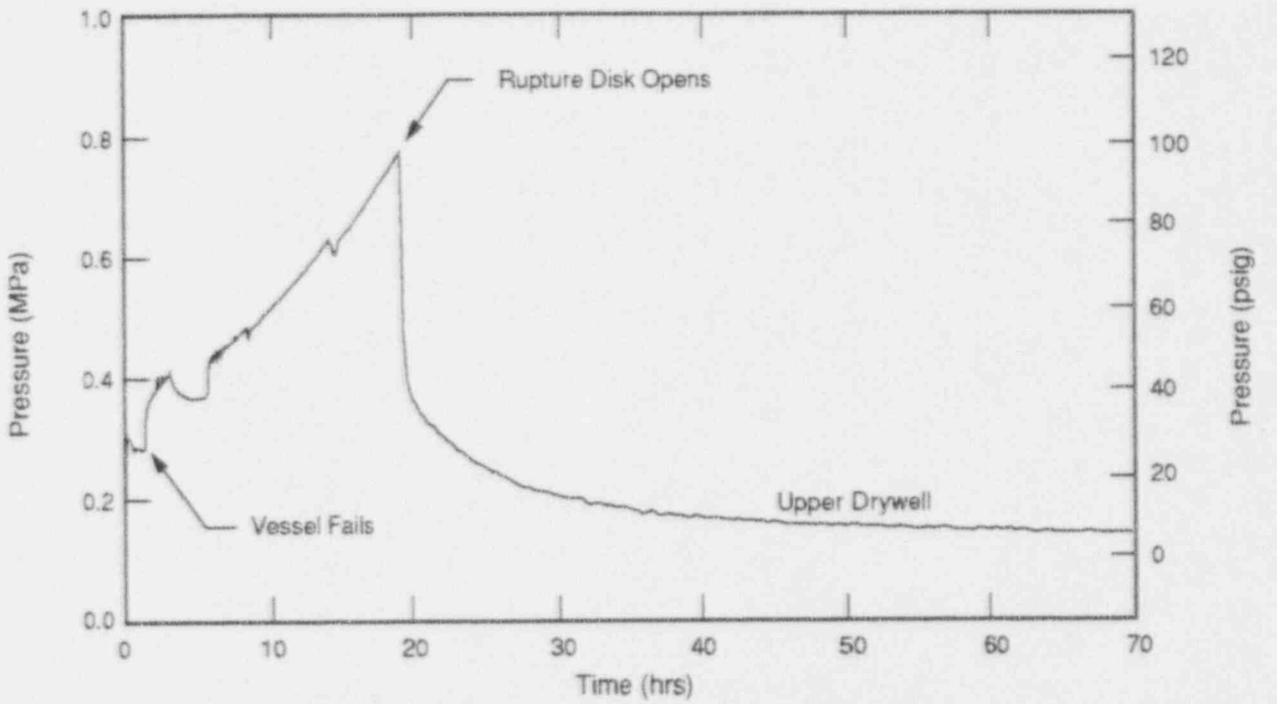


Figure 19E.2-9A LBLC-PF-R-N: LARGE BREAK LOCA WITH LOSS OF ALL CORE COOLING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: DRYWELL PRESSURE

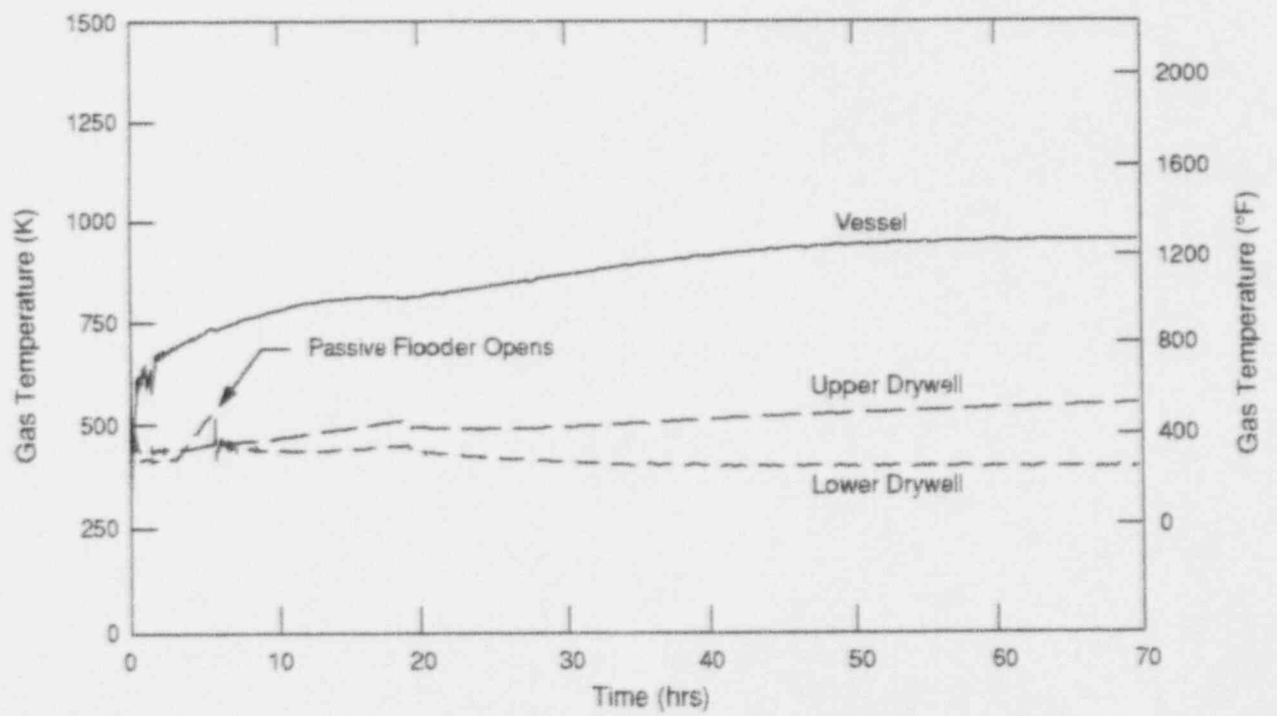


Figure 19E.2-9B LBLC-PF-R-N: LARGE BREAK LOCA WITH LOSS OF ALL CORE COOLING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: GAS TEMPERATURE

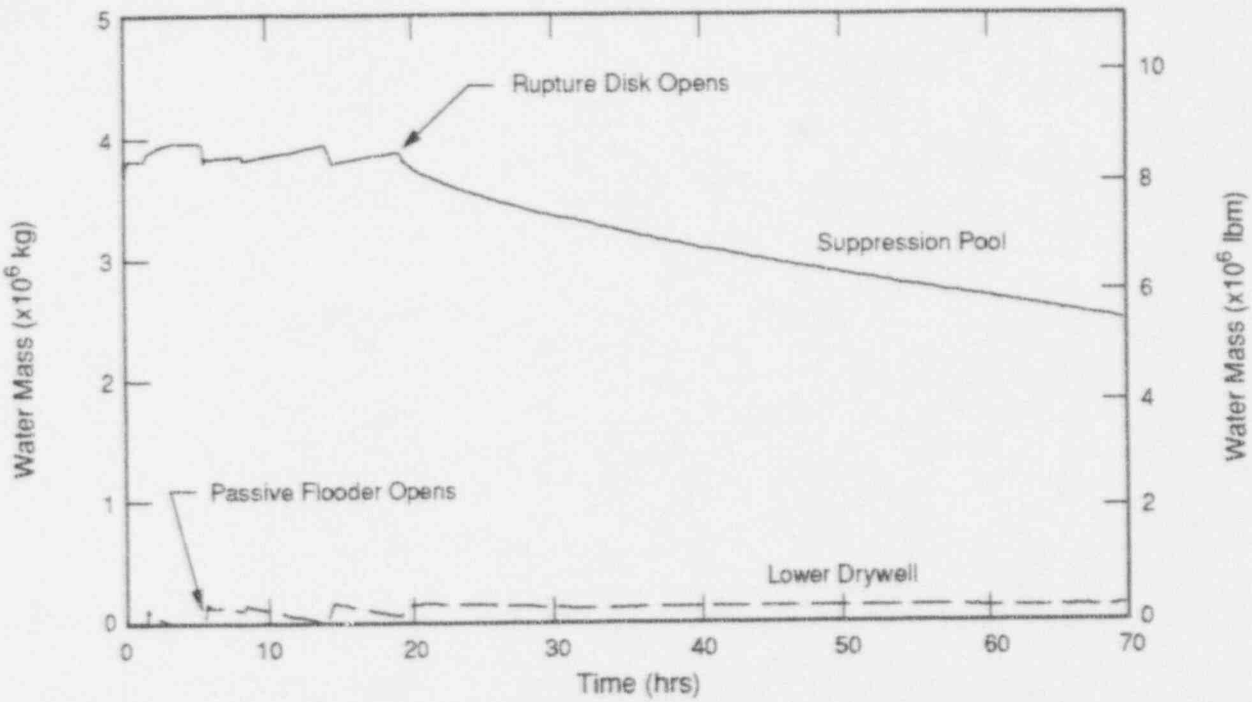


Figure 19E.2-9C LBLC-PF-R-N: LARGE BREAK LOCA WITH LOSS OF ALL CORE COOLING, PASSIVE FLOODER OFERATES AND RUPTURE DISK OPENS: WATER MASS

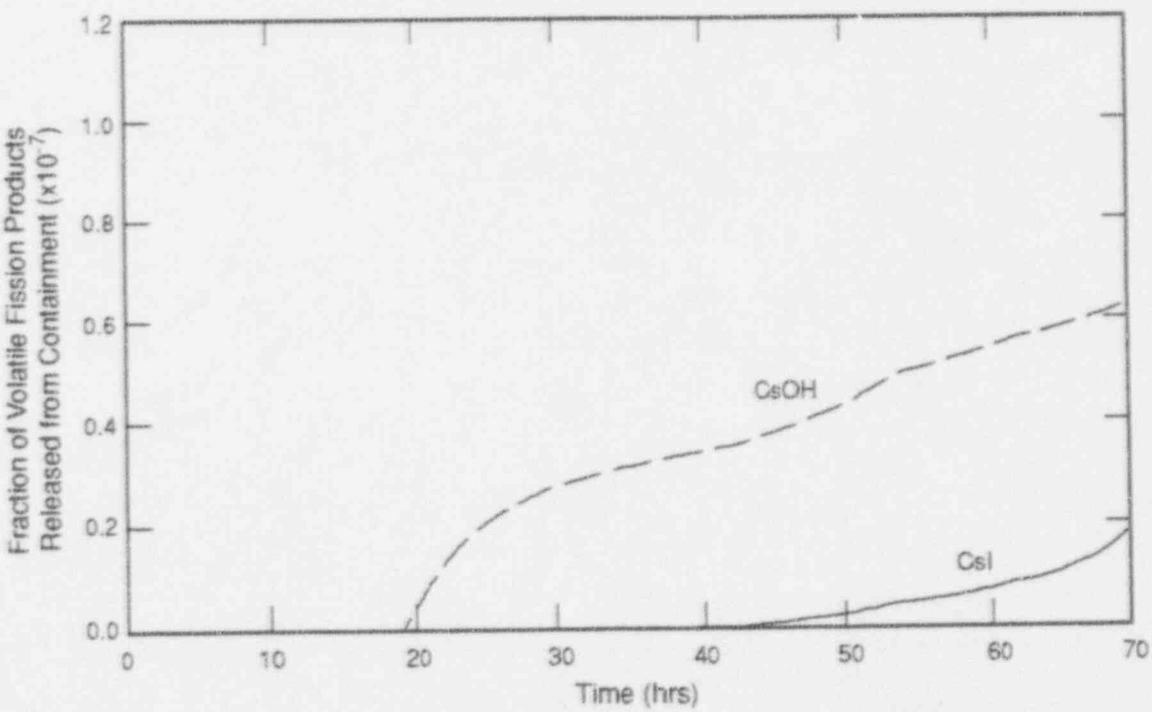


Figure 19E.2-9D LBLC-PF-R-N: LARGE BREAK LOCA WITH LOSS OF ALL CORE COOLING, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: VOLATILE FISSION PRODUCT RELEASE

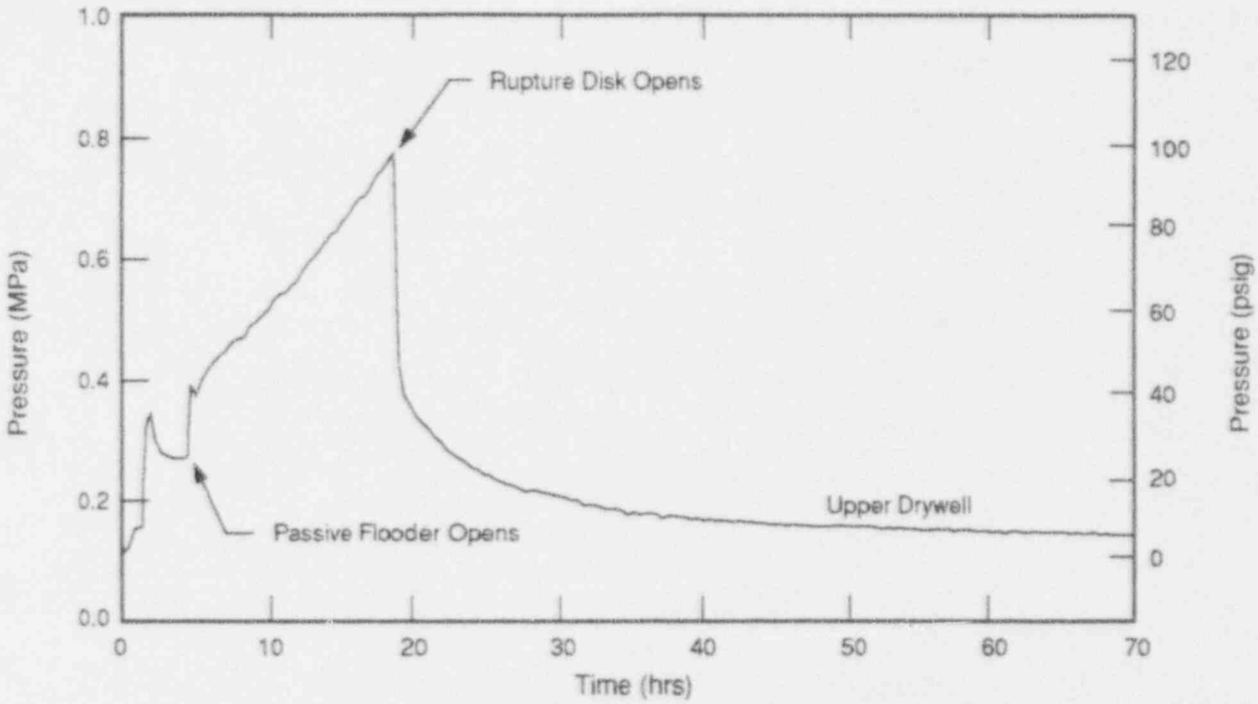


Figure 19E.2-10A NSCL-PF-R-N: CONCURRENT LOSS OF ALL CORE COOLING AND ATWS WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER AND RUPTURE DISK: DRYWELL PRESSURE

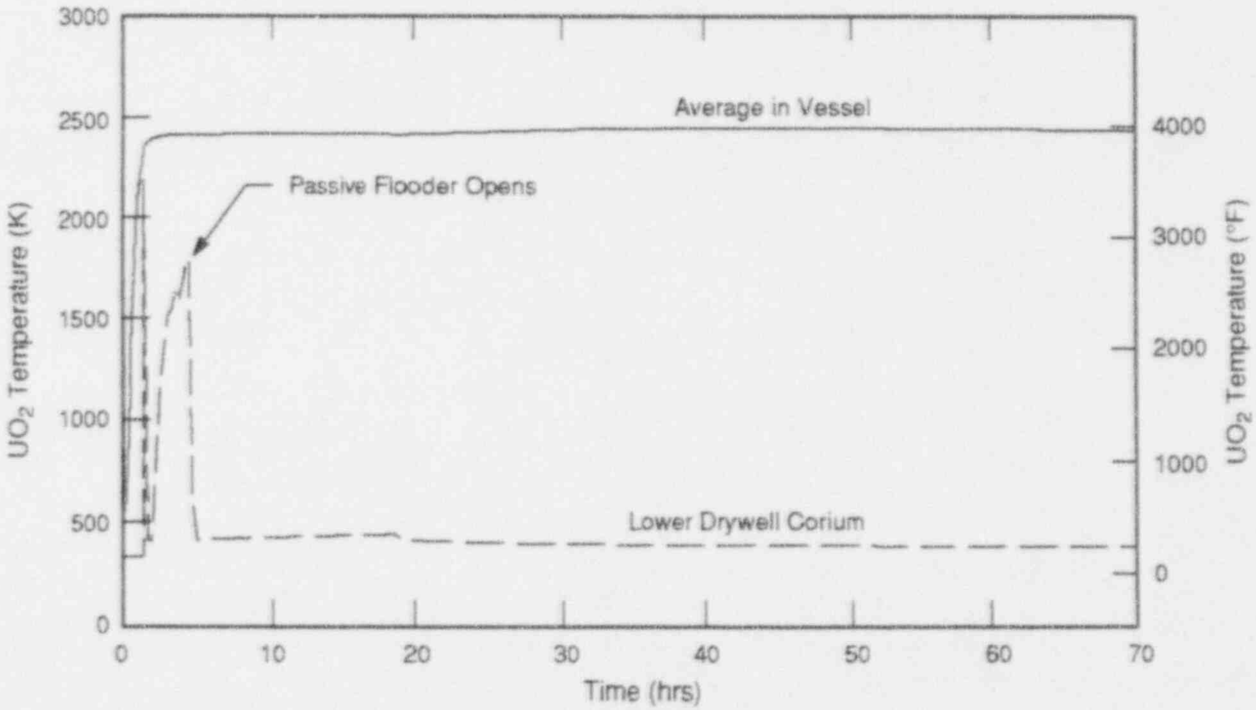


Figure 19E.2-10B NSCL-PF-R-N: CONCURRENT LOSS OF ALL CORE COOLING AND ATWS WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER AND RUPTURE DISK: UO_2 TEMPERATURE

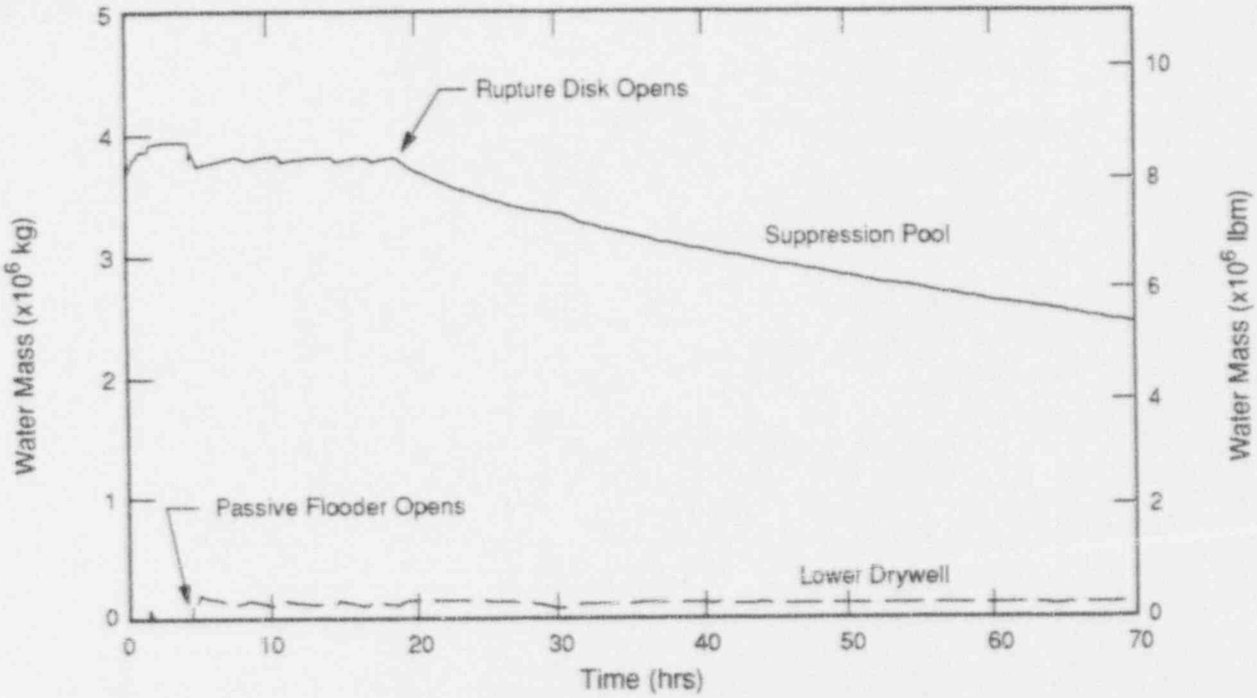


Figure 19E.2-10C NSCL-PF-R-N: CONCURRENT LOSS OF ALL CORE COOLING AND ATWS WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER AND RUPTURE DISK: WATER MASS

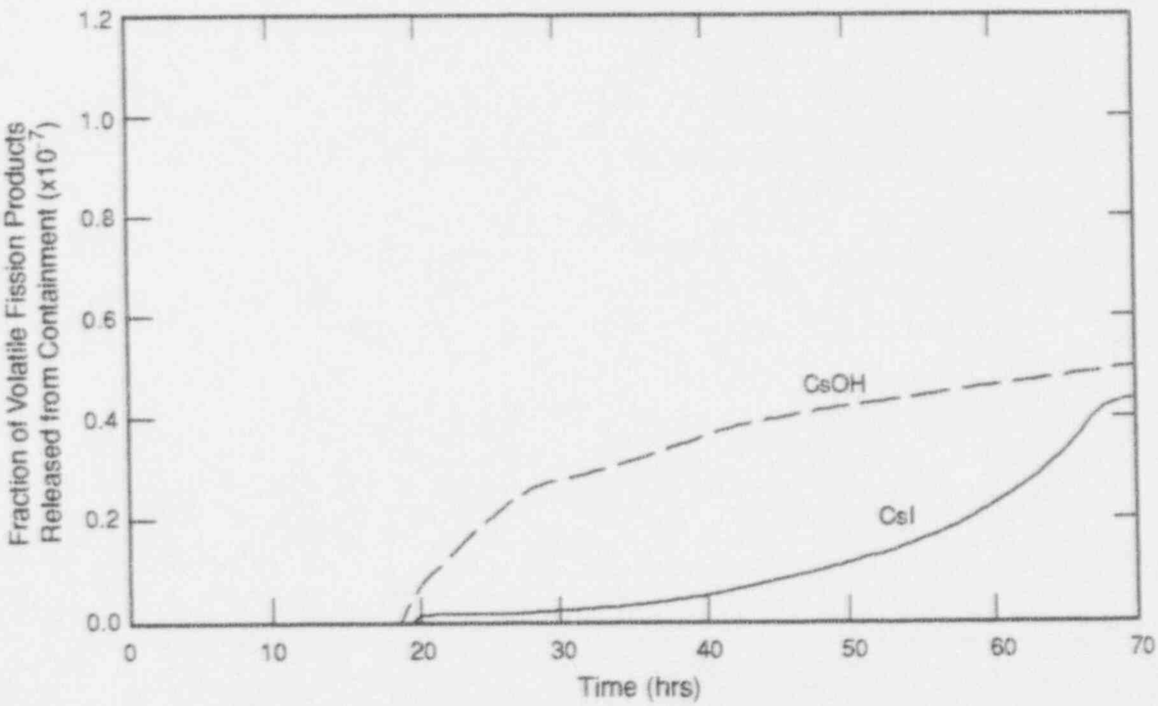


Figure 19E.2-10D NSCL-PF-R-N: CONCURRENT LOSS OF ALL CORE COOLING AND ATWS WITH VESSEL FAILURE AT LOW PRESSURE, PASSIVE FLOODER AND RUPTURE DISK: VOLATILE FISSION PRODUCTS

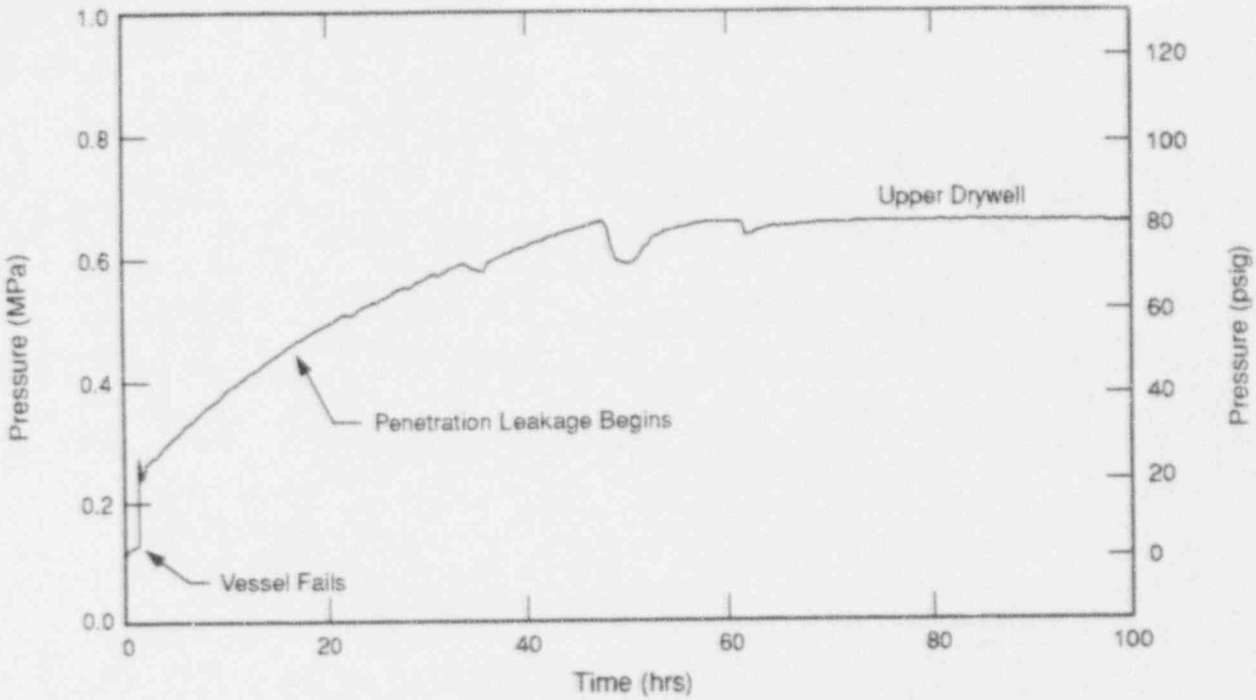


Figure 19E.2-11A NSCH-PF-P-M: CONCURRENT LOSS OF ALL CORE COOLING AND ATWS WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER, PENETRATION LEAKAGE: DRYWELL PRESSURE

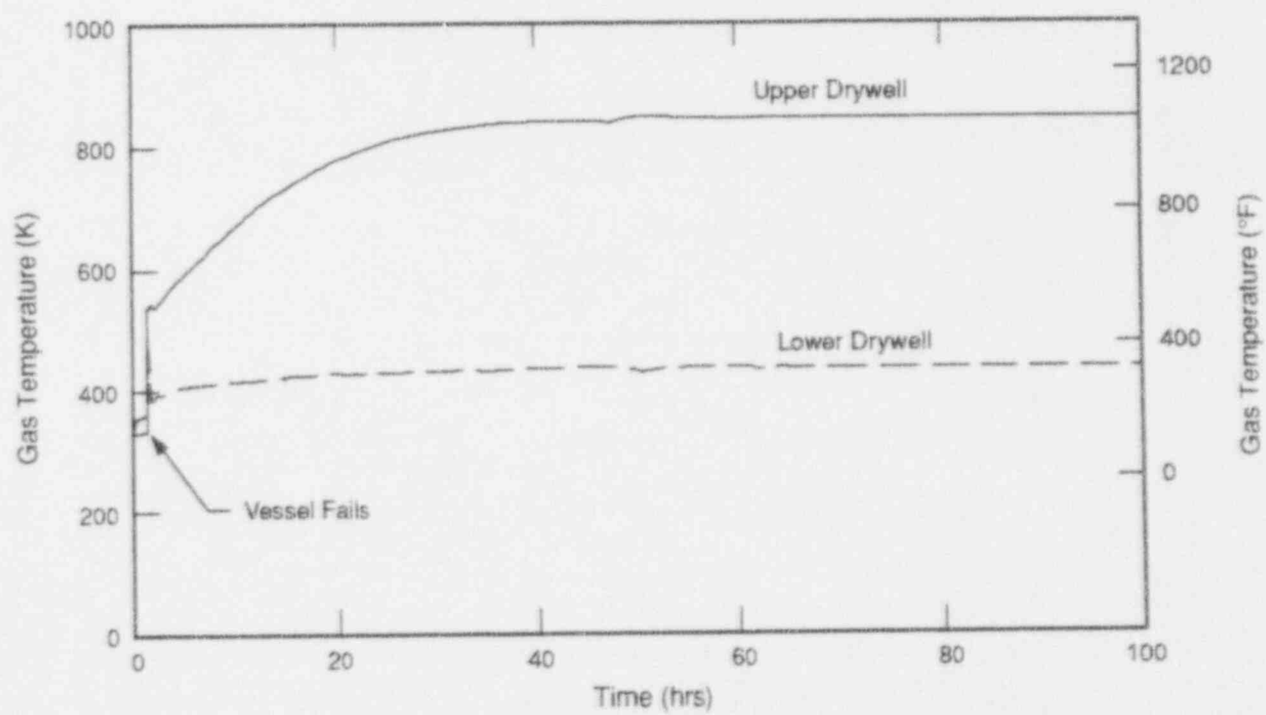


Figure 19E.2-11B NSCH-PF-P-M: CONCURRENT LOSS OF ALL CORE COOLING AND ATWS WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER, PENETRATION LEAKAGE: GAS TEMPERATURE

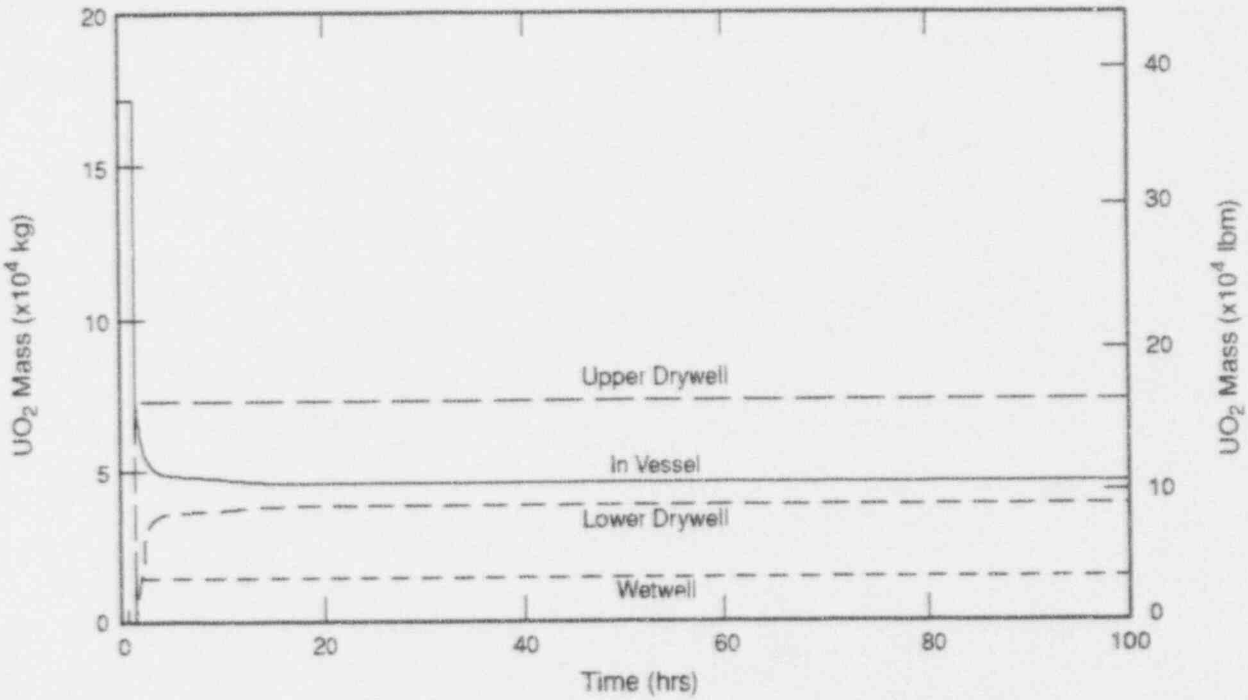


Figure 19E.2-11C NSCH-PF-P-M: CONCURRENT LOSS OF ALL CORE COOLING AND ATWS WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER, PENETRATION LEAKAGE: UO_2 MASS

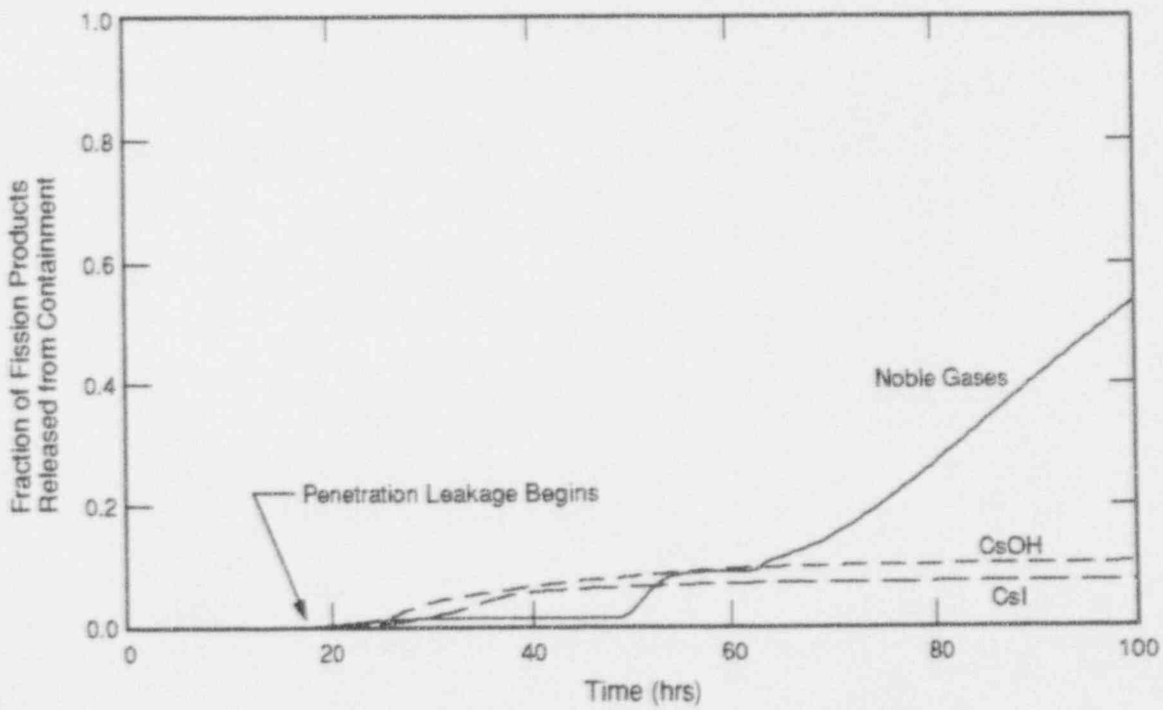


Figure 19E.2-11D NSCH-PF-P-M: CONCURRENT LOSS OF ALL CORE COOLING AND ATWS WITH VESSEL FAILURE AT HIGH PRESSURE, PASSIVE FLOODER, PENETRATION LEAKAGE: FISSION PRODUCTS

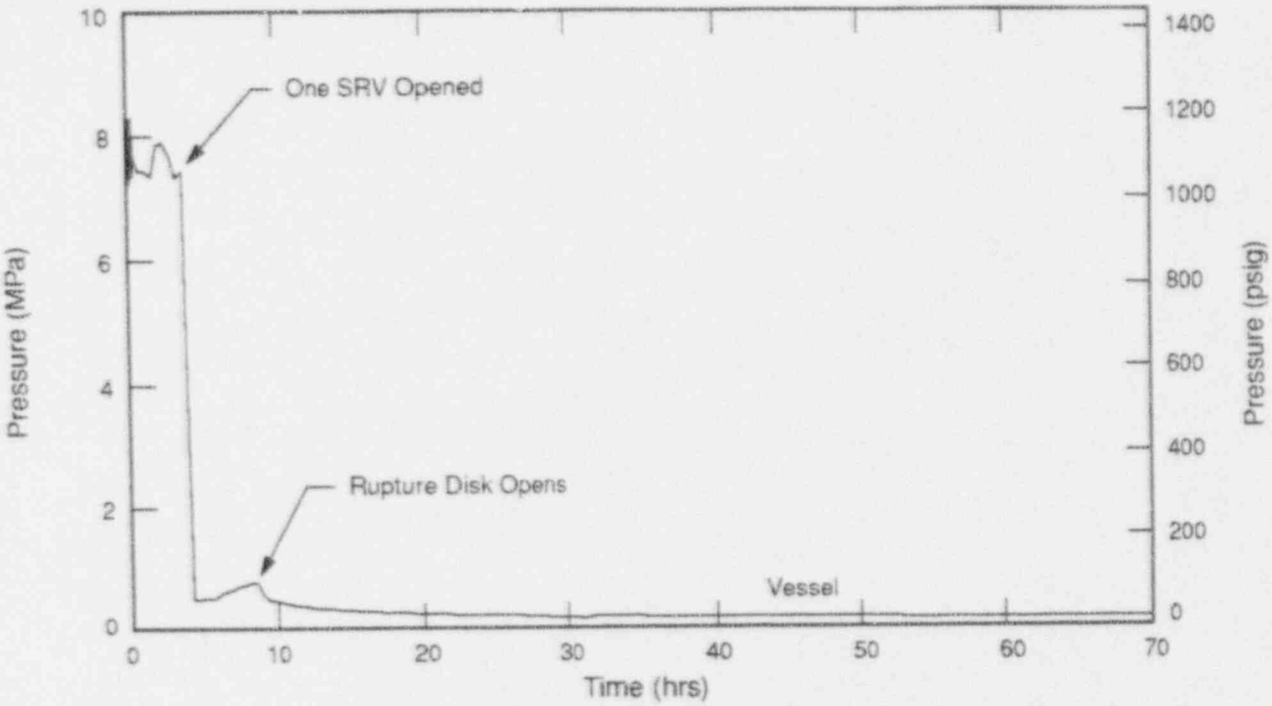


Figure 19E.2-12A NSRC-PF-R-N: CONCURRENT STATION BLACKOUT WITH ATWS, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: VESSEL PRESSURE

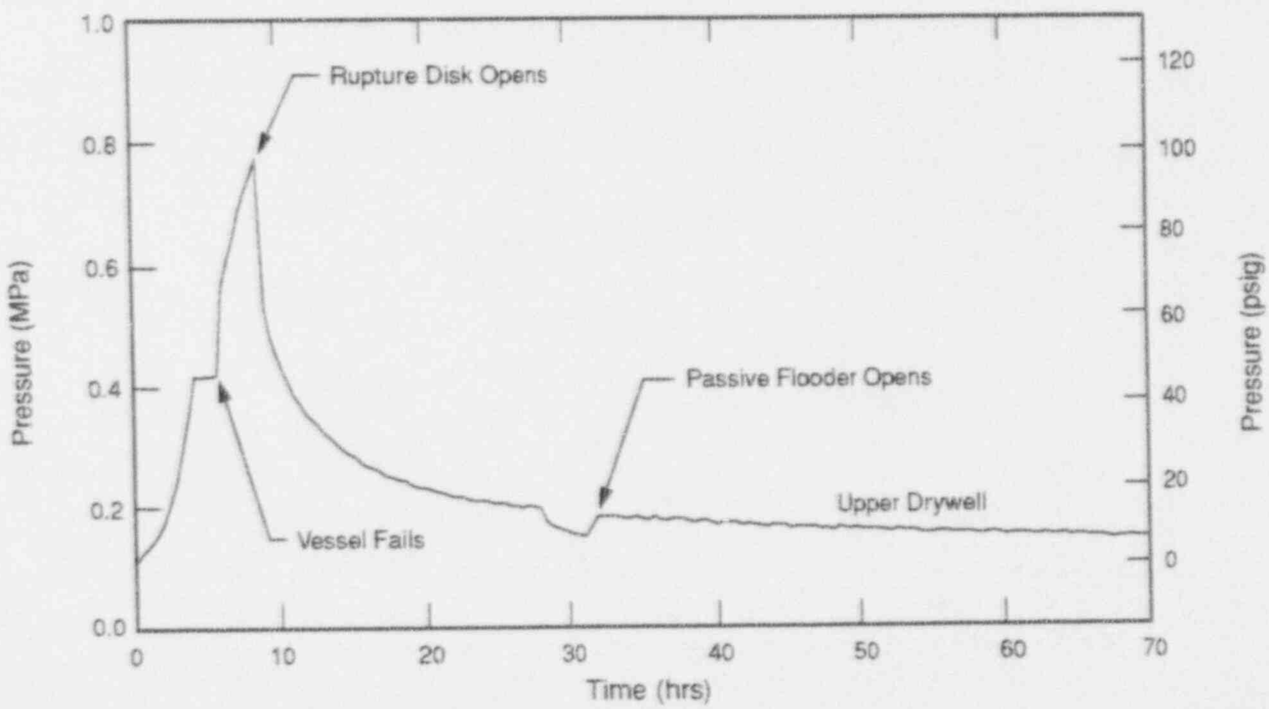


Figure 19E.2-12B NSRC-PF-R-N: CONCURRENT STATION BLACKOUT WITH ATWS, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: DRYWELL PRESSURE

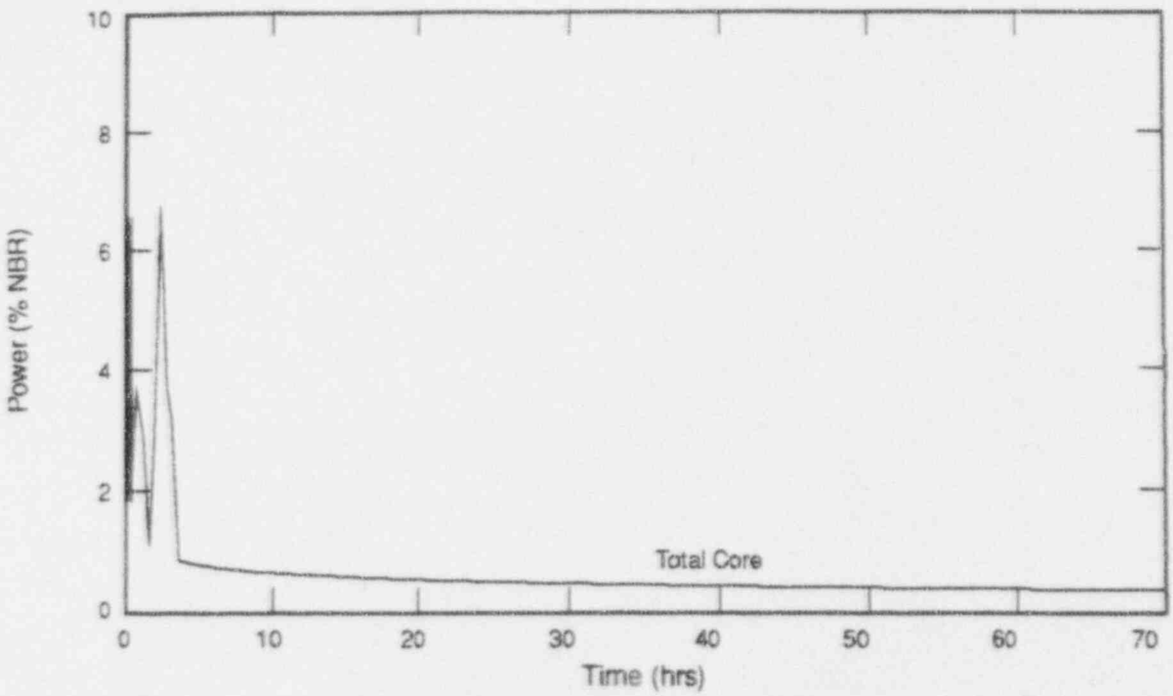


Figure 19E.2-12C NSRC-PF-R-N: CONCURRENT STATION BLACKOUT WITH ATWS, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: POWER

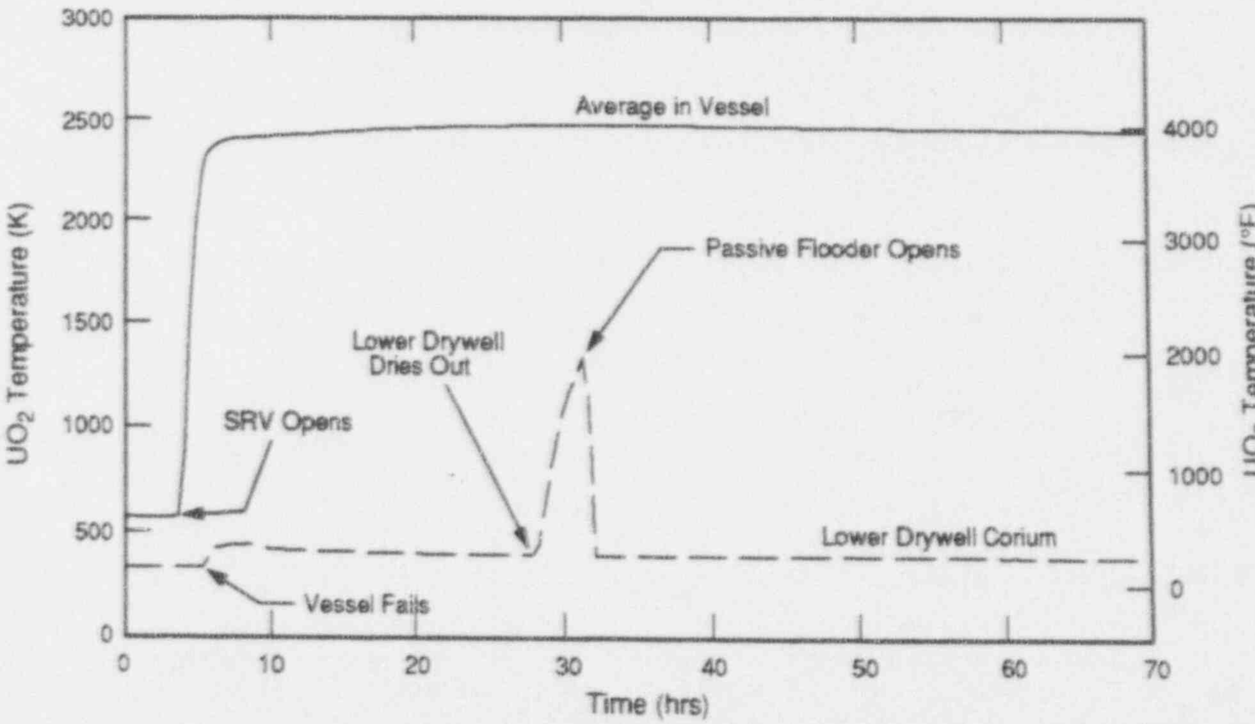


Figure 19E.2-12D NSRC-PF-R-N: CONCURRENT STATION BLACKOUT WITH ATWS, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: UO_2 TEMPERATURE

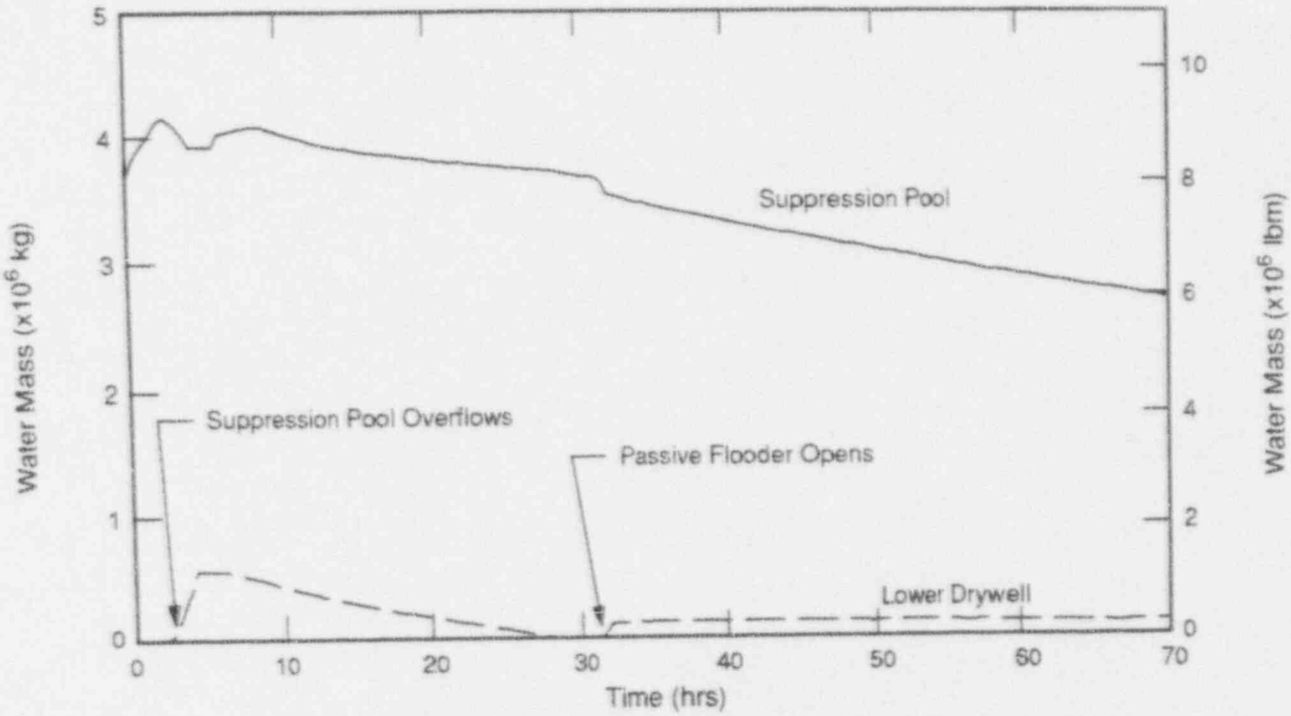


Figure 19E.2-12E NSRC-PF-R-N: CONCURRENT STATION BLACKOUT WITH ATWS, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: WATER MASS

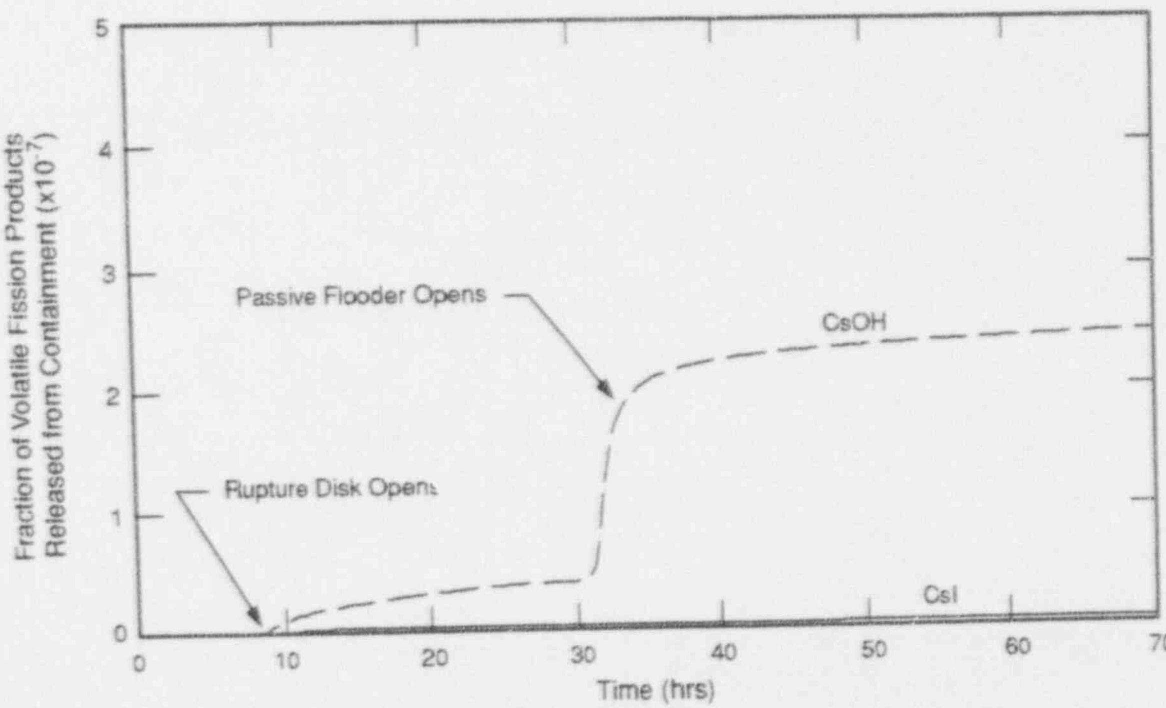


Figure 19E.2-12F NSRC-PF-R-N: CONCURRENT STATION BLACKOUT WITH ATWS, PASSIVE FLOODER OPERATES AND RUPTURE DISK OPENS: VOLATILE FISSION PRODUCT RELEASE

Figure 19E.2–13 STEAM EXPLOSION PROCESS

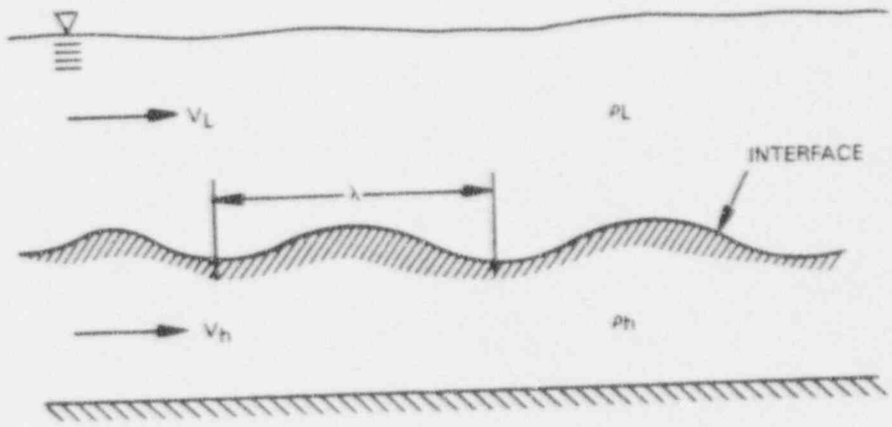


Figure 19E.2–14A INTERFACIAL INSTABILITY

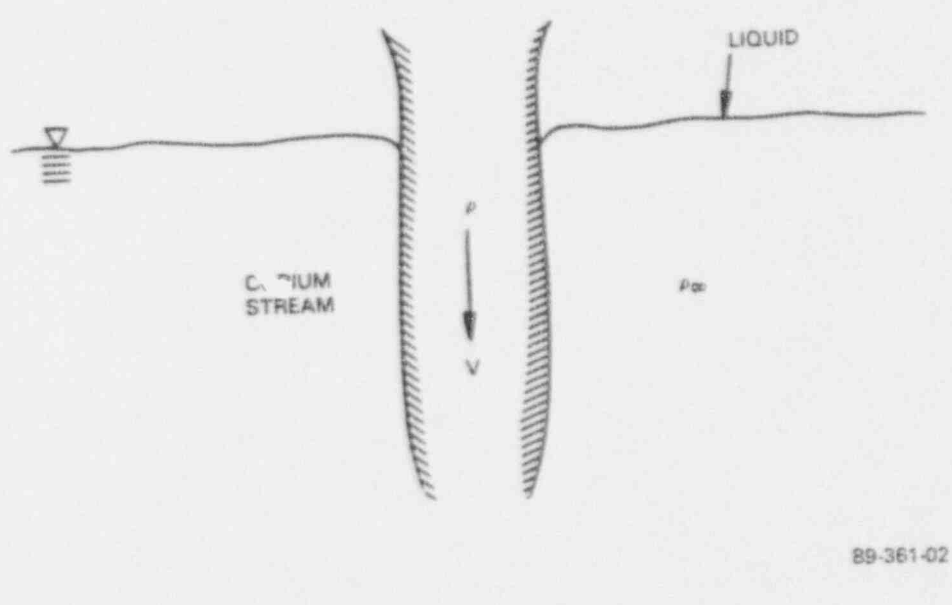


Figure 19E.2–14B CORIUM STREAM IN LIQUID

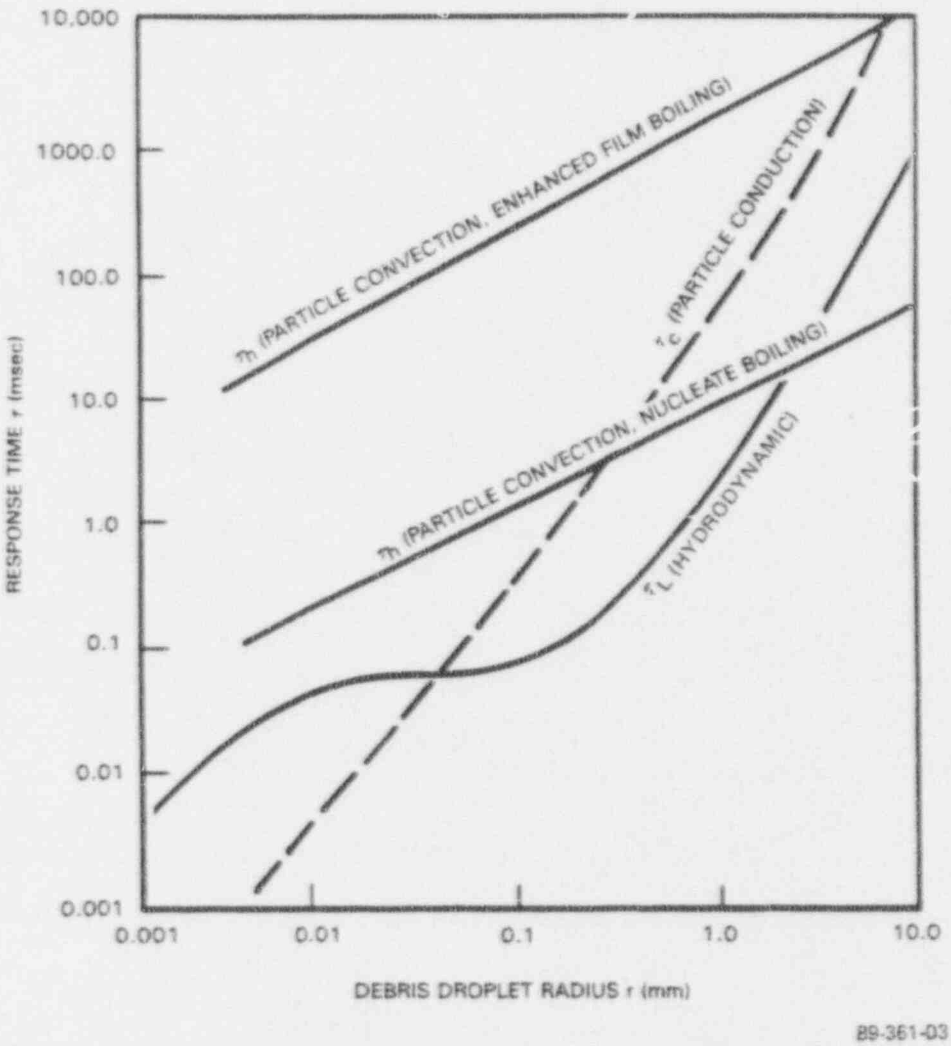


Figure 19E.2–15 IMPORTANT RESPONSE TIMES

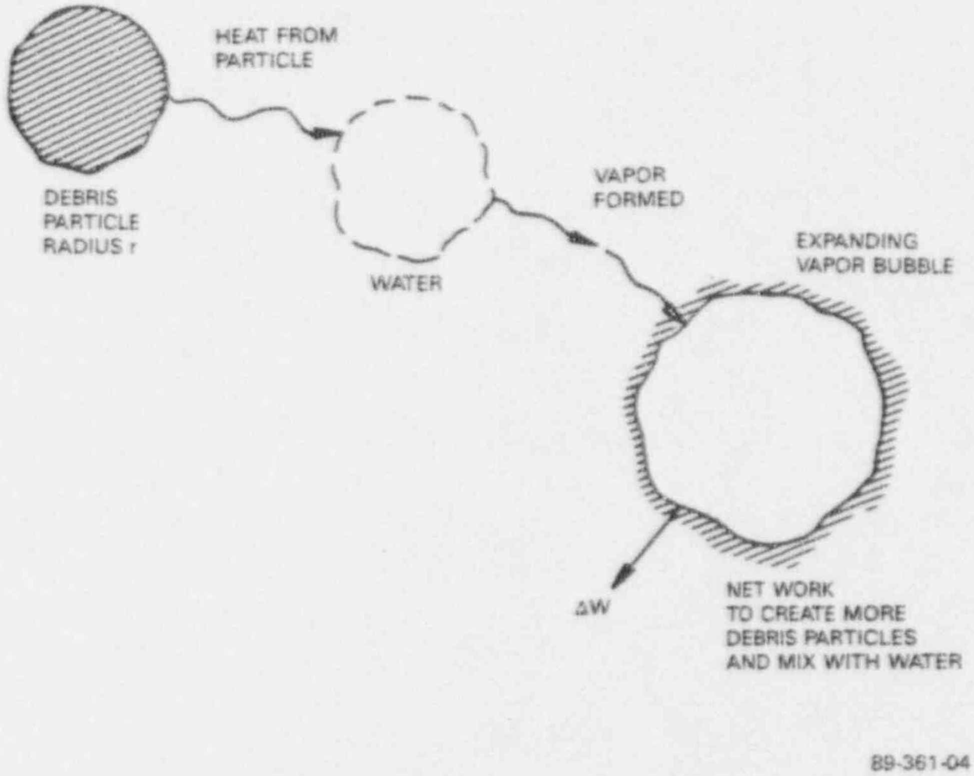


Figure 19E.2–16 SELF-TRIGGERING PROCESS

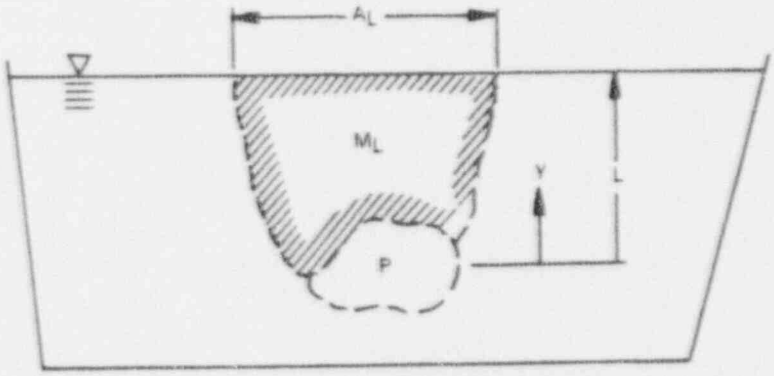


Figure 19E.2—17 CONDITIONS FOR STEAM EXPLOSION

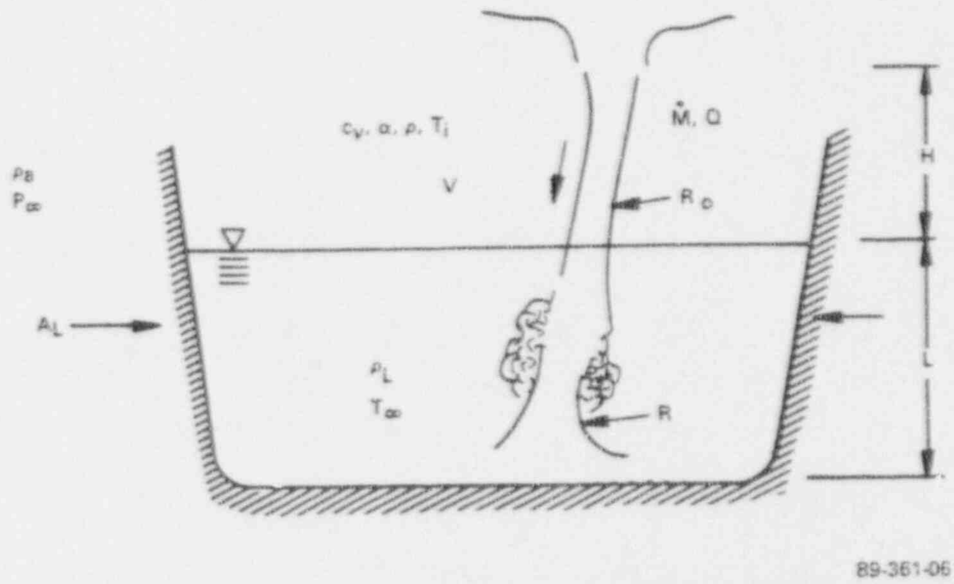


Figure 19E.2–18 APPLICATION TO ABWR

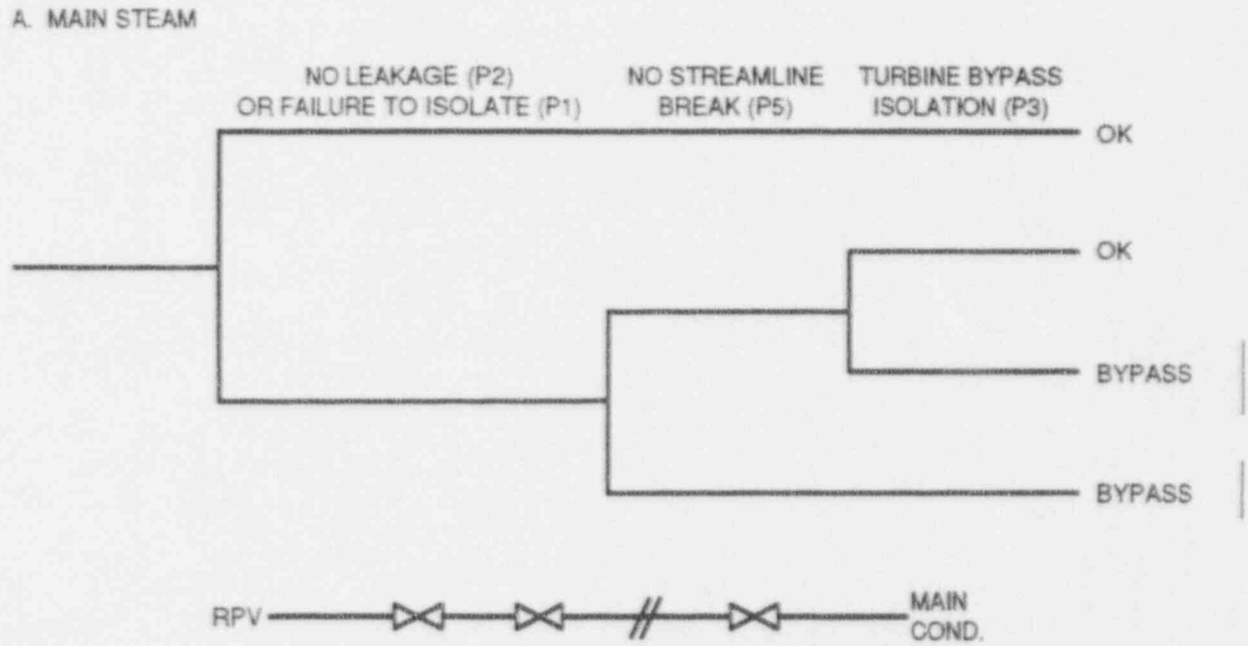


Figure 19E.2-19A SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

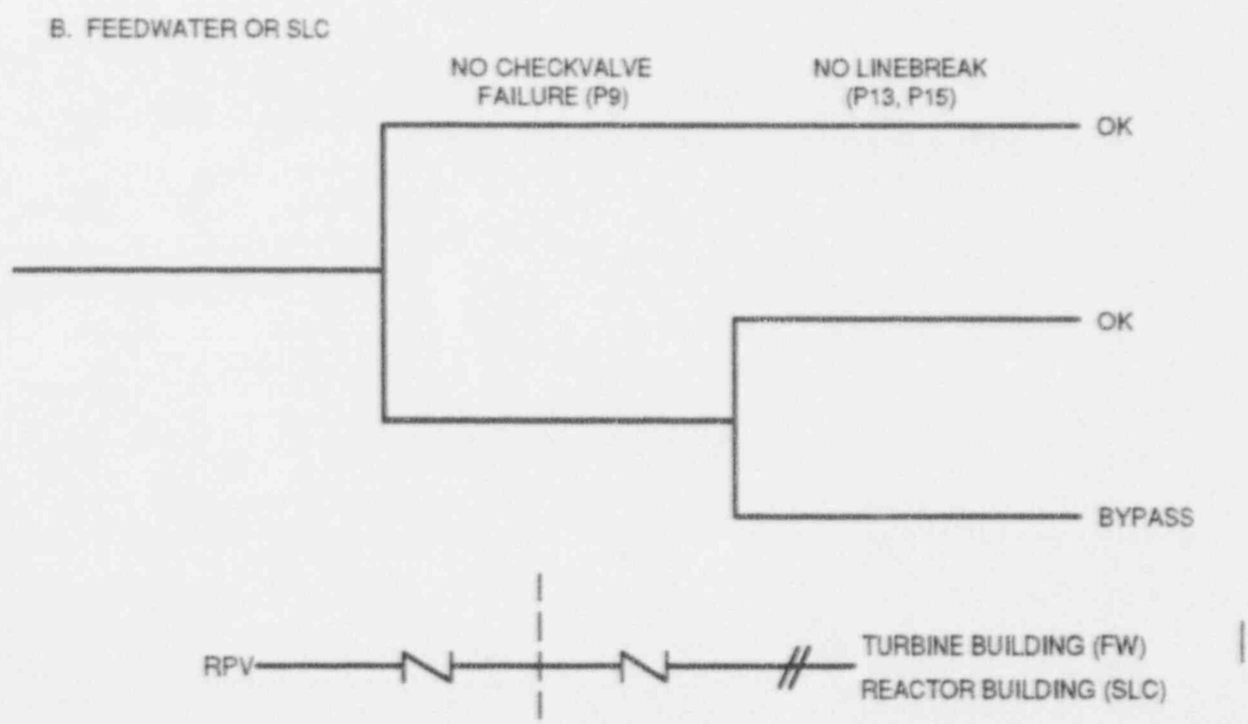


Figure 19E.2-19B SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

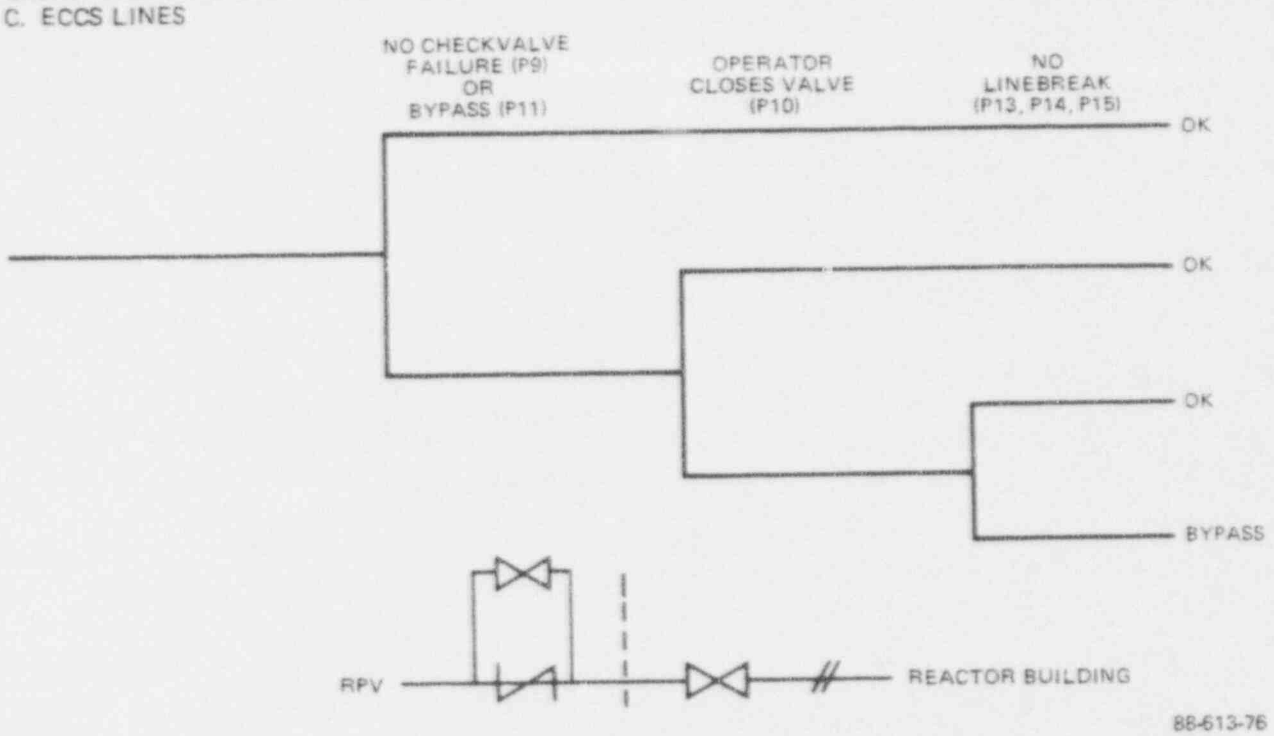


Figure 19E.2–19C SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

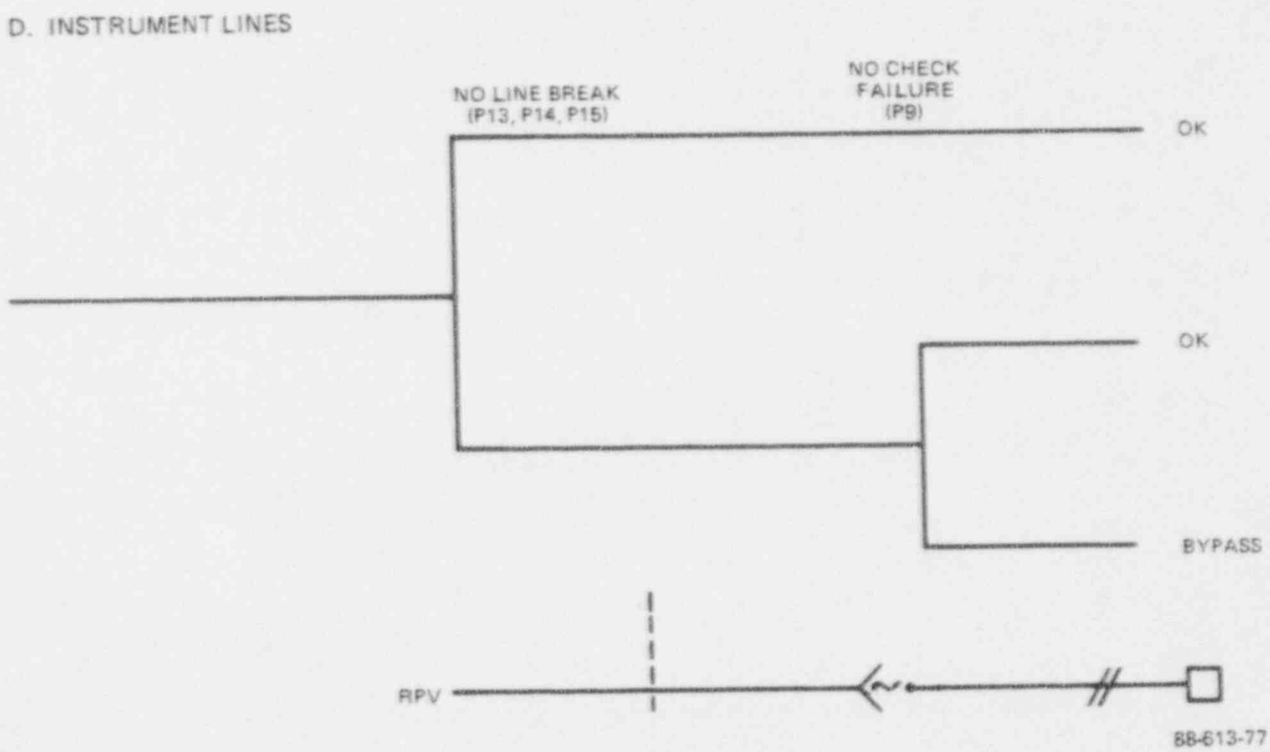


Figure 19E.2–19D SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

E. STATION BLACKOUT AFFECTED LINES

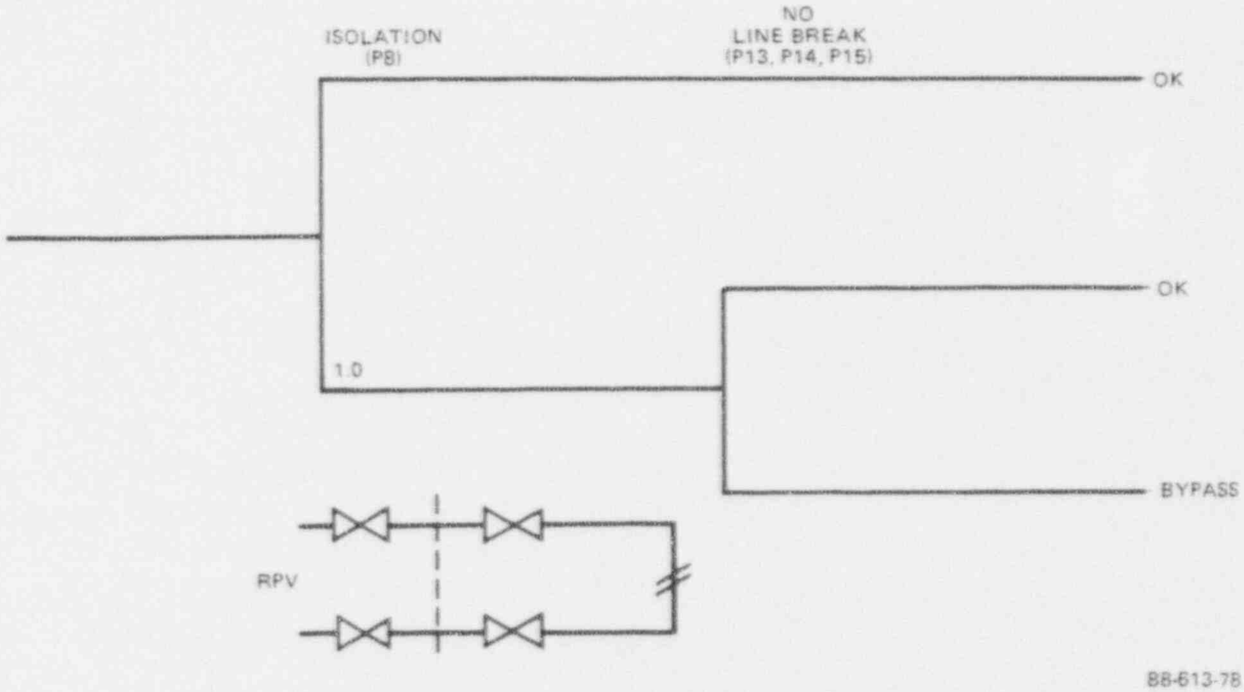


Figure 19E.2–19E SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

F. CONTAINMENT ATMOSPHERIC MONITOR

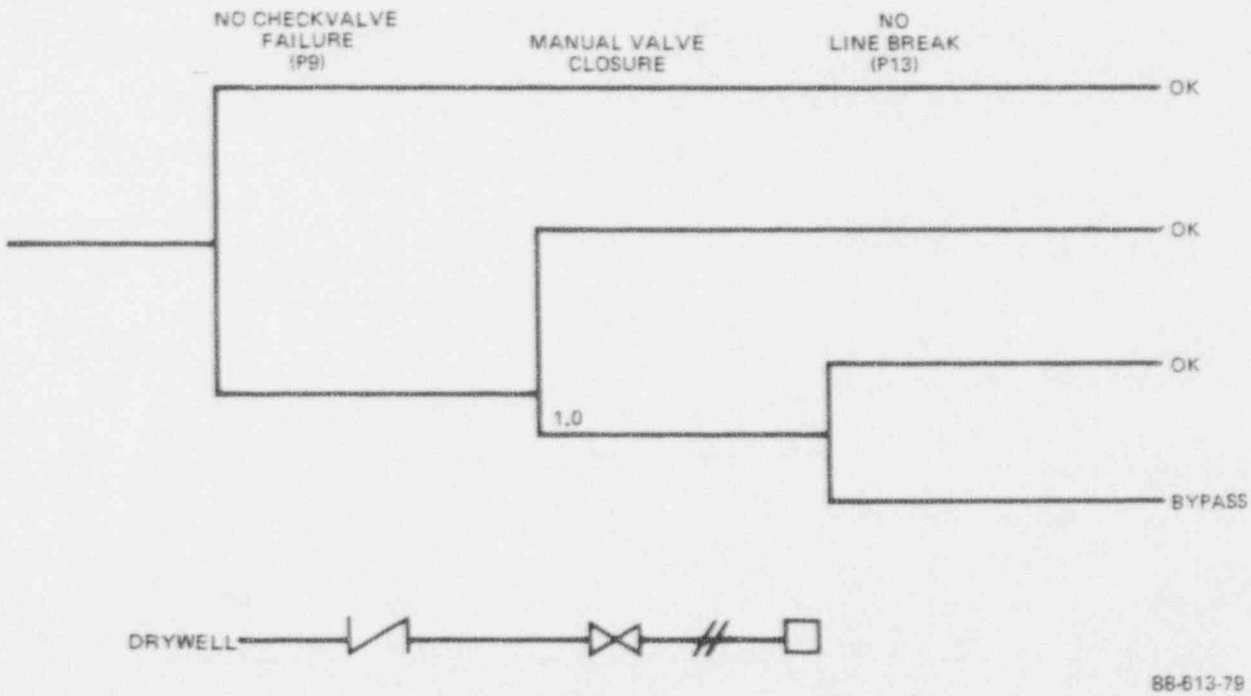


Figure 19E.2–19F SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

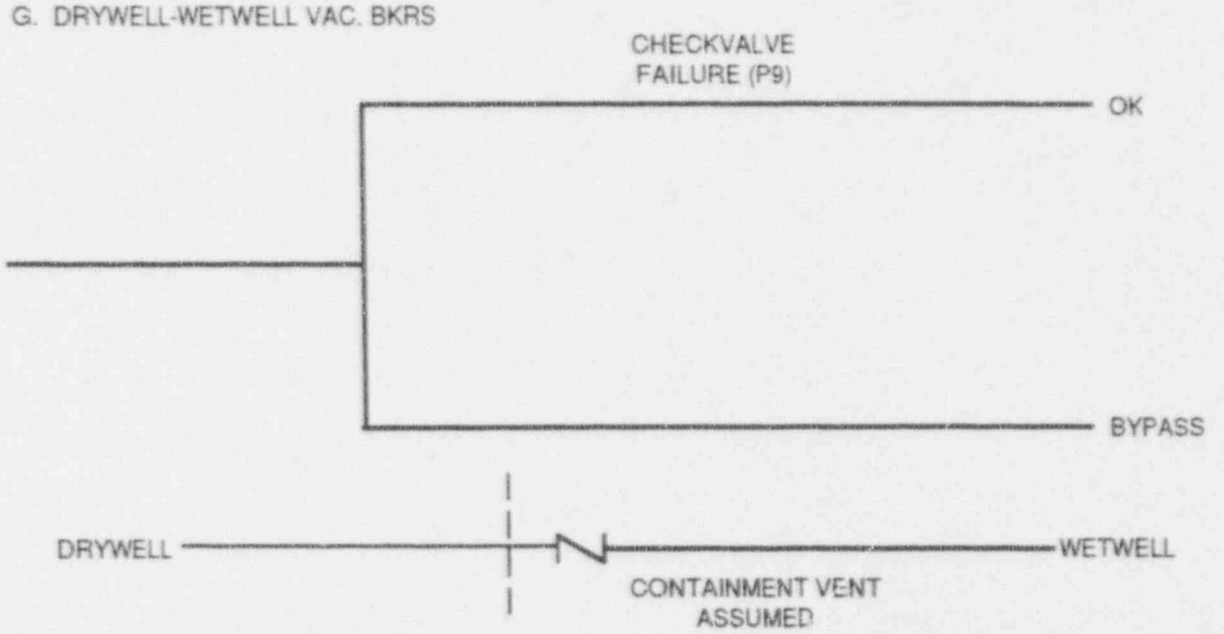


Figure 19E.2-19G SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

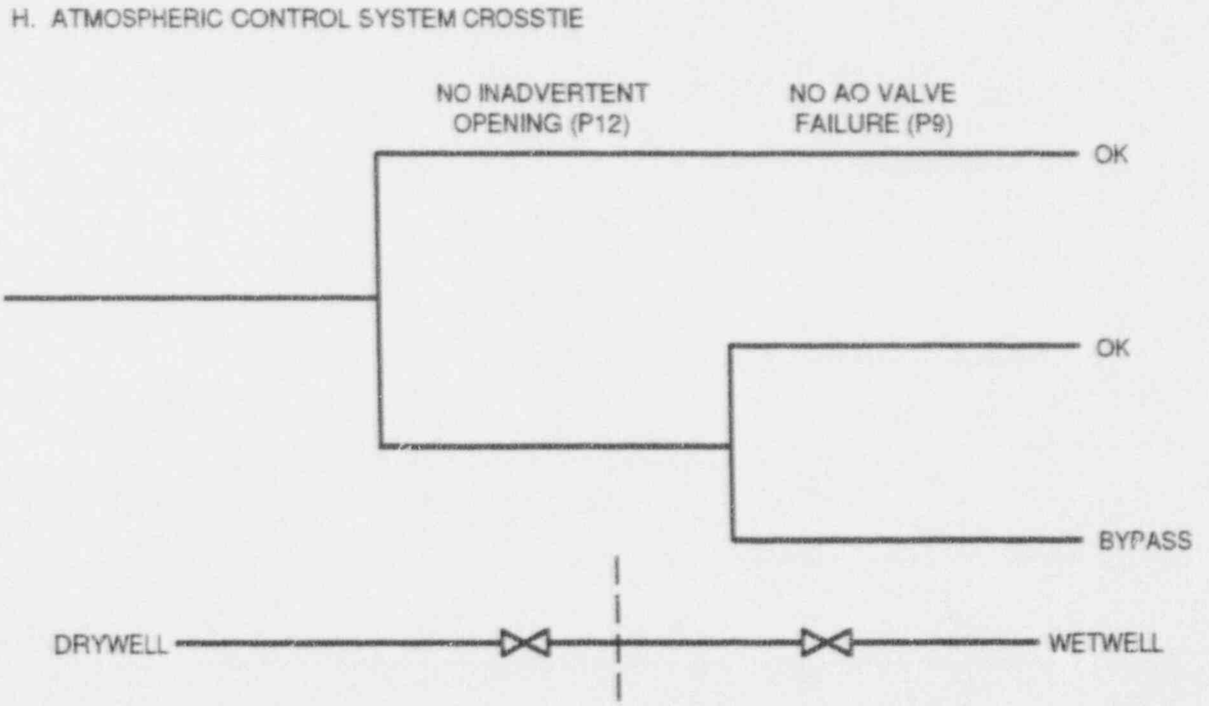


Figure 19E.2-19H SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

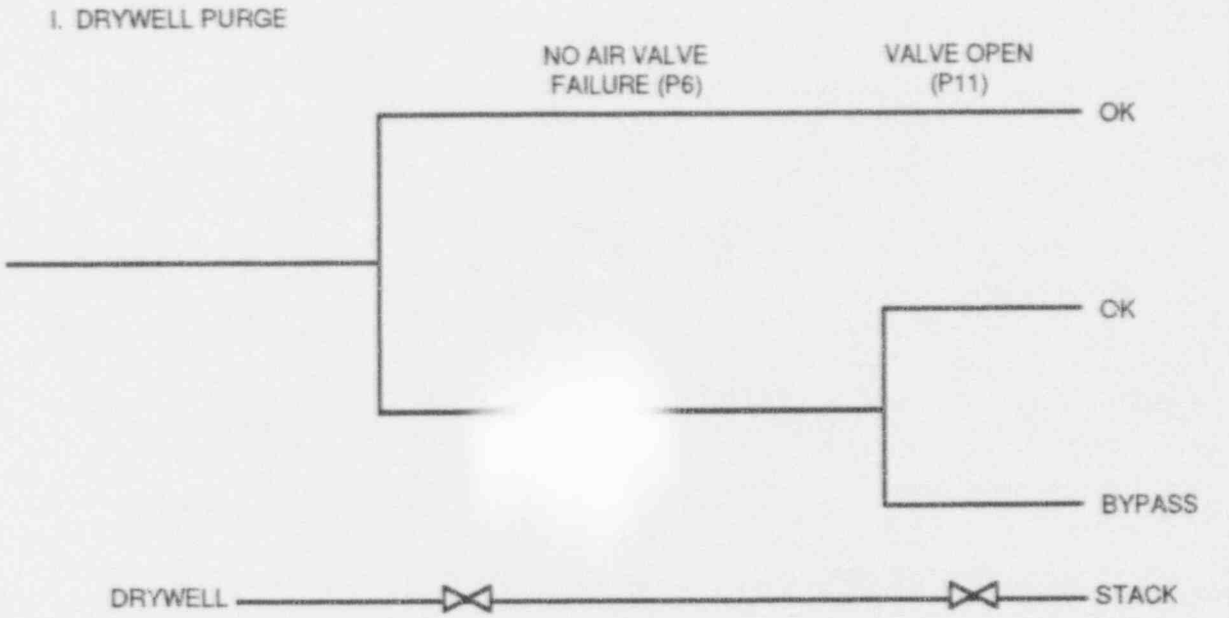


Figure 19E.2-19I SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

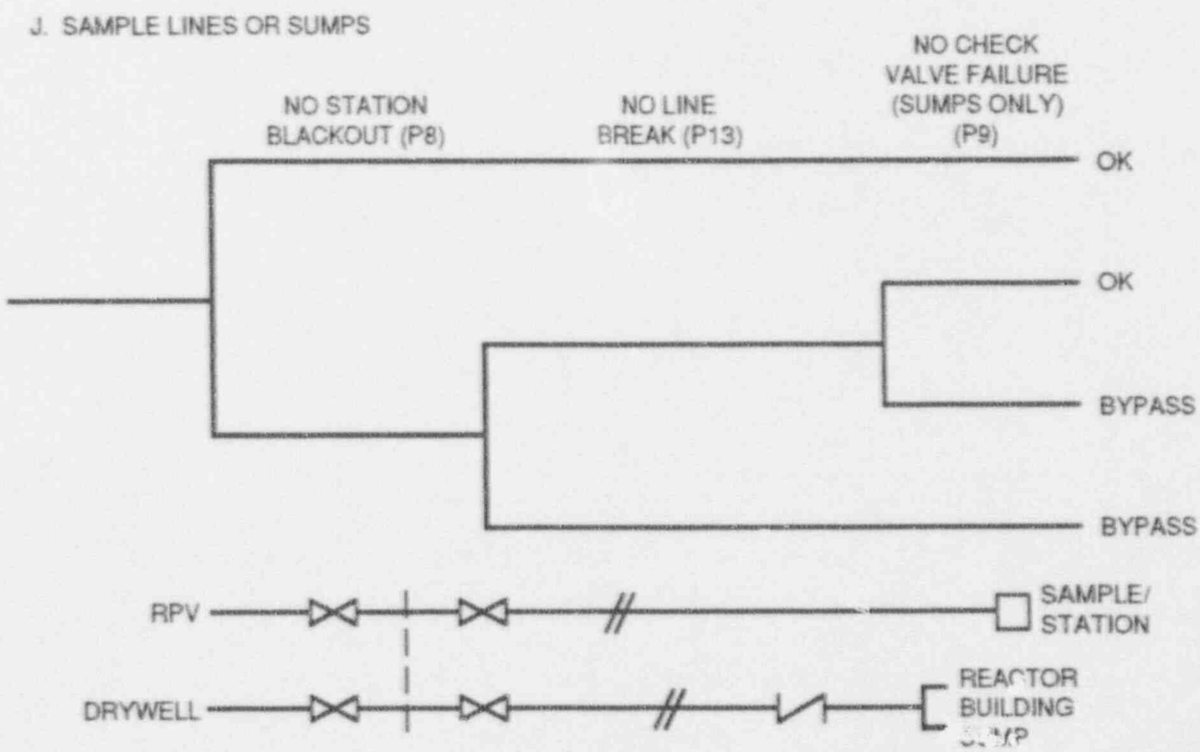


Figure 19E.2-19J SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

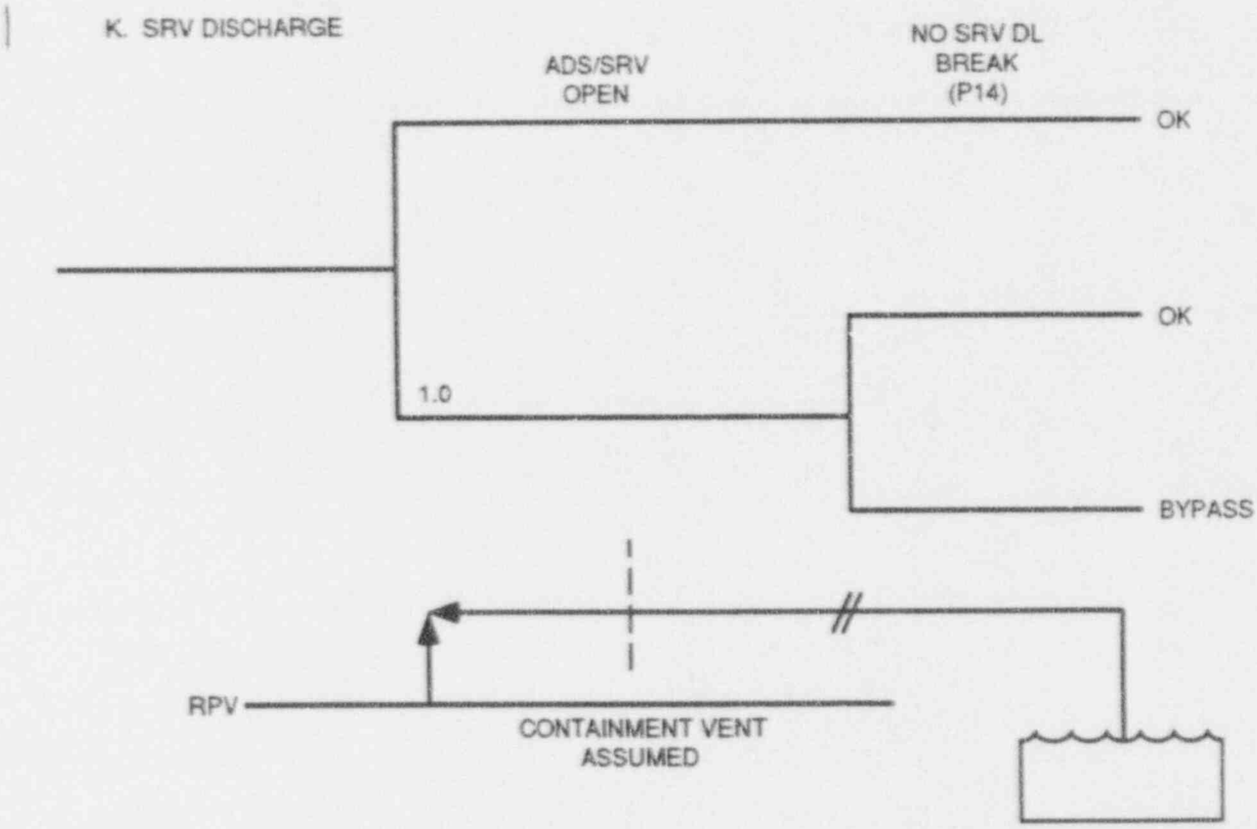


Figure 19E.2-19K SUPPRESSION POOL BYPASS PATHS AND CONFIGURATIONS

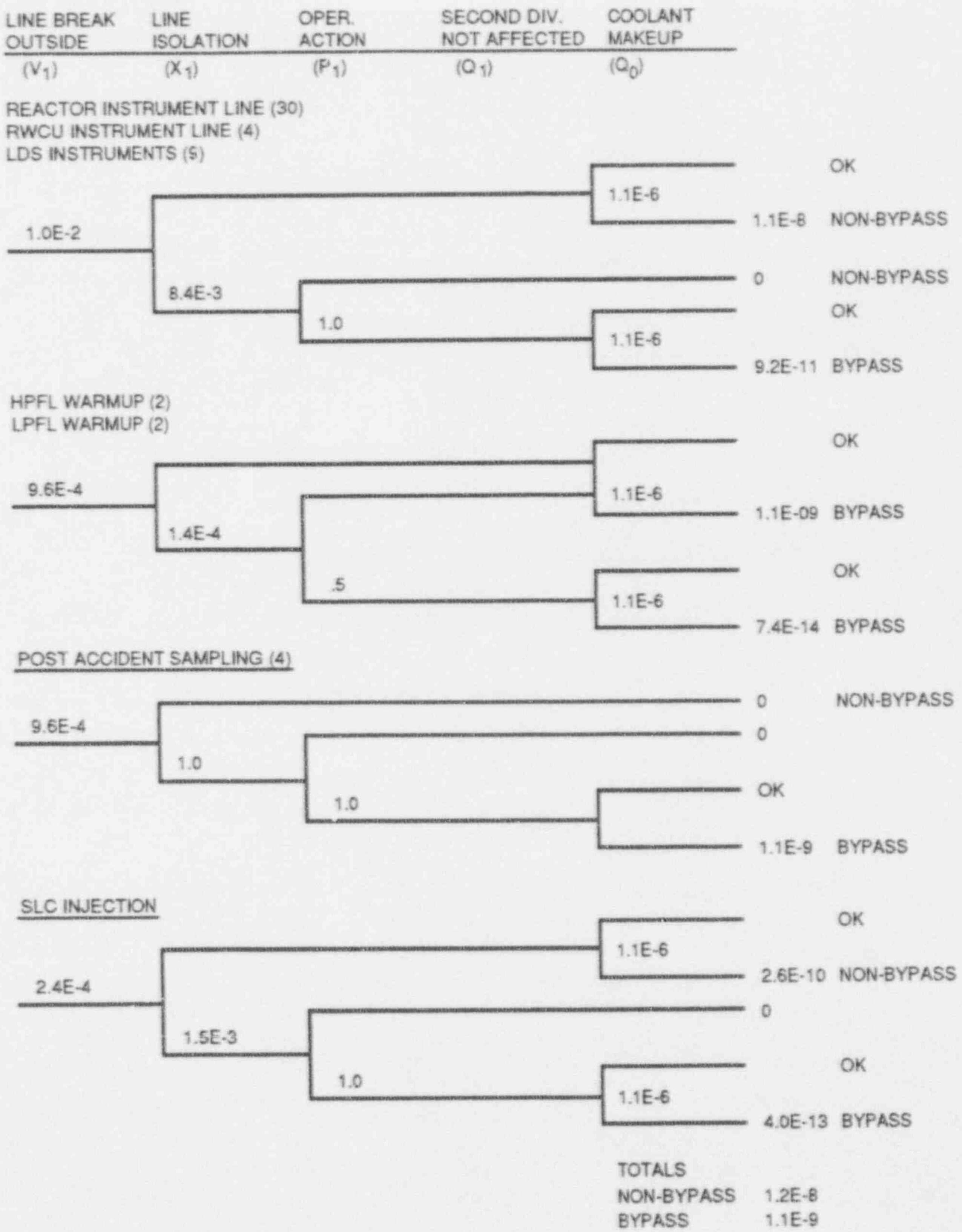


Figure 19E.2-20A SMALL LOCAS OUTSIDE CONTAINMENT

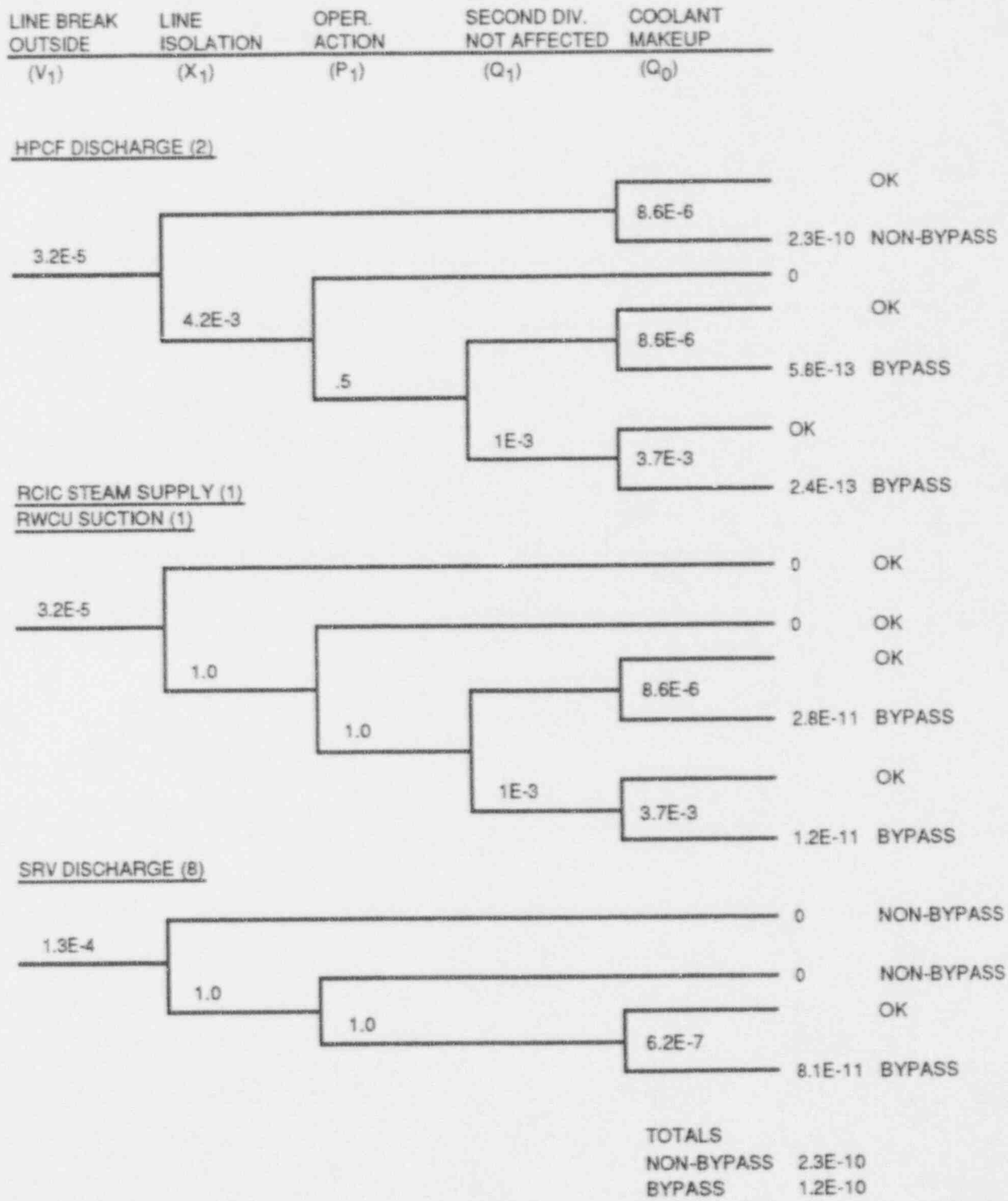


Figure 19E.2-20B MEDIUM LOCAS OUTSIDE CONTAINMENT

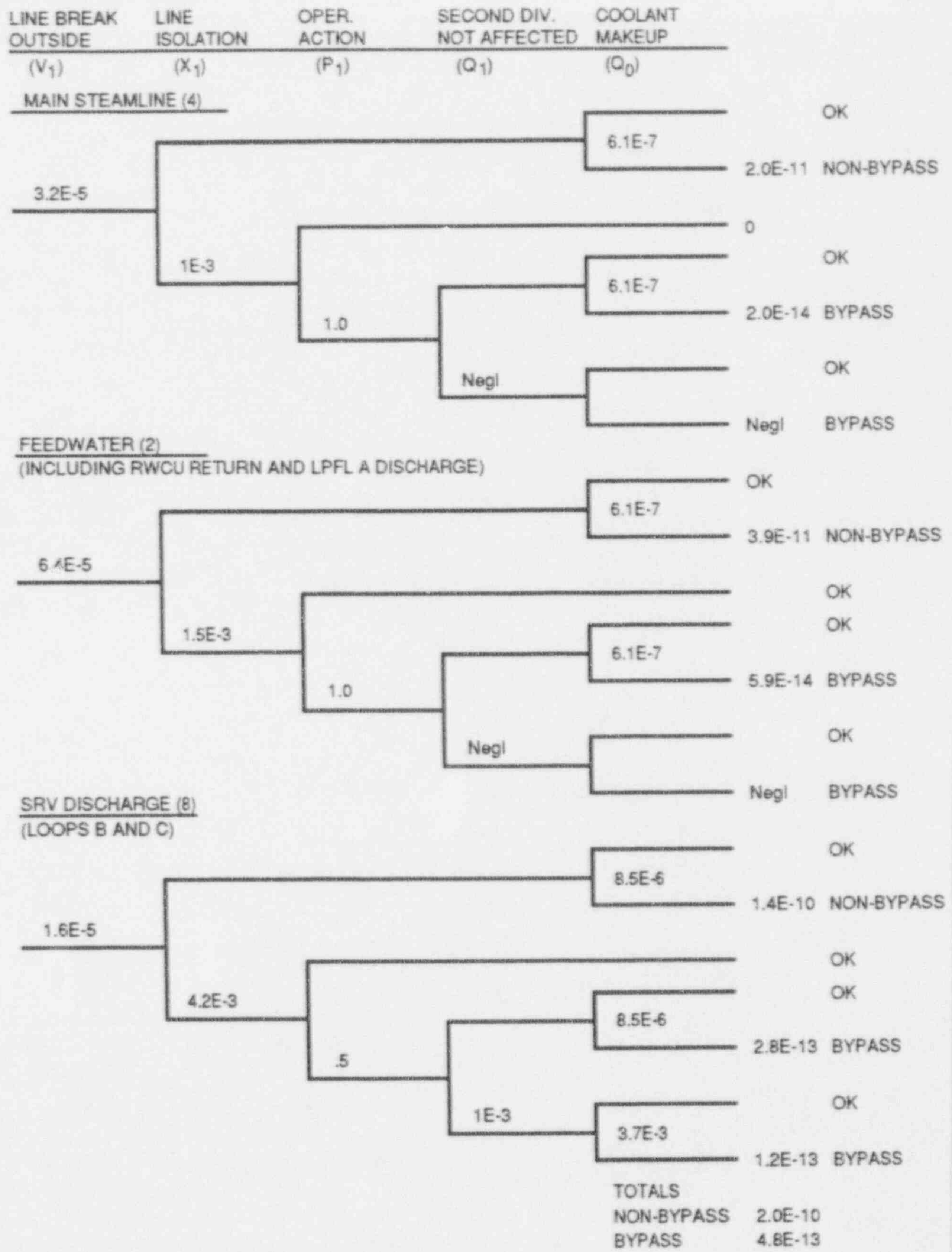
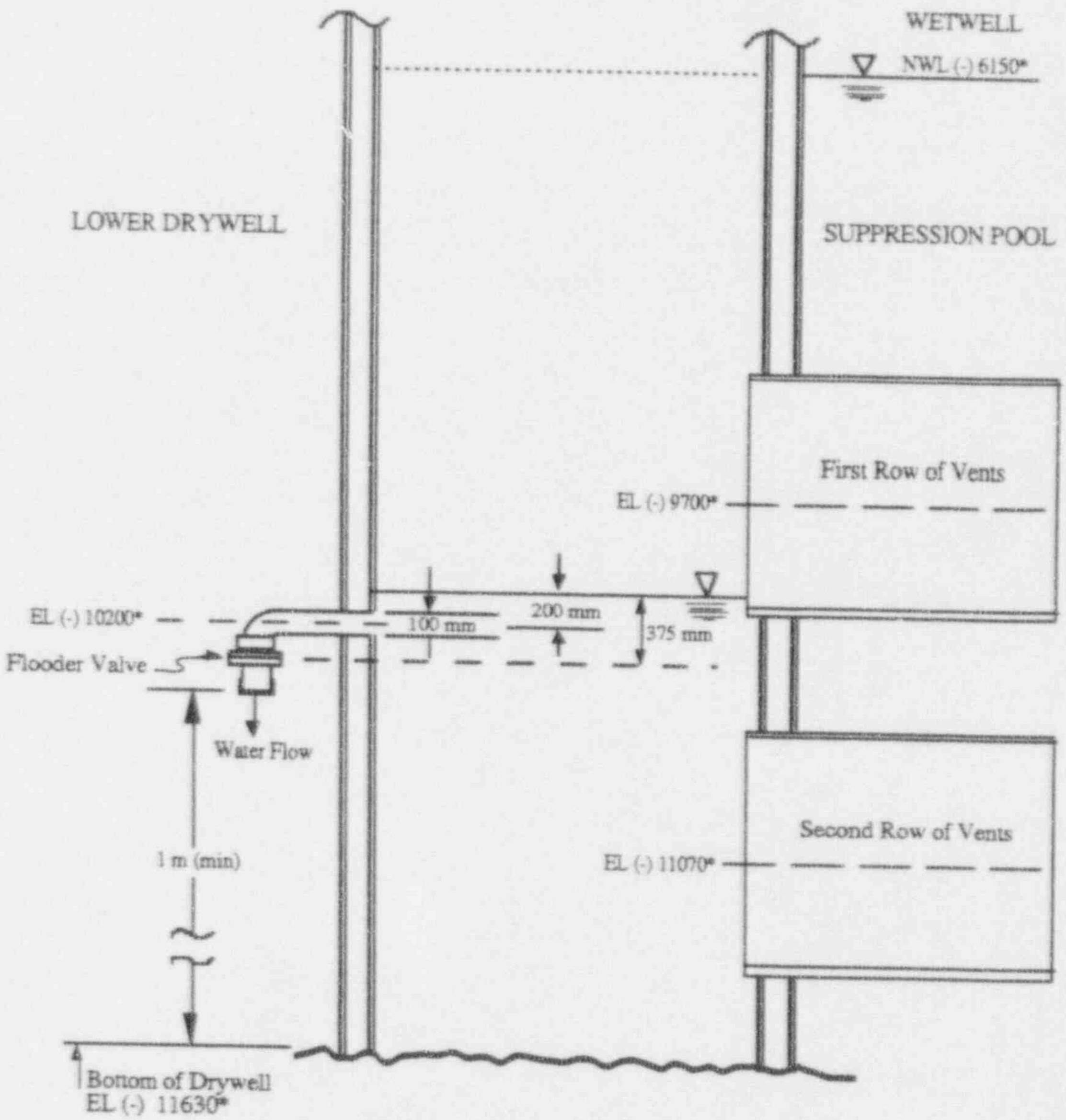


Figure 19E.2-20C LARGE LOCAS OUTSIDE CONTAINMENT



* Elevations based on RPV bottom at EL 0.

Figure 19E.2-23
LOWER DRYWELL FLOODER SYSTEM

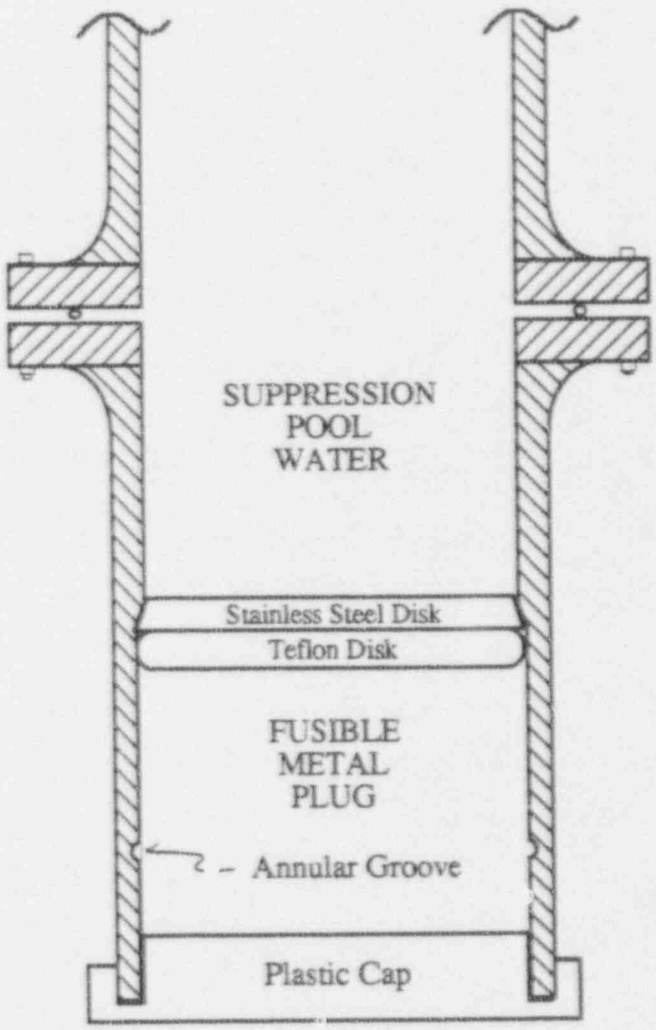


Figure 19E.2-24
FLOODER VALVE ASSEMBI Y

SECTION 19E.3

CONTENTS

SECTION 19E.3

TABLES

SECTION 19E.3

FIGURES

19E.3 CONSEQUENCE ANALYSIS

This section 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 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 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 (eg, 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 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 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 Figure 3-1 of Reference 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 98.6% of the risk calculated in the GESSAR II PRA (Reference 5). This accident data set is given in Table 19E3-1. 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 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 (see 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 tables describe 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 19 E.3.1. The plant input parameters are described below with the subsection of the SSAR 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:

Release Category Name, LNAME(j) - Abbreviated name given to release which results from the event. (Section 19E.2)

Release Probability, P(j) - the probability per year associated with release LNAME(j). (Section 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. (Section 19E.2)

DR(j) - duration of initial release (hr) 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). (Section 19E.2)

TLL(j) - warning time (hr) 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 6, Appendix I require a site area emergency be delayed when "delayed core with possible loss of coolable geometry" occurs.

FPR(j) - Sensible heat release rate in calories/sec 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. (Section 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 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 percent) 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 one 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 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 percent) 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 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 9.

3. Radiation Dose Goal. The probability of exceeding a whole body dose of 25 Rem at a distance of 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 sections.

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 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

1. Ritchie, L.T., et al, *Calculation of Reactor Accident Consequences Version 2 CRAC2: Computer Code*, NUREG/CR-2326, Feb 1983.

2. Ritchie, L.T., et al, *CRAC2 Model Description*, NUREG/CR-3552, March 1984.

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

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

5. *General Electric Company GESSAR II BWR/6 Nuclear Island Design* (22A7007), March 1980.

6. *Criteria for preparation and Evaluation of Radiological Emeregency Response Plans and Preparedness in Support of Nuclear Power Plants*, NUREG-0654.

7. Murley, T.E., *Advanced Boiling Water Reactor Licensing Review Basis*, Project No. 671, Aug 7, 1987.

8. *Accident Facts, 1988*, National Safety Council.

9. *1986 Cancer Facts & Figures*, American Cancer Society, 90Park Ave, New York, NY 10016.

Table 19E.3-1

Gessar Reactor Release Parameters

(see Subsection 19E.3.2.2 for definition of parameters)

Category	P(i)	TL(i)	DR(i)	TLL(i)	FPR(i)	RH(i)		Isotopic Release Fractions by Group						
							Group =	1	3	4	5	6	7	8
C1-TR-E2	2.09E-07	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	1.316E-06	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	6.92E-07	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-I2	7.59E-07	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-I3	1.66E-06	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

Table 19E.3-2

Population Density for Each Geographical Region (people per square mile). Taken from Reference 4, Table 3-2.

Radial Interval (mi)	Mean Population by Geographic Sector				
	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

Table 19E.3-2

Population Density for Each Geographical Region (people per square mile). Taken from Reference 4, Table 3-2.

Radial Interval (mi)	Mean Population by Geographic Sector				
	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

Table 19E.3-3

Evacuation parameters
(See Subsection 19E.3.1.3 for additional description of parameters)

Parameter	Strategy 1	Strategy 2
Fraction of Population Evacuating	0.95	0.05
Time Delay Before Evacuation - hrs.	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)	11260 (7)	0
Sheltering Radius - m (mi)	24140 (15)	0

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	4	1.5 mi.

Table 19E.3-6

Event Release Parameters
(see paragraph 19E3.2.2 for definition of parameters)

Accident	P(i)	TL	DR	TLL	FPR	RH	Release Fractions*		
							NG	Iodine	Cesium
NCL	1.34E-7	2.7	10	1.7	3.3E+5	37	.005	3.8E-6	5.1E-5
CASE 1	2.08E-8	20	1	19.2	3.3E+5	37	1	1.5E-07	1.3E-05
LCHPFSRN									
LCHPPSRN									
LBLCFSRN									
SBRCPFRN									
LCLPPFRN									
LCPFSRN									
CASE 2	$<10^{-10}$	19	1	18.2	3.3E+5	37	1	5.0E-06	5.0E-06
LCLPPFCR									
LCLPFSCR									
CASE 3	$<10^{-10}$	50	10	49.2	3.3E+5	37	1	2.8E-04	2.2E-03
LCHPFSD90									
CASE 4	$<10^{-10}$	20	1	19.2	3.3E+5	37	1	1.6E-03	1.6E-03
DF100FSR									
DF100PFR									
CASE 5	$<10^{-10}$	19	1	19.2	3.3E+5	37	1	6.0E-03	5.3E-04
LBLCPFRN									
CASE 6	$<10^{-10}$	19	10	18.2	3.3E+5	37	1	3.1E-02	7.7E-02
LCHPPSD90									
LBLCPFD90									
LBLCFSD90									
CASE 7	4.4E-10	20	10	19.2	3.3E+5	37	1	8.9E-02	9.9E-02
LCLPFSD90									
LCHPPFPM									
LCLPPFD90									
CASE 8	2.1E-10	2	10	1.2	1.0E+6	37	1	1.9E-01	2.5E-01
LCHPPFEH									
CASE 9	1.7E-10	23.6	10	12.2	3.3E+5	37	1	3.7E-01	3.6E-01
SBRCPFD90									

* Group 5-8 negligible release

Table 19E.3-7

Consequence Goals and Results

Goal	Numerical Goal	ABWR
Individual Risk	$<3.9 \times 10^{-7}$ (0.1%)	1.4×10^{-13}
Societal Risk	$<1.7 \times 10^{-6}$ (0.1%)	8.4×10^{-13}
Radiation Dose Probability at 25 Rem	$<10^{-6}$	$<10^{-9}$

Table 19E.3-8

Event Release Probabilistics-Rupture Disk Sensitivity Study

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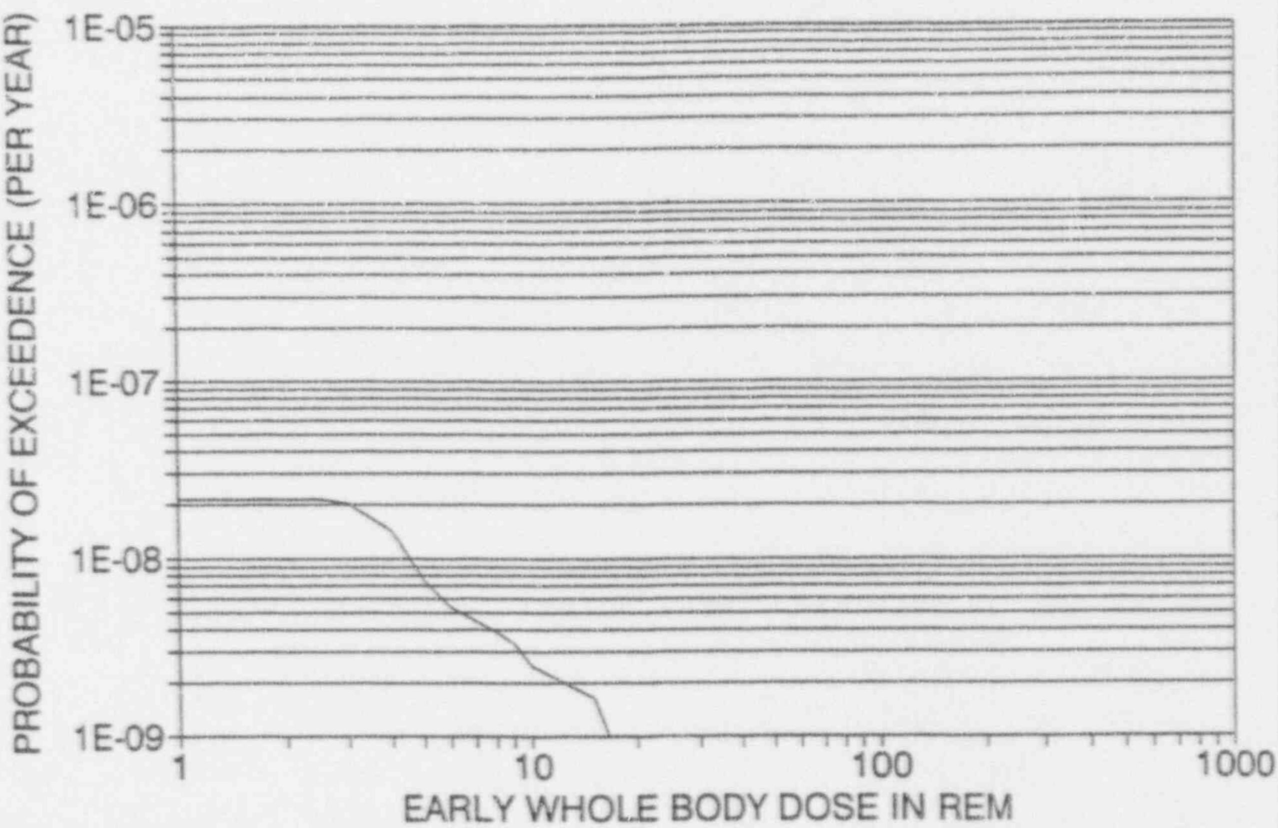

Figure 19E.3-1 WHOLE BODY DOSE AT 1/2 MILE AS PROBABILITY OF EXCEEDENCE

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Figure 19E.3-2 WHOLE BODY DOSE AT 1/2 MILE AS PROBABILITY OF EXCEEDENCE-RUPTURE DISK SENSITIVITY STUDY

APPENDIX 19E
ATTACHMENT 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 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, 7.3.1.1.1.1 and 19E.2.1.2.2.2.

Subsection 7.3.1.1.1.1 (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 requires 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 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.

Section 5.2.5 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 100 psig. (Determination of a new setpoint is not completed following strengthening of the drywell head.) 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. The following 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.2-1, 19EA.2-2, and 19EA.2-3) and three supplemental decomposition event trees (DETs) (Figures 19EA.2-4, 19EA.2-5, and 19EA.2-6). 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 Section 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 on 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 calculations (described in Section 19EA.2.2 and summarized on Table 19EA.2-1) 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 2).

Class	Pressure Range	Examples
Low	15 – 30 psia (Nominal = 1.5 atm)	Non ATWS sequences with operable DHR or with rapid core damage (i.e. all in-vessel injection failed)
Inter	30 – 45 psia (Nominal = 2.5 atm)	Large LOCAS with early failure of DHR. SBO with RCIC and failure of DHR.
High	> 45 psia (Nominal = 4 atm)	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 from elevated pressure contributes directly to containment pressurization. High RPV pressures

promote entrainment of the debris from the lower drywell and debris fragmentation.

Section 19EA.3.1 describes the mechanism for entrainment and the potential for debris dispersal in the ABWR.

Two pressure regimes are considered:

High (> 200 psia),

Low (≤ 200 psia).

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 Section 19EA.3.1, entrainment of the debris is an essential prerequisite for DCH. The entrainment of debris from the lower drywell occurs due 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 or 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 3 and 17). Even though the best-estimate studies indicate a small penetration failure is expected, this analysis addresses hole sizes up to 2 m^2. The probability of a larger failure is judged to be quite low based on References 3 and 17. Therefore, the probability of large lower-head failure modes has been decreased to 0.1 in this analysis. Thus,

P(Small) = 0.9,

P(Large) = 0.1.

These 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% 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 appear to be also 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 18). Table 19EA.2-2 compares the approximate debris masses released from the vessel at selected intervals after the vessel has failed. The ABWR DCH analysis column shows the values used for a *small* mass. It should be noted that debris entrainment will occur only until the vessel has depressurized to about 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 assumed that these results can also be applied to the ABWR. Additional discussion is provided in Section 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 18) indicate that the RPV depressurizes before any substantial amount of core material is expelled from the vessel:

P (HPME) = 0.8,

P (No HPME) = 0.2.

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 seriously 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), 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.2-7. 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 degree. 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 Section 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:

$$f_{frag} = f_{downcomer} \times f_{impact} \times f_{split\text{-}updw} \times f_{part} \quad (1)$$

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 (estimated uncertainty range 0.25 – 0.75),

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) (estimated uncertainty range 0.5 – 1),

$f_{split\text{-}updw}$ = the fraction of the gas flow which goes upward toward the upper drywell (as opposed to the wetwell) (estimated uncertainty range 0.5 – 1),

f_{part} = the fraction of the debris entrained into the upper drywell which enters in the form of small particles (estimated uncertainty range 0.75 – 1).

The uncertainty ranges for these four parameters were chosen based on the physical layout of the lower drywell along with engineering judgment. A median value of 0.5 has been chosen to represent the amount of material expelled from the vessel which exits the lower drywell. As described earlier, this represents the potential for material to be trapped in the stagnant region above the horizontal vent pipes. The ANL experiment described in Section 19EA.3 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, a median value of 0.5 appears reasonable for $f_{downcomer}$.

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 a median value of 0.75 would provide a conservative estimate for the fraction of debris that 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 Section 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, a median value of 0.75 was conservatively selected to represent the fraction of debris that 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 Section 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.2-8. This distribution for f_{frag} has the following characteristics:

median value	0.22,
80th percentile	0.34,
99th percentile	0.60.

Based on the above results, three regimes were selected to represent this parameter in the event tree:

Low ($f_{frag} \le 0.35$), Intermediate ($0.35 < f_{frag} \le 0.60$), and High ($f_{frag} > 0.60$). In the deterministic DCH pressure calculations described in Reference (1), the nominal values for f_{frag} used to represent these three regimes were 0.25, 0.5, and 0.75. The estimated probabilities for each of these discrete regimes for this parameter are shown below:

P (Low) = 0.8,

P (Inter) = 0.19,

P (High) = 0.01.

Figure 19EA.2-9 shows the comparison of the calculated cumulative distribution, as determined from the above parameters, and the distribution assumed for the 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 Section 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 below in Attachment 19FA.

Table 19EA.2-1
CONTAINMENT PRESSURE AT RPV FAILURE

MAAP Calculation	Pressure MPa	(psia)
LCLP-PF-D-M	0.13	(19)
LCLP-FS-D-L	0.14	(20)
LCHP-PS-D-N	0.14	(20)
LCHP-PF-P-H	0.14	(20)
SBRC-PF-D-H	0.24	(35)
LBLC-PF-D-M	0.29	(42)
NSCL-PF-D-H	0.13	(19)
NSCH-PF-P-H	0.13	(19)
NSRC-PF-D-H	0.43	(62)

Table 19EA.2-2
COMPARISON OF ASSUMED DEBRIS DISCHARGE WITH BWRSAR RESULTS

	Integrated Debris Discharged					
	ABWR DCH Analysis			BWRSAR		
Time	Zr	Metals	Oxides	Zr	Metals	Oxides
Vessel Failure	0	0	0	0	0	0
+2 minutes	6500	(1)	17,000	926	3880	0
+4 minutes	6500	(1)	17,000	2316	8172	0

(1) Only a small portion of the core plate was assumed to be added to the debris.

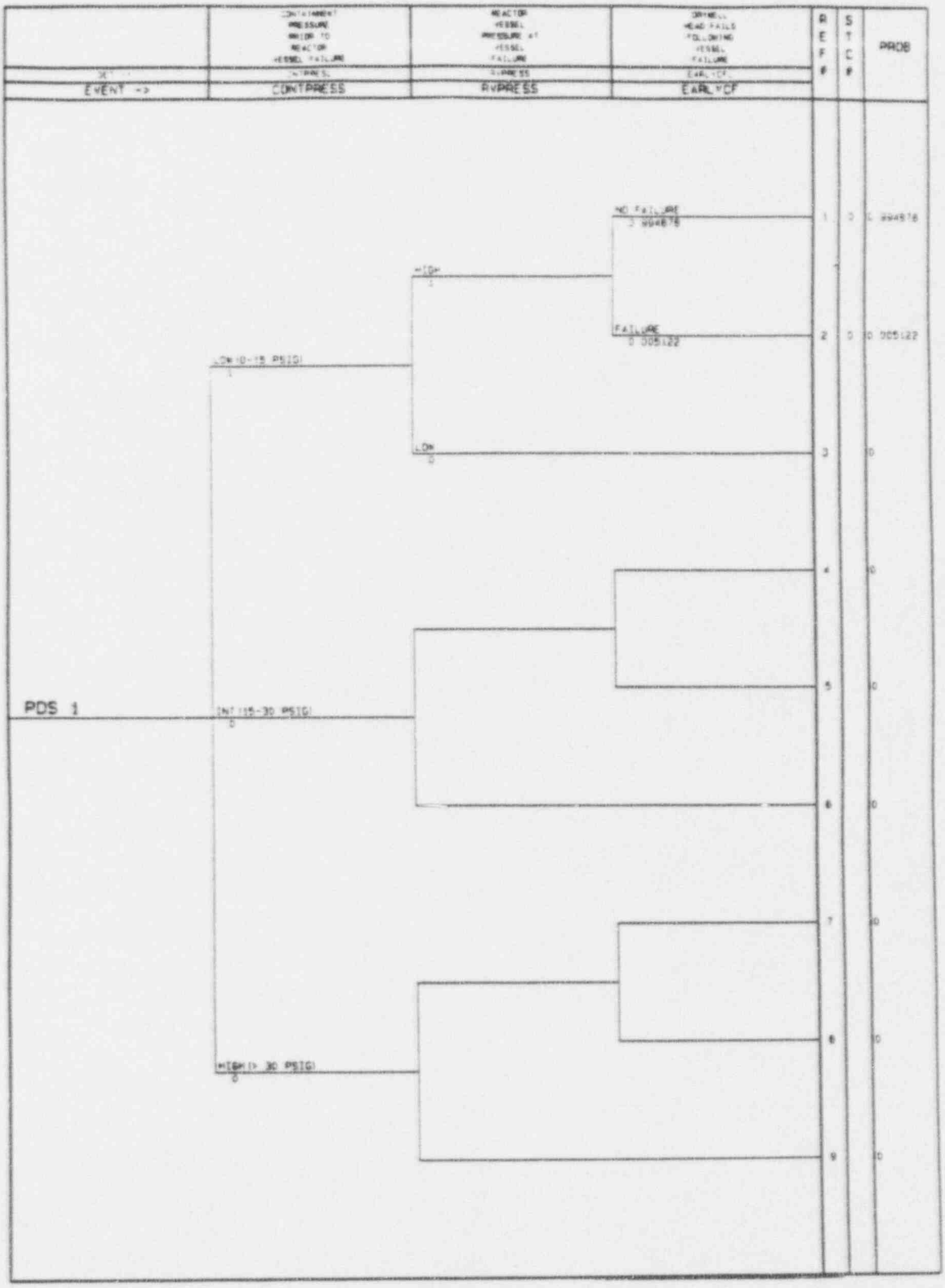


Figure 19EA.2-1
DCH EVENT TREE FOR SEQUENCES WITH LOW CONTAINMENT PRESSURE

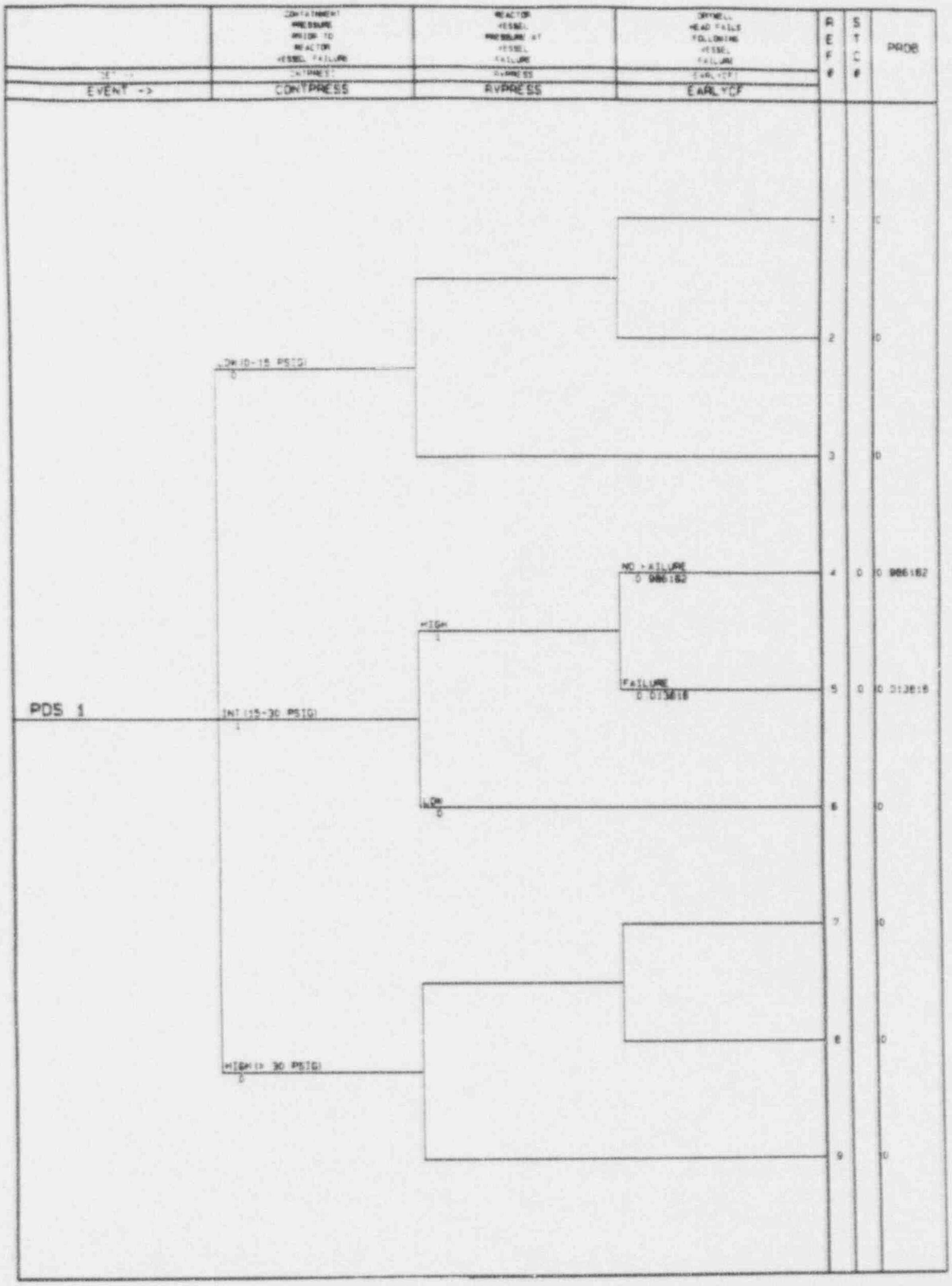


Figure 19EA.2-2

DCH EVENT TREE FOR SEQUENCES WITH INTERMEDIATE CONTAINMENT PRESSURE

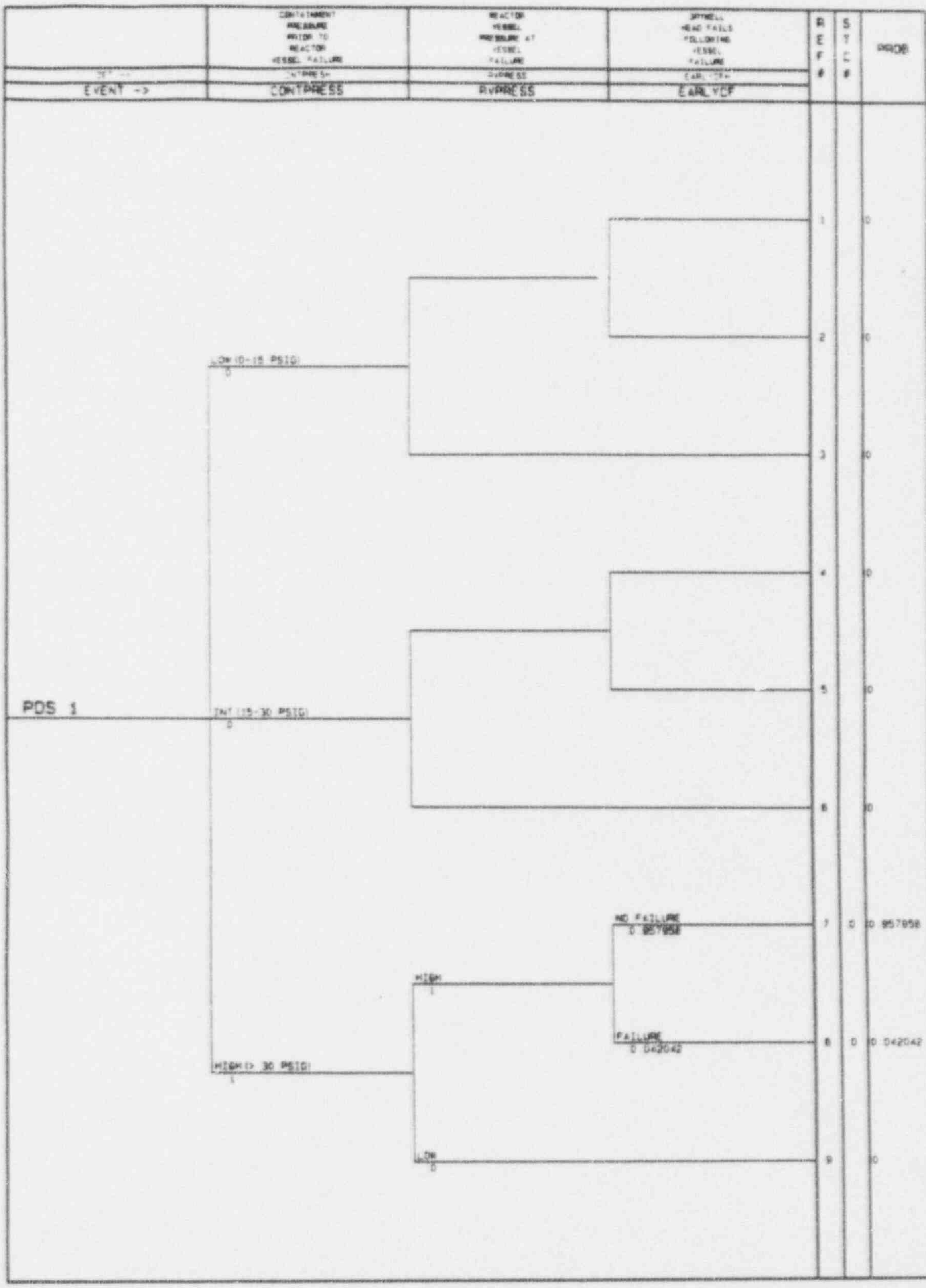


Figure 19EA.2-3

DCH EVENT TREE FOR SEQUENCES WITH HIGH COINTAINMENT PRESSURE

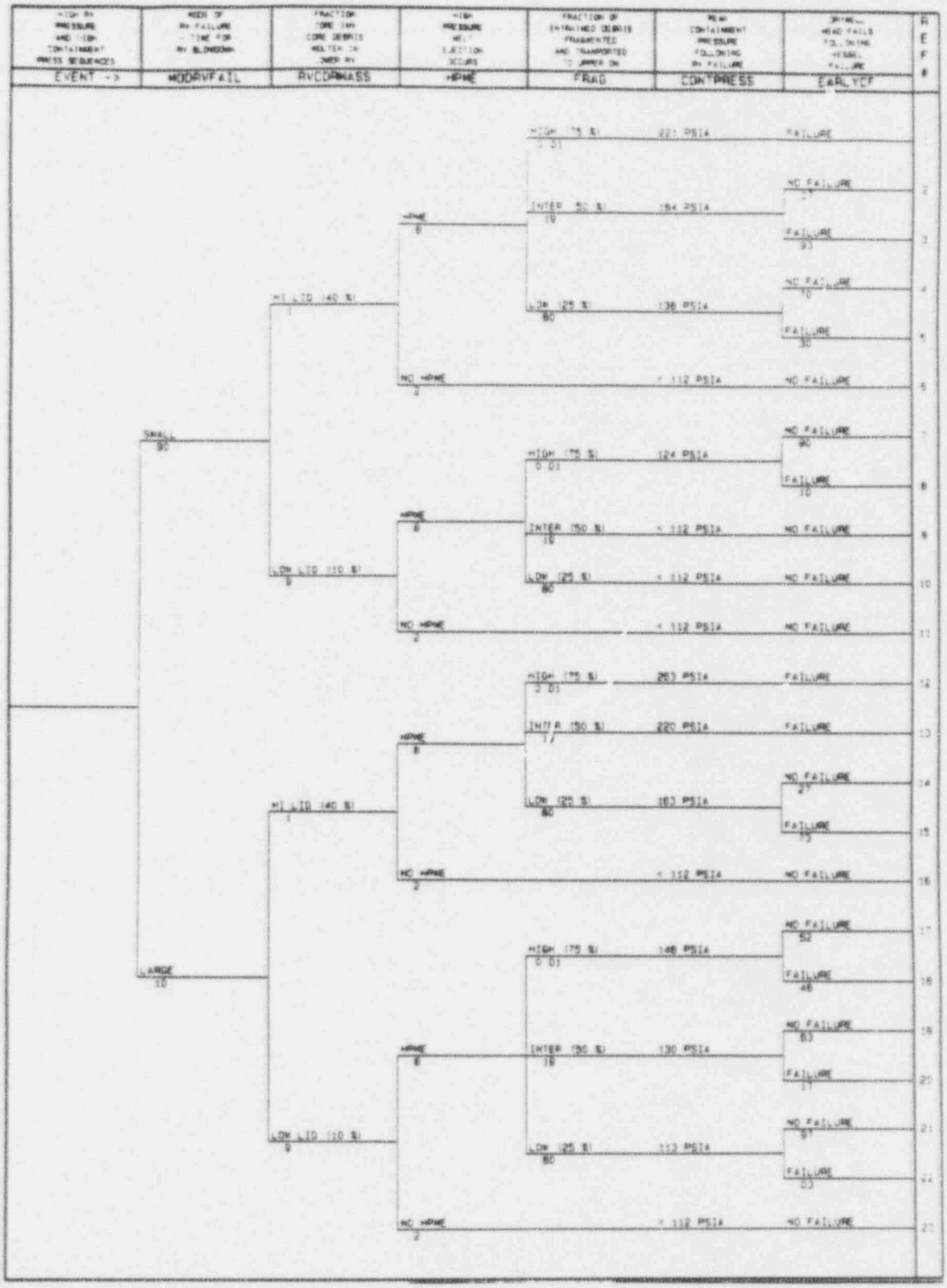


Figure 19EA.2-4

DET FOR PROBABILITY OF EARLY CONTAINMENT FAILURE - HIGH RV PRESS AND HIGH CONT PRESS SEQUENCES

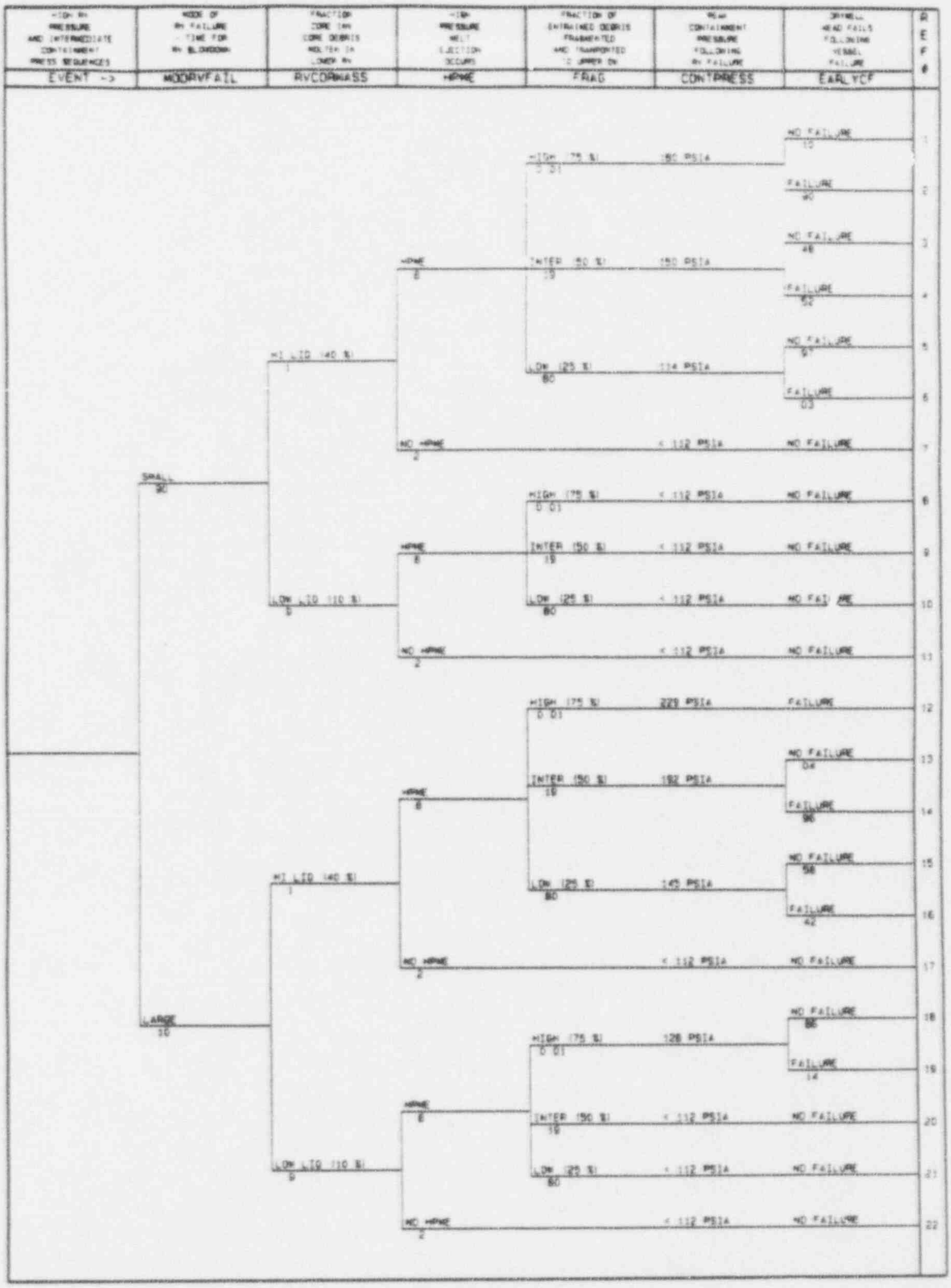


Figure 19EA.2-5

DET FOR PROBABILITY OF EARLY CONTAINMENT FAILURE - HIGH RV PRESS AND INTER CONT PRESS SEQUENCES

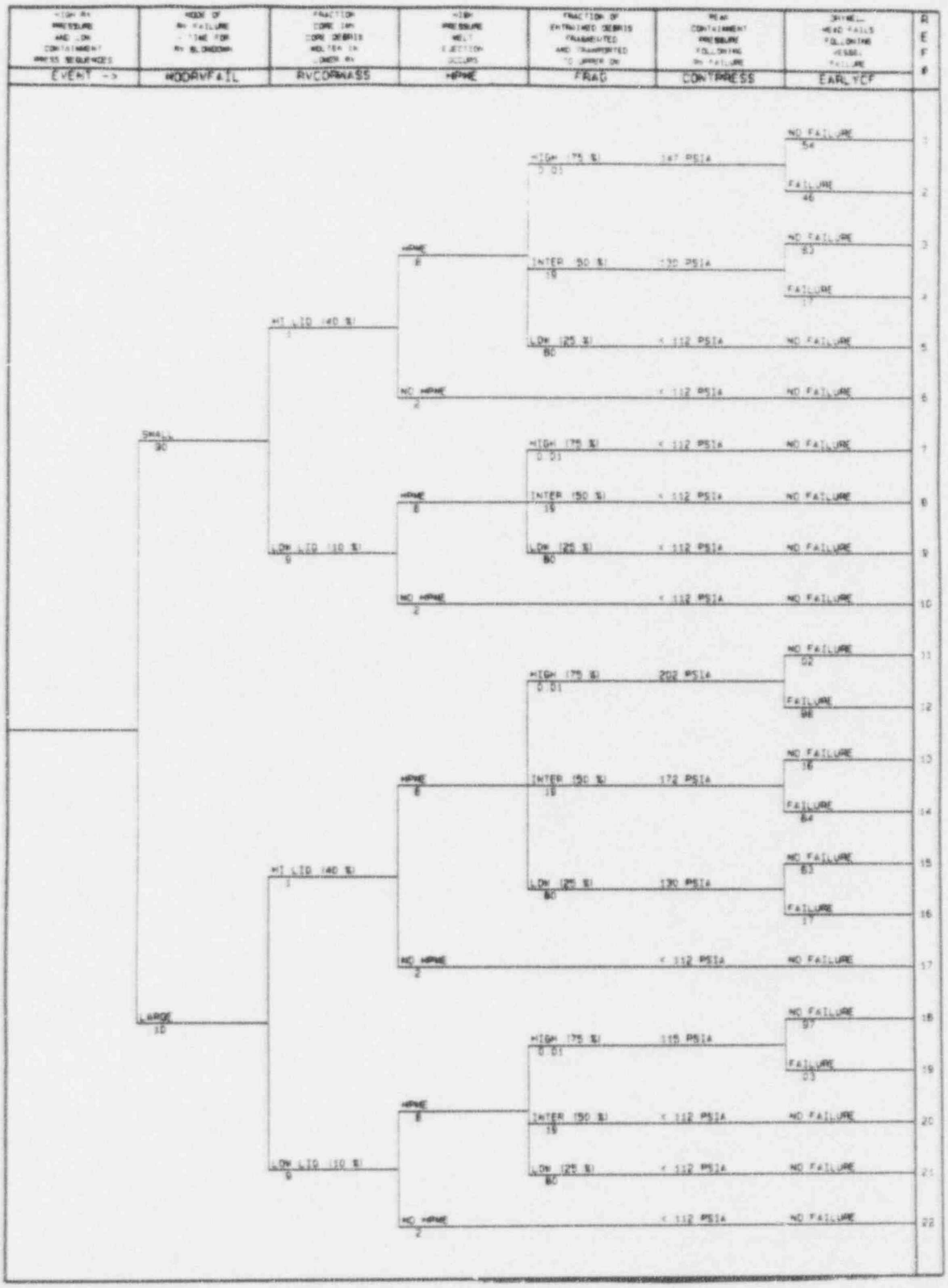


Figure 19EA.2-6

DET FOR PROBABILITY OF EARLY CONTAINMENT FAILURE - HIGH RV PRESS AND LOW CONT PRESS SEQUENCES

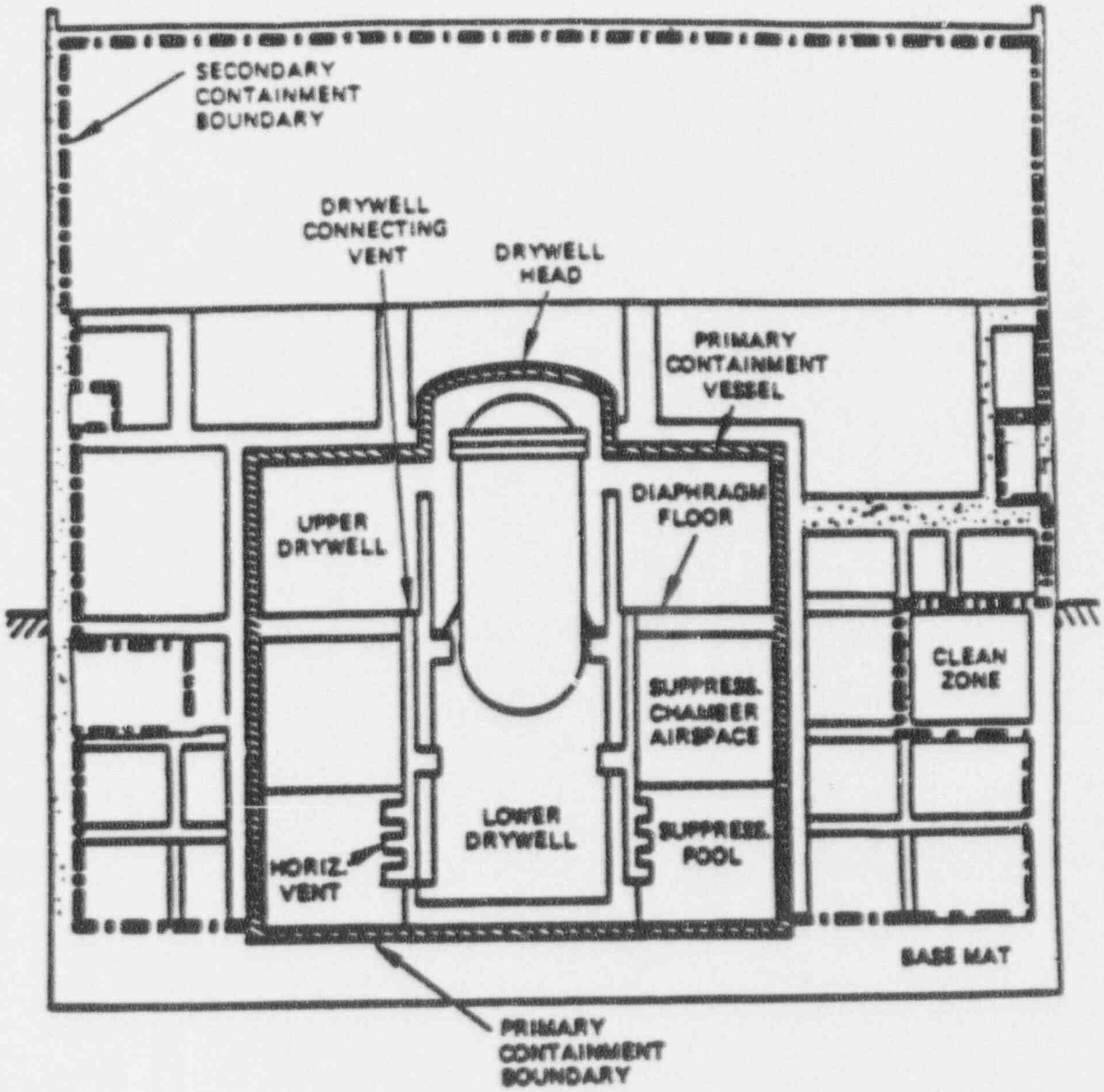


Figure 19EA.2-7
ABWR CONTAINMENT BOUNDARY NOMENCLATURE

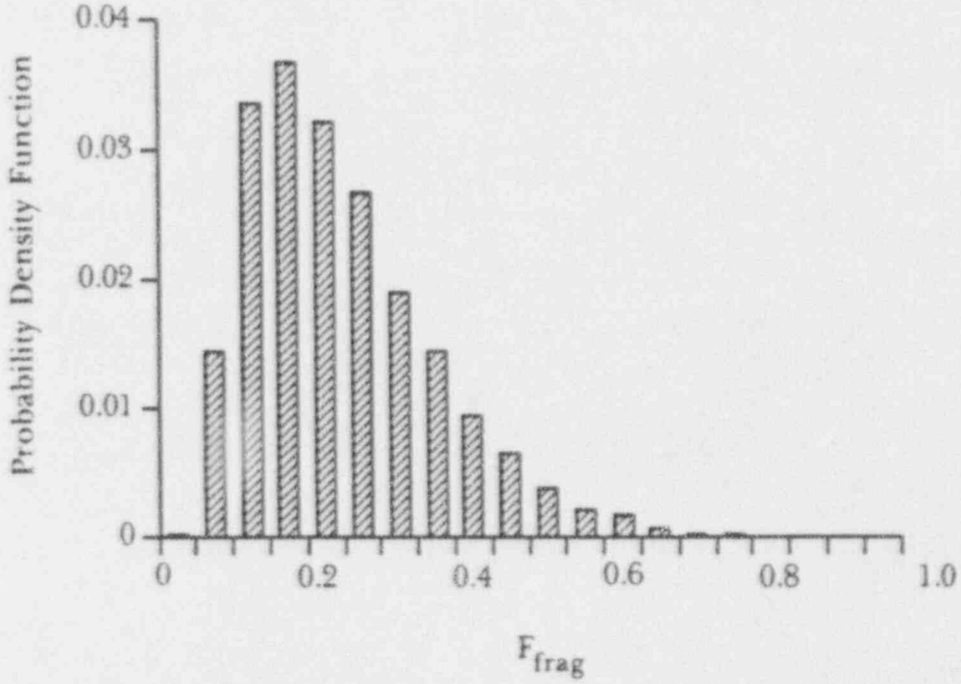


Figure 19EA.2-8

CALCULATED PROBABILITY DISTRIBUTION FUNCTION FOR DCH PARAMETER F_{frag}

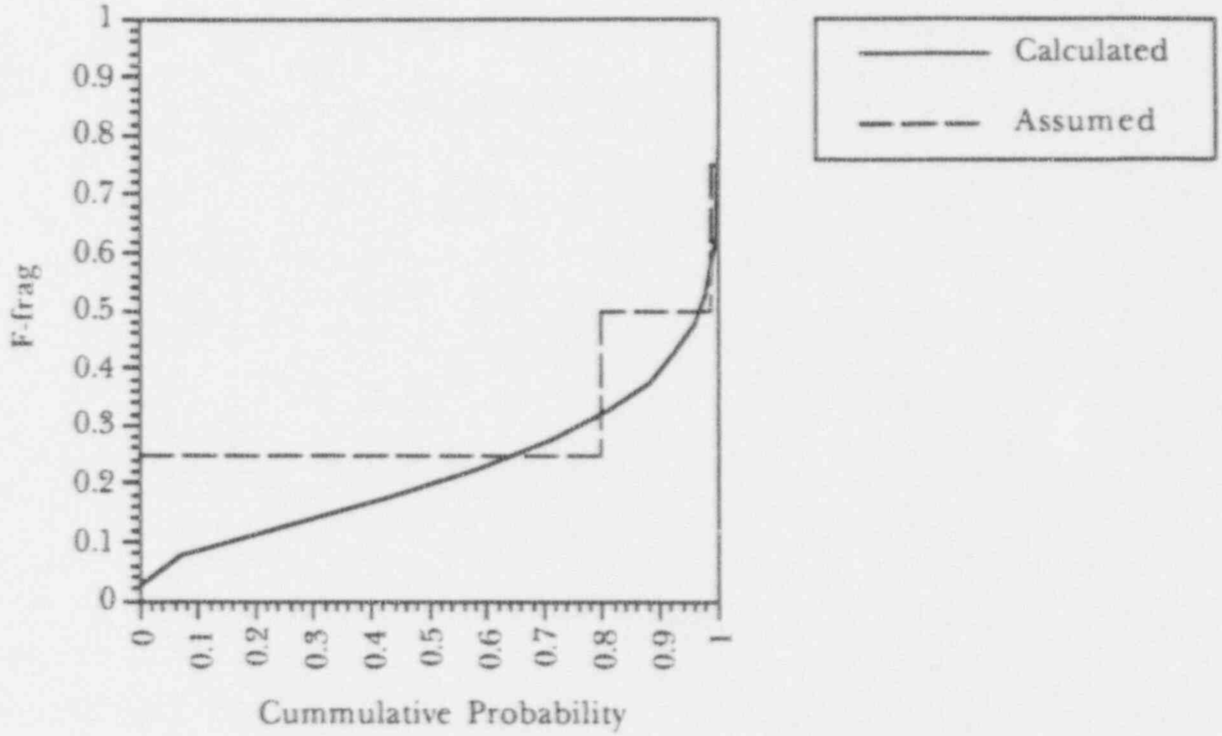


Figure 19EA.2-9

COMPARISON OF CALCULATED AND ASSUMED F_{frag} DISTRIBUTIONS

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 section 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 4). To lift a particle of radius r against the force of gravity requires a velocity given by:

$$(\rho_f - \rho_g)\frac{4}{3}\pi r^3 g = c_d \pi r^2 \frac{\rho_g u_g^2}{2} \qquad (2)$$

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:

$$We = \frac{2c_d \rho_g u_g^2 r}{\sigma} \qquad (3)$$

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 5) 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 6).

If we substitute Equation (3) into (2) and neglect the gas density compared to the liquid density we obtain:

$$u_g^4 = \left(\frac{4}{3}\frac{We}{c_d^2}\right)\frac{\rho_f \sigma_g}{\rho_g^2} \qquad (4)$$

Define the Kutateladze number by:

$$Ku = \frac{u_g}{\left(\frac{\rho_f \sigma_g}{\rho_g^2}\right)^{1/4}} \qquad (5)$$

so that:

$$Ku = \left(\frac{4We}{3c_d^2}\right)^{1/4} \qquad (6)$$

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:

$$Ku = 2.7 \qquad (7)$$

When substituted into Equation (5), this gives a velocity close to the experimentally measured value required to entrain particles off a free surface (Reference 7).

Consider a different situation in which gas is sparging a pool from below. If we consult Figure 19EA.3-1 (Reference 4), we note that the drag coefficient for a particle bed is in the range 20-40. If we use 30 and leave the We number at 8, we obtain:

$$Ku = 0.2 \qquad (8)$$

This does yield just the experimentally measured velocity required to fluidize a pool (Reference 8).

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 8, 9, 10, and 11). 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.3-2 shows a schematic view of the Zion reactor cavity. Experiments conducted with this geometry (Reference 9) 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 (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 (12), 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 (2), the drag coefficient necessary to explain the observed velocity threshold for debris dispersal of ~16 m/sec (Reference 11) 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.3-3 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 (8), this configuration is reminiscent of a pool sparged from below. Using this reasoning, Equation (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/sec. By consulting Figure 19EA.3-1, 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.2-7, 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:

$$\dot{P} = \frac{M_g R \dot{T}_g}{MW_g V} + \frac{\dot{M}_g R T_g}{MW_g V} \qquad (9)$$

where: M_g = Total mass of gas in containment (steam and non-condensible 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 13). Constant specific heat is used for the non-condensible 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 13). The pressurization of the wetwell due to any addition of non-condensible 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 increased linearly over the time constant for the event (discussed in Section 19EA.3.4). The fraction of the debris allowed to oxidize is a user input (discussed in Section 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 2500K. 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 14), 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 15) 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 16) 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:

$$m_e = m_0\left(1 - e^{-t/2.5}\right) \quad (10)$$

where: t = time in seconds since the blowdown begins;

(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 a debris removal e-folding time of 0.5 seconds, or the linear debris removal assumed over a 2-second period. This value appears to be conservative, but not remarkably so. Figure 19EA.3-4 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 Section 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; 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 Section 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. Section 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.2-4, 19EA.2-5, and 19EA.2-6.

19EA.3.6 Sensitivity to Various DCH Parameters

As indicated in Section 19EA.2, the DET methodology addresses the variation of several key DCH parameters. This section 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 constant — 2 seconds,

(4) Initial containment pressure — 1.5 atm.

The result of the analysis indicates a peak drywell pressure of 131 psia. Referring to the containment failure curve, this has a failure probability of about 0.17.

19EA.3.6.2 Dispersal Time Constant

The above case was re-run assuming that the dispersal time constant was 1 and 3 seconds, respectively. The results are:

	Peak Drywell Pressure (psia)
Base Case	131
Time Const = 1	163
Time Const = 3	110

Section 19EA.3.4 provides the justification for the 2-sec dispersal time and indicates that it may be somewhat conservative. However, a 50% change in the dispersal time resulted in only a 20% change in the peak containment pressure. This does not represent a very significant change.

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 133 psia vs. the base case value of 131 psia.

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 assumed in the DCH model 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). 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 111 psia vs 131 psia in the base case. This simply 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 U02 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 150 psia vs. the base case value of 131 psia. 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 results of the study indicate that this is true, although the rise in the peak pressure is small. The peak pressure for this scenario is 142 psia, as compared to a peak of 131 psia 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 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 15 psid relative to the base case pressure of 131 psia. 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 33 psid. 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 25 psid. 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 32 psid compared to 131 psia 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.

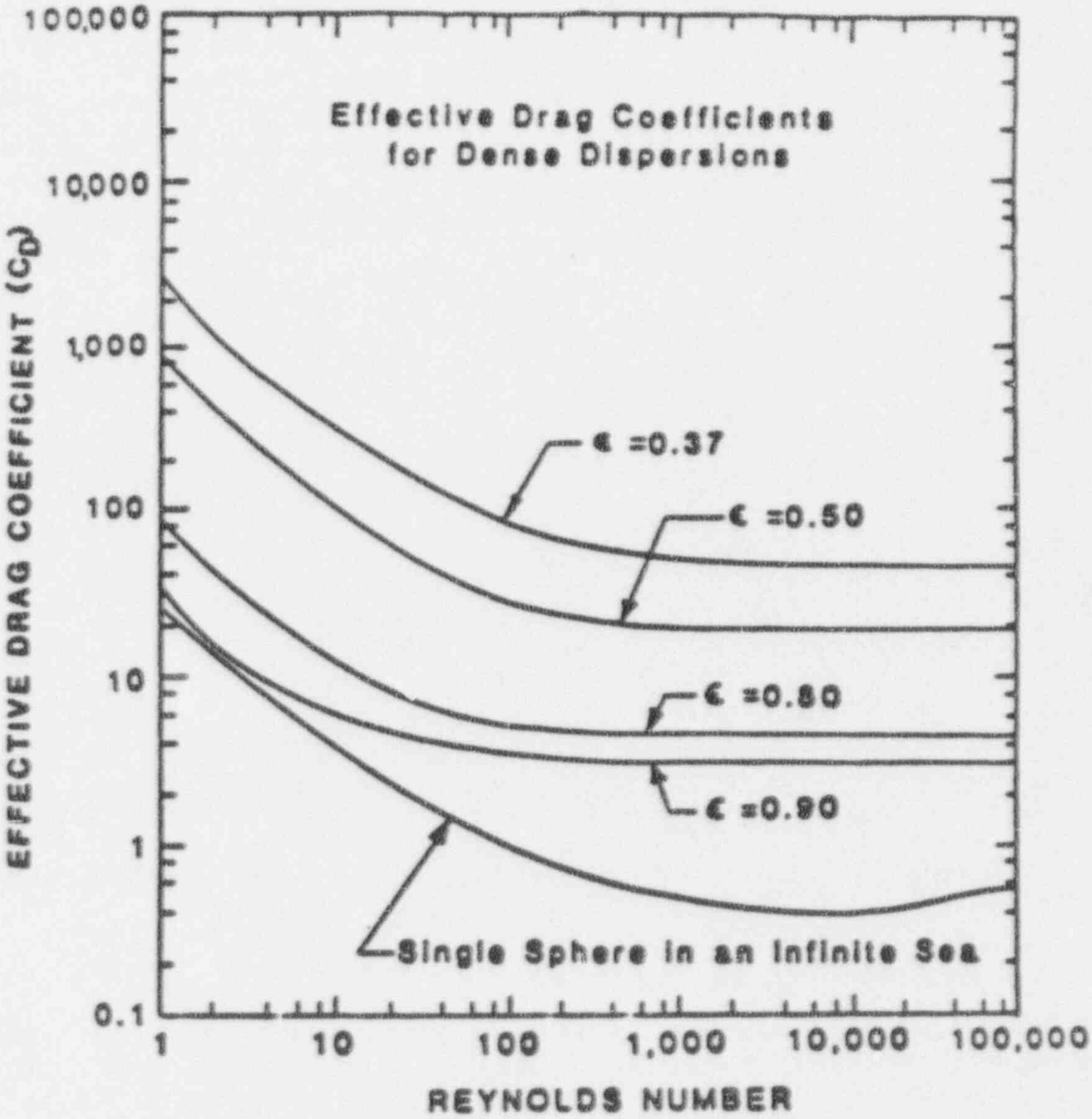


Figure 19EA.3-1
EFFECTIVE DRAG COEFFICIENT FOR DENSE DISPERSIONS

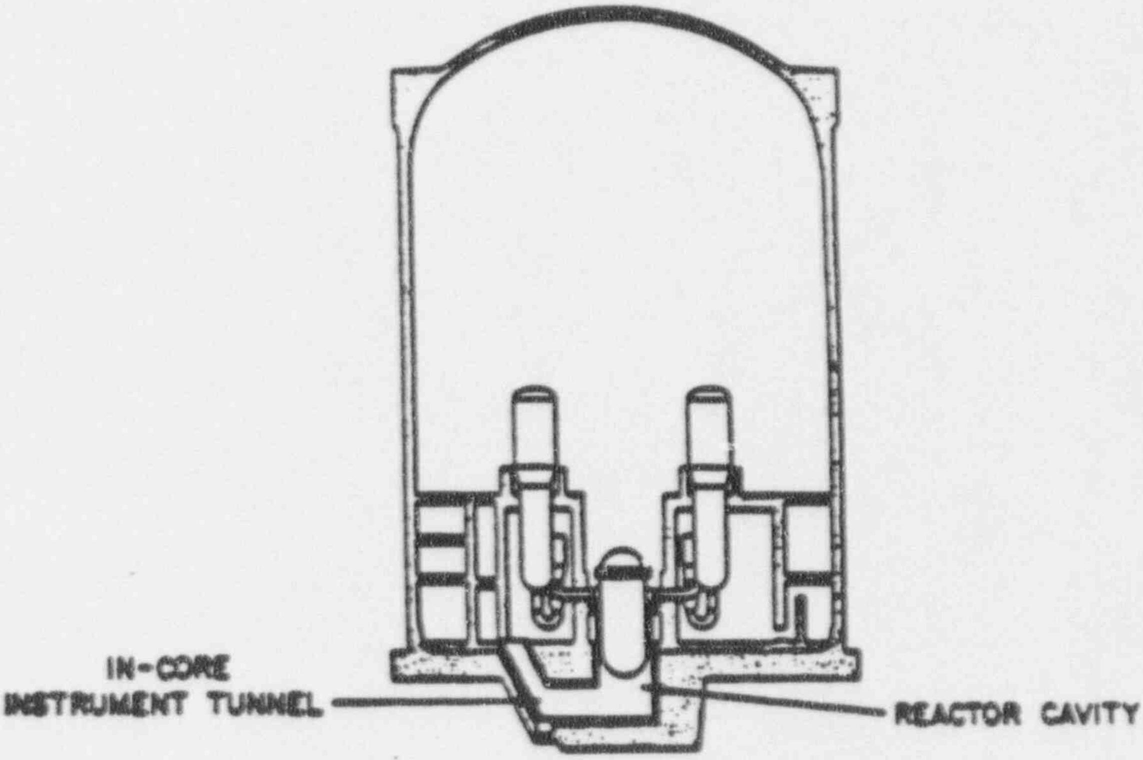


Figure 19EA.3-2
ZION REACTOR BUILDING

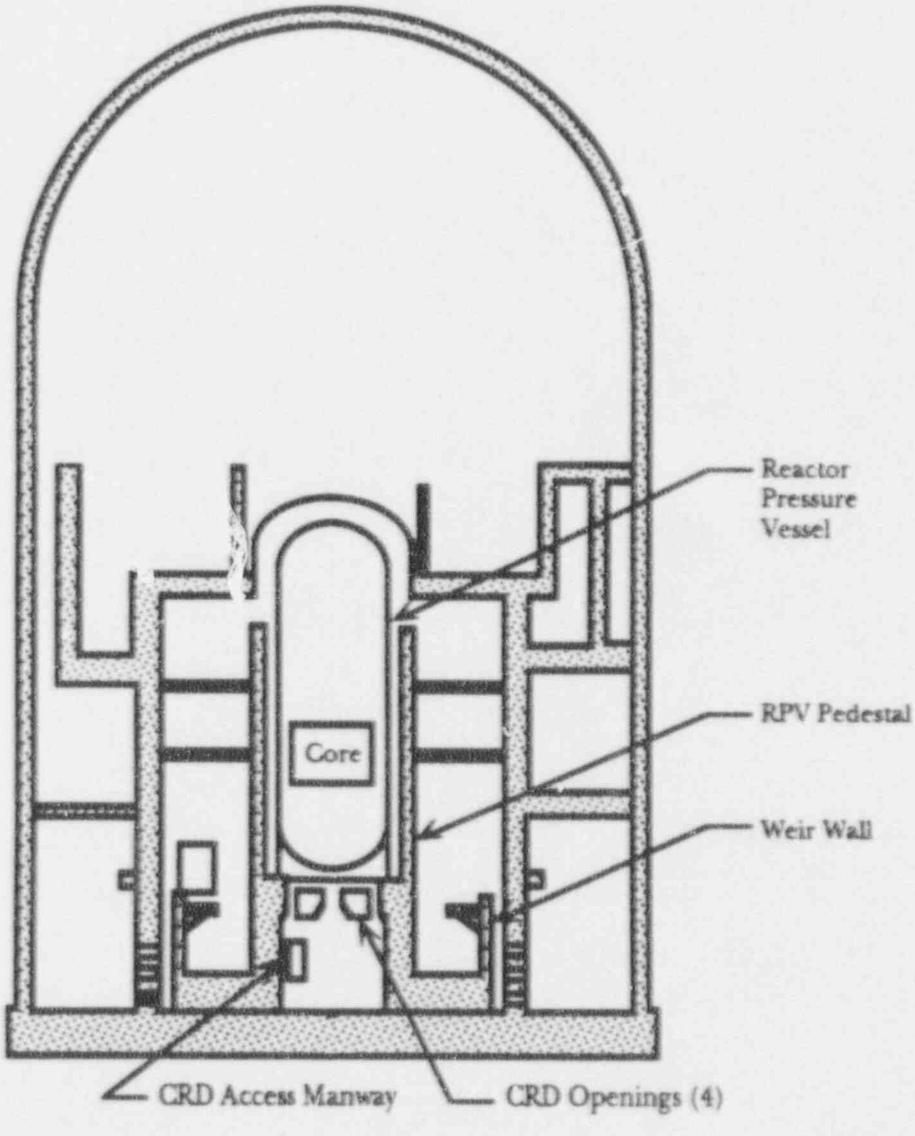


Figure 19EA.3-3
SCHEMATIC OF GRAND GULF CONTAINMENT

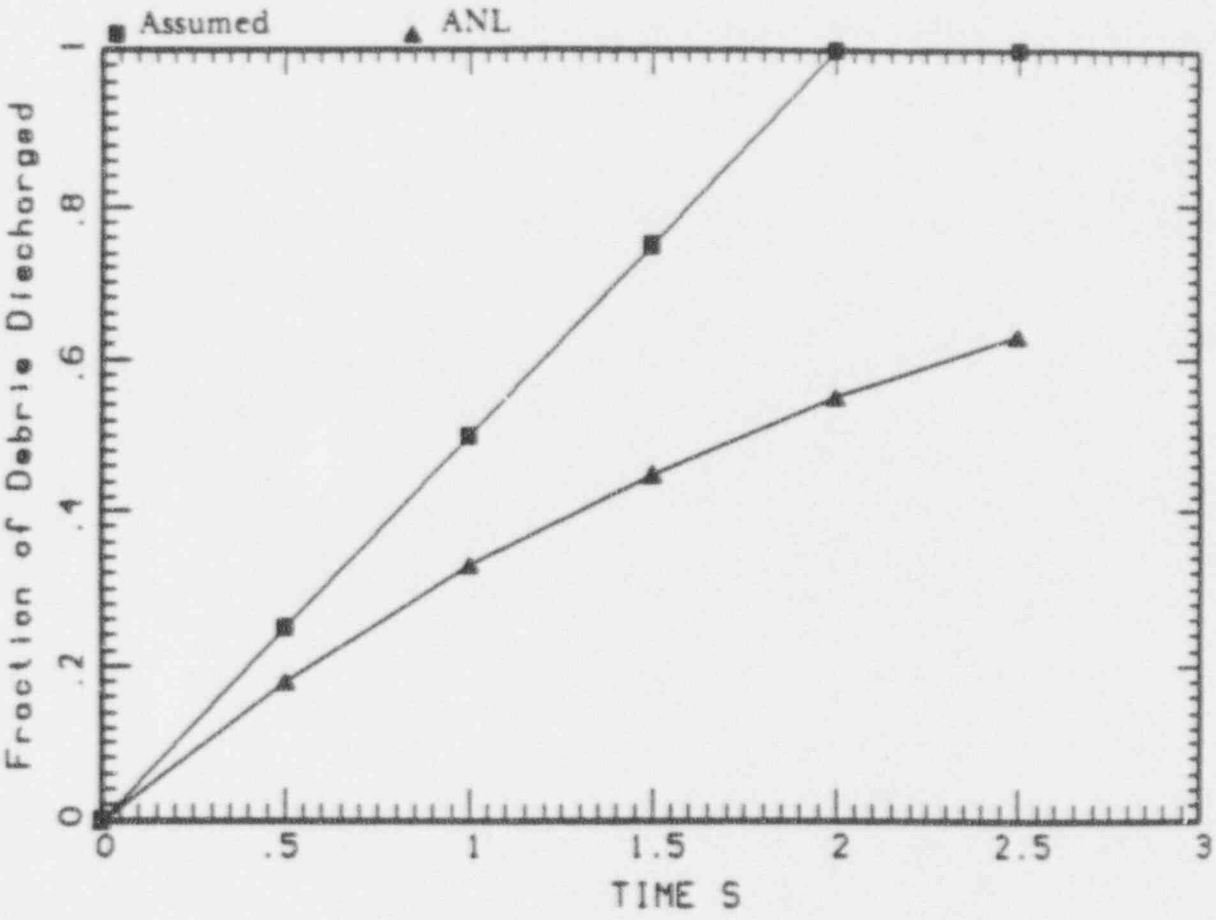


Figure 19EA.3-4
COMPARISON OF ASSUMED DEBRIS DISCHARGE TO ANL DATA FIT

19EA.4 SUMMARY OF RESULTS

19EA.4.1 Quantification of Decomposition Event Trees

The quantified decomposition event trees are shown in Figures 19EA.2-1 through 19EA.2-6. The relationship between the pressure and the cumulative probability distribution are shown in Figure 19EA.4-1. 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.4-1, 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. The analysis results are summarized below:

	Conditional Probability of Drywell Failure
High RPV Pressure – Low Cont. Pressure	0.005
High RPV Pressure – Inter. Cont. Pressure	0.014
High RPV Pressure – High Cont. Pressure	0.042

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 0.273. Furthermore, the conditional probabilities of high, intermediate and low pre-existing containment pressure at vessel failure for these high RPV pressure sequences are:

P (Low) = 0.9998,

P (Inter.) = 2×10^{-4}.

P (High) < 10^{-4}.

Combining the above probabilities results in a calculated probability of early containment failure (from direct containment heating) of 1×10^{-3} conditional on core damage.

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 30 psid. 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 was increased from 1×10^{-3} to 7×10^{-3}.

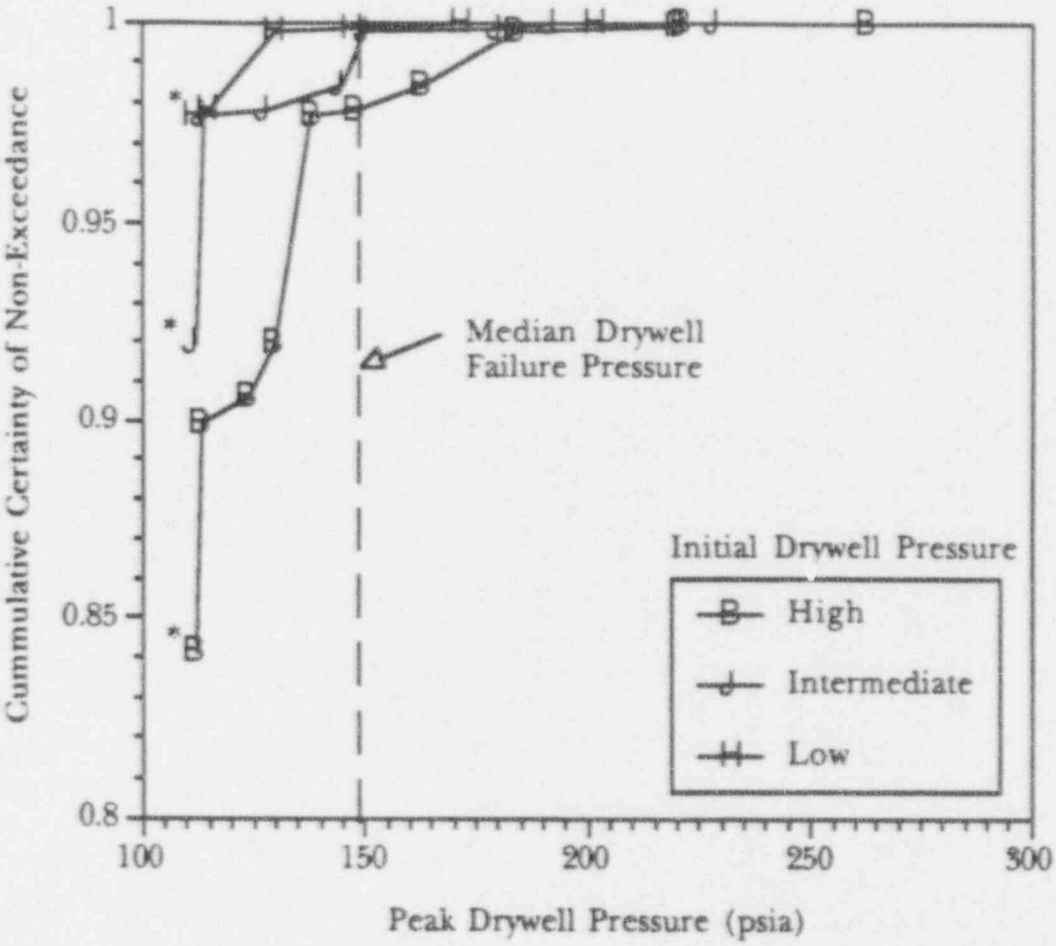
The second sensitivity case assumed that the containment would be at intermediate pressure for all cases. This represents potential uncertainties in the hydrogen production during the in-vessel portion of the accident. For this case, the containment failure probability due to DCH increases to 3×10^{-3}. Although the conditional probability of failure by DCH is a factor of 3 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 30 psid. The results show that for this conservative case, the conditional containment failure probability for DCH would be about 1.5×10^{-2}. Thus, even with these very conservative assumptions, the DCH containment failure probability of 0.015 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.4-1) 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 50% and 95% values from Figure 19EA.4-1, one may draw the dose curves shown in Figure 19EA.4-2. This figure shows, that for the accident frequencies and certainty levels of interest, DCH has no detectable impact on the offsite dose.



* Distributions truncated below 112 psia
Containment not expected to fail

Figure 19EA.4-1
CUMULATIVE DISTRIBUTION FOR PEAK PRESSURE DUE TO DCH

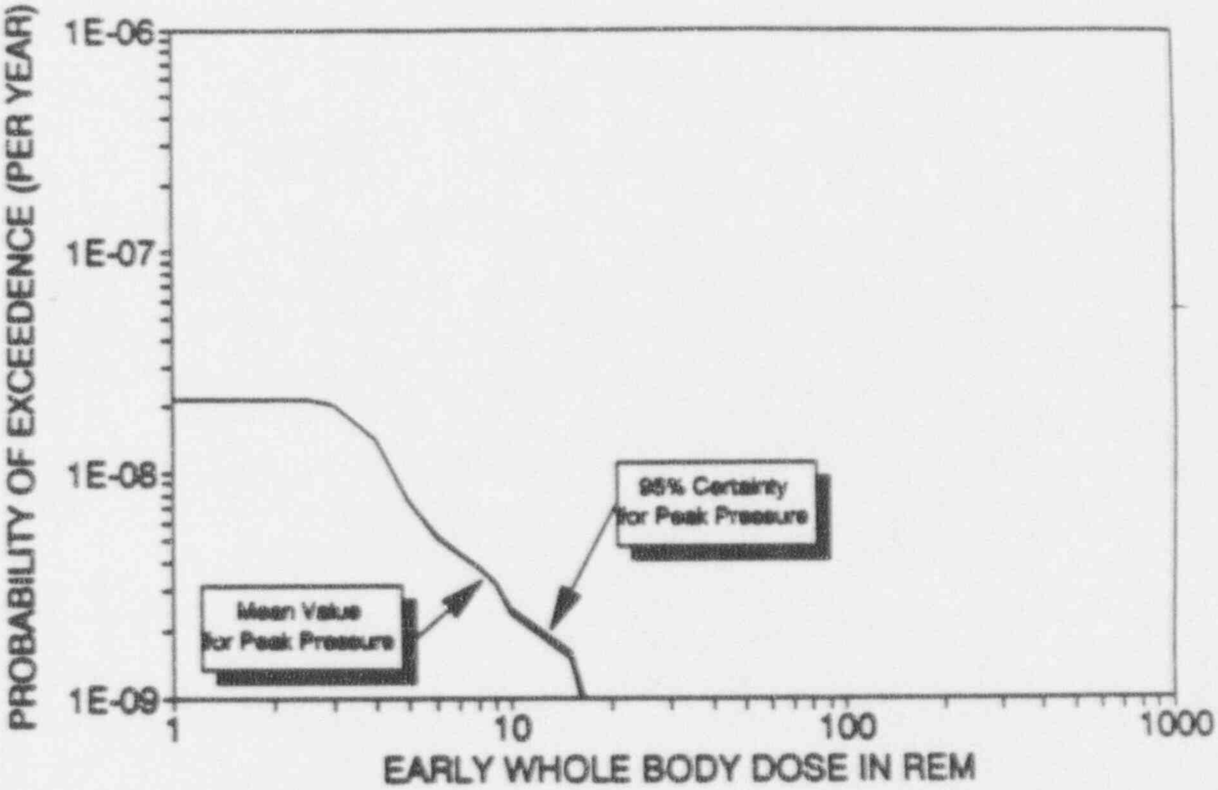


Figure 19EA.4-2
UNCERTAINTY IN WHOLE BODY DOSE AT 1/2 MILE DUE TO DCH

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.

19EA.6 REFERENCES

1. U.S. Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG-1150, June 1989.

2. Brown, T.D, et. al., *Evaluation of Severe Accident Risks: Grand Gulf Unit 1*, NUREG/CR-4551, Vol. 6 Rev. 1, Part 2, December 1990.

3. *BWR Lower Head Failure Assessment for CSNI Comparison Exercise*, EGG-EAST-9609, April 1991.

4. Fauske and Associates, *Key Phenomenological Models for Assessing Non-Explosive Steam Generation Rates*, IDCOR Technical Report 14.1B, June 1983.

5. R.E. Henry, personal communication.

6. M. Pilch, et. al., *Acceleration Induced Fragmentation of Liquid Drops*, NUREG/CR-2247, USNRC, August 1981 quoted in J. Sienicki and B. Spencer, *A Multifluid and Multiphase Flow and Heat Transfer for the Prediction of Sweepout from a Reactor Cavity*, Proc. Fourth Miami Inter. Symposium on Multi-Phase Transport and Particulate Phenomena, Miami Beach, December 15-17 1986.

7. S. Kutateladze, *Elements of the Hydrodynamics of Gas-Liquid Systems*, Fluid Mechanics-Soviet Research, 1, 4, 1972.

8. B. Spencer, S. Baronowsky, and D. Kilsdonk, *Hydrodynamic Sweepout Thresholds in BWR Mark III Reactor Cavi.y Interactions*, ANL/LWR/SAF-84-1, April 1984.

9. B. Spencer, et. al., *Sweepout Thresholds in Reactor Cavity Interactions*, ANL/LWR/SAF 82-1, April 1982.

10. B. Spencer, D Kilsdonk, and J. Sienicki, *Hydrodynamics Aspects of Ex-Vessel Debris Dispersal in Zion-Type Containment Designs*, ANL/LWR/SAF 83-1.

11. B. Spencer, D Kilsdonk, J. Sienicki, and G.R. Thomas, *Phenomenological Investigation of Cavity Interactions Following Postulated Vessel Meltthrough*, Proc. Inter. Meeting on Thermal Nuclear Reactor Safety, NUREG/CP-0027, Chicago, August 1982.

12. R.E. Henry, *Fission Product Release During High-Pressure Melt Ejection*, Task 3.4.6 report, Advanced Reactor Severe Accident Program, November 1988.

13. MAAP 3.0 B Computer Code Manual, EPRI NP-7071-CCML, Volume 2, November 1990.

14. Frederick Moody, *Introduction to Unsteady Thermofluid Mechanics*, 1990.

15. R.E. Henry, Modifications for the Development of the MAAP-DOE Code, DOE/ID-10216 Vol. IV, U.S. Department of Energy, November 1988.

16. S. Levy, *Debris Dispersal from Reactor Cavity During Lower Temperature Simulant Tests of Direct Containment Heating (DCH)*, Paper presented at 18th Water Reactor Safety Information Meeting, Gaithersburg MD, 1990.

17. S.A. Hodge and L.J. Ott, *Failure Modes of BWR Reactor Vessel Bottom Head*, ORNL/M-1019, Letter Report, May 10, 1989.

18. S.R. Greene, S.A. Hodge, C.R. Hyman, M.L. Tobias, *The Response of BWR Mark II Containment to Station Blackout Severe Accident Sequences*, NUREG/CR-5565, ORNL/TM-11548, May 1991.

APPENDIX 19E
ATTACHMENT 19EB
FCI

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 sections examine the available experimental data base 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 explosives.

19EB.1.1 Probability of Pre-flooded Lower Drywell

The configuration of the ABWR containment, shown in Figure 19EB.6-1, 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 flow from the upper drywell to enter the lower drywell, it would have to fall almost 9 m 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 2.2E6 kg (4.8E6 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) have a frequency of 9.7E-11, or 0.06% 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. The frequency of these sequences is 1.66E-10, or 0.11% 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 about 0.16%.

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 5.1E-10, equivalent to 0.3% 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 section 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 section 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 1 and 2) and Higgins (Reference 3), identified the requirements for considerable mixing of the molten debris and water. Higgins and Lemmon (Reference 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 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 msec.

The final, intermediate scale test performed at Sandia (Reference 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 msec. 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-condensible 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 configuration for a large 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 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.

Protoytpic 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.2-1. 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 Section 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.2 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.2-1. 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 facility 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 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 demonstrated in Section 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 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.2-2. 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.

Table 19EB.2-1
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/sec

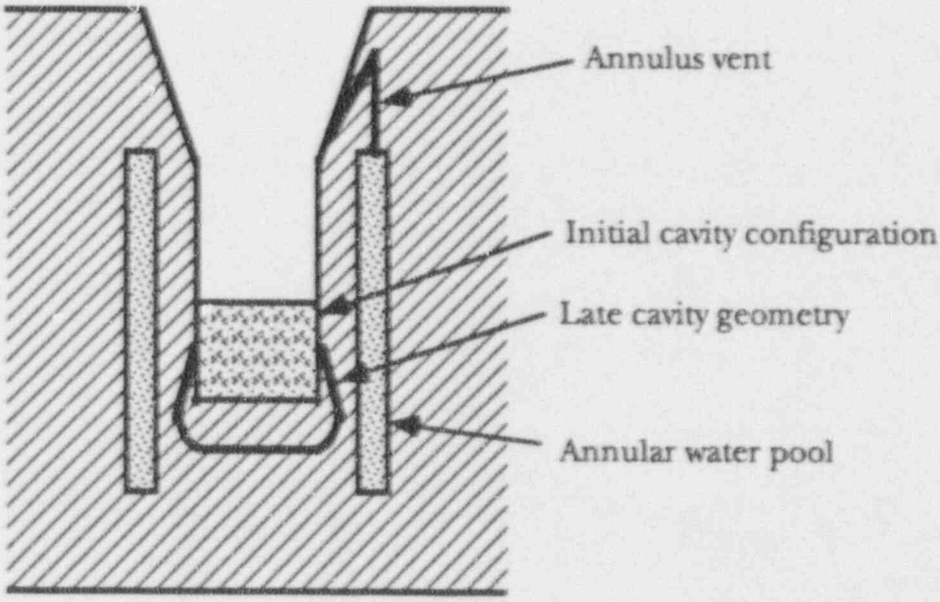


Figure 19EB.2-1
BETA V6.1 CONFIGURATION

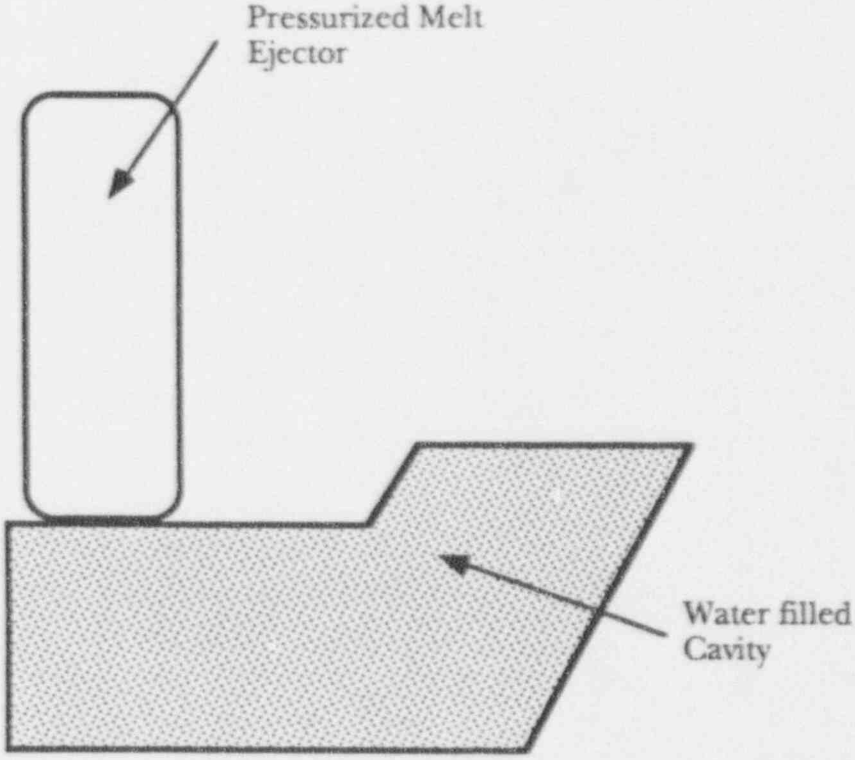


Figure 19EB.2-2
HIPS EXPERIMENTAL CONFIGURATION

19EB.3 EXPLOSIVE STEAM GENERATION

This section 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 refered 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, 2600K. Therefore, the maximum temperature in either the pour or stratified geometries will be 2600K. 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 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 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:

$$\dot{m}_{g,d} = \frac{HA_d\left(T_{ci} - T_\infty\right)}{h_{fg}} e^{-t/\tau_h} \tag{1}$$

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:

$$\begin{aligned} H &= H_c + H_r \\ &= H_c + \frac{\sigma\left(T_{ci}^4 - T_\infty^4\right)}{\left(T_{ci} - T_\infty\right)}\varepsilon \end{aligned} \tag{2}$$

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 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:

$$N\left(\frac{4}{3}\pi r^3\right) = \frac{M_c}{\rho_c} \tag{3}$$

where: ρ_c = density of corium.

The total steam generation rate of N corium droplets is:

$$\dot{m}_g = N\dot{m}_{g,d} = \dot{m}_{g,max} e^{-t/\tau_h} \tag{4}$$

where the maximum generation rate is:

$$\dot{m}_{g,max} = \frac{3M_c H\left(T_{ci} - T_\infty\right)}{\rho_c h_{fg} r} \tag{5}$$

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 section 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 10) determined the maximum pressure increase at the site of an FCI based on the steam generation rate given in Equation (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 (5) is:

$$\Delta P_{max} = 0.178\left[\rho_l \frac{\left(R_g T_\infty \dot{m}_{g,max}\right)^2}{R_o^4}\right]^{1/3} \qquad (6)$$

where: ρ_l = density of saturated water at the ambient pressure,

R_g = Universal 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:

$$\frac{4}{3}\pi R_o^3 = \frac{M_c}{\rho_c} + \frac{M_c c_c\left(T_{ci} - T_\infty\right)}{h_{fg}\rho_l} \qquad (7)$$

where: ρ_c = density of corium,

c_c = specific heat of corium.

The maximum pressure predicted by Equation (6) is shown in Figure 19EB.4-1 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.4-1. The pressure predicted by Equation (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 12).

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 simulates were reviewed in Section 19EB.2.2. Pulse widths were observed to be of the order 5 ms 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 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 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:

$$R_m = \frac{\sigma_y \Delta_w}{R_w} \tag{8}$$

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:

$$T = 2\pi\sqrt{\frac{\rho_w R_w^2}{E_w}} \tag{9}$$

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 ms.

The maximum response of elastic-plastic one-degree systems (undamped) due to rectangular load pulses is shown in Figure 19EB.4-2. 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 Section 19EB.4.1.

As discussed previously, the impulse duration of a FCI is expected to be approximately 5 ms, see Section 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.4-2 is 2.0. Using this ratio, the maximum pressure rise the pedestal can withstand is estimated to be 0.85 MPa. The uncertainty in pulse duration (assumed to be 5 ms) 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 0.85 MPa. The amount of corium which must participate in a FCI to achieve this pressure can be obtained from the analysis presented in Section 19EB.4.1 and summarized in Figure 19EB.4-1. 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 13). The smallest impulse load expected to fail the pedestal was reported to be 3.5 psi-sec (0.024 MPa-sec). 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 pressure predicted by the elastic-plastic analysis (0.85 MPa) can be applied for 28 milliseconds before an impulse load of 3.5 psi-sec 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 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 Section 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.

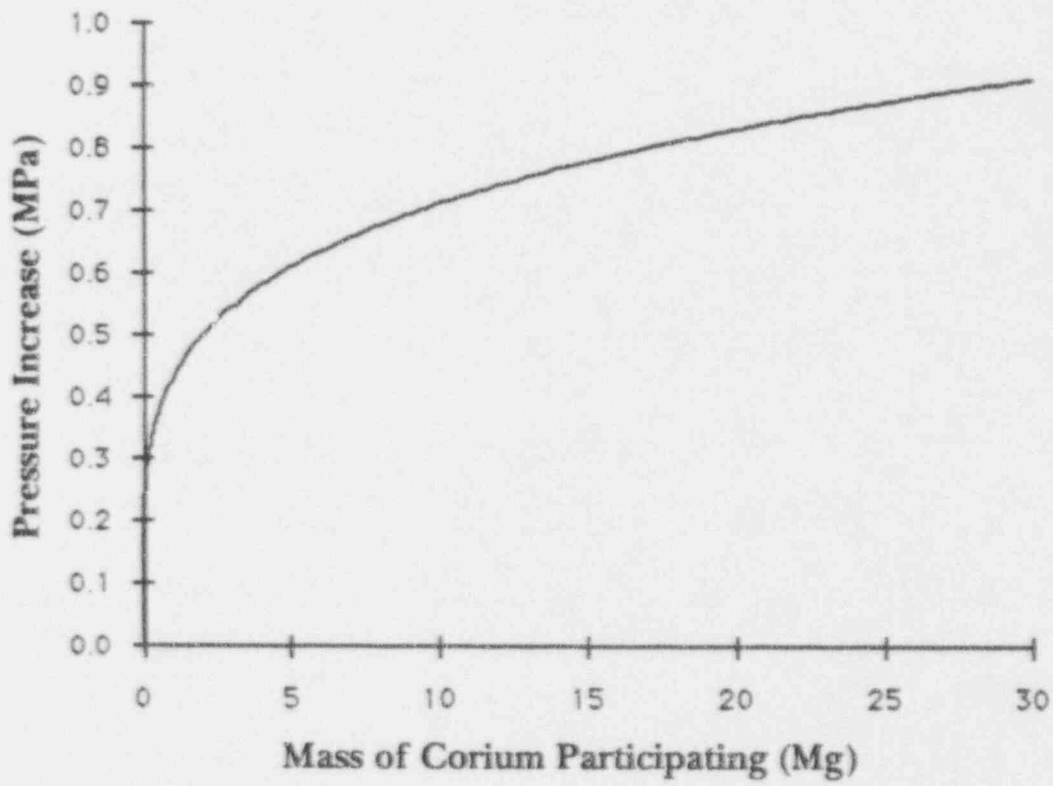


Figure 19EB.4-1
PEAK IMPULSE PRESSURE FROM FCI

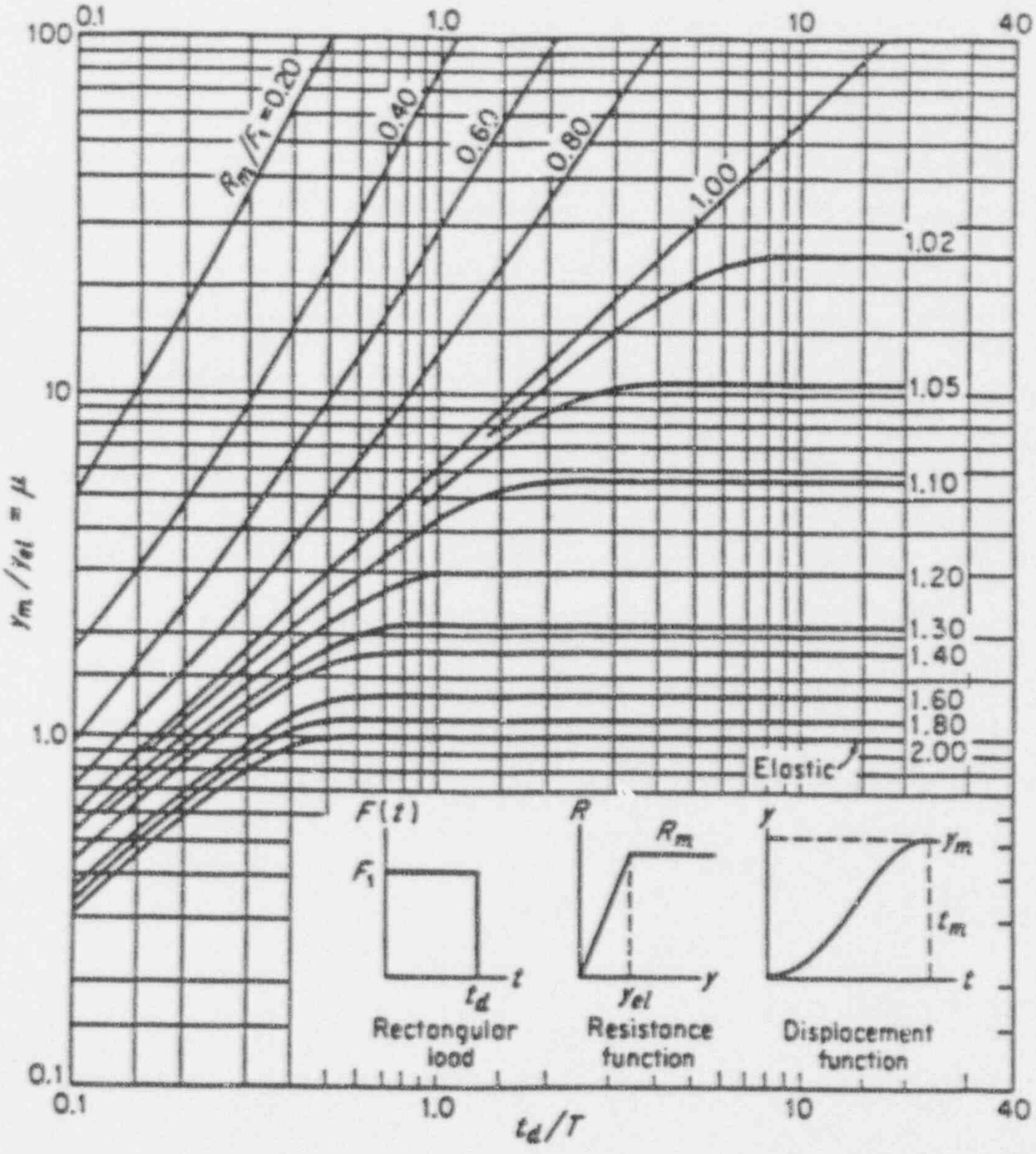


Figure 19EB.4-2

MAXIMUM RESPONSE OF ELASTIC-PLASTIC ONE-DEGREE SYSTEMS (UNDAMPED) DUE TO RECTANGULAR LOAD PULSES

(Reference 14)

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 section 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 10) used the steam generation rate determined in Section 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:

$$\dot{R}_\infty = \frac{3}{5}\left[\frac{5}{2}\,\frac{R_g T_\infty \dot{m}_{g,max}}{4\pi\rho_l R_\infty^2}\right]^{1/3} \tag{10}$$

where: R_∞ = equilibrium steam bubble radius.

It is equal to:

$$R_\infty = \left[\frac{4}{3\pi}\,\frac{M_c c_c \left(T_{ci} - T_\infty\right)}{h_{fg}\rho_g}\right]^{1/3} \tag{11}$$

where: ρ_g = vapor density.

Balancing the kinetic and potential energies of a water missile yields:

$$\Delta y_{max} = \frac{\dot{R}_\infty^2}{2g} \tag{12}$$

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.5-1 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 6.05 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, see Figure 19EB.5-1. 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.

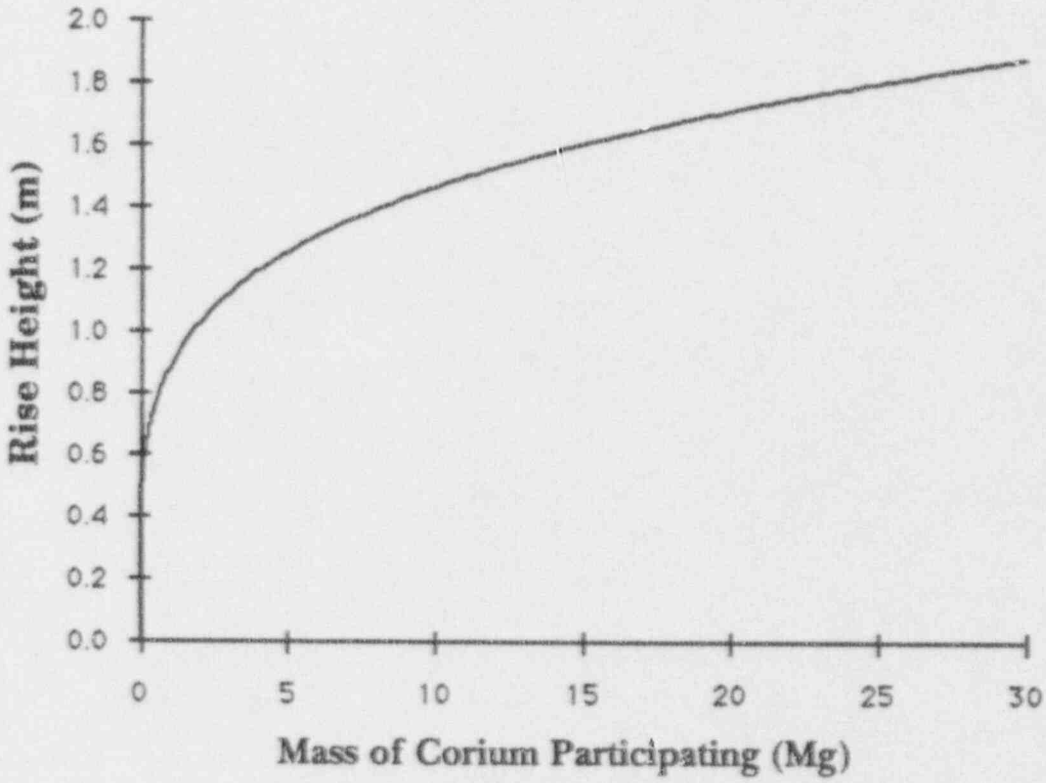


Figure 19EB.5-1
RISE HEIGHT OF WATER MISSILE

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.6-1. 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 Section 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, 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 (Section 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 vast 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 UO2 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.

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 15). 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.6-3, 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.6-2.

The flow from the wetwell into the lower drywell is d en by the difference in the water height, h, between the connecting vents and the flooder. The flow rate is given by:

$$\dot{m} = \rho A\sqrt{2gh} \qquad (13)$$

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 (.n^3),

g = acceleration of gravity (9.81 m/s^2),

h = drving 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 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, the height of water above the passive flooder is h= 4.753, which yields a maximum flow rate of $\dot{m}$= 780 kg/s. The flow rate will typically be less than this maximum because the DW press is greater than WW and the first row of vents are 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 does not rely on AC power, so it is available even during a station blackout event. The ECCS is dependant 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 3800 gpm (230 kg/s), and three LPFL systems with flow of 4200 gpm (265 kg/s). The RCIC system is not considered since the vessel will be depressurized. The total water addition rate to the lower drywell is 1250 kg/sec.

119EB.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 16 and 17). Once there is some flow through a penetration, the molten material will begin to ablate the hole. Since there is little 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 (as well as MELCOR) uses the Pilch model for the ablation of the penetration (Reference 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 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.6-3 and 19EB.6-4. The maximum rate of debris ejection from the vessel is about 6000 kg/sec. Assuming this material quenches as it is ejected, the steam generation rate is about 2800 kg/sec.

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/sec. 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 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/sec (see Section 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.6-1. 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, even though this region has a much higher ultimate strength than the drywell head (see 19F.3.1). 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 (21 psid).

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. Since the pressure difference between the upper- and lower-drywell regions is small, the entire drywell volume may be considered when calculating the pressure rise during this period.

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 18). This model requires the pressurization rate for the drywell. A constant ramp 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 0.26 MPa (38 psid) 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 (23 psid).

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 0.41 MPa (59 psi) 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 30 psia (0.2 MPa). Even if the wetwell pressure were near the COPS setpoint of 90 psig (0.72 MPa), the lower drywell would be below its estimated ultimate capability of 180 psig. 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.6-1 indicates that the maximum pressure in the upper drywell is the wetwell pressure plus 38 psi. 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 pressure (97 psig). 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.

Table 19EB.6-1
MAXIMUM STEAM GENERATION FOR STEAM SPIKES

<u>Water Limited Cases</u>	
Flow from lower plenum at the time of vessel failure	1020 kg/s
Passive flooder	780 kg/s
Recovered ECCS	1250 kg/s
<u>Debris Limited Case</u>	
Debris falling into cavity is quenched instantaneously	2800 kg/s
Experimentally limit for debris poured into water	80 kg/s
<u>Explosive Steam Generation</u>	
Scoping result for shock wave capability	4100 kg/s

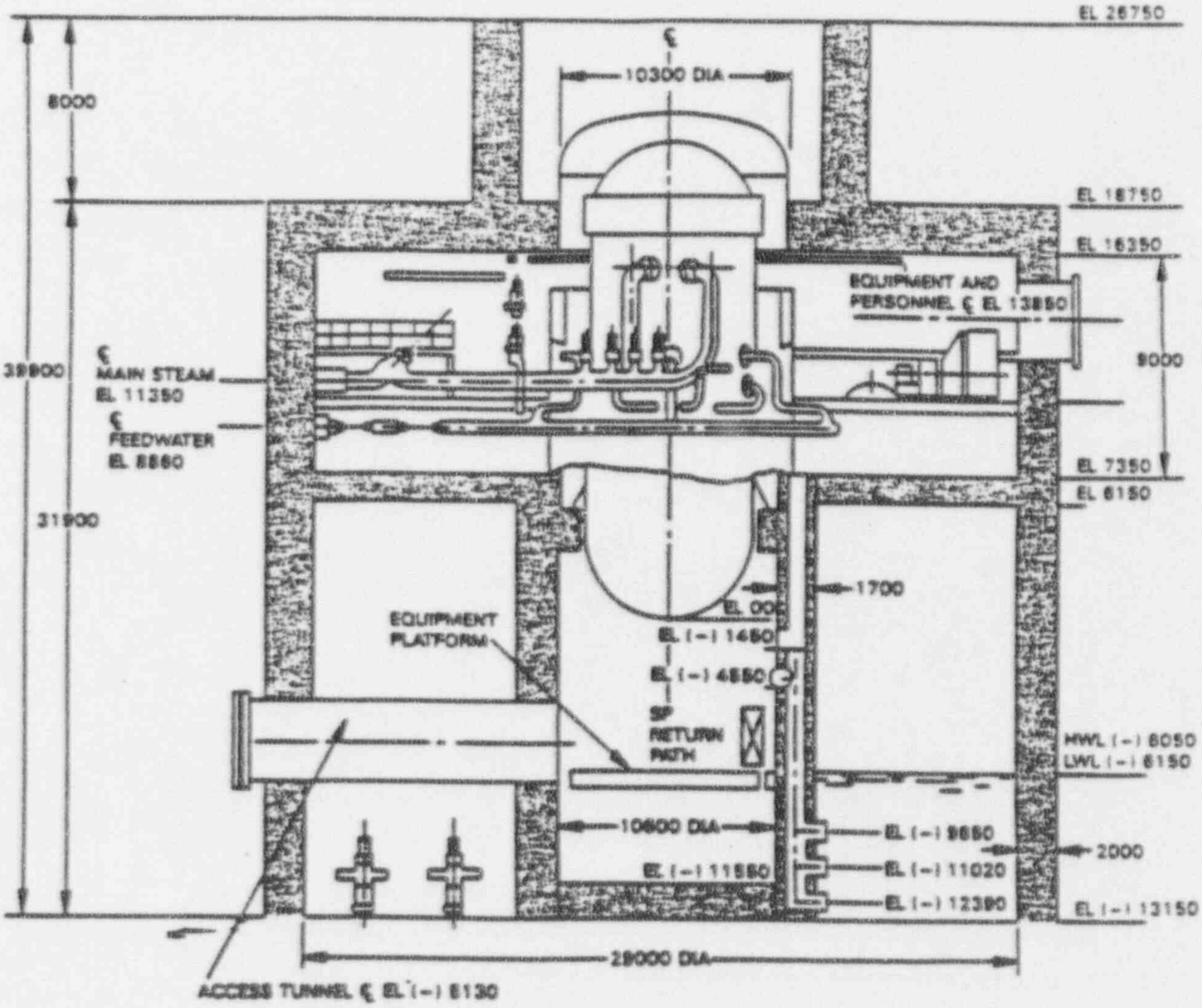


Figure 19EB.6-1
ABWR CONTAINMENT CONFIGURATION

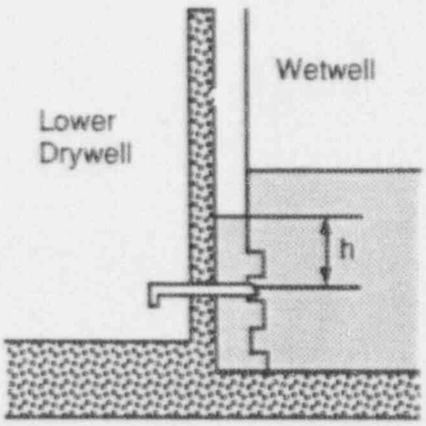


Figure 19EB.6-2

PRESSURE HEAD FOR LOWER DRYWELL FLOODER FLOW

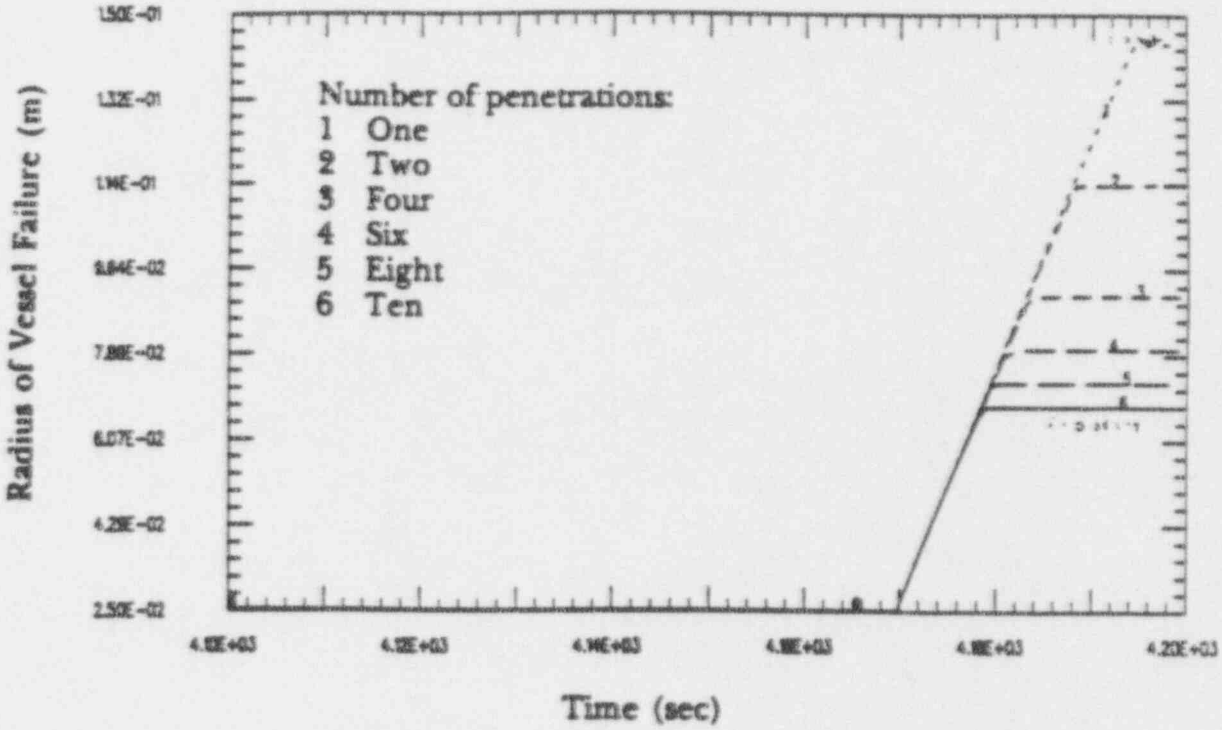


Figure 19EB.6-3
ABLATED RADIUS OF VESSEL FAILURE

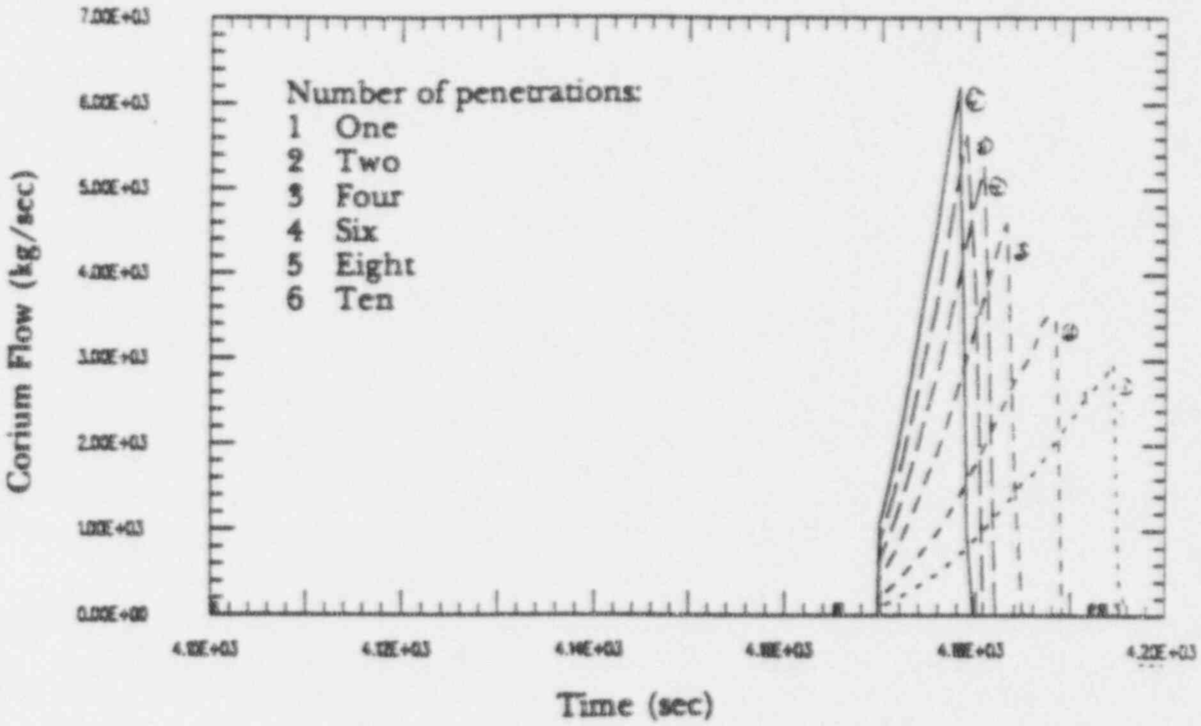


Figure 19EB.6-4

MASS FLOW OF CORE DEBRIS THROUGH VESSEL FAILURE

19EB.7 REFERENCES

1. G. Long, *Explosions of Aluminum and Water*, Aluminum Company of America, New Kensington, Pennsylvania, ALCOA Report 2-50-33, August 1950.

2. G. Long, *Explosions of Molten Aluminum and Water*, Metal Progress, Volume 71, p. 107, May 1957.

3. H.M. Higgins, *The Reaction of Molten Uranium and Zirconium Alloys with Water*, Aerojet Report 2914-2, Azusa, California, April 1955.

4. A.W. Lemmon, *Explosions of Molten Aluminum and Water*, Light Metals, 1980, (C. Minn, Ed.), P. 817, Proceedings of Technical Sessions Sponsored by TMS Light Metals Committee at the 190th AIME Annual Meeting.

5. L.D. Buxton and W.B. Benedick, *Steam Explosion Efficiency Studies*, Sandia National Laboratory, SAND/79-1399, NUREG/CR-0947, November 1979.

6. L.D. Buxton, W.B. Benedick and M.L. Corradini, *Steam Explosion Efficiency Studies: Part II - Corium Experiments*, NUREG/CR-1746, SAND/80-1324, Sandia National Laboratory, October 1980.

7. K.H. Bang and M.L. Corradini, *Vapor Explosions in a Stratified Geometry*, Nuclear Science and Engineering, Volume 108, Number 1, May 1991.

8. M.L. Corradini, personal communication, June 24, 1992.

9. W.W. Tarbell, et. al., *Pressurized Melt Ejection into Water Pools*, NUREG/CR-3916, SAND84-1531, Sandia National Laboratories, March 1991.

10. F.J. Moody, R. Muralidharan, S.S. Dua, *Assessment of Ex-Vessel Steam Pressure Spikes in BWR MARK II Containments*, 17th Water Reactor Safety Information Meeting, NUREG/CP-0104, Oct, 1989.

11. MAAP 3.0 B Computer Code Manual, EPRI NP-7071-CCML, Volume 2, November 1990.

12. M.L. Corradini, et. al., *Ex-Vessel Steam Explosions in the MARK II Containment*, NUREG-1079 Appendix C, December 1985.

13. *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants*, NUREG–1150 Appendix C.

14. US Army Corps of Engineers, Design of Structures to Resist the Effects of Atomic Weapons, Manual 1110-345-415, 1957

15. J. L. Rempe, *Light Water Reactor Lower Head Failure Analysis (Draft)*, NUREG/CR-5642, December 1990.

16. J. L. Rempe, *BWR Lower Head Failure Assessment for CSNI Comparison Exercise*, EGG-EAST-9609, April 1991.

17. S. A. Hodge and L. J. Ott, *Failure Modes of the 19BWR Reactor Vessel Bottom Head*, ORNL/M-1019, May 1989.

18. Frederick Moody, Introduction to Unsteady Thermofluid Mechanics, 1990.

APPENDIX 19E
ATTACHMENT 19EC
DEBRIS COOLABILITY AND CORE CONCTRETE INTERACTION

19EC DEBRIS COOLABILITY AND CORE CONCRETE INTERACTION

Appendix 19E of the ABWR PRA discusses core concrete interaction. In particular, in Section 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 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 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.

Theofanous and Saito - 1980 (Reference 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.

Greene - 1988 (Reference 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.

FRAG (Reference 4)

This series of tests performed at Sandia National Laboratories 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.

SWISS (Reference 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).

Mark I Shell Failure Experiments (Reference 6)

Several experiments were carried out Fauske and Associates 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.

MACE (Reference 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

(1) employ prototypic corium melt materials,

(2) employ prototypic concrete types,

(3) obtain realistic melt temperatures,

(4) obtain realistic MCCI initial conditions,

(5) include prototypic chemical and internal heating, and by the increased size,

(6) 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 [illegible] rate (a few mm/hr). 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.2-1, was developed to assess the likelihood of debris coolability. This section 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 (see Subsection 19EA.2.1.4). These regimes are:

Low 0 - 20% (nominal 10%)	0.9
High 20 - 40% (nominal 40%)	0.1

19EC.2.1.2 Amount of Initial Debris Superheat (SUPERHEAT)

This event is used 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 Section 19EC.1 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%).

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. Hence, the following probabilities were assigned for this case:

Case 1 Small Debris Mass in Lower Drywell Early

Low (Superheat)	0.9
High (Superheat)	0.1

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, probabilities of 0.5 are assigned to each branch to represent this large uncertainty.

Case 2 Large Debris Mass in Lower Drywell Early

Low (Superheat)	0.5
High (Superheat)	0.5

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 probability is the initiated from 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 reactor drain line 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, and

(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 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 case 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.

Quench	0.99
No Quench	0.01

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 by residual RPV coolant inventory for this case than for case 1.

Quench	0.95
No Quench	0.05

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 by residual RPV coolant inventory.

Quench	0.75
No Quench	0.25

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.

Quench	0.50
No Quench	0.50

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 diesel driven 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 valves, located at the ends of the pipes near the drywell floor, melt. The fusible valves on the passive flooder system are assumed to open when the lower drywell gas temperature reaches 500 F (533 K). 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 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 the following branch probabilities were estimated.

Case 1 Small Debris Mass in Lower Drywell Early

After Late Injection	0.9
Before Late Injection	0.1

Case 2 Large Debris Mass in Lower Drywell Early

After Late Injection	0.5
Before Late Injection	0.5

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, and

(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.

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 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.

CHF Limit	0.95
Film Boiling	0.045
Conduction Limit	0.005

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.

CHF Limit	0.9
Film Boiling	0.09
Conduction Limit	0.01

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.

CHF Limit	0.5
Film Boiling	0.4
Conduction Limit	0.1

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.

CHF Limit	0.1
Film Boiling	0.6
Conduction Limit	0.3

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.

No CCI	0.0
Wet CCI	0.0
Dry CCI	1.0

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.

No CCI	1.0
Wet CCI	0.0
Dry CCI	0.0

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 the following probabilities were assigned.

No CCI	0.75
Wet CCI	0.25
Dry CCI	0.0

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.

No CCI	0.0
Wet CCI	1.0
Dry CCI	0.0

19EC.2.2 Pedestal Resistance to CCI

This section 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.2-2. 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, and

(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 branch taken in the previous CET event.

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 initially flooded by either active injection or the passive flooder. The probabilities are assigned based on judgement.

SP Ingression	0.95
No SP Ingression	0.05

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.

WW Debris	0.7
No WW Debris	0.3

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 probability of 0.1 is assigned to the 1/1 erosion branch.

Radial to axial erosion ratio 1/5	0.45
Radial to axial erosion ratio 1/3	0.45
Radial to axial erosion ration 1/1	0.1

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 Section 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 Section 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.

No Pedestal Failure	1.0
Pedestal Failure	0.0

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, the following estimates are made for the probability of pedestal failure:

RAD_EROS	1/5	1/3	1/1
No Pedestal Failure	1.0	1.0	0.5
Pedestal Failure	0.0	0.0	0.5

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 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, the split fractions are are assigned based on the pedestal capability:

RAD_EROS	1/5	1/3	1/1
No Pedestal Failure	1.0	0.99	0.0
Pedestal Failure	0.0	0.01	1.0

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. Therefore, the split fractions assigned are:

RAD_EROS	1/5	1/3	1/1
No Pedestal Failure	1.0	0.99	0.5
Pedestal Failure	0.0	0.01	0.5

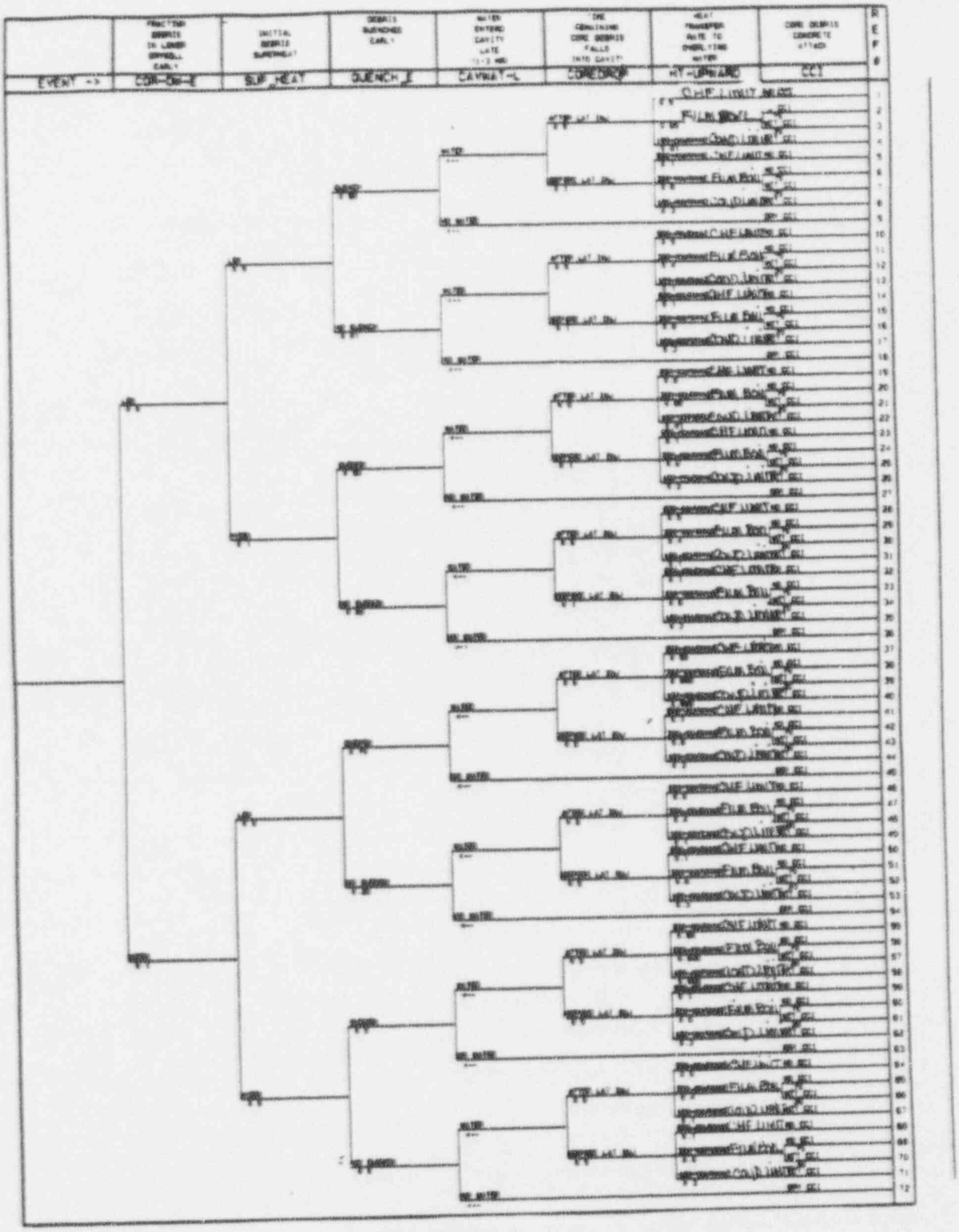


Figure 19EC.2-1
CORE DEBRIS CONCRETE ATTACK DET

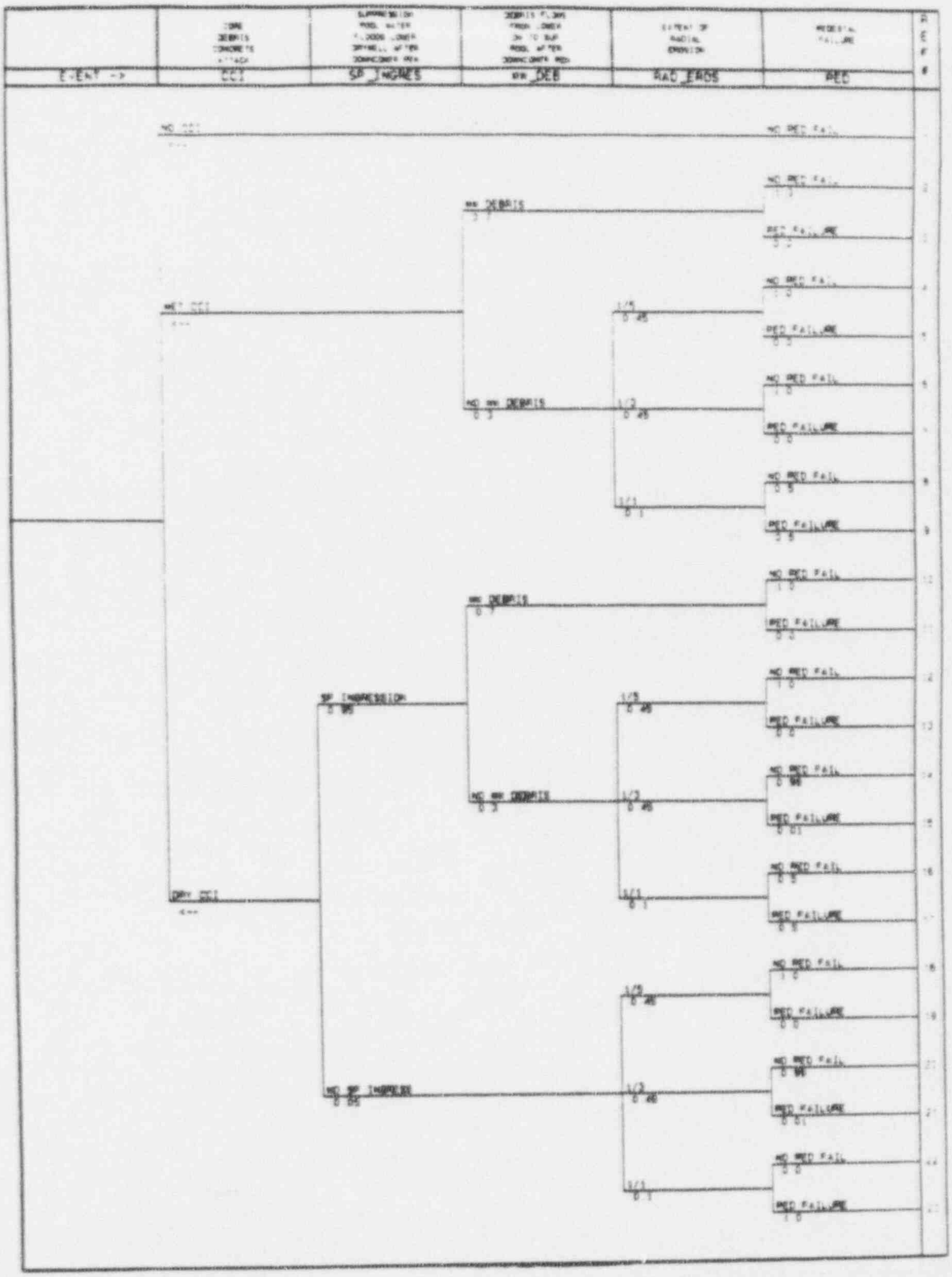


Figure 19EC.2-2
CONTAINMENT EVENT EVALUATION DET FOR PEDESTAL FAILURE

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 over-pressure 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 - 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 determing 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, or

(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 been observed that the smallest of surface imperfections or contaminants would quickly result 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 a homogeneous debris mixture, is given by:

$$k\frac{\partial^2 T}{\partial x^2} + q''' = 0 \qquad (1)$$

where: k = thermal conductivity (3.5 W/mK),

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 244,000 kg, this implies and average initial volumetric heat generation rate:

$$q''' = 1.3\frac{MW}{m^3}$$

In a one-dimensional flat geometry, integrating Equation (1) twice yields:

$$T = \frac{-q'''x^2}{2k} + C_1 x + C_2 \qquad (2)$$

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) assume molten debris is at uniform temperature, and

(4) impose the condition that the debris not ablate concrete,

we have as boundary conditions:

$$C_1 = 0$$

$$C_2 = 1550K$$

$$T(\delta_{lim}) = 450K$$

where: δ_{lim} = debris thickness.

Substituting into Equation (2), we have for the limiting debris thickness for coolability:

$$\delta_{lim} = 0.08 \text{ m}$$

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 8 cm. The surface heat flux is:

$$q'' = q'''\delta_{lim} = 100 \text{ MW}/m^2$$

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 crust 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 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 Section 19EC.3. This section 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. Table 19EC.5-1 provides a chronology of the events up until the vessel is failed.

Table 19EC.5-2 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 critical heat flux limits 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.5-2 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 Section 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 (Section 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 fire water 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.5-1 through 19EC.5-5. 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 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,

$$Nu = \frac{hD}{k} = 1.14\sqrt{Re} \qquad (3)$$

or

$$h = 1.14 k_{cm} \sqrt{\frac{\rho_{cm} u_c}{\mu_{cm} D_{jet}}} \qquad (4)$$

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,

$$u_c = u_0 + gt_{fall} \qquad (5)$$

where: u_0 = Velocity of the corium expelled from the reactor vessel,

g = Acceleration of gravity,

and t_{fall} is defined by

$$u_0 t_{fall} + \frac{1}{2} g t_{fall}^2 = z_v \qquad (6)$$

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

$$u_{cn} = \frac{h\left(T_{cm} - T_{cnp}\right)}{\rho_{cn}\left[c_{pcn}\left(T_{cnp} - T_0\right) + \gamma_{cn}\right]} \qquad (7)$$

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/sec. 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.

Table 19EC.5-1

SUMMARY OF TIMING FOR CORE CONCRETE INTERACTION BASE CASE

Time (Secs)	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.5-2
SUMMARY OF CCI DETERMINISTIC ANALYSIS FOR ABWR

Case #	Containment Press. at Vessel failure	Upward Heat Trans. (kw/m^2)	Debris Mass at Vess. Fail (Frac. of Tot. Inventory)	Radial Attack at 50 hrs. (meters)	H2 Generated at 50 hrs. (Kg)	Time of FP Release (hours)	Mode of Release	Fission Product Release Fraction from Containment NG	CsI	Sr
ABWR100	1 Atm.	100	0.6	0.22	1813	19.1	COPS	1.0	2E-06	3E-09
ABWR300	1 Atm.	300	0.6	9E-07	122	23.3	COPS	1.0	2E-10	2E-12
ABWR900	1 Atm.	900	0.6	7E-06	122	23.2	COPS	1.0	3E-11	2E-12
FMX100	1 Atm.	100	1.0	0.25	2130	17.6	COPS	1.0	1E-06	1E-08
FMX300	1 Atm.	300	1.0	7E-03	154	19.3	COPS	1.0	1E-08	3E-15
FMX900	1 Atm.	900	1.0	7E-04	111	19.1	COPS	1.0	1E-08	2E-14
FMXCSP (1)	1 Atm.	100	1.0	0.25	2126	15.7	COPS	1.0	4E-07	3E-10
SENSITIVITY RUNS										
DRY	1 Atm.	N/A	0.6	0.50	4990	19.8	DWT	0.34	4E-03	1E-05
DWFAIL	1 Atm.	300	1.0	7E-03	154	19.3	DWF	1.0	8E-04	2E-10
FIRE	1 Atm.	100	1.0	0.25	2131	24.6	COPS	1.0	5E-06	4E-10
FMX1P (2)	1 Atm.	100	1.0	0.31	2762	17.6	COPS	1.0	1E-06	1E-08
NFlood (3)	1 Atm.	100	0.6	0.25	2127	17.4	COPS	1.0	8E-07	5E-10
LATE (4)	1 Atm.	N/A	1.0	0.31	2697	20.0	DWT	0.23	6E-03	9E-09

Notes:

COPS - Containment Overpressure Protection System
DWT - Drywell Leakage occurs through penetrations.
DWF - Drywell Failure (0.0973 m^2)
(1) - FMX100 Run with five times steel mass
(2) - Penetration into connecting vents does not cause debris spread.
(3) - Flooder not operational, Radial attack results in penetration to WW and debris spread.
(4) - Vessel failure assumed to occur after lower plenum water boiled away and debris reheats.

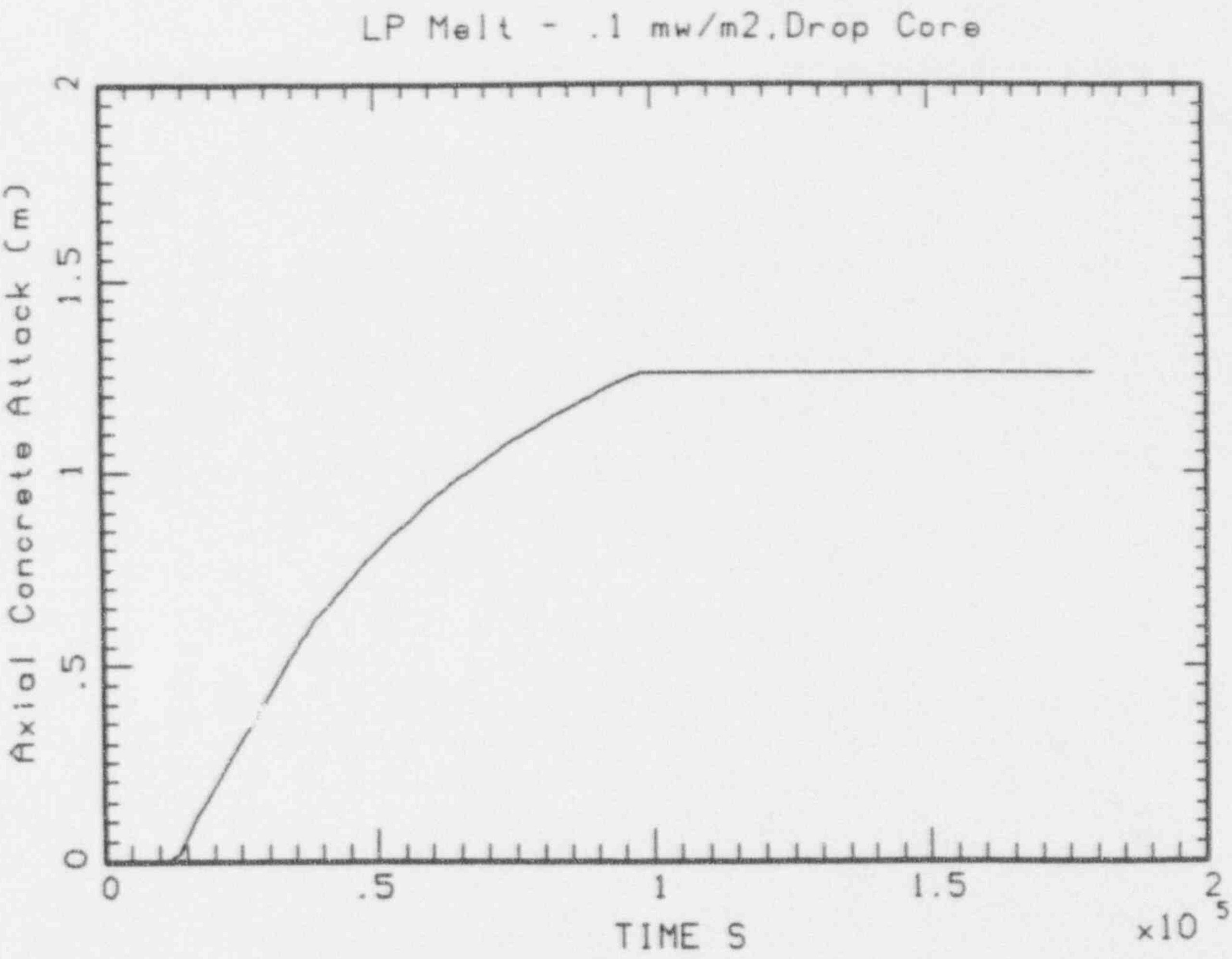


Figure 19EC.5-1
SAMPLE CALCULATION FOR CORE CONCRETE INTERACTION
UPWARD HEAT FLUX 100 kW/m^2: AXIAL CONCRETE ATTACK

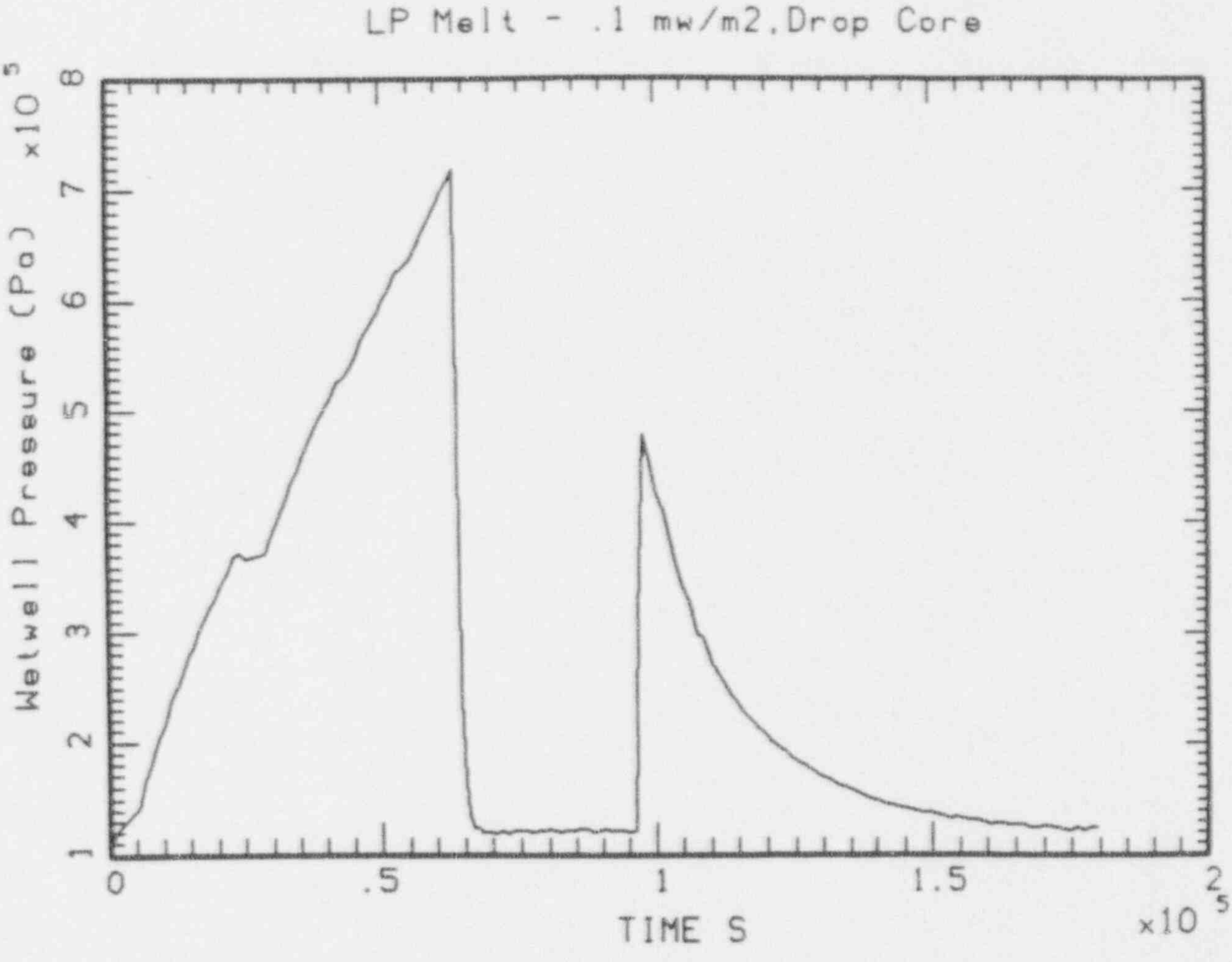


Figure 19EC.5-2
SAMPLE CALCULATION FOR CORE CONCRETE INTERACTION UPWARD HEAT FLUX 100 kW/m²: WETWELL PRESSURE

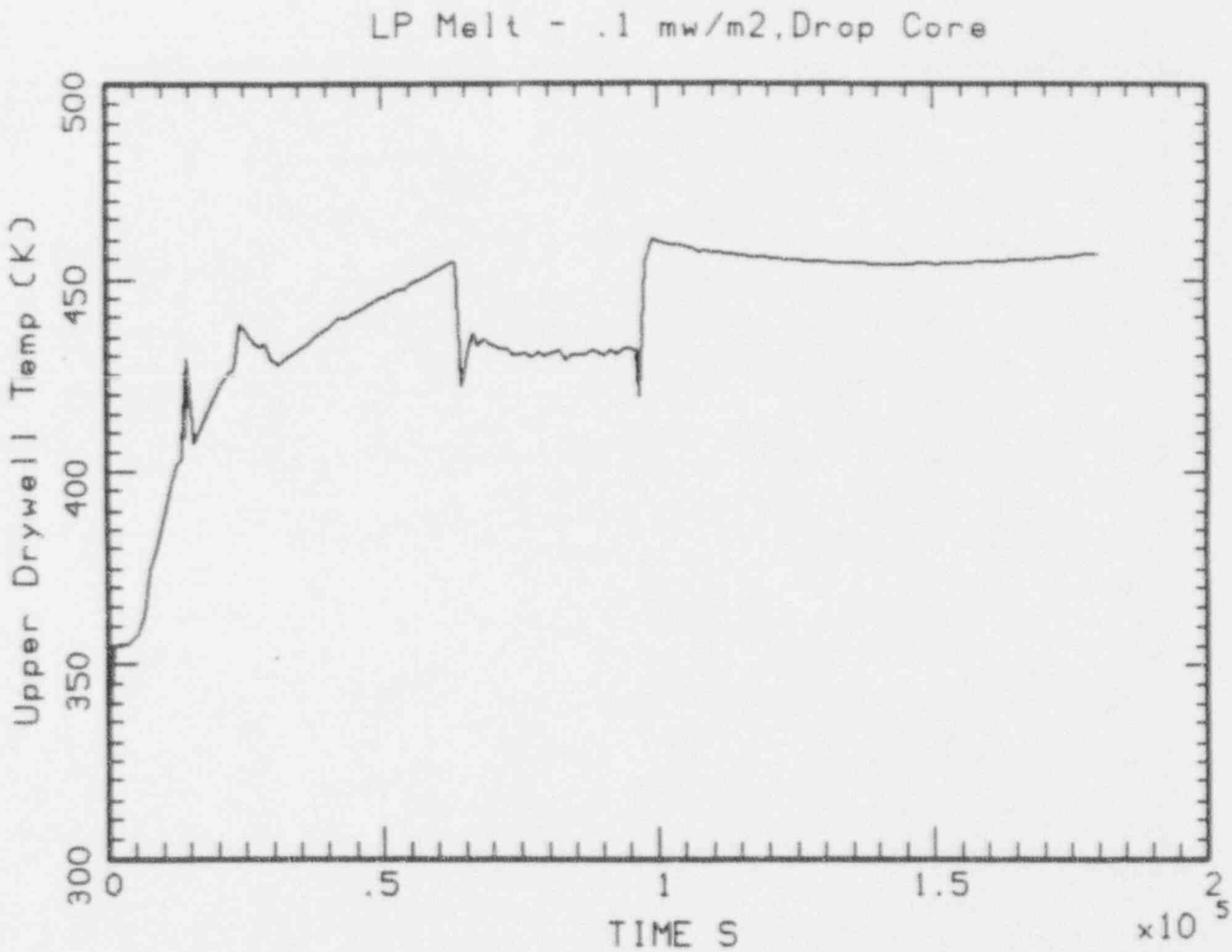


Figure 19EC.5-3
SAMPLE CALCULATION FOR CORE CONCRETE INTERACTION
UPWARD HEAT FLUX 100 kW/m²: UPPER DRYWELL TEMPERATURE

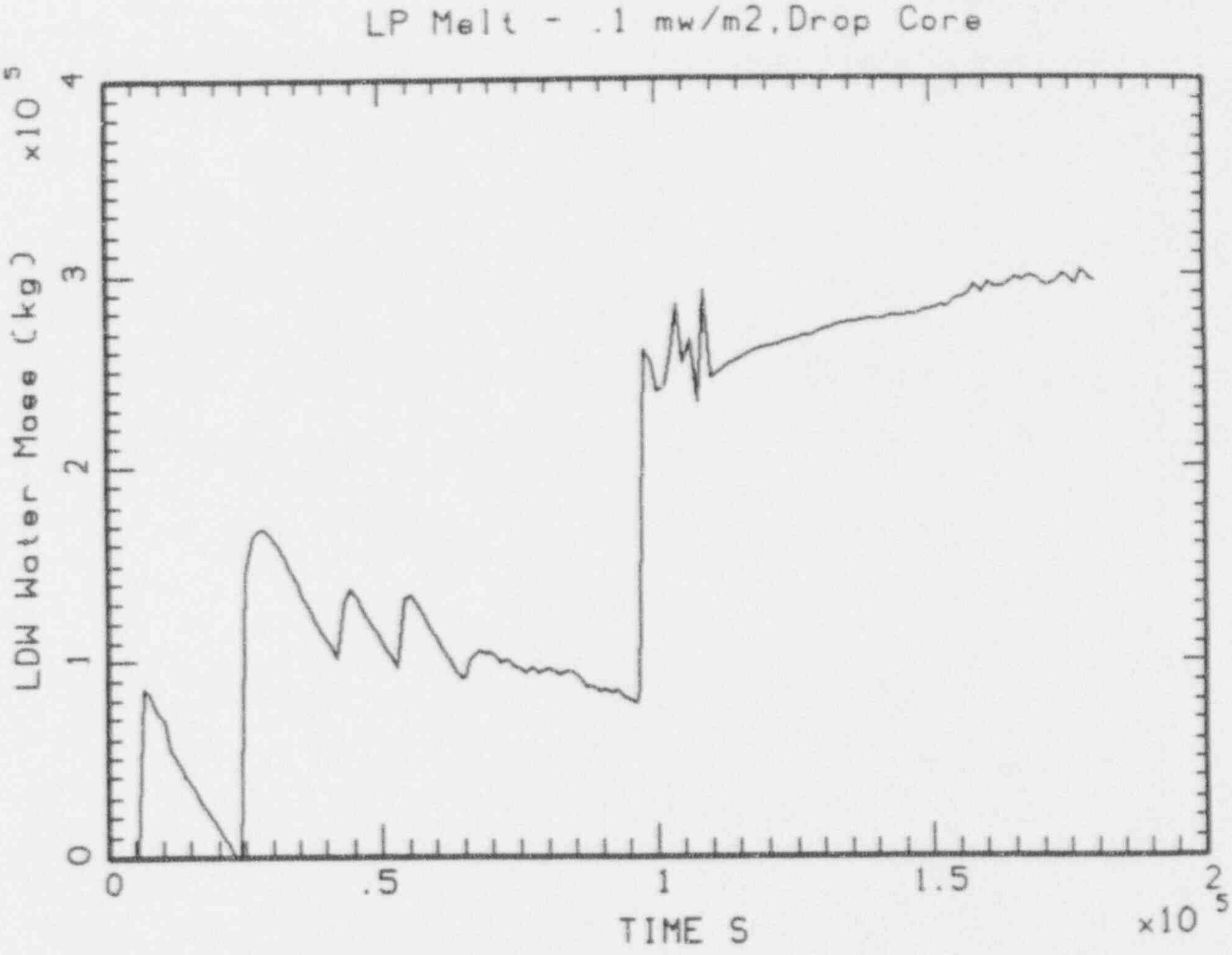


Figure 19EC.5-4
SAMPLE CALCULATION FOR CORE CONCRETE INTERACTION UPWARD HEAT FLUX 100 kW/m^2: LDW WATER MASS

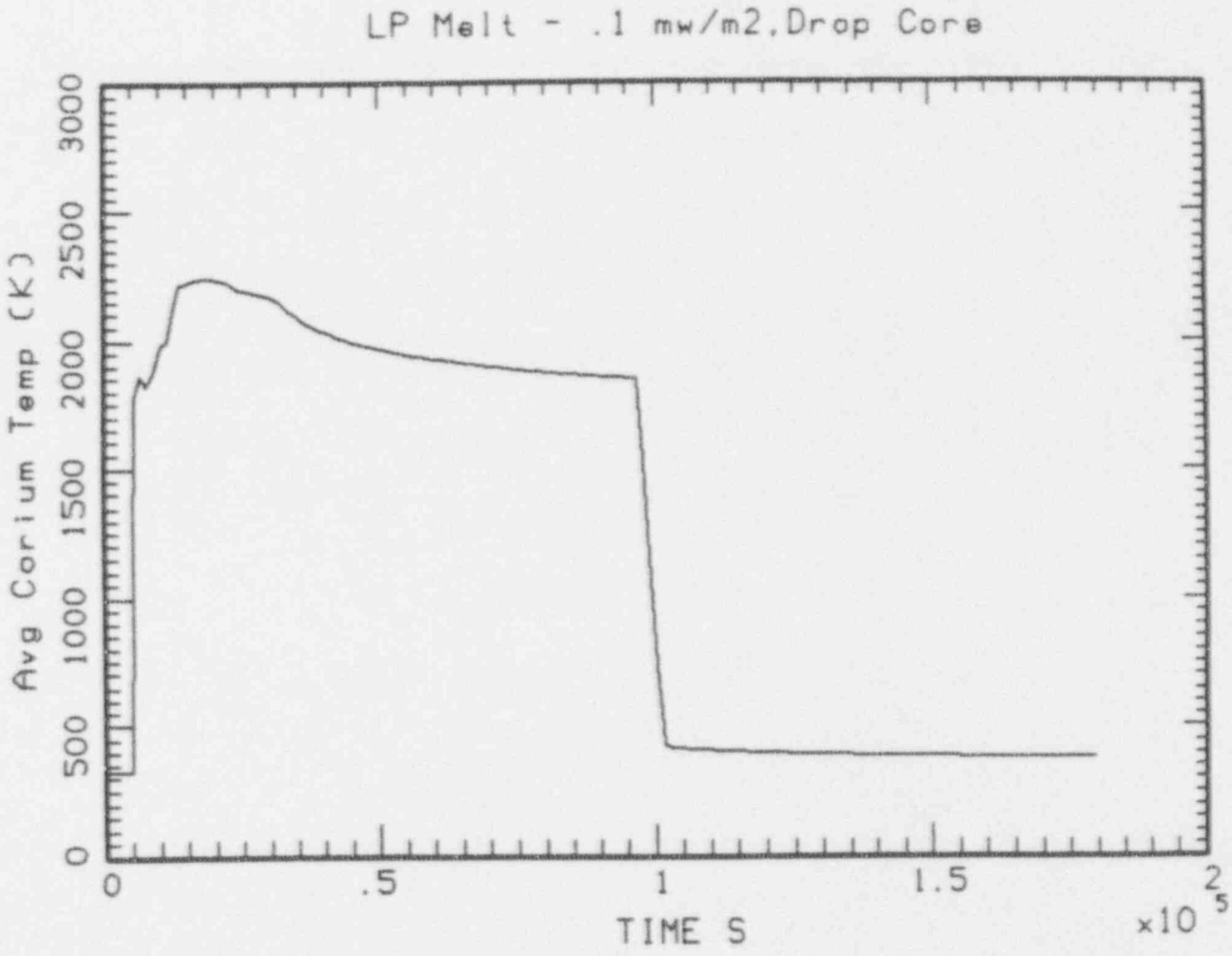


Figure 19EC.5-5
SAMPLE CALCULATION FOR CORE CONCRETE INTERACTION UPWARD HEAT FLUX 100 kW/m^2: AVG CORIUM TEMPERATURE

19EC.6 SENSITIVITY TO VARIOUS PARAMETERS

Also included in Table 19EC.5-2 are other analyses that address possible sensitivities to modelling 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 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 no 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 occured 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 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. Basaltic concrete 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 rate would be slower than that presented in Table 19EC.5-2. Therefore, the times estimated in that analysis for the time 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-condensible gas. Limestone-common sand concrete produces more non-condensible gas than does basaltic concrete. However, this will not have a significant impact on theis analysis because the surface area of the sidewall will be only ten to fifteen percent 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-condensible 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.7 Impact on Offsite Dose

The effect of the maximum core concrete interaction source term on a relaease with operation of the rupture disk is shown in Figure 19EC.7-1. 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.) As the Figure 13 clearly shows, CCI does not have significant impact on the offsite dose.

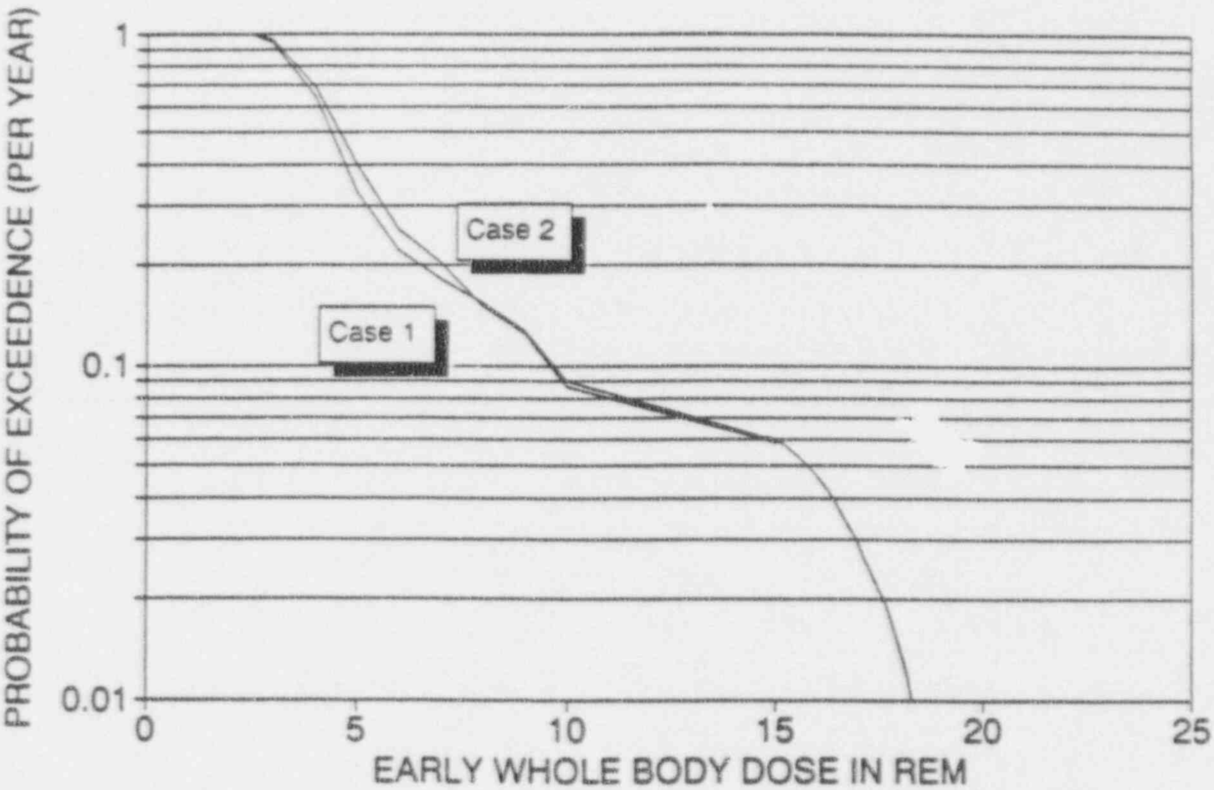


Figure 19EC.7-1
WHOLE BODY DOSE AT 1/2 MILE AS A PROBABILITY OF EXCEEDENCE

19EC.8 Conclusions

This section 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 determinstic analysis 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, 90% 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 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

1. Final Report on Core Debris Coolability, IDCOR Task 15.2

2. An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution, to be published as NUREG/CR, Draft 1991.

3. G.A. Greene, C. Finfrock and S.B. Burson, *Phenomenological Studies on Molten Core-Concrete Interactions*, Nuclear Engineering and Design, 108, 167-177, 1988.

4. M.W. Tarbell, D.R. Bradley, R.E. Blose, J.W. Ross, and D.W. Gilbert, *Sustained Concrete Attack by Low-Temperature Fragmented Core Debris*, NUREG/CR-3024, SAND82-2476 R3, R4, July 1987.

5. R.E. Blose, J.E. Gronager, A.J. Suo-Antilla, and J.E. Brockman, *Sustained Heated Metallic/Melt Concrete Interactions with Overlaying Water Pools*, NUREG/CR-4727, SAND85-1546 R3, R4, R7, July 1987.

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.

7. B.R. Sehgal, *ACE Program Phase D: Melt Attack and Coolability Experiments (MACE) Program*, presentation at CSARP meeting, May 1992.

8. S.R. Greene, S.A. Hodge, C.R. Hyman, M.L. Tobias, *The Response of BWR Mark II Containment to S.ation Blackout Severe Accident Sequences*, NUREG/CR-5565, ORNL/TM-11548, May 1991.

9. MAAP 3.0 B Computer Code Manual, EPRI NP-7071-CCML, Volume 2, November 1990.

APPENDIX 19E
ATTACHMENT 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/MW_{th} 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 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 Section 5.2.5.

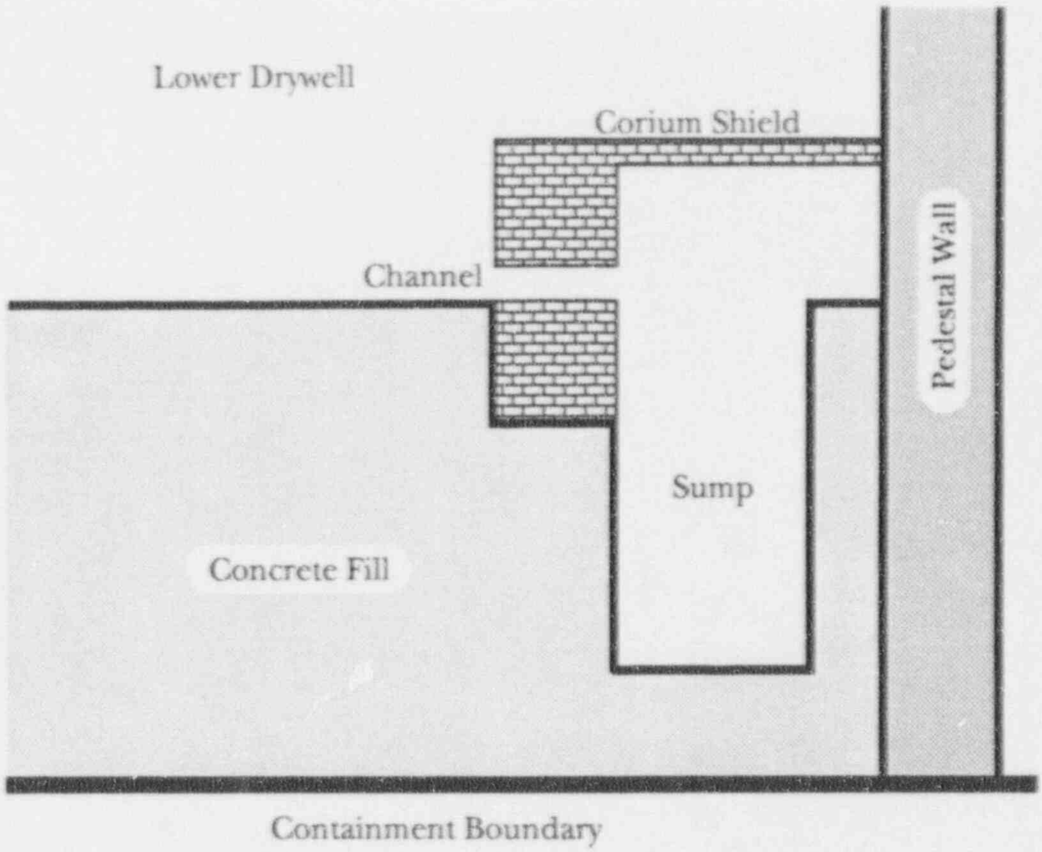
19ED.2 PROPOSED DESIGN

A protective layer of refractory bricks—a corium shield—could be built around the sumps to prevent corium ingression. The shield for equipment drain sump would be solid except for the inlet and outlet piping which would go through its roof. The shield for the floor drain sump would 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 height of the channels would be chosen so that any molten debris which reaches the inlet will freeze before it exited and spilled into the sump. The width and number of the channels would be chosen so that the required water flow rate during normal reactor operation is achievable. A sketch of a concept for floor drain sump shield is shown in Figure 19ED.2-1.

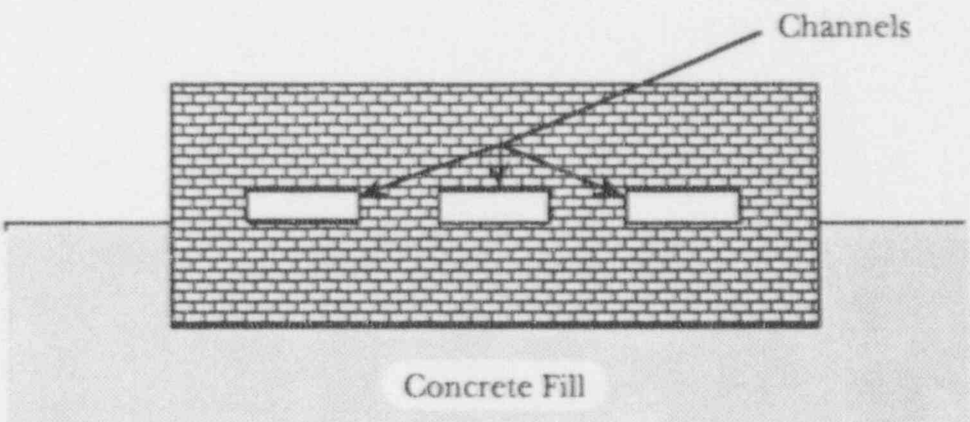
The walls of the equipment drain sump shield (solid shield) only have to be thick enough to withstand ablation, if any is expected to occur, for the chosen wall material. The walls of the floor drain sump shield (channeled shield) must be thicker so that molten debris flowing through the channels has enough residence time to ensure debris solidification.

Both shields would extend above the LD floor to an elevation greater than the expected maximum height of core debris. Thus, no significant amounts of debris will collect on the shield roofs. The solid shield can be placed directly on top of the LD floor. The channeled shield will have refractory bricks embedded into the LD floor beneath the shield to prevent core-concrete interaction involving the molten debris in the channels.

The analyses presented in Sections 19ED.4 and 19ED.5 provide a basis for sizing the proposed design of the floor drain sump corium shield.



a) side view



b) front view

Figure 19ED.2-1
CONCEPTUAL DESIGN OF LOWER DRYWELL FLOOR DRAIN SUMP SHIELD

19ED.3 SUCCESS CRITERIA FOR PROPOSED DESIGN

For the proposed design to be considered successful, it must satisfy the following requirements:

(1) Melting Point of Shield Material Above Initial Contact Temperature

The shield wall material shall be chosen so that its melting temperature is greater than the interface temperature between the debris and the shield wall.

(2) Channel Length

The length of the channels in the 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.

(3) Shield Height, H_{uw}, Above Lower Drywell Floor

The shield height above the lower drywell floor shall be chosen to ensure long term debris solidification. The freezing process will be complete during the time frame when the shield walls are behaving as semi-infinite solids. In addition, the shield must be tall enough to prevent debris from accumulating on the roof of the shield.

(4) Shield Depth, H_{lw}, Below Lower Drywell Floor

The shield depth of the below the lower drywell floor shall be chosen to ensure long term debris solidification.

(5) Water Flow Rate

The total flow area of the shield channels 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.

(7) Seismic Adequacy

The seismic adequacy of the corium shields will be determined in the detailed design phase. Adequacy should be easily meet 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.

Section 19ED.6 contains an example of success calculations for requirements (1) through (4) for a chosen channel height of 1 cm.

19ED.4 ANALYSIS OF SHIELD FREEZING ABILITY

Heat transfer and phase change analyses are presented in this section to determine the feasibility of a channeled shield to prevent molten debris ingression into the floor drain sump. Two time frames were considered. First, a freeze front analysis was performed for early times (seconds or less) to determine the time required to form a plug. The long term ability of a plug to remain solid was determined using a steady-state analysis.

19ED.4.1 Assumptions

The major assumptions invoked in the analyses 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 and can result in eutectic formations. The melting temperature of core debris which has undergone little interaction is approximately 2500 K. Significant interaction with the concrete floor reduces the debris melting temperature to approximately 1700 K.

(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 heat conduction coefficient in the solidified debris is significantly larger than that of the shield material,

b) the depth of the solidified debris is considerably less than the height of the shield.

(3) Heat transfer within the channel and shield is one-dimensional.

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 properties of shield cause it to be 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 at least two hours after accident initiation at which time the decay heat level is approximately one percent of rated power.

Core debris will not enter the lower drywell before about two hours for any credible severe accident (Section 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 can be controlled in the final 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 molten corium has a chance to traverse the channel and reach the sump. Molten debris enters the channel at a significantly elevated temperature (2500 K to 1700 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 leading edge of the freeze front will stay at the melting temperature of the debris. 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.4-1.

(1) Freezing Time

The temperature profile in the crust (Reference 1), assuming it quickly reaches its steady state shape, is:

$$T_c(x) = \frac{\dot{q}L_c^2}{2k_f}\left(1-\frac{x^2}{L_c^2}\right) + \frac{T_s - T_{f,m}}{2}\frac{x}{L_c} + \frac{T_s + T_{f,m}}{2} \tag{1}$$

where: $T_c(x)$ = temperature within the crust,

x = crust coordinate measured from the crust centerline,

$\dot{q}$ = heat density of the crust,

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:

$$q''_{lh} = -k_f \frac{dT_c}{dx}\Big|_{x=-L_c} \tag{2}$$

where: q''_{lh} = the latent heat flux.

The latent heat flux is:

$$q''_{lh} = \frac{dx_c}{dt}\rho_{cm}h_{lh} \tag{3}$$

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:

$$\frac{dx_c}{dt} = \frac{1}{\rho_{cm}h_{lh}}\left[\frac{k_f}{x_c}\left(T_{f,m} - T_s\right) - \dot{q}\frac{x_c}{2}\right] \tag{4}$$

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 an semi-infinite solid body only experiencing conduction. The contact temperature between the debris and the channel wall (Reference 2), assuming semi-infinite bodies, is:

$$T_s = \frac{T_{f,m}\sqrt{(k\rho c)_{cm}} + T_i\sqrt{(k\rho c)_w}}{\sqrt{(k\rho c)_{cm}} + \sqrt{(k\rho c)_w}} \tag{5}$$

where: c = specific heat,

cm = debris material properties,

w = wall material properties.

Using the debris properties found in the Table

entitled *Important Parameters for Steam Explosion Analysis* (19E.2-17) and representative wall properties found in Table 19ED.4-1, the interface temperature is estimated to be 1390 K.

The debris energy generation density can be found by assuming a decay heat level and a total amount of corium. The density is:

$$\dot{q} = \frac{Q_{dh}\rho_{cm}}{m_{cm}} \qquad (6)$$

where: Q_{dh} = decay heat level,

m_{cm} = total mass of corium, 235 Mg.

Evaluating this two hours after accident initiation (decay heat level equals approximately one percent of rated power) yields:

$$\dot{q} = 1.5 \times 10^6 \ \mathrm{MW/m^3}$$

The two terms inside the brackets in Equation (4) can now be evaluated. For a channel height of 1 cm ($x_{c,max} = 0.5$ cm) and a debris melting temperature of 1700 K, these values are:

$$\frac{k_f}{x_c}\left(T_{f,m} - T_s\right) = 1.86 \times 10^6 \ \mathrm{W/m^2}$$

$$\dot{q}\frac{x_c}{2} = 3.8 \times 10^3 \ \mathrm{W/m^2}$$

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.4-1. Equation (4) can be simplified to:

$$\frac{dx_c}{dt} = \frac{k_f}{\rho_{cm}h_{lh}x_c}\left(T_{f,m} - T_s\right) \qquad (7)$$

Solving this equation with the initial condition that $x_c(t=0) = 0$, reveals:

$$x_c = \sqrt{\frac{2k_f\left(T_{f,m} - T_s\right)t}{\rho_{cm}h_{lh}}} \qquad (8)$$

This equation can be rearranged to determine the time required to freeze debris in a channel of height H_o. The freezing time is:

$$t_{freeze} = \frac{H_o^2\rho_{cm}h_{lh}}{8k_f\left(T_{f,m} - T_s\right)} \qquad (9)$$

(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:

$$q''_{crust} = -k_f \left.\frac{dT_c}{dx}\right|_{x=x_c/2} \qquad (10)$$

which evaluates to:

$$q''_{crust} = \frac{\dot{q}x_c}{2} + \frac{k_f}{x_c}\left(T_{f,m} - T_s\right) \qquad (11)$$

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:

$$q''_{crust} = \frac{k_f}{x_c}\left(T_{f,m} - T_s\right) \qquad (12)$$

Inserting the expression for x_c in Equation (8) and rearranging yields:

$$q''_{crust} = \sqrt{\frac{k_f\rho_{cm}h_{lh}\left(T_{f,m} - T_s\right)}{2t}} \qquad (13)$$

The heat flux (Reference 3) absorbed by the channel wall can be approximated by that which a semi-infinite solid body can absorb. This flux is:

$$q''_w = \frac{k_w\left(T_s - T_i\right)}{\sqrt{\pi\alpha_w t}} \qquad (14)$$

where: α_w = thermal diffusivity of the wall material.

Equating (13) and (14) produces an equation governing the interface temperature. It is:

$$\frac{T_s - T_i}{\sqrt{T_{f,m} - T_s}} = \left(\frac{\pi k_f\rho_{cm}h_{lh}\alpha_w}{2k_w^2}\right)^{1/2} \qquad (15)$$

Solving this equation for T_s using the quadratic formula yields:

$$T_s = \frac{-(c_o - 2T_i) \pm \sqrt{(c_o - 2T_i)^2 - 4(T_i^2 - c_o T_{f,m})}}{2} \tag{16}$$

where: c_o = the square of the right hand side of Equation (15).

Negative solutions of this equation are physically impossible. For a $T_{f,m}$ of 1700 K and a T_i of 330 K, the interface temperature is 1560 K. Similarly, the interface temperature is 2180 K for $T_{f,m}$ = 2500 K and T_i = 330 K.

Since this temperature is higher than the value for two semi-infinite solid bodies coming into contact, the dominance of the temperature difference term in Equations (4) and (11) should be reverified. The heat-generation and temperature-difference terms for a interface temperature of 1560 K and channel half-height of 0.5 centimeters are:

$$\frac{k_f}{x_c}(T_{f,m} - T_s) = 8.4 \times 10^5 \ \mathrm{W/m^2}$$

$$\dot{q}\frac{x_c}{2} = 3.8 \times 10^3 \ \mathrm{W/m^2}$$

Even though the dominance is not as great as before, the temperature-difference term is still significantly greater than the heat-generation term and the assumptions made previously are still valid.

19ED.4.3 Required Channel Length to Insure Freezing

The propagation rate of the freeze front was determined in Section 19ED.4.2. This allowed determination of the time to completely freeze the debris 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 section, 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.

The entrance velocity will be 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. A conservative overprediction of debris depth can be obtained by multiplying the maximum expulsion rate by time and dividing by area. The upper bound of the expulsion rate was shown in Section 19EB.6.2.2 to be 6000 kg/sec.

The velocity in the channel without area reduction due to debris freezing can be conservatively overpredicted by ignoring frictional effects. This velocity is:

$$v_e(t) = \sqrt{2g\Delta z(t)} \tag{17}$$

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:

$$v_e(t) = \sqrt{\frac{2g\dot{m}_{ves}t}{\rho_{cm}A_{ld}}} \tag{18}$$

where: $\dot{m}_{ves}$ = maximum ejection rate of corium from a failed vessel,

A_{ld} = floor area of the lower drywell ($79 m^2$).

(2) Channel Area Decrease Resulting From Debris Freezing

Since the entrance velocity is assumed to remains constant, 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.4-2. Conservation of mass requires that the mass flow rate of corium entering the channel per unit length is constant throughout the channel. The mass flow rate at the entrance of the channel and at the location downstream where the debris front has just arrived is:

$$\dot{m}_i(t) = \rho_{cm}v_e(t)H_i(t) = \rho_{cm}v_o(t)H_o \tag{19}$$

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.

This equation requires that:

$$v_o(t) = \frac{v_e(t)}{H_o} H_i(t) \qquad (20)$$

The entrance flow height is:

$$H_i(t) = H_o - 2x_c(t) \qquad (21)$$

Inserting the relationship for x_c found in Equation (8) into this expression yields:

$$H_i(t) = H_o - \sqrt{\frac{8k_f\left(T_{f,m} - T_s\right)t}{\rho_{cm} h_{lh}}} \qquad (22)$$

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 (20) and (22). It is:

$$v_o(t) = v_e(t)\left(1 - \frac{1}{H_o}\sqrt{\frac{8k_f\left(T_{f,m} - T_s\right)t}{\rho_{cm} h_{lh}}}\right) \qquad (23)$$

The average velocity of debris between the entrance of the channel and the leading edge of molten corium is:

$$\overline{v}(t) = \frac{\int_0^t t v_o(t)dt}{\int_0^t t dt} \qquad (24)$$

Evaluating this integral yields:

$$\overline{v}(t) = a_o\sqrt{t} - \frac{a_o b_o}{H_o} t \qquad (25)$$

where:

$$a_o = \frac{4}{5}\sqrt{\frac{2g\dot{m}_{ves}}{\rho_{cm} A_{ld}}}$$

$$b_o = \frac{5}{3}\sqrt{\frac{2k_f\left(T_{f,m} - T_s\right)}{\rho_{cm} h_{lh}}}$$

This is the average velocity of the molten debris into the shield channel.

(4) 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:

$$L_{freeze} = \overline{v}(t_{freeze})t_{freeze}$$
$$= a_o t_{freeze}^{3/2} - \frac{a_o b_o}{H_o} t_{freeze}^2 \qquad (26)$$

Table 19ED.4-1
MATERIAL PROPERTIES OF A REPRESENTATIVE REFRACTORY BRICK AND CONCRETE

Property	Representative Brick (Reference 6)	Concrete
Melting Temperature (K)	> 2200	1450
Density (kg/m^3)	2700	2300
Thermal Conductivity (W/mK)	4	1.3
Specific Heat (J/kgK)	1000	800
Thermal Diffusivity (m^2/s)	1.48×10^{-6}	7.5×10^{-7}

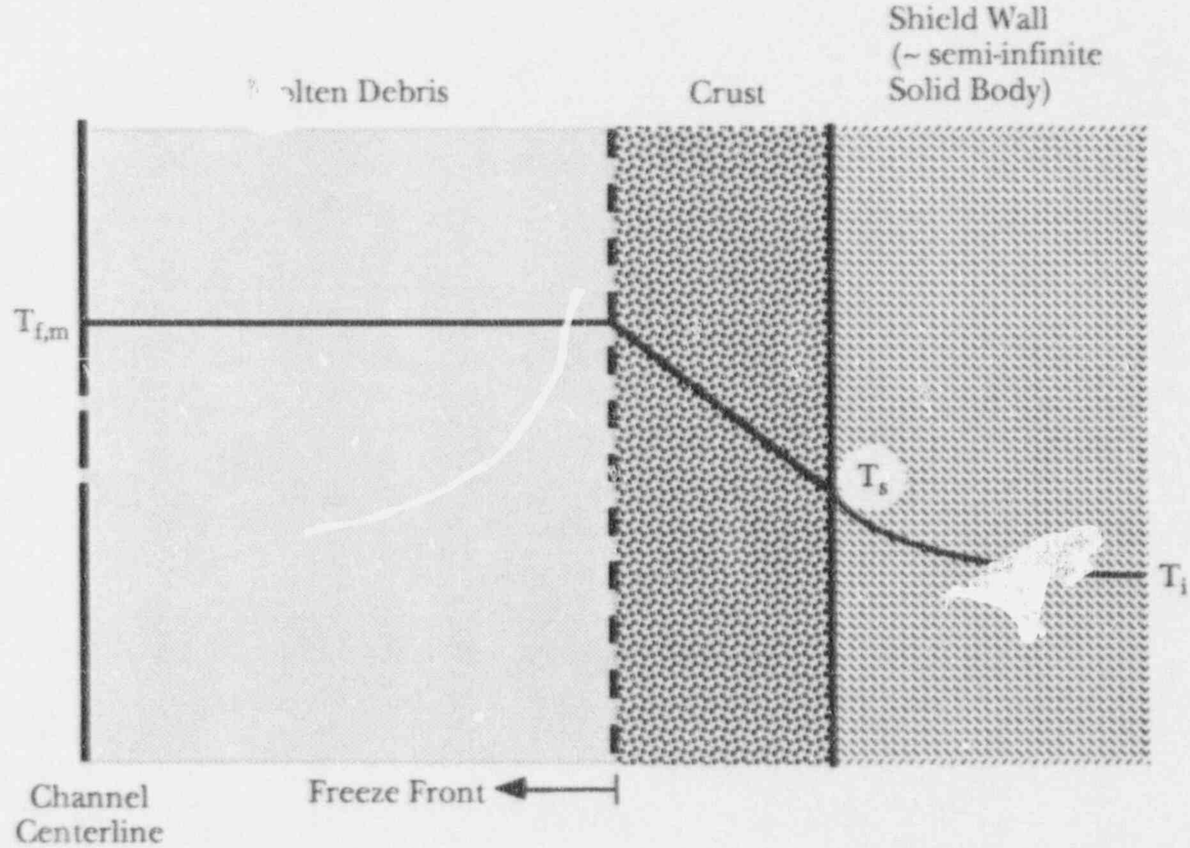


Figure 19ED.4-1
TEMPERATURE PROFILE IN CHANNEL REGION

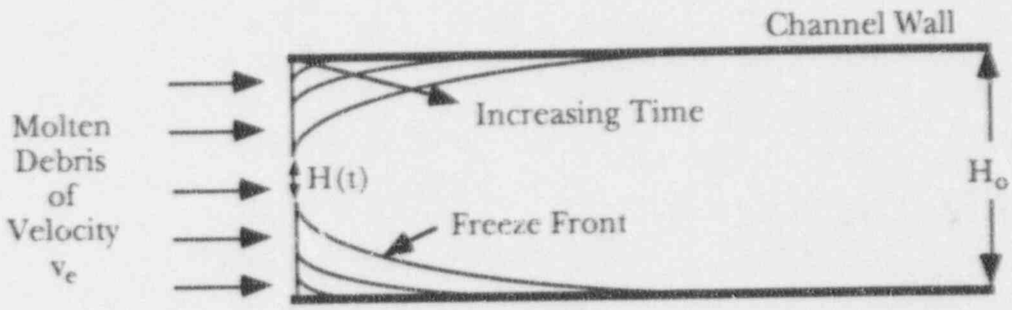


Figure 19ED.4-2
CHANNEL FLOW HEIGHT REDUCTION
DURING FREEZE PROCESS

19ED.5 LONG-TERM ABILITY OF DEBRIS TO REMAIN SOLID

Initial debris solidification was considered in Section 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 section. The height of the upper shield wall (above the lower drywell floor) and depth of the lower shield wall (below the lower drywell floor) will be specified.

19ED.5.1 Upper Shield Wall (Above Lower Drywell Floor)

The roof of the upper shield wall should be free, or at least nearly so, of debris to provide long-term cooling to the debris frozen in the channel. No significant amount of debris will splatter on the roof during ejection from the vessel because the sump is near the periphery of the lower drywell. 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:

$$H_{uw} \geq \frac{m_{cm,tot}}{\rho_{cm} A_{ld,min}} \tag{27}$$

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:

$$H_{uw} \geq 0.33 \text{ m} \tag{28}$$

In the long term (at least minutes after debris solidification), the lower drywell will be filled with either saturated steam or water. Heat transfer from the shield to the environment is less effective when steam is present. Therefore, only steam will be considered in the remainder of this analysis. A shield wall sized to perform its function when steam is present will also perform its function when water fills the lower drywell.

The maximum steam temperature in the lower drywell is that of saturated steam at the ultimate containment pressure (180 psig). The steady-state heat flux through the upper shield wall is:

$$q''_{uw} = \frac{k_w}{H_{uw}}\left(T_i - T_o\right) \tag{29}$$

where: q''_{uw} = steady state heat flux through the upper shield wall,

H_{uw} = height of the upper wall,

T_i = temperature of the upper wall in contact with debris,

T_o = temperature of the upper wall in contact with the lower drywell environment.

Natural convection governs the temperature of the wall in contact with the lower drywell environment. The heat flux from the top of the wall can be written as:

$$q''_{uw} = \bar{h}\left(T_o - T_{ld}\right) \tag{30}$$

where: $\bar{h}$ = natural convection heat transfer coefficient,

T_{ld} = temperature of the lower drywell environment.

The natural convection heat transfer coefficient depends on the Rayleigh number. The Rayleigh number is:

$$Ra_L = \frac{g\beta\left(T_o - T_{ld}\right)L_s^3}{\nu\alpha} \tag{31}$$

where: Ra_L = Rayleigh number,

β = thermal expansion coefficient of steam = $1/T_{ld}$ assuming ideal gas behavior,

ν = kinematic viscosity of steam,

α = thermal diffusivity of steam,

L_s = characteristic length of the shield top.

The characteristic length of a horizontal heated plate is one-half its width (Reference 4). The floor drain sump is approximately one meter wide; therefore, the characteristic length of the shield roof is 0.5 meters. Evaluating the Rayleigh number for saturated steam at ultimate containment pressure (180 psig, 190 C)

yields:

$$Ra_L = 5.5x10^8 K^{-1}(T_o - T_{ld}) \tag{32}$$

For $10^7 \leq Ra_L \leq 10^{11}$, the Nusselt number (Reference 4) for an upward-facing heated plate undergoing natural convection is:

$$Nu_L = 0.15 Ra_L^{1/3} \tag{33}$$

The average natural convection heat transfer coefficient is:

$$\bar{h} = \frac{k}{L_s} Nu \tag{34}$$

where: k = thermal conductivity of steam in the lower drywell.

Combining Equations (30) and (32) through (34) yields:

$$q''_{uw} = 8.79(T_o - T_{ld})^{4/3} \ \ W/m^2K^{4/3} \tag{35}$$

which can be rearranged to:

$$T_o = T_{ld} + \left(\frac{q''_{uw}}{8.79\ W/m^2}\right)^{3/4} K \tag{36}$$

Inserting this into Equation (30) yields:

$$q''_{uw} = \frac{k_w}{H_{uw}}\left[T_i - T_{ld} - \left(\frac{q''_{uw}}{8.79\ W/m^2}\right)^{3/4} K\right] \tag{37}$$

This equation can be solved iteratively to determine the heat flux which can be transferred through the upper wall of a given wall height. The wall height requirement for transferring a given heat flux is:

$$H_{uw} \leq \frac{k_w}{q''_{uw}}\left[T_i - T_{ld} - \left(\frac{q''_{uw}}{8.79\ W/m^2}\right)^{3/4} K\right] \tag{38}$$

The decay heat level in the ABWR 24 hours after accident initiation is approximately 0.6%. The volumetric heat generation rate of debris at this time can be determined using Equation (6). It is 0.9 MW/m^3. The debris/wall interface temperature which will guarantee that the debris remains frozen is 1700 K. The temperature of saturated steam at ultimate containment pressure is 190 C. The height of the upper shield wall, which will transfer all of the heat generated in the channel for these conditions, is:

$$H_{uw} \leq \frac{k_w}{8.52\ W/m^2K} \tag{39}$$

If the upper shield wall satisfies this inequality, it will be capable of transferring all of the heat generated by debris in the channel; and, as a result, guarantee long term debris solidification even if the lower drywell has not been flooded. To be acceptable, the height of the shield wall must satisfy the inequalities in Equations (39) and (28).

19ED.5.2 Lower Shield Wall (Below Lower Drywell Floor)

One side of the lower shield wall is in contact with debris and the other is in direct contact with the basemat. The basemat is constructed of concrete. A conservative estimate of the lower shield wall depth can be made by assuming that concrete acts like a perfect insulator. Thus, no heat is allowed to pass from the shield wall to the basemat. The boundary condition between the debris and the wall is conservatively assumed to be constant heat flux. The initial burst of energy into the shield wall, caused by debris freezing, has ample time to distribute itself throughout the wall. With these boundary conditions, the temperature distribution in the lower wall can be determined analytically.

The analytical solution will provide a means for determining the time required for each of the interfaces to reach their allowable temperature limits for a given heat flux. The wall/basemat interface temperature should not exceeded the melting point of concrete (1450 K). Continued debris solidification is guaranteed if the wall/basemat interface temperature does not exceed 1700 K. The wall will be sized so that the limits are not exceeded during the first 24 hours after initial debris solidification. The upper shield wall will be sized so that it can transfer the full decay heat load after 24 hours has elapsed, as discussed in Section 19ED.5.1.

The temperature distribution in a slab (Reference 5), subjected to constant heat flux at one surface ($x = H_{lw}$) and insulated at the other ($x = 0$), is:

$$T_{lw}(x,t) - T_{i,lw} = \frac{q''_{lw}t}{\rho_w c_{p,w} H_{lw}} + \frac{q''_{lw} H_{lw}}{k_w} * \left\{ \frac{3x^2 - H_{lw}^2}{6H_{lw}^2} - \frac{2}{\pi^2} \sum_{n=1}^{\infty} \frac{(-1)^n}{n^2} e^{-k_w n^2 \pi^2 t / H_{lw}^2} \cos\frac{n\pi x}{H_{lw}} \right\} \tag{40}$$

where: T_{lw} = temperature distribution in the lower shield wall,

$T_{i,lw}$ = adjusted initial temperature of the shield wall,

q''_{lw} = heat flux through the lower shield wall,

$c_{p,w}$ = specific heat of the shield wall,

H_{lw} = depth of the lower shield wall below the lower drywell floor.

The maximum temperature at each interface is achieved as $t \rightarrow \infty$. The maximum temperatures at the wall/debris interface, $T_{w/d}$, and the wall/basemat interface, $T_{lw/b}$, are:

$$T_{w/d} = T_{i,lw} + \frac{q''_{lw}t}{\rho_w c_{p,w} H_{lw}} + \frac{q''_{lw} H_{lw}}{3k_w} \tag{41}$$

and:

$$T_{w/b} = T_{i,lw} + \frac{q''_{lw}t}{\rho_w c_{p,w} H_{lw}} - \frac{q''_{lw} H_{lw}}{6k_w} \tag{42}$$

The heat flux through the lower wall is bounded by one-half of the heat flux generated in the channel when the sizes of the upper and lower wall are comparable. The actual heat flux will be less because the upper wall is free to convect to the lower drywell environment and will accept more heat flux than the lower wall. The maximum heat generation in the channel corresponds to a decay heat level of one-percent. Since decay heat decreases with time, using the maximum value bounds the temperature response of the lower shield wall. Using Equation (6), one-half of the heat flux generated in the channel is:

$$q''_{1/2\ chan} = \frac{Q_{1\%,dh} \rho_{cm} H_o}{2m_{cm}} = \left(q''_{lw}\right)_{limiting} \tag{43}$$

where: m_{cm} = total corium mass.

The initial temperature of the shield wall should be adjusted to account for the energy it absorbs during the debris freezing process. If both shield walls have the same thickness, the adjusted temperature is:

$$T_{i,lw} = T_i + \frac{\rho_{cm} h_{lh} H_o}{2\rho_w c_{p,w} H_{lw}} \tag{44}$$

Equations (41) through (44) can be used to determine if a chosen lower shield wall depth will satisfy the requirement of keeping the debris in the channel frozen for at least 24 hours. After 24 hours has elapsed, the upper shield wall will be able to remove the entire amount of heat generated in the channel (Section 19ED.5.1).

The process for determining an acceptable wall depth proceeds as follows. First a wall depth is chosen

which is comparable to the upper shield wall height. Then, the adjusted initial temperature and heat loads are calculated using Equations (44) and (43), respectively. The interface temperatures at 24 hours are determined by Equations (41) and (42). If $T_{w/d} < 1700$ K and $T_{w/b} < 1450$ K, the chosen depth is acceptable. If not, a new depth is chosen and the process repeated until an acceptable depth is determined. An example of this procedure is given in part (4) of Section 19ED.6.

19ED.6 EXAMPLE CALCULATION

The sizing requirements for the floor drain corium shield were set forth in Sections 19ED.4 and 19ED.5 based on a chosen channel height. An example sizing exercise is presented in this section. The selected channel height, H_o, is one centimeter. Representative shield wall material properties are shown in Table 19ED.4-1.

(1) Melting Point of Shield Material Above Initial Contact Temperature

The initial contact temperature between the debris and the channel wall given in Equation (16) is:

$$T_s = \frac{-(c_o - 2T_i) \pm \sqrt{(c_o - 2T_i)^2 - 4(T_i^2 - c_o T_{f,m})}}{2} \tag{16}$$

where:

$$c_o = \left(\frac{\pi k_f \rho_{cm} h_{lh} \alpha_w}{2k_w^2}\right)$$

The parameters required to evaluate this equation are:

- T_i = initial temperature of the shield wall, 330 K,
- $T_{f,m}$ = debris freezing temperature ranges from 1700 K to 2500 K,
- k_f = debris thermal conductivity, 30 W/m^{2}K,
- ρ_{cm} = density of corium, 9000 kg/m^3,
- h_{lh} = debris latent heat of fusion, 2.7 x 10^5 J/kg,
- α_w = thermal diffusivity of the shield wall material, a representative value of 1.48 x 10^{-6} m^2/sec will be used,
- k_w = thermal conductivity of shield wall material, a representative value of 4 W/mK will be used.

Evaluating Equation (16) yields interface temperatures of 1560 K and 2180 K for debris melting temperatures of 1700 K and 2500 K, respectively. The melting temperature of the representative shield material is over 2200 K; therefore, it passes this test.

(2) Channel Length

The equations, needed to determine the channel length required to ensure that a plug is formed at the entrance of the channel before debris spills into the sump, are (9) and (26). These equations combine to give:

$$L_{freeze} = a_o t_{freeze}^{3/2} - \frac{a_o b_o}{H_o} t_{freeze}^2 \tag{45}$$

where:

$$t_{freeze} = \frac{H_o^2 \rho_{cm} h_{lh}}{8k_f(T_{f,m} - T_s)}$$

$$a_o = \frac{4}{5}\sqrt{\frac{2g\dot{m}_{ves}}{\rho_{cm} A_{ld}}}$$

$$b_o = \frac{5}{3}\sqrt{\frac{2k_f(T_{f,m} - T_s)}{\rho_{cm} h_{lh}}}$$

and H_o is the assumed channel height (0.01m).

The maximum length results when $T_{f,m}$ = 1700 K. The contact temperature was shown previously to be 1560 K for this freezing temperature. The other parameters required to evaluate these equations are:

- $\dot{m}_{ves}$ = ejection rate of debris from a failed vessel, 6000 kg/sec (conservative maximum),
- $A_{ld,min}$ = minimum floor area of the lower drywell, specified as 0.02 m^2/MWth in the EPRI ALWR Requirements Document; it is equal to 79 m^2.

Using these parameters, the plug formation time is 7.2 seconds and the required channel length is 1.06 meters. This length was determined using a highly conservative corium discharge rate. The analysis assumed a constant discharge rate equal to maximum discharge rate predicted using a highly conservative model. The actual discharge rate will be lower. If the length requirement is highly

restrictive, the discharge rate could be refined with additional effort.

(3) Shield Height, H_{uw}, Above Lower Drywell Floor

The height requirements for the upper shield wall are given in Equations (28) and (39). These equations are:

$$H_{uw} \geq 0.33 \text{ m} \tag{28}$$

and:

$$H_{uw} \leq \frac{k_w}{8.52 \text{ W/m}^2\text{K}} \tag{39}$$

For a wall conductivity of 4 W/mK, these inequalities require:

$$0.33 \text{ m} \leq H_{uw} \leq 0.47 \text{ m} \tag{46}$$

A height of 0.4 meters is chosen.

(4) Shield Depth, H_{lw}, Below Lower Drywell Floor

The lower shield wall should be sized according to Equations (41) through (44). An initial height of 0.4 meters is chosen to begin the determination of acceptability. The adjusted initial temperature of the lower shield wall accounting for energy absorption during debris freezing is:

$$T_{i,lw} = T_i + \frac{\rho_{cm} h_{lb} H_o}{2\rho_w c_{p,w} H_{lw}} = 341 \text{ K} \tag{44}$$

where: ρ_w = density of wall material,

$c_{p,w}$ = specific heat of wall material.

The limiting heat flux through the lower wall is:

$$(q''_{lw})_{limiting} = \frac{Q_{1\%,dh} \rho_{cm} H_o}{2m_{cm}} = 7520 \text{ W/m}^2 \tag{47}$$

where: $Q_{1\%,dh}$ = 1% of decay heat, 39.26 Mw;

m_{cm} = mass of corium, 235 Mg.

The wall/debris, $T_{w/d}$, and the wall/basemat, $T_{w/b}$, interface temperatures are given by:

$$T_{w/d} = T_{i,lw} + \frac{q''_{lw} t}{\rho_w c_{p,w} H_{lw}} + \frac{q''_{lw} H_{lw}}{3k_w} \tag{41}$$

and:

$$T_{w/b} = T_{i,lw} + \frac{q''_{lw}}{\rho_w c_{p,w} H_{lw}} - \frac{q''_{lw} H_{lw}}{6k_w} \tag{42}$$

where: t = time assumed to be 24 hours.

Evaluating these expressions yields $T_{w/d}$ = 1190 K and $T_{w/b}$ = 820 K. Since these temperatures meet the requirements for long-term debris solidification ($T_{w/d}$ < 1700 K and $T_{w/b}$ < 1450 K), the chosen wall depth is acceptable.

(5) Summary of Shield Requirements

A proposed floor drain sump corium shield with a specified channel height of one centimeter and wall material properties shown in Table 19ED.4-1 will prevent corium ingression into the sump if it meets the following requirements:

Minimum melting point of shield material: 2180 K,

Channel Length: 1.06 m,

Height above lower drywell floor: 0.4 m,

Depth below lower drywell floor: 0.4 m.

19ED.7 DETAILED DESIGN ISSUES

During detailed design of the ABWR, the exact shield material and shield dimensions will be chosen. Cursory examination of material properties indicate alumina may be an acceptable wall material. The requirements for the shield are stated in Section 19ED.3. Example calculations of the requirements are shown in Section 19ED.6. Interference with under-vessel servicing equipment will be considered in determining if the proposed dimensions are acceptable; if not, the sizing process will be redone for a new channel height and/or a new shield wall material. The number and width of channels in the shield will be chosen to meet the design requirements for water flow into the sump during normal ABWR plant operation.

19ED.8 REFERENCES

1. Frank P. Incropera and David P. DeWitt, *Fundamentals of Heat and Mass Transfer, 2nd Ed.*, John Wiley and Sons, 1985, pp. 85-86.

2. Glen E. Myers, *Analytical Methods in Conduction Heat Transfer*, Genium Publishing Corp., Schenectady, NY, 1987, p. 202.

3. Frank P. Incropera and David P. DeWitt, *Fundamentals of Heat and Mass Transfer, 2nd Ed.*, John Wiley and Sons, 1985, p. 203.

4. Frank P. Incropera and David P. DeWitt, *Fundamentals of Heat and Mass Transfer, 2nd Ed.*, John Wiley and Sons, 1985, pp. 433-435.

5. H.S. Carslaw and J.C. Jeager, *Conduction of Heat in Solids, 2nd Ed.*, Oxford University Press, 1959, pp. 112-113.

6. *Mark's Standard Handbook for Mechanical Engineers, 8th Ed.*, Theodore Baumeister, Editor-in Chief, McGraw-Hill Book Company, 1978, pp. 6-171 to 6-177.

APPENDIX 19E
ATTACHMENT 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 passes through the vacuum breakers.

The ABWR contains eight vacuum breakers. ABWR vacuum breakers are swing check valves designed to open passively when wetwell pressure exceeds drywell pressure by 0.0035 MPa (0.5 psid). 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 pressure differential will produce a large force tending to close the vacuum breaker valves. A pressure differential of +0.048 MPa (+7 psid) 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 (+14 psid) 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 access the impact of vacuum breaker leakage on the performance of the ABWR during a severe accident. The event tree analysis is contained in Section 19EE.2. Section 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.2-1. 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 catastrophically fails, a large pathway is established between the drywell and wetwell. The deterministic analysis described in Section 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 Section 19E.2.3.3 assumed a failure probability for vacuum breaker full reverse flow of 6.7E-2/demand 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.

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 open, 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, the depressurization transients which lead to opening of the vacuum breakers are very mild. 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 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 by:

$$P(VB) = \frac{8N_{close}T_{stroke}}{T_{close}} \quad (1)$$
$$= 3.9E-3/\text{demand}$$

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 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 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 Section 19EE.3. The BWR operating history described in the previous section (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 BWR's 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, and 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 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 which corresponds 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 by:

$$P(VB\_LEAK) = 1 - \left(1 - \frac{N_{leak} T_{test}}{T_{leak}}\right)^8 \tag{2}$$
$$= 0.17 / \text{demand}$$

The value used in the quantification of the containment event trees is 0.18/demand. The difference between this value and the value calculated above, using a more detailed analysis, results in a slight conservatism in the analysis.

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 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 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 1.63 cm. Therefore, high probability, 0.9, 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.

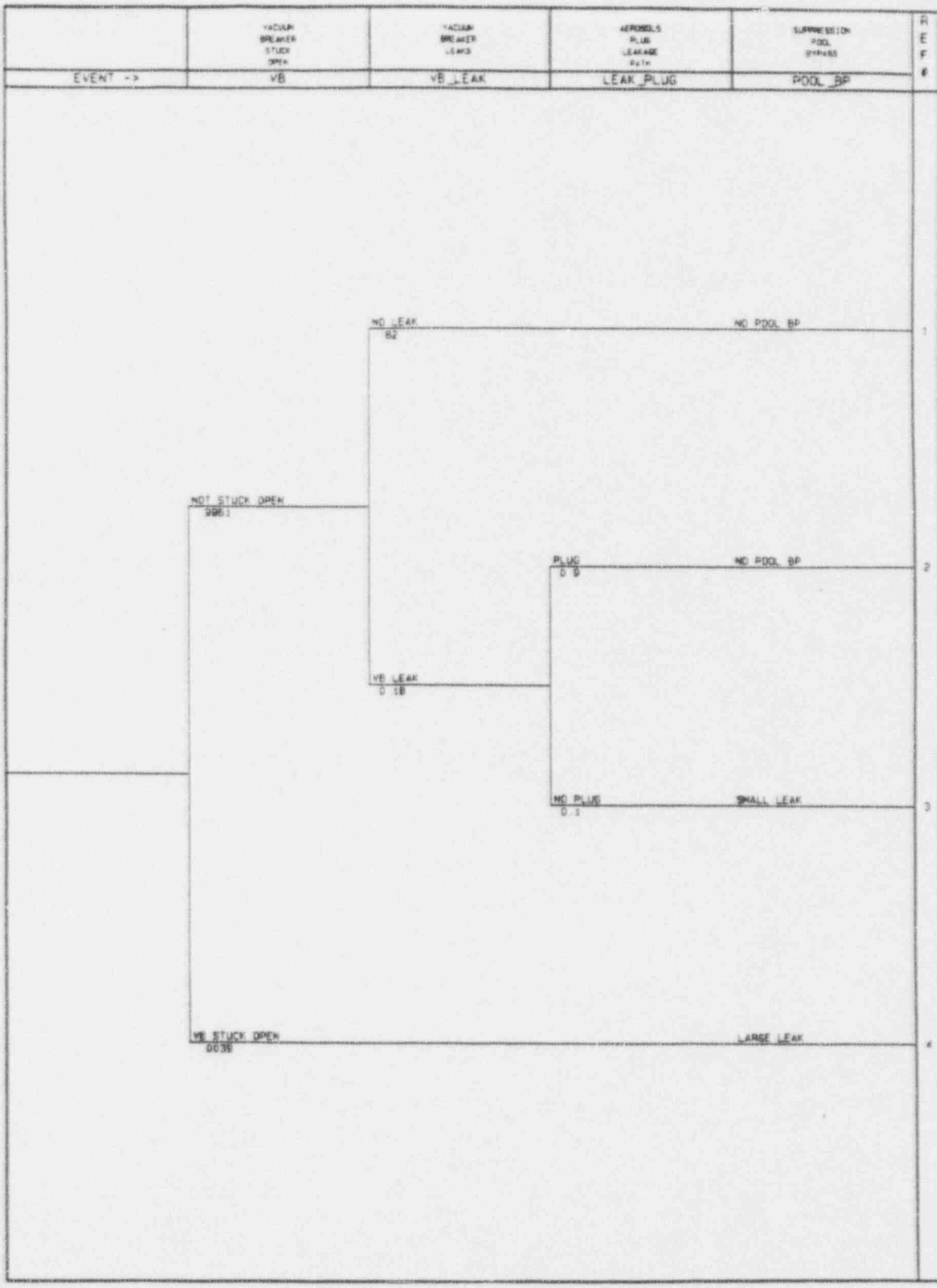


Figure 19EE.2-1

CONTAINMENT EVENT EVALUATION DET FOR SUPPRESSION POOL BYPASS

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. Five variations were analyzed. In each case the overpressure relief rupture disk opened when the wetwell pressure reached 0.72 MPa (90 psig). 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 MAAP3.0B aerosol plugging model developed by E.U. Vaughan (Reference 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 (623 to 3115 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 (3115 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 2), the NRC asserted that the data cited by Morewitz (Reference 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 2).

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 (6.45 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.3-1. 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.3-1 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 cases analyzed. 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 (7.13 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 (90 psig) 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 flowrate 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. 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 decomposition event tree analysis in Section 19EE.2.

Table 19EE.3-1

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 (hrs)**										
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%

* Plugging presumed to be ineffective
*** Not calculated

19EE.4 SUMMARY OF RESULTS

19EE.4.1 Quantification of DET

The quantified event tree is shown in Figure 19EE.2-1. The probabilities for different leakage areas are transferred to containment event trees. The probabilities are listed below:

No Leakage	0.9782,
Small Leakage	0.0179,
Large Leakage	0.0039.

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 Section 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 ~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.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, 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 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 (approximately 98 percent) of the accident demands. Small amounts of leakage have a probability of 1.8 percent per event, and can result in medium volatile fission product releases (one to ten percent of initial inventory). Volatile fission product releases on the order of 10 to 20 percent 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 only 0.39 percent.

19EE.6 REFERENCES

1. Vaughan, E.U., *Simple Model for Plugging of Ducts by Aerosol Deposits*, Trans. Am. Nuclear. Soc., 28, 507, 1978.

2. NRC/IDCOR Technical Issue 13A, 1986.

3. Morewitz, H.A, *Leakage of Aerosols from Containment Buildings*, Health Physics, Vol. 42, No. 2, 1982, pp. 195-207.

APPENDIX 19F

TABLE OF CONTENTS

APPENDIX 19F.1

ILLUSTRATIONS

SECTION 19F.1

GE PROPRIETARY - provided under separate cover

Page	Amendment
19F.1-1	28
19F.1-2	28
19F.1-3	28
19F.1-4	4
19F.1-5	4

SECTION 19F.2

CONTENTS

SECTION 19F.2

TABLES

ILLUSTRATIONS

SECTION 19F.2

ILLUSTRATIONS (Continued)

Figure	Title	Page

SECTION 19F.2

ILLUSTRATIONS (Continued)

SECTION 19F.2

GE PROPRIETARY - provided under separate cover

Page	Amendment
19F.2-1	28
19F.2-2	28
Deleted 19F.2-3	
Thru 19F.2-33	

SECTION 19F.3

CONTENTS

SECTION 19F.3

TABLES

ILLUSTRATIONS

SECTION 19F.3

GE PROPRIETARY - provided under separate cover

Page	Amendment
19F.3-1	28
19F.3-2	28
19F.3-2.1	28
19F.3-3	28
19F.3-4	28
19F.3-4.1	28
19F.3-5	28
19F.3-5.1	28
19F.3-6	28
19F.3-7	4
19F.3-8	4
19F.3-9	28
19F.3-10	28

19F.4 REFERENCES

1. Deleted

2. Deleted

3. Pfeiffer, P. A., Kennedy, J. M., and Marchertas, A. H., *Thermal Effects in Concrete Containment Analysis*, Fourth Containment Integrity Workshop, June 15-17, 1988.

4. Shield, R. T., and Drucker, D. C., *Design of Thin-Walled Torispherical and Toriconical Pressure-Vessel Heads, Journal of Applied Mechanics*, Transaction of ASME, June 1961.

5. Galletly, G. D., and Radhamohan, S. K., "Elastic-Plastic Buckling of Internally-Pressurized Thin Torispherical Shells," *Journal of Pressure Vessel Technology*, ASME, Vol. 101, August 1979.

6. Construction Technology Laboratories, *Concrete Containment Tests, Phase 2: Structural Elements with Liner Plates*, EPRI NP-4867M, August 1987.

7. Anatech International Corporation, *Methods for Ultimate Load Analysis of Concrete Containments: Second Phase*, EPRI NP-4869M, March 1987.

8. Argonne National Laboratory, *Characterization of Nuclear Reactor Containment Penetrations Final Report*, NUREG/CR-3855, February 1985.

9. Clauss, D. B., von Riesemann, W. A., and Parks, M. B., *Containment Penetrations*, SAND88-0331C.

10. Galletly, G. D., and Blachnut, J., *Torispherical Shells Under Internal Pressure - Failure Due to Asymmetric Plastic Buckling or Axisymmetric Yielding*, Proc. of Institution of Mech. Engineers, Vol. 199, No. C3, 1985

11. Galletly, G.D., *A Simple Design Equation for Preventing Buckling in Fabricated Torispherical Shells under Internal Pressure*, ASME Journal of Pressure Vessel Technology, Vol. 108, November 1986.

12. 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.

ATTACHMENT A

APPENDIX 19F

UNCERTAINTY ANALYSIS

ATTACHMENT A

CONTENTS

TABLES

ILLUSTRATIONS

ATTACHMENT A

CONTENTS

TABLES

ILLUSTRATIONS

SECTION 19FA.0

GE PROPRIETARY - provided uner separate cover

Page	Amendment
19FA.0-4	21

SECTION 19FA.1

GE PROPRIETARY - provided under separate cover

Page	Amendment
19FA.1-1	28
19FA.1-1.1	28
19FA.1-2	21
19FA.1-3	28

APPENDIX 19H
SEISMIC CAPACITY ANALYSIS

APPENDIX 19H

TABLE OF CONTENTS

19H.1 INTRODUCTION

This section 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 of Chapter 3. 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 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 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.

SECTION 19H.2

CONTENTS

ILLUSTRATIONS

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:

$$A = A_m \epsilon_\gamma \epsilon_\mu \qquad (19H\text{-}1)$$

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.

$$P_f(A < a \; Q) = \phi\left[\frac{1}{\beta_\gamma} \ln\left(\frac{a}{A_m}\right) + \frac{\beta_\mu}{\beta_\gamma} \phi^{-1}(Q)\right] \qquad (19H\text{-}2)$$

where $\phi(.)$ is the standard Gaussian cumulative distribution function. Figure 19H.2-1 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 Fig. 19H.2-1 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.

$$\beta_c = \sqrt{\beta_\gamma^2 + \beta_\mu^2} \qquad (19H\text{-}3)$$

The fragility curve corresponding to the median value A_m with associated composite logarithmic standard deviation can be computed by the following equation:

$$P_f(A \leq a) = \phi\left[\frac{1}{\beta_c} \ln\left(\frac{a}{A_m}\right)\right] \qquad (19H\text{-}4)$$

This composite fragility curve is also called the mean fragility curve and is shown as the dashed curve in Fig. 19H.2-1 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.

$$A_m = F A_d \qquad (19H\text{-}5)$$

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.

$$F = \prod_i F_i \qquad \text{(19H-6)}$$

The overall composite logarithmic standard deviation is SRSS of the composite logarithmic standard deviations in the individual factors of safety.

$$\beta_c = \sqrt{\sum_i \beta^2_{ci}} \qquad \text{(19H-7)}$$

Knowing the median peak ground acceleration (Am) and associated logarithmic standard deviation (β_c), the HCLPF capacity is obtained using the equation below.

$$\text{HCLPF} = \text{Am} \exp(-2.326\,\beta_c) \qquad \text{(19H-7a)}$$

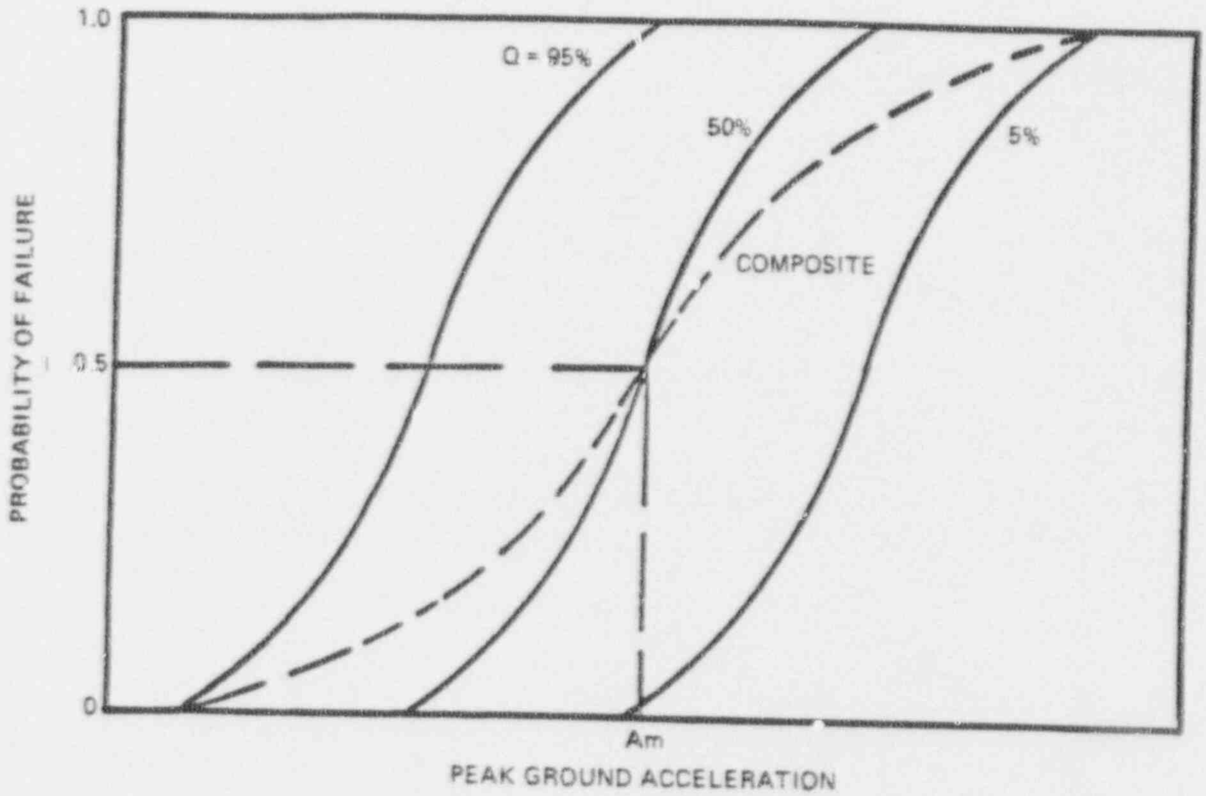


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Figure 19H.2–1 TYPICAL FRAGILITY CURVES

SECTION 19H.3

CONTENTS

TABLES

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 six 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 of 3g (see Table 19H.4-6).

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};

$$F = F_c F_{rs} \qquad (19H\text{-}8)$$

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:

$$F_c = F_s F_u \qquad (19H\text{-}9)$$

where F_s represents the ultimate structural strength margin above the design SSE load, and F_u is 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.

a. Strength Factor (F_s)

The strength factor associated with seismic load can be calculated using the following equation.

$$F_s = \frac{P_u - P_n}{P_s} \qquad (19H\text{-}10)$$

where P_u is the actual ultimate strength, P_n is the normal operating and operation transient (i.e., SRV) loads, and P_s is 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 4000 psi. The specified yield strength of reinforcing steel of ASTM A615, Grade 60 is 60,000 psi. The structural steel material for the pedestal shells and stiffeners is A572, Gr. 50, for which the specified yield strength is 50,000 psi. 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:

(1) To meet the design specifications, the contractor attempts to create a mix that has an "average" strength somewhat above the design strength, and

(2) 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 5). 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 5 and 6).

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 7).

$$v_u = v_c + v_s \qquad (19H\text{-}11)$$

$$= 8.3\sqrt{f'_c} - 3.4\sqrt{f'_c}\left(\frac{h}{w} - \frac{1}{2}\right) + \frac{N}{4wt}$$

$$+ \rho_{se} f_y$$

where

v_c	=	shear strength provided by concrete
v_s	=	shear strength provided by reinforcing steel
f'_c	=	concrete compressive strength in psi
h	=	wall height
w	=	wall length
N	=	bearing load
fy	=	yield strength of reinforcing steel in psi
t	=	wall thickness
ρse	=	$A\rho_v + B\rho_h$
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 Eq. 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 7) accounting for both concrete and reinforcing steel material strength variabilities. There is also an uncertainty associated with Eq. 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 7). 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.

b. 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, u. It is defined as the ratio of the maximum displacement to the displacement at yield. Newmark, Reference 8, has shown that in the amplified acceleration range (approximately 2 to 8 Hz) the inelastic energy absorption factor F_u can be estimated by

$$F_u = \epsilon \sqrt{2\mu-1} \qquad (19H\text{-}12)$$

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 9). For rigid structures (fundamental frequency above 20 Hz), the following equation given by Reference 9 may be used.

$$F_u = \epsilon \mu^{0.13} \qquad (19H\text{-}13)$$

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 5, 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 Eq. 19H-12. The associated logarithmic standard deviation is 0.25 (Reference 5). 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 10) 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.

$$F_{rs} = F_{sa} F_d F_{ssi} F_m F_{mc} F_{ecc} \quad \text{(19H-14)}$$

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

a. 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 11) 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.

$$F_{sa} = AF_d / AF_m \quad \text{(19H-15)}$$

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
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 2. No variability exists for frequencies above 33 Hz.

b. 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 11.

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 11). 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.

$$F_d = AF_{dd} / AF_{md} \qquad (19H\text{-}16)$$

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.

$$\beta_c = \ln(AF_{bd} / AF_{md}) \qquad (19H\text{-}17)$$

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

c. 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 of Chapter 3. 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.

d. 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.

e. 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.

f. 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.3-1 through 19H.3-3 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 0.3g.

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.3-4 shows the margin in each of the strength and response factors. The resulting fragility is 4.1g median peak ground acceleration with a lograrithmic standard deviation of 0.44.

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

capacity is estimated for the ABWR non-category I building structures. The associated logarithmic standard deviation is estimated to be 0.5 (which is larger than the median value of 0.45 in the data base).

Table 19H.3-1

SEISMIC FRAGILITY FOR REACTOR BUILDING

Component: Shear Walls
Failure Mode: Shear

	Factor of Safety		Median Value	β_c
F_c	F_s :	strength margin	3.32	0.17
	F_u :	inelastic energy absorption	2.00	0.25
F_{rs}	F_{sa} :	spectral shape margin	1.34	0.20
	F_d :	damping margin	1.19	0.18
	F_{ssi} :	soil-structure interaction	1.00	0.10
	F_m :	modeling factor	1.00	0.15
	F_{mc} :	modal combination	1.00	0.05
	F_{ecc} :	earthquake component combination	1.00	0.05
	F :	**overall factor**	**10.59**	**0.45**

Median Peak Ground Acceleration = $F * A_d = 10.59 * 0.3 = 3.2g$

HCPLF = 1.12g

Table 19H.3-2

SEISMIC FRAGILITY FOR CONTAINMENT

Component: Containment
Failure Mode: Shear

			Factor of Safety	Median Value	β_c
F_c	F_s	:	strength margin	3.22	0.15
	F_u	:	inelastic energy absorption	2.00	0.25
F_{rs}	F_{sa}	:	spectral shape margin	1.34	0.20
	F_d	:	damping margin	1.19	0.18
	F_{ssi}	:	soil-structure interaction	1.00	0.10
	F_m	:	modeling factor	1.00	0.15
	F_{mc}	:	modal combination	1.00	0.05
	F_{ecc}	:	earthquake component combination	1.00	0.05
	F	**:**	**overall factor**	**10.27**	**0.44**

Median Peak Ground Acceleration = $F * A_d = 10.27 * 0.3 = 3.1g$

HCLPF = 1.11g

Table 19H.3-3

SEISMIC FRAGILITY FOR RPV PEDESTAL

Component: RPV Pedestal
Failure Mode: Flexural

			Factor of Safety	Median Value	β_c
F_c	F_s	:	strength margin	4.31	0.14
	F_u	:	inelastic energy absorption	2.65	0.25
F_{rs}	F_{sa}	:	spectral shape margin	1.27	0.20
	F_d	:	damping margin	1.15	0.18
	F_{ssi}	:	soil-structure interaction	1.00	0.10
	F_m	:	modeling factor	1.00	0.15
	F_{mc}	:	modal combination	1.00	0.05
	F_{ecc}	:	earthquake component combination	1.00	0.05
	F	:	**overall factor**	**16.68**	**0.44**

Median Peak Ground Acceleration = $F * A_d$ = 16.68 * 0.3 = 5.0g

HCPLF = 1.8g

Table 19H.3-4

SEISMIC FRAGILITY FOR CONTROL BUILDING

Component: Shear Walls
Failure Mode: Shear

Factor of Safety				Median Value	β_c
F_c	F_s	:	strength margin	4.71	0.17
	F_u	:	inelastic energy absorption	2.00	0.25
F_{rs}	F_{sa}	:	spectral shape margin	1.27	0.20
	F_d	:	damping margin	1.15	0.15
	F_{ssi}	:	soil-structure interaction	1.00	0.10
	F_m	:	modeling factor	1.00	0.15
	F_{mc}	:	modal combination	1.00	0.05
	F_{ecc}	:	earthquake component combination	1.00	0.05
	F	**:**	**overall factor**	**13.75**	**0.44**

Median Peak Ground Acceleration = $F * A_d = 13.75 * 0.3 = 4.1g$

HCPLF = 1.47g

SECTION 19H.4

CONTENTS

TABLES

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 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 2-1/2" 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. Its median ground acceleration capacity is 5.0g with a logarithmic standard deviation of 0.33. The individual factors contributing to the median capacity are shown in Table 19H.4-1.

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.4-2 through 19H.4-5 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 inserting 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. The resulting seismic fragility is 1.4g median peak ground acceleration with a logarithmic standard deviation of 0.35.

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
Battery chargers/Inverters
Electric equipment (chatter failure mode)

Switchgear/Motor control centers
Transformers (not off-site transformers)
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 valves are summerized in Table 19H.4-6. These generic seismic capacities are selected from a review of ALWR recommendation (Reference 2) and other PRA studies (Reference 3 and 4).

Table 19H.4-1

SEISMIC FRAGILITY FOR REACTOR PRESSURE VESSEL

Component: RPV
Failure Mode: Support

			Factor of Safety	Median Value	β_c
F_c	F_s	:	strength margin	12.08	0.14
	F_u	:	inelastic energy absorption	1.00	0.00
F_{rs}	F_{sa}	:	spectral shape margin	1.27	0.20
	F_d	:	damping margin	1.08	0.12
	F_{ssi}	:	soil-structure interaction	1.00	0.10
	F_m	:	modeling factor	1.00	0.15
	F_{mc}	:	modal combination	1.00	0.05
	F_{ecc}	:	earthquake component combination	1.00	0.05
	F	**:**	**overall factor**	**16.57**	**0.33**

Median Peak Ground Acceleration = $F * A_d = 16.57 * 0.3 = 5.0g$

HCPLF = 2.32g

Table 19H.4-2

SEISMIC FRAGILITY FOR SHROUD SUPPORT

Component: Shroud Support
Failure Mode: Buckling

		Factor of Safety		Median Value	β_c
F_c	F_s	:	strength margin	4.99	0.20
	F_u	:	inelastic energy absorption	1.00	0.00
F_{rs}	F_{sa}	:	spectral shape margin	1.27	0.20
	F_d	:	damping margin	1.08	0.12
	F_{ssi}	:	soil-structure interaction	1.00	0.10
	F_m	:	modeling factor	1.00	0.15
	F_{mc}	:	modal combination	1.00	0.05
	F_{ecc}	:	earthquake component combination	1.00	0.05
	F	**:**	**overall factor**	**6.84**	**0.36**

Median Peak Ground Acceleration = $F * A_d = 6.84 * 0.3 = 2.0g$

HCPLF = 0.87g

Table 19H.4-3

SEISMIC FRAGILITY FOR CRD GUIDE TUBES

Component: CRD Guide Tubes
Failure Mode: Buckling

			Factor of Safety	Median Value	β_c
F_c	F_s	:	strength margin	4.16	0.20
	F_u	:	inelastic energy absorption	1.00	0.00
F_{rs}	F_{sa}	:	spectral shape margin	1.18	0.20
	F_d	:	damping margin	1.25	0.10
	F_{ssi}	:	soil-structure interaction	1.00	0.10
	F_m	:	modeling factor	1.00	0.15
	F_{mc}	:	modal combination	1.00	0.05
	F_{ecc}	:	earthquake component combination	1.00	0.05
	F	**:**	**overall factor**	**6.14**	**0.36**

Median Peak Ground Acceleration = $F * A_d = 6.14 * 0.3 = 1.8g$

HCPLF = 0.78g

Table 19H.4-4

SEISMIC FRAGILITY FOR CRD HOUSINGS

Component: CRD Housings
Failure Mode: Plastic Yielding

			Factor of Safety	Median Value	β_c
F_c	F_s	:	strength margin	7.92	0.35
	F_u	:	inelastic energy absorption	1.00	0.00
F_{rs}	F_{sa}	:	spectral shape margin	1.18	0.20
	F_d	:	damping margin	1.25	0.10
	F_{ssi}	:	soil-structure interaction	1.00	0.10
	F_m	:	modeling factor	1.00	0.15
	F_{mc}	:	modal combination	1.00	0.05
	F_{ecc}	:	earthquake component combination	1.00	0.05
	F	:	**overall factor**	**11.68**	**0.46**

Median Peak Ground Acceleration = F * A_d = 11.68 * 0.3 = 3.5g

HCPLF = 1.2g

Table 19H.4-5

SEISMIC FRAGILITY FOR FUEL ASSEMBLIES

Component: Fuel Assemblies
Failure Mode: Channel Excessive Deflection

			Factor of Safety	Median Value	β_c
F_c	F_s	:	strength margin	2.36	0.00
	F_u	:	inelastic energy absorption	1.32	0.21
F_{rs}	F_{sa}	:	spectral shape margin	1.43	0.20
	F_d	:	damping margin	1.06	0.06
	F_{ssi}	:	soil-structure interaction	1.00	0.10
	F_m	:	modeling factor	1.00	0.15
	F_{mc}	:	modal combination	1.00	0.05
	F_{ecc}	:	earthquake component combination	1.00	0.05
	F	**:**	**overall factor**	**4.72**	**0.35**

Median Peak Ground Acceleration = $F * A_d = 4.72 * 0.3 = 1.4g$

HCPLF = 0.62g

Table 19H.4-6

SEISMIC CAPACITY SUMMARY

Structure/Component	Failure Mode	Fragility Capacity[a] Am (g)	Combined[b] Uncertainty	HCPLF (g)
Reactor Building	Wall Shear	3.2	0.45	1.12
Containment	Shear	3.1	0.44	1.11
RPV Pedestal	Flexural	5.0	0.44	1.80
Control building	Shear	4.1	0.44	1.47
Service Water Pump House	Structural	1.7	0.45	0.60
Reactor pressure vessel	Support	5.0	0.33	2.32
Shroud support	Buckling	2.0	0.36	0.87
CRD guide tubes	Buckling	1.8	0.36	0.78
CRD housing	Plastic yielding	3.5	0.46	1.20
Fuel Assemblies	Channel deflection	1.4	0.35	0.62
Hydraulic Control Unit	LOF	2.0	0.50	0.63
Cable trays	Support	3.0	0.60	0.74
Large flat-bottom storage tanks	Anchorage	2.1	0.45	0.79
Air-operated valves	Stem binding/Air line	3.0	0.60	0.74
Heat Exchanger	Anchorage	2.0	0.45	0.70
Off-site power	Ceramic insulators	0.3	0.55	0.08
Batteries and battery racks	Anchorage/LOF	3.3	0.46	1.13
Battery chargers/Inverters	LOF	2.2	0.46	0.75
Electric equipment (chatter)				
function req'd during event	Relay chattering[c]	N/A	N/A	N/A
function req'd after event	Relay chattering[c]	2.0	0.50	0.63
Switchgear/Motor control centers	Functional/Structural[c]	1.8	0.46	0.62
Transformers	Functional/Structural	1.8	0.46	0.62
Diesel generators & support systems	Support	1.8	0.46	0.62
Turbine-driven pumps	Anchorage	2.0	0.45	0.70
Motor-driven pumps	Anchorage/Impeller defle	1.8	0.46	0.62
Small tanks	Anchorage	1.8	0.46	0.62
Motor-operated valves	Operator distortion	3.0	0.60	0.74
Safety relief, & check valves	Internal damage	3.0	0.60	0.74
Manual valves	Internal damage	3.6	0.60	0.89
HVAC ducting	Support	3.0	0.60	0.74
Air handling units/Room A.C.	Blade rubbing	2.0	0.50	0.63
Piping	Support	3.0	0.60	0.74
Diesel-driven pumps	Support	1.8	0.46	0.62

Table 19H.4-6

SEISMIC CAPACITY SUMMARY (Cont.)

Notes:

a. Capacities are in terms of median peak ground acceleration.

b. Combined uncertainties are composite logarithmic standard deviations of uncertainty and randomness components.

c 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 (1.8) was more representative than the specific relay chatter value (2.0) 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 capacity above 2.0. 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.

19H.5 COL LICENSE INFORMATION

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.4-6. It should be noted that only the capacities of important contributors (see Section 19.8) need to be determined and compared.

The HCLPF calculations can be made using fragility analysis or the conservative deterministic failure margin (CDFM) approach recommended in EPRI report NP-6041. The location effects should be taken into account in determining the limiting capacity of the same component on different locations.

For structures and components other than the generic components mentioned above, HCLPFs specified in Table 19H.4-6 can be considered achieved when the seismic design adequacy is confirmed if, according to the procedure described in Subsection 2.3.1.2 of Chapter 2, the site-dependent conditions are within the site envelope parameters or the site-specific SSE responses are bounded by those considered in the standardized design. Otherwise, site-specific HCLPF capacities for these structures and components need to be established.

It is not necessary that in each case the HCLPF exceed the value assumed in the margins analysis of the standardized design. 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) beyond the SSE level in accordance with the approach recommended in EPRI report NP-6041.

In order to increase confidence that the seismic capacities of as-designed structures and components are realized in the final constructed plant, a seismic walkdown as well as a review of construction drawings and documents shall be performed by the COL applicant. The walkdown procedure should follow the guidelines described in EPRI report NP-6041, including an assessment of potential seismic vulnerabilities, such as marginal anchorage of equipment and gross deviations from the design documents, and spatial interactions (e.g., operators being disabled due to the failure of the control room suspended ceiling in a seismic event).

19H.6 REFERENCES

1. *NUREG/CR-2300, PRA Procedures Guide,* January 1983.

2. *ALWR Requirements Document, Appendix A* PRA Key Assumptions and Groundrules, EPRI, October 1988.

3. 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.

4. Campbell, R. D., Ravindra, M. K., and Bhatia, A., *Compilation of Fragility Information from Available Probabilistic Risk Assessments, LLNL,* September 1985.

5. *Report on Quantification of Uncertainties, Report of Seismic Analysis Main Committee, ASCE,* March 15 1983.

6. *Severe Accident Risk Assessment - Limerick Generating Station,* NUS Report, April 1983.

7. NUREG/CR-3558, *Handbook of Nuclear Power Plant Seismic Fragilities, Seismic Safety Margins Research Program,* June 1985.

8. 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.

9. Kennedy, R. P., and Ravindra, M. K., *Seismic Fragilities for Nuclear Power Plant Risk Studies, Nuclear Engineering and Design,* PP4-68, (79) 1984.

10. *Structural Analysis and Design of Nuclear Plant Facilities, Manual and Reports on Engineering Practice,* No. 58, ASCE, 1980.

11. NUREG/CR-0098, *Development of Criteria for Seismic Review of Selected Nuclear Power Plants,* May 1978.

19J.4 OPERATOR ACTIONS

Three types of operator actions are critical to the SCETs. The first is the use of the firewater system to mitigate the effects of the accident. The second is the manual initiation of the RHR system after some loss of power cases. The third case is the closure of the RHR suction valves after the earthquake causes a pipe break in this system.

19J.4.1 Firewater Injection Initiation

The firewater system is used in two ways in this analysis. The first mode is to inject water into the vessel following loss of all core cooling. Except in cases where the RCIC is initially available there is a limited period of time available to begin injection into the vessel, about 20 minutes to diagnose and begin firewater injection. Based on this time a value of 0.9 was selected for the combined reliability of the operator and the system to perform this action.

Secondly, for the cases where the RCIC is initially available, about 8 hours is available for the operator to prepare to initiate the firewater system to prevent core damage. Under these conditions the operator is aware that the RCIC may not operate for more than 8 hours, so he would prepare to initiate the firewater system when required. This would include ensuring that pumping power was available by acquiring a fire truck if necessary. Therefore, a value of 0.999 was used for the operator reliability under these circumstances (See Subsection 19K.4.2.5). The availability of the firewater system to inject if called upon is based on the results from the seismic accident events analysis.

The third type of firewater use is that which follows a loss of all core cooling where the operator failed to begin the firewater addition system before vessel failure occurred. In this case the firewater system is to be used in drywell spray mode. Several hours are allowed for this action. Using the logic about for vessel injection where the RCIC was initially available, a value of 0.999 was used for the operator reliability. The firewater system availability is based on the seismic accident events analysis.

19J.4.2 Power Transformer Bypass

A mechanism for loss of core cooling following an earthquake is the loss of the divisional 480V AC power or the divisional 120V AC power. The weakest elements in these systems are identified in the seismic fault tree, Figure 19I.2-7, as the transformer, ETR6C1, with median capacity 1.5g and the inverter, E1V0F1H, with median capacity 1.3g. These elements are required to power initiation of the ECC and RHR systems. However, it is possible to manually initiate either of these systems as described below.

The first step that the operator should take is to manually open the required valves for the system to operate. In the case of the RHR system, the operator should open the injection valves of the RHR system and the appropriate service water valves. These actions would normally be provided automatically using 120V AC.

After the valves are properly aligned, the operator should manually close the breakers to provide power to the pumps. This action, normally initiated automatically using 480V AC may be performed from the remote shutdown panel if necessary. The pumps then run directly from the 6.9kV power.

The actions required of the operators to perform these actions are very similar to those to initiate the firewater addition system described above. Therefore, the reliability used for the firewater system is also used for this transformer bypass. For the same reason, no credit is taken for the firewater system if the operator fails to perform the transformer bypass.

19J.4.3 RHR Isolation to Prevent Suppression Pool Drain

If the RHR heat exchanger anchorage fails, it is possible for a pipe to break, allowing flow from the suppression pool to the pump room. If power is available the operator has sufficient means and indication to isolate the RHR system, preventing the suppression pool from being drained. The reliability of the operator to isolate the RHR is discussed in the seismic event tree analysis, Subsection 19I.4.2.6.

CHAPTER 19L
ABWR SHUTDOWN RISK

APPENDIX 19L

TABLE OF CONTENTS

APPENDIX 19L

TABLE OF CONTENTS (Continued)

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 ris 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 the ABWR Standard Safety Analysis Report (SSAR). The internal event PRA (Section 19.3 of SSAR) provided an extensive analysis of transients and accidents that initiate during power operation. The seismic PRA (Section 19.4 of SSAR) also consisted of events that initiate during power operation. In both PRAs, it was judged that the risks during cold shutdown conditions would be low with respect to those during power operations for several reasons: most of the transients that disturb power operations do not apply to the shut down plant; low system pressure reduces the already small frequency of loss of coolant events due to pipe break; and 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 (Section 19L.11, Reference 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 (Section 19L.11, Reference 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) (Section 19L.11, Reference 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 (Section 19L.11, Reference 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 (Section 19L.11, Reference 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 a conditional core damage probability of less than 1.0E-05 per year. 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, and how ABWR features could mitigate past events at operating BWRs.

SECTION 19L.4

CONTENTS

TABLES

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.4-1.

The ABWR PRA (SSAR 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.

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 External Events

Seismic events are the only external events analyzed in detail as part of the SSAR submittal. Similarly, in this shutdown risk study, only seismic events are reviewed and other external events (such as fire, flood, tornado, etc.) are excluded since they are judged to be negligible contributors to core damage risk.

The impact of loss of offsite power during shutdown is considered as part of the scope discussed in Subsection 19L.4.1.

19L.4.3 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 SSAR 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.4.4 Summary of Types of Events Considered

The types of events considered in this shutdown risk study are summarized as follows:

(1) Reactivity Excursion Events (Section 19L.5)

(2) Reactor Pressure Vessel Draining Events (Section 19L.6)

(3) Loss of Core Cooling (Section 19L.7)

(4) Loss of Decay Heat Removal Events (Section 19L.8)

(5) External Events (Section 19L.9)

(6) NonCore-Related Events (Section 19L.10)

Table 19L.4-1

ABWR MODES OF OPERATION

Mode[a]	Title	Reactor Mode Switch Position	Average Reactor Coolant Temperature, °C
1	Power Operation	Run	Any temperature
2	Startup	Startup/Hot Standby	Any temperature
3	Hot Shutdown	Shutdown	> 93.3°C
4	Cold Shutdown	Shutdown	≤ 93.3°C
5	Refueling	Shutdown or Refuel	≤ 93.3°C[b]

(a) 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.

(b) Technical specification states "any temperature", but in this mode, the temperature will be below boiling point.

SECTION 19L.5

CONTENTS

TABLES

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.5-1.

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 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 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.5-1).

(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 coincident with 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.5-1. 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 a control rod for testing;

(2) a second adjacent control rod sticks and unsticks at specific times;

(3) Class 1E separation detection of the second control rod or rod block fails;

(4) operator tries to withdraw the second control rod and the interlock fails;

(5) operator ignores alarm and continues withdrawal of the second rod; and

(6) hollow piston latch of the second rod fails.

For Failure Mode 2:

(1) operator withdraws a control rod for testing;

(2) a second adjacent control rod sticks and unsticks at specific times;

(3) positive bayonet coupling of the second control rod experiences structural failure;

(4) Class 1E separation detection of second control rod or rod block fails;

(5) operator tries to withdraw the second control rod and the interlock fails; and

(6) operator ignores alarm and continues withdrawal of the second rod.

For Failure Mode 3:

(1) operator withdraws a control rod for testing;

(2) operator installs a second adjacent control rod drive without coupling, and fails to detect the error during procedural coupling checks;

(3) the second control rod sticks and unsticks at specific times;

(4) Class 1E separation detection of the second rod or rod block fails; and

(5) operator ignores alarm and continues withdrawal of the second 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 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.5-2. 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 at the maximum fuel grapple speed 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.5-3 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) With mode switch in the REFUEL position, only one rod can be withdrawn. Any attempt to withdraw a second rod results in a rod block initiated by the refueling interlock.

(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. This is done by physically preventing the automatic refueling machine from moving over the core with one control blade withdrawn (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 refueling interlock 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) (Section 19L.7, Reference 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 to be <<1.0E-8 per reactor year and therefore negligible. The probability of this accident is also expected to be negligible for the ABWR plants.

19L.5.4 Rod Withdrawal Error

During refueling, 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) With mode switch in the REFUEL position, only one rod can be withdrawn. Any attempt to withdraw a second rod results in a rod block by the refueling interlock.

(2) If the interlock 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 the refueling operations.

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 the second control rod without inserting the first control (i.e. operator does not follow procedures).

(3) The second control rod is adjacent to the first control rod 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 control rods.

(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.

Table 19L.5-1

CONTROL ROD DROP ACCIDENT

Cause/Event: Hardware	Cause/Event: Operator	Preventive and Mitigative Features
Failure Mode 1		
1. -	One control rod withdrawn for test.	-
2. A second adjacent rod sticks (still coupled to hollow piston)	-	-
3. -	The second control rod is withdrawn	Interlock prevents withdrawal of the second control rod
4. Separation of ballnut and hollow piston in the second control rod	Operator misses alarm & continues withdrawal of the second control rod	Class 1E separation detection
5. The second control rod unsticks and drops	-	Rod block + hollow piston latch
Failure Mode 2		
1. -	One control rod withdrawn for test.	-
2. A second adjacent control rod sticks	-	-
3. -	The second control rod is withdrawn	Interlock prevents withdrawal of second control rod
4. Rod to hollow piston separation occurs in the second control rod	Operator misses alarm & continues withdrawal of the second control rod	1. Positive bayonet coupling 2. Class 1E separation detection
5. The second control rod unsticks and drops	-	Rod block

Table 5-1 (Continued)

CONTROL ROD DROP ACCIDENT

Cause/Event: Hardware	Cause/Event: Operator	Preventive and Mitigative Features
Failure Mode 3		
1. -	One control rod withdrawn for test.	-
2. -	A second adjacent control rod is installed without coupling	-
3. -	Error in the second control rod not detected during coupling check	
4. -	The second control rod withdrawn	Interlock prevents withdrawal of second control rod
5. The second rod sticks	-	-
6. Rod to hollow piston separation in the second control rod	Operator misses alarm & continues withdrawal of the second control rod	Class 1E separation detection
7. The second control rod unsticks and drops	-	Rod block

Table 5-2

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. -	One control rod with-drawn 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. -	One control rod with-drawn 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 5-3

REFUELING ERROR

Cause: Hardware Failure	Operator Error/Action	Preventive and Mitigative Features
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. -	First 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

SECTION 19L.6

CONTENTS

SECTION 19L.6

CONTENTS

TABLES

ILLUSTRATIONS

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 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. Next 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. 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. In addition, at least two emergency core cooling systems are required to be available during this mode of operation, which will mitigate the consequences of draining.

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.6-1.

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 6mm until the impeller backseats to prevent leakage of reactor water when the motor cover is removed. A secondary seal is then provided with the help of an inflatable seal. 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 inflatable seal and 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 nozzle 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 100 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 cover plate is bolted. In the most likely failure scenario, it is possible that the sealing between the impeller and its 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 (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), the inflatable seal must fail and finally the bottom plate must also fail. Because of multiple failures required, it is judged that the probability of

draining the vessel during RIP maintenance is negligibly low.

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.6-2, and shown in Figure 19L.6-1. 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 32mm diameter, 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 CRD's 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, one or 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.6-3 and shown in Figure 19L.6-2. 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 50mm diameter pipe. 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 (50mm diameter) size of the drain line, 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 50mm diameter pipe, will allow sufficient time to correct the situation. Should no corrective action be taken manually, on reaching reactor water level 3, 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 suppression pool. 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 transmitter 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 3 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 transmitter 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 3 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 50mm diameter pipe, the slow drainage rate will allow sufficient time to correct the situation before level 3 is reached at which point the path will terminate on reactor level 3 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 suppression pool. 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 transmitter 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 3. 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 3, 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 1.7 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.6-4, and depicted in Figure 19L.6-3. 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 1.7 meters of water over the top of the active fuel.

19L.6.5.1 Path 1

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 water

level in the control room and correctly responds to control room indicators and alarms; and the drain path is only a 50mm 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 150mm 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 300mm 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 250mm 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 100mm 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, 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 450mm 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.

Table 19L.6-1

POTENTIAL FOR DRAINING RPV DURING RIP MAINTENANCE

Cause		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 bottom plate is bolted
		4) Impeller removal results in the loss of the backseat seal, but cap is inserted in the impeller cavity to provide additional protection

Table 19L.6-2

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 32mm 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

Table 19L.6-2 (Cont'd)

POTENTIAL FOR DRAINING RPV THROUGH CONTROL ROD DRIVE HYDRAULIC SYSTEM AT SHUTDOWN

PATH	EQUIPMENT FAILURE	OPERATOR ERROR	RESULTS	PREVENTIVE/MITIGATIVE MEASURES
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
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.6-3

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 50mm 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 and 3 A	Valve F055A fails open or Valve F055B fails open		RPV water drainage into reactor building	Drain Line is only 50mm 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 and 3 B		Valve F055A open or Valve F055B open	See 2A	See 2A

Table 19L.6-3 (Cont'd)

POTENTIAL FOR DRAINING RPV THROUGH REACTOR WATER CLEANUP SYSTEM

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.6-3 (Cont'd)

POTENTIAL FOR DRAINING RPV THROUGH REACTOR WATER CLEANUP SYSTEM

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.6-4

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 50mm 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 150mm 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

Table 19L.6-4 (Cont'd)

POTENTIAL FOR DRAINING RPV THROUGH RESIDUAL HEAT REMOVAL SYSTEM

PATH	EQUIPMENT FAILURE	OPERATOR ERROR	RESULTS	PREVENTIVE/MITIGATIVE MEASURES
2B		[A] + Valve F029 inadvertently open + Valve F030 inadvertently open	See 2A	See 2A
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

Table 19L.6-4 (Cont'd)

POTENTIAL FOR DRAINING RPV THROUGH RESIDUAL HEAT REMOVAL SYSTEM

PATH	EQUIPMENT FAILURE	OPERATOR ERROR	RESULTS	PREVENTIVE/MITIGATIVE MEASURES
4A	[A] + Valve F014 fails open + Valve F015 fails open		Drain RPV water to fuel fool via 300mm 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.6-4 (Cont'd)

POTENTIAL FOR DRAINING RPV THROUGH RESIDUAL HEAT REMOVAL SYSTEM

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 250mm 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

Table 19L.6-4 (Cont'd)

POTENTIAL FOR DRAINING RPV THROUGH RESIDUAL HEAT REMOVAL SYSTEM

PATH	EQUIPMENT FAILURE	OPERATOR ERROR	RESULTS	PREVENTIVE/MITIGATIVE MEASURES
6A	[A] + Valve F019 fails open		Drain RPV water to wetwell spray via 100mm diameter pipe	1. Requires valve failure/opening 2. Indicators in control room will F017 open 3. F010 & F011 will isolate on reactor level 3 4. Operator monitors level in control room and takes corrective actions
6B		[A] + Valve F019 inadvertently opened	See 6A	See 6A
7A	[A] + Valve F016 fails open		Drain RPV water to fuel pool via 50mm 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

Table 19L.6-4 (Cont'd)

POTENTIAL FOR DRAINING RPV THROUGH RESIDUAL HEAT REMOVAL SYSTEM

PATH	EQUIPMENT FAILURE	OPERATOR ERROR	RESULTS	PREVENTIVE/MITIGATIVE MEASURES
8A	[A] + Valve F011 fails open		Drain RPV water to suppression pool via 450mm 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
9A	Minimum flow valve F021 opens during low flow during shutdown pool 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

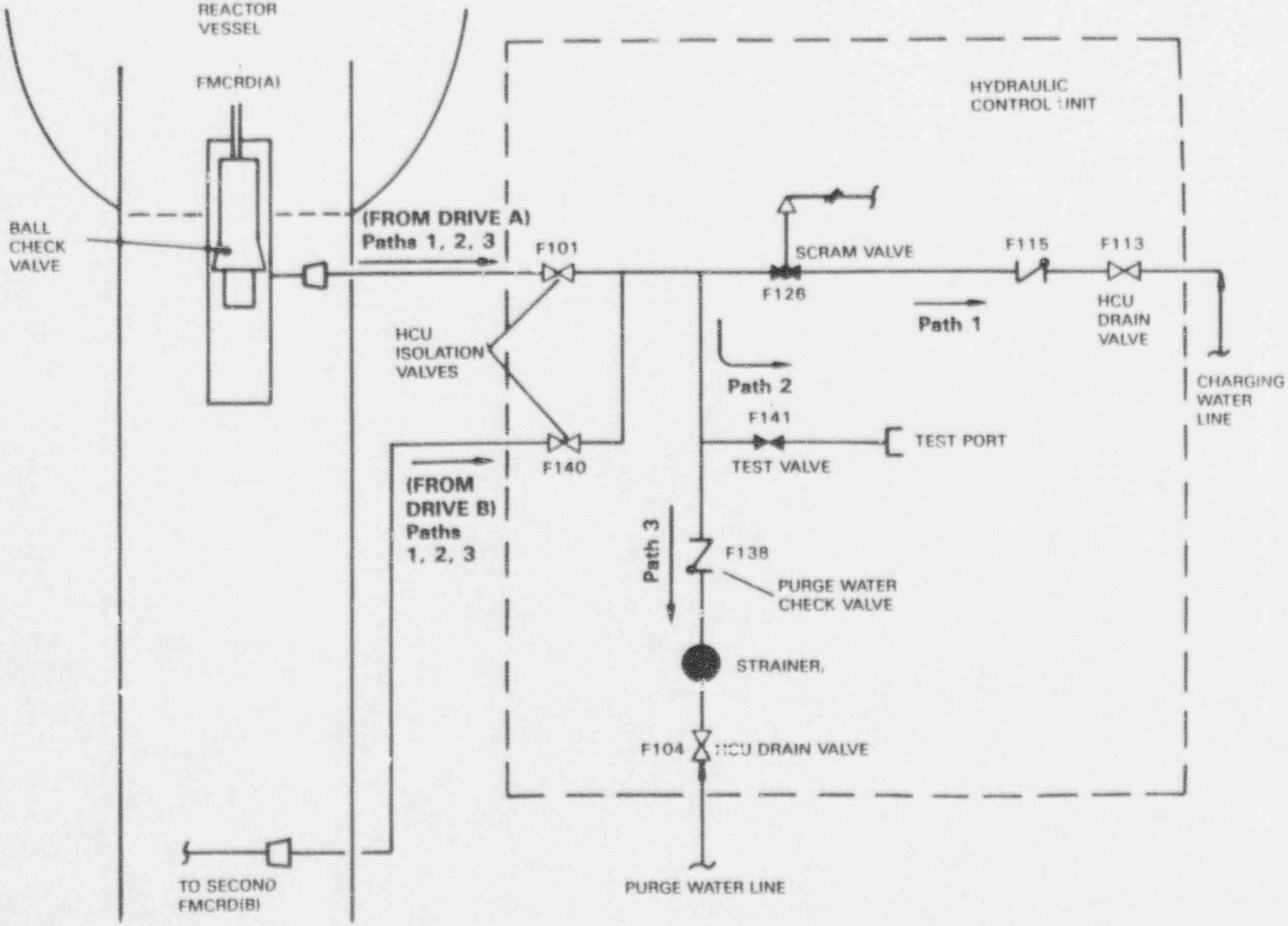

Figure 19L.6-1 POTENTIAL PATHS FOR DRAINING RPV THROUGH CONTROL ROD DRIVE HYDRAULIC SYSTEM

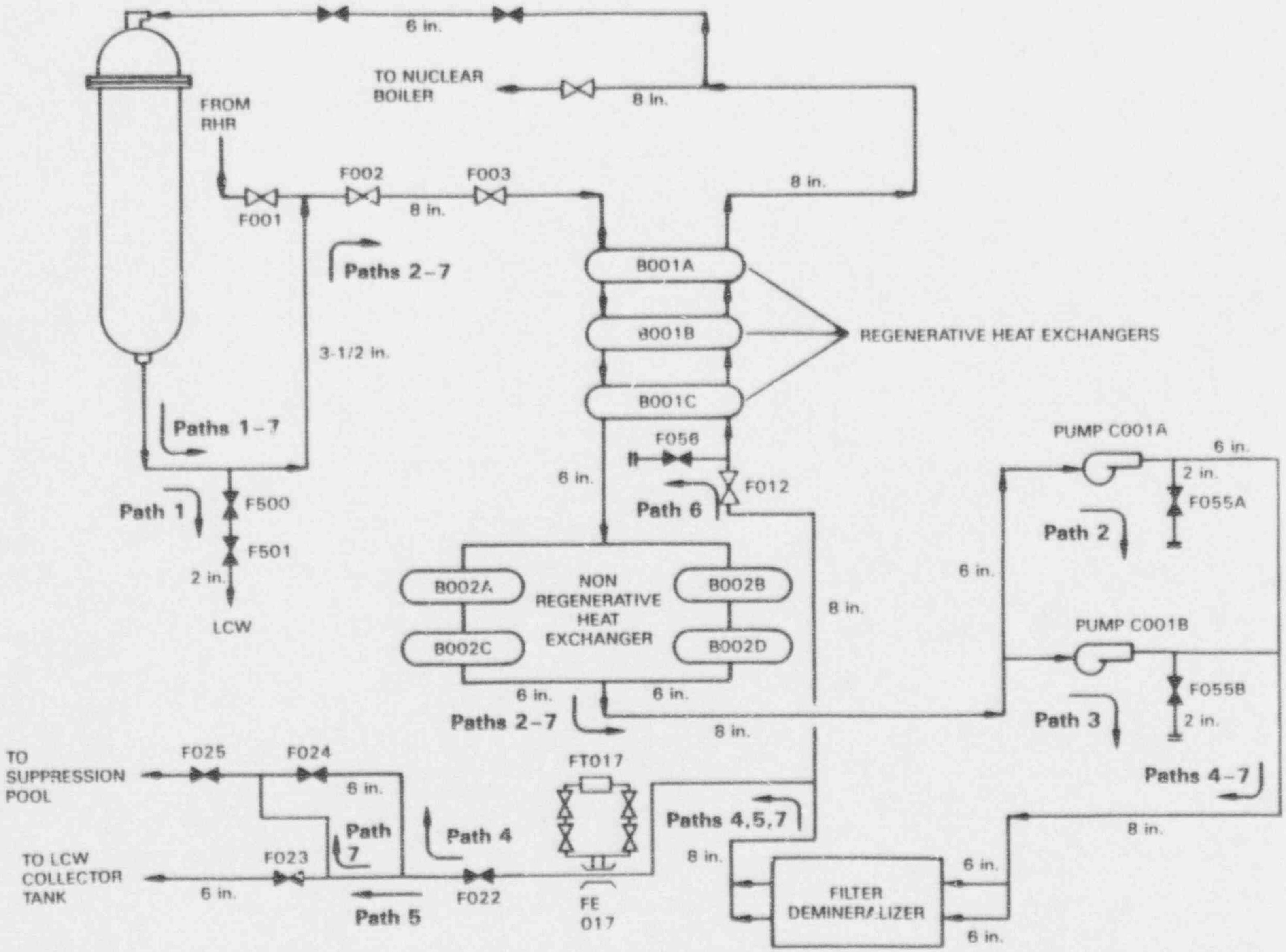

Figure 19L.6-2 POTENTIAL PATH FOR DRAINING RPV THROUGH REACTOR WATER CLEANUP SYSTEM

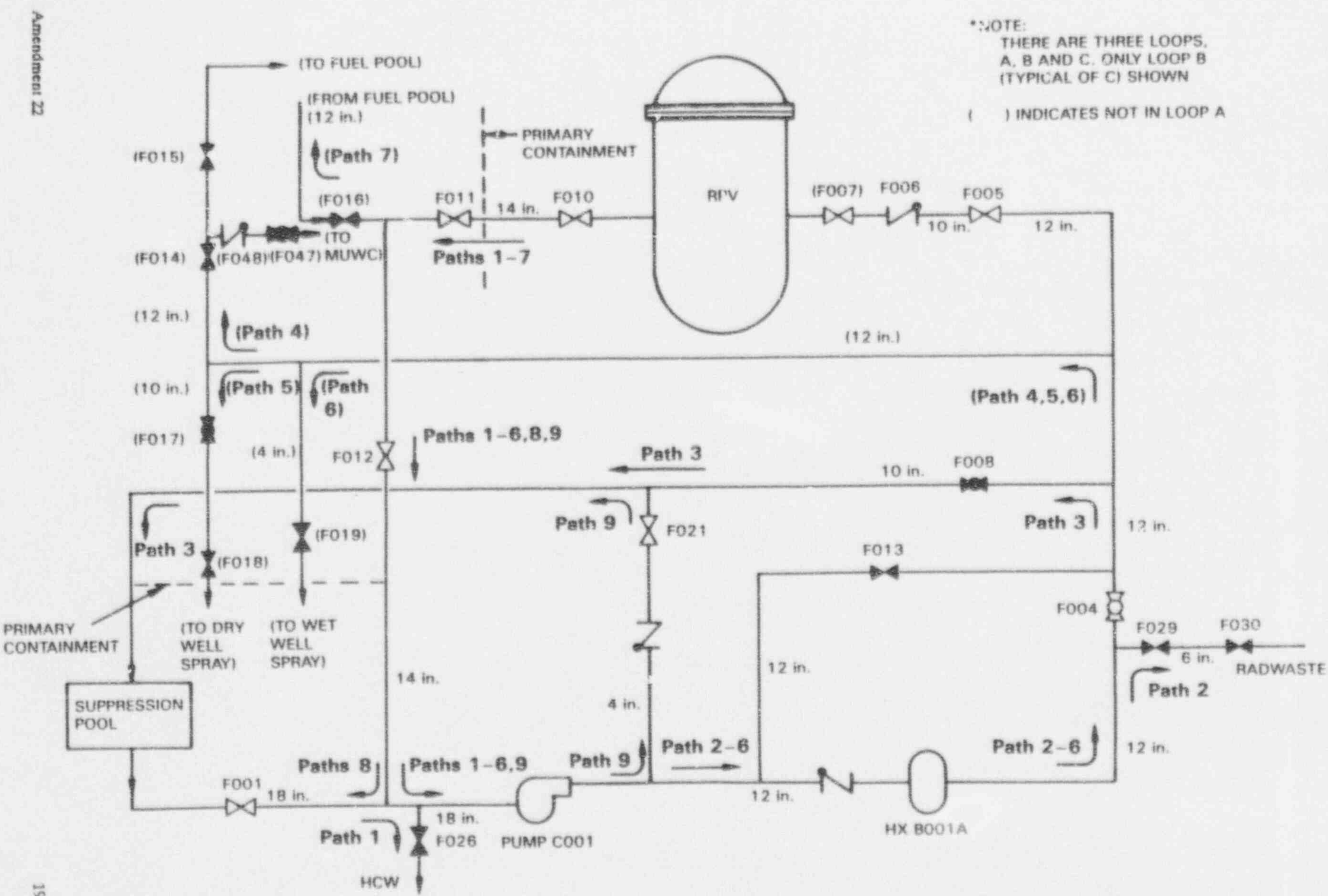

Figure 19L.6-3 POTENTIAL PATH FOR DRAINING RPV THROUGH RESIDUAL HEAT REMOVAL SYSTEM (PUMP ON)

SECTION 19L.7

CONTENTS

TABLES

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 Section 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.

Appendix 19Q 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.7-1 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 one diesel generator be available. In addition, the gas turbine 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 off-site power during the first one week period of operating in mode 4 with no recovery, and

b) Failure of one diesel generator, and

c) Combustion turbine 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.

Given the loss of offsite power frequency of 0.1 per year, frequency of (a) is 2.0E-3 (without any credit for recovery of offsite power). Failure probability of diesel generator (b) is 2.5E-02/demand. Combustion turbine failure probability (c) is 5.0E-2/demand. And (d), the operator failure to initiate the AC independent water addition system, is estimated to be 1.0E-3/demand. Even without estimating (e), it can be seen that the combined failure probability of all these systems is (2.5E-09 per year) negligible.

It is recognized that there are other cut-sets 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.

Table 19L.7-1

DEPENDENCY OF CORE COOLING SYSTEMS ON ELECTRICAL POWER

	POWER SYSTEMS								
SYSTEM	OFFSITE POWER	COMBUSTION TURBINE	DG1	DG2	DG3	DIV 1 DC	DIV 2 DC	DIV 3 DC	FIRE DIESEL GENERATOR
RCIC						XX			
HPCF (B)	OR	OR		OR			XX		
HPCF (C)	OR	OR			OR			XX	
FW (A)	OR	OR	OR*	OR*	OR*				
FW (B)	OR								
CRD (A)	OR	OR	OR*	OR*	OR*				
CRD (B)	OR	OR							
LPFL (A)	OR*	OR	OR			XX			
LPFL (B)	OR*	OR		OR			XX		
LPFL (C)	OR*	OR			OR			XX	
FIREWATER**	OR*	OR			OR*			XX	OR
CONDENSATE									
(A)	OR		OR*	OR*	OR*				
(B)	OR								
(C)	XX								
(D)	XX								

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
* - Assumes feedback capability for combustion turbine distribution system
** - AC independent water addition system

SECTION 19L.8

CONTENTS

TABLES

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 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.

Appendix 19Q 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 SSAR 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 SSAR 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.8-1. It should be noted that even though the reactor is at low pressure, main condenser and RWCU 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 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.8-1.

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.8-2 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 one diesel generator be available. In addition, the gas turbine 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 generator and the gas turbine fail to start, then the only heat removal system available is the overpressure relief rupture disk. The combined probability of event sequence is calculated as follows:

Loss of offsite power during one week	= 2.0E-3/year (See Subsection 19L.7.3)
Offsite power not recovered in 24 hours	= 2.3E-4/demand (19L.11, Ref 6)
Failure of one diesel generator	$\simeq$ 2.5E-02/demand
Combustion turbine failure to start	$\simeq$ 5.0E-02/demand
Failure of overpressure rupture disk	$\simeq$ 1.0E-3/demand

The combined failure probability of this sequence of events is then calculated to be negligible ($\simeq$ 6.0E-13 per year). 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.

Table 19L.8-1

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(1) or Normal heat removal using main condenser(2) or Reactor water cleanup(3) or Overpressure relief rupture disc(4)

Notes:

1) 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.

2) Reactor will have to be pressurized and MSIVs opened for establishing this path.

3) Reactor may have to be pressurized to use the RWCU system efficiently to remove decay heat or reactor water could be drained to the main condenser hotwell through the RWCU system and reactor water makeup obtained from HPCF, feedwater, CRD hyrdraulic system, or the AC independent water addition system.

4) 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 3 above.

Table 19L.8-2

DEPENDENCY OF HEAT REMOVAL SYSTEMS ON ELECTRICAL POWER

SYSTEM	OFFSITE POWER	GAS TURBINE	DG1	DG2	DG3 DC	DIV 1 DC	DIV 2 DC	DIV 3
RHR (A)	OR*	OR	OR			XX		
RHR (B)	OR*	OR		OR			XX	
RHR (C)	OR*	OR			OR			XX
RWCU A	OR*		OR					
RWCU B	OR*			OR				
OVERPRESSURE RELIEF **								
MAIN CONDENSER	XX							

Notes:

DG1: Diesel generators
OR: Redundant supply to other ORs
XX: Loss of this power supply means loss of system
*: 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.

APPENDIX 19K

PRA BASED RELIABILITY AND MAINTENANCE

APPENDIX 19K

TABLE OF CONTENTS

APPENDIX 19K

TABLE OF CONTENTS (Continued)

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 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) Section 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 reviev 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 off-site release. Also considered were the initiating events that had significant impact on CDF. From this review (Section 19K.3), the most important plant features were identified.

The PRA was further reviewed (Sections 19K.4 through 19K.10) for other important features, the failure of which was not addressed directly in Section 19K.3, to supplement the above list. Finally (Section 19K.11), the individual features identified in Sections 19K.3 through 19K.10 were reviewed to determine appropriate maintenance and surveillance actions.

SECTION 19K.3

TABLES

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 19 SSCs of greatest importance, in that they had FV importance greater than 1%, contributed more than 60% of the sum of the importances and they are shown in Table 19K.3-1. Also shown for each SSC is its risk achievement worth, and five additional SSCs with risk achievement worth greater than 20 were considered. Not shown in Table 19K.3-1 are several human error contributions. Significant human errors are addressed in Subsection 19D.7.

The 24 SSCs in Table 19K.3-1 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 Section.

The remaining 20 designated SSCs of Table 19K.3-1 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 Section 19K.11.

A second table, 19.K.3-2, was prepared to show those SSCs with risk achievement worth between 5 and 20. 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 risk achievement worth of 12 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.3-3. There are five such events which are shown with their frequency and their total and relative contribution to CDF. The three most significant events, accounting for more than 70% of the CDF, are all station blackout events. The next two events, contributing 11% and 7% of CDF, respectively, are isolation/loss of feedwater and manual reactor shutdown. All other initiating events contribute less than 5% of CDF each.

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.11-1.

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 Sections 19K.4 through 19K.10, and RAP activities are in Section 19K.11.

Table 19K.3-1

ABWR SSCs of Greatest Importance for CDF, Level 1 Analysis

SSC	Fussell-Vesely Importance %	Risk Achievement Worth
RCIC System (Unavailable, Test or Maintenance)	21.8	12.
Multiplex Transmission Network (CCF)	12.1	204,400.
RCIC Turbine	12.0	12.
RCIC Pump	7.4	12.
Trip Logic Units	6.0	204,300.
Remote Multiplexing Units	6.0	204,300.
RCIC Turbine Lubrication System	4.6	12.
Station Batteries (CCF)	3.3	13,160.
Single Offsite Power Line (1)	3.1	4.1
RCIC Min Flow Bypass Valve E51-F011 (NOFO)	2.0	12.
RCIC Min Flow Bypass Valve E51-F011 (NCFC)	1.9	12.
RCIC Injection Valve E51-F004 (NCFC)	1.9	12.
RCIC Steam Supply Valve E51-F037 (NCFC)	1.9	12.
HPCF Maintenance Valve E22-F005B (2)	1.7	2.7
Combustion Turbine Generator (1)	1.7	1.3
RCIC Isolation Signal Logic	1.5	12.
Both Offsite Power Sources	1.3	14.
HPCF Pump (1)	1.1	2.6
SRVs (1)	1.0	4.3
RHR Flow Transmitters (CCF Miscalibration)	0.2	32.
SRV (CCF)	< 0.1	189.
Level 2 Sensors (CCF)	< 0.1	273.
Level 8 Sensors (CCF Miscalibration)	< 0.1	28.
Digital Trip Modules (CCF)	<< 0.1	281.

(1) SSCs with low FV importance and low risk achievement worth. Not considered further for RAP.

(2) SSC with low FV importance and low risk achievement worth, but retained because of human factor importance.

Table 19K.3-2

ABWR SSCs With Risk Achievement Worth Between 5 & 20 For CDF, Level 1 Analysis

SSC	Fussell-Vesely Importance %	Risk Achievement Worth
RCIC Turbine Exhaust Isolation Valve F039 Limit Switch Fails	0.74	12.
RCIC Steam Supply Bypass Valve F045 Limit Switch Fails	0.74	12.
Div 1 Transmission Ntwk Failure (EMS)	0.70	13.
All 3 Diesel Generators, CCF	0.56	11.
1st ESF RMU Div 1 Fails	0.34	13.
2nd ESF RMU Div 1 Fails	0.34	13.
RCIC Flow Sensor E51-FT007-2 Fails	0.32	12.
RCIC Isolation Valve F036 Fails (NOFC)	0.18	12.
RCIC Isolation Valve F035 Fails (NOFC)	0.18	12.
RCIC Isolation Valve F039 Fails (NOFC)	0.18	12.
RCIC Check Valve E51-F003 Fails to Open	0.15	12.
RCIC Check Valve F038 Fails to Open	0.15	12.
RCIC Outboard Check Valve F005 Fails to Open	0.15	12.
NBS Isolation Check Valve B21-F003B (FW Isolation) Fails Closed	0.15	12.
NBS Isolation Check Valve B21-F004B (FW Isolation) Fails Closed	0.15	12.
NBS Manual Valve B21-F005B (FW Isolation) Fails Closed (NOFC)	0.14	12.
RCIC Pres Sensor PIS-Z605 Miscalibrated	0.054	12.
RCIC Flow Sensor FT-007-2 Miscalibrated	0.054	12.
RCIC Pressure Sensor E51-PIS-Z605 Fails	0.013	12.
Failure of Division 1 Distribution Panel	0.0064	12.
SP Temp High (Loss of Pump Head)	0.00055	6.6
SLU/EMS Link for Div 1 SLU 1 Fails (RCIC Fails)	0.00046	12.
SLU/EMS Link for Div 1 SLU 2 Fails (RCIC Fails)	0.00046	12.

Notes: EMS = Essential Multiplexing System
ESF = Engineered Safety Feature
RMU = Remote Multiplex Unit
SLU = Safety System Logic Unit

Table 19K.3-3

ABWR Initiating Event Contribution to CDF, Level 1 Analysis

Initiating Event	Events Per Year	Total CDF X 10[8]	Percent CDF Contribution
Station Blackout for Less Than Two Hours	1.22E-6	6.67	42.7
Station Blackout for Two to Eight Hours	4.46E-7	2.57	16.5
Station Blackout for More Than Eight Hours	1.62E-8	1.71	11.0
Isolation/Loss of Feedwater	0.18	1.70	10.9
Unplanned Manual Reactor Shutdown	1.00	1.15	7.4

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 and the diesel-driven pump, 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 drywell floor.

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 failure in the wetwell, as opposed to the drywell. 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 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 valves of the RHR system that control this spray function are included in RAP. The wetwell spray function of the RHR is used for control of bypass leakage by keeping containment pressure low. It does not play as important a role in the analyses performed as does the drywell spray, so its components will not be a part of the RAP.

The RAP activities for important SSCs identified by this Level 2 analysis are given in Table 19K.11-1.

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 following features were identified as having high confidence, low probability of failure (HCLPF) capacities less than 0.60 (twice SSE).

- The diesel generators, 480Vac transformers, and motor control centers of the ac power system
- The batteries, battery racks and inverters of the dc power system
- The motor driven pumps, heat exchangers, and room air conditioning units of the service water system
- The motor driven pumps of the high pressure core flooder system
- The motor driven pumps of the residual heat removal system
- The SLC tank and the motor driven pumps of the standby liquid control system
- The motor driven pumps of the fire water system

The RAP activities for important SSCs identified by this seismic analysis are given in Table 19K.11-1.

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.11-1.

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 and turbine building/service building 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, which limit the amount of water spilled into the control building, and circuit breakers that will trip RSW pumps, which also limits the amount of water spilled into the control building.

The RAP activities for important SSCs identified by this flood analysis are given in Table 19K.11-1.

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, 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 LPFL, HPCF and CRD systems.

The RAP activities for important SSCs identified by this shutdown analysis are given in Table 19K.11-1.

19K.9 Reliability and Maintenance Actions

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.11-1.

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 from outside the control room; (3) closing water tight doors that are open (if there is flooding in the intact RSW 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 connection of the ac-independent water addition system; (6) manual bypass of the regenerative heat exchanger in the reactor water cleanup system; and (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.11-1.

SECTION 19K.11

CONTENTS

TABLES

19K.11 Reliability and Maintenance Actions

The individual SSCs identified as being "important" in Sections 19K.3 through 19K.10 were reviewed to determine the appropriate reliability and maintenance actions. These actions are defined in this section.

The important SSCs are tabulated in Table 19K.11-1, 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 unavail- ability assumption is met.

19K.11.1 Component Inspections and Maintenance

The system of greatest FV importance is the RCIC system, which has been assigned 2% 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 2% assumption annually.

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.11-1.

The turbine of the RCIC system is an important component, as identified in Table 19K.3-1. 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.11-1.

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 on-line, self-test feature of ABWR solid-state logic. Additional off-line, 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 on-line system. The TLU tests that are suggested as part of the RAP are given in Table 19K.11-1.

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.11-1 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.11-1.

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.11-1.

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. Suggested RAP activities are given in Table 19K.11-1.

Common cause failures (CCFs) of RHR flow meter calibration, Level 8 sensor calibration, Level 2 sensor calibration, and of digital trip modules (DTMs) 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. The procedure for testing DTMs should include a warning about their importance to safety. Suggested RAP activities are given in Table 19K.11-1.

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.11-1 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.11-1.

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 as reliable as assumed for the analysis 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.11-1 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.11-1 with the other RAP activities.

19K.11.4 Lower Drywell Flooder (LDF)

Activities suggested for RAP are given in Table 19K.11-1 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, Section 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.

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.11-1.

1. On-site 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 NFPA1901, "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 90%. 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 percent rated capacity, preferably at draft, at 150 psi (1039 kPa) net pump pressure;

   (b) Ten minutes of pumping 70 percent rated capacity at 200 psi (1379 kPa) net pump pressure; and

   (c) Ten minutes of pumping 50 percent rated capacity at 250 psi (1724 kPa) 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 165 psi (1138 kPa) 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 90% reliability for supporting the water injection function.

3. Once every two years the RHR nonsafety 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 in-service testing program.

4. Once every four years the ac independent water addition (ACIWA) 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 Section 19K.4 as important to limiting fission product release. Suggested system component testing as part of RAP is identified in Table 19K.11-1. 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 other 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 indication 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.11-1.

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 2% of the rated value. Every five years, the disks should be tested and replaced. These tests are included in Table 19K.11-1.

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, 480Vac transformers and motor control centers of the ac power system; batteries, battery racks and inverters of the dc power system; motor driven pumps, heat exchangers and room air conditioning units of the service water system; the SLC tank and motor driven pumps of the standby liquid control system; and motor driven pumps of the HPCF system, the RHR system and the fire water system. For this equipment, the following seismic related inspections should be conducted once every 10 years or after any earthquake equal to or greater than the operating basis earthquake (OBE):

1. Repeat the seismic walkdown which was conducted after construction in the general area of the equipment. (The examination for seismic vulnerabilities described in EPRI NP-6041, not including a repeat of the calculational portion of the walkdown, is one acceptable method of accomplishing the desired inspection. This examination will include such issues as component functionality or integrity, component anchorage, and secondary component interaction.)

2. Visually inspect all related supporting devices and supporting structures.

19K.11.8 Plant Structures

No maintenance activities other than those already associated with the in-service 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 the OBE, 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 1.108, "Periodic Testing of Diesel Generators", and with the surveillance requirements described in the Technical Specifications (Section 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 monthly operation at rated speed and rated load until temperatures reach steady state values, approximately one hour. Also monthly 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 and fuel, oil and air filters should be replaced and there should be a thorough inspection of the entire assembly, including assurance 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.11-1.

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.11-1.

19K.11.13 Flood Protection

The important SSCs for flood protection are the water tight doors in the turbine building and in ECCS rooms, the RSW isolation valves and the circuit breakers that trip RSW pumps (Subsection 19K.7). Periodically room water barriers should be inspected to assure that they will prevent the spread of flooding, RSW isolation valves (MOVs) should be stroke tested (normally accomplished by switching from one pump to the standby pump in a given loop), and the ability of RSW pump circuit breakers to trip upon receipt of a trip signal should be demonstrated. These RAP activities are included in Table 19K.11-1.

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 and CRD systems. RAP activities for SRVs are covered in Subsection 19K.11.3. Testing of valves and pumps of the RHR system and for the HPCF and 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.3-2).

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.11-1, 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, which 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 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.

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 Section 19K.11.1, as are periodic checks on station batteries. Testing of emergency diesel generators and the combustion turbine generator are covered in Sections 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.

TABLE 19K.11-1

FAILURE MODES & RAP ACTIVITIES

COMPONENT	FAILURE MODE/CAUSE	RECOMMENDED MAINTENANCE	TEST OR MAINTENANCE INTERVAL	BASIS	UNAVAIL-ABILITY, FAILURE RATE
RCIC System	System failure	See following items	See below	See below	6.3E-2
RCIC System	Unavailable due to test or maintenance	Monitor unavailable time, compare with assumed 2%	Annually	Level 1 analysis	Note (1)
Multiplexers	Common cause failure of all MUX to give proper signals	System functional test	3 months	Experience	5.9E-7
		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	3.0E-4
		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*	Note (1)
		Turbine inspection, refurbishment	5 years	Experience*	3.0E-7
RPS Trip Logic Units	Failure to trip upon demand	System functional test	3 months	Experience	
		Complete system test, error check	R/M outage	Experience	Note (1)
RCIC Turbine Lube System	Lube oil pump failure	Lube oil pump operation and oil pressure check	3 months	Experience	Note (1)

* These types of valves and turbines have been used in operating BWRs, so there is much experi[illegible] to guide owners/operators in care of the equipment.

Note (1):RCIC component failure rates are included within the system unavailability.

TABLE 19K.11-1

FAILURE MODES & RAP ACTIVITIES (Continued)

COMPONENT	FAILURE MODE/CAUSE	RECOMMENDED MAINTENANCE	TEST OR MAINTENANCE INTERVAL	BASIS	UNAVAIL-ABILITY, FAILURE RATE
RCIC Check Valve F038	Failure to open	Open and close during system test	3 months	Experience*	Note (1)
RCIC Check Valves F003 & F004	Failure to open	Open and close test	R/M outage	Experience*	Note (1)
RCIC Isolation Signal Logic	Failure to provide isolation signal when conditions warrant	Logic functional test	3 months	Experience	Note (1)
RCIC Min Flow Bypass Valve (NOFO or NCFC)	Failure to operate because of mechanical problems	Stroke test	3 months	Experience*; ASME Code ISI	Note (1)
		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*	Note (1)
RCIC Injection Valve and Turbine Steam Supply Valve	Failure to open because of mechanical problems	Stroke test	3 months	Experience*; ASME Code ISI	Note (1)
		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*	

* 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.

Note (1): RCIC component failure rates are included within the system unavailability.

TABLE 19K.11-1

FAILURE MODES & RAP ACTIVITIES (Continued)

COMPONENT	FAILURE MODE/CAUSE	RECOMMENDED MAINTENANCE	TEST OR MAINTENANCE INTERVAL	BASIS	UNAVAILABILITY, FAILURE RATE
RCIC Isolation Valves (NOFC)	Spurious failure because of mechanical problems	Stroke test	3 months	Experience*; ASME Code ISI	Note (1)
		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*	Note (1)
RCIC Flow Sensor FT-007-2	Sensor fails	Calibration of sensor	R/M outage	Experience	Note (1)
	Miscalibration	Review calibration procedures for note about potential safety considerations	R/M outage	Judgment	Note (1)
RCIC Pressure Sensor PIS-Z605	Sensor fails	Calibration of sensor	R/M outage	Experience	Note (1)
	Miscalibration	Review calibration procedures for note about potential safety considerations	R/M outage	Judgment	Note (1)
NBS Isolation Check Valves 003B & 004B	Fails to open	Leak rate test and subsequent operation of valves	R/M outage	Experience	1.4E-4

Note (1): RCIC component failure rates are included within the system unavailability.

TABLE 19K.11-1

FAILURE MODES & RAP ACTIVITIES (Continued)

COMPONENT	FAILURE MODE/CAUSE	RECOMMENDED MAINTENANCE	TEST OR MAINTENANCE INTERVAL	BASIS	UNAVAIL-ABILITY, FAILURE RATE
NBS Manual valve F?05B (NO/FC)	Normally open valve fails closed	Stroke test	R/M outage	Experience	1.3E-4
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	1.0E-2
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.	3 years	Experience	N/A
RHR Flow Meters	Common mode miscalibration	Review calibration procedures for note about potential safety considerations	Annual	Judgment	5.0E-5
Level 2 Sensors	Common mode miscalibration	Review calibration procedures for note about potential safety considerations	Annual	Judgment	2.4E-6
Level 8 Sensors	Common mode miscalibration	Review calibration procedures for note about potential safety considerations	Annual	Judgment	2.0E-5
Digital Trip Modules	Common cause failure to trip	Review trip unit test procedure to assure note about potential safety considerations	Annual	Judgment	3.0E-7
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	5.0E-6

Note (2): ACIWA component failure rates are included within the system unavailability.

TABLE 19K.11-1

FAILURE MODES & RAP ACTIVITIES (Continued)

COMPONENT	FAILURE MODE/CAUSE	RECOMMENDED MAINTENANCE	TEST OR MAINTENANCE INTERVAL	BASIS	UNAVAILABILITY, FAILURE RATE
		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	Leakage	Inspect for leakage	R/M outage	Judgment	1E-3
	Failure to open at temperature	Two of ten plugs replaced; tested to verify temperature setpoint	2 R/M outages	Judgment	
ACIWA System	System unavailable	See following items	See below	See below	1E-2
ACIWA Flow Instrumentation	Failure to accurately monitor flow	Measure zero flow and full system flow	4 years	Judgment	Note (2)
ACIWA Manual Valves	Failed closed following maintenance	Review procedure to assure note regarding safety considerations	2 years	Experience	Note (2)
Firewater System Pumps on Fire Truck	Failure of pumps to provide required flow at pressure	20 min pump at 100% rated flow, 150 psi	1 year	Judgment	Note (2)
		10 min pump at 70% rated flow, 200 psi	1 year	Judgment	
		10 min pump at 50% rated flow, 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	Note (2)
		Test system flow with ac-driven and diesel driven pumps, water from tanks & from UHS	4 years	Judgment	

Note (2): ACIWA component failure rates are included within the system unavailability.

Note (3): COPS component failure rates are included within the system unavailability.

TABLE 19K.11-1

FAILURE MODES & RAP ACTIVITIES (Continued)

COMPONENT	FAILURE MODE/CAUSE	RECOMMENDED MAINTENANCE	TEST OR MAINTENANCE INTERVAL	BASIS	UNAVAILABILITY, FAILURE RATE
ACIWA Diesel Pump	Failure to pump on demand	Pump start test	3 months	Experience	Note (2)
		Pump flow test	4 years	Experience	
RHR Non-Safety Related Valve	Failure to open on demand	Manually open and close valve	2 years	Experience	Note (2)
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	Note (2)
		Piping support visual inspection to assure structural adequacy	5 years	Judgment	
COPS System	System unavailable	See following items	See below	See below	1E-2
COPS AOVs	Inadvertently left open following maintenance	Stroke test; position indication check, verification of local and control room indication following test	R/M outage	Experience	Note (3)
COPS Rupture Disks	Failure to open on demand	Disk replacement	5 years	ASME Code	Note (3)
		Verification of actuation within 2% of rated pressure	5 years	ASME Code	
COPS Flow Lines	Flow blockage	Flow test to assure no blockage in line	5 years	Judgment	Note (3)
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

Note (3): COPS component failure rates are included within the system unavailability.

TABLE 19K.11-1

FAILURE MODES & RAP ACTIVITIES (Continued)

COMPONENT	FAILURE MODE/CAUSE	RECOMMENDED MAINTENANCE	TEST OR MAINTENANCE INTERVAL	BASIS	UNAVAILABILITY, FAILURE RATE
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
Turbine Building & ECCS Room Water Barriers	Failure to retain integrity	Inspection of water barriers, including penetrations	1 year & after major maintenance	Judgment	N/A
RSW Isolation Valves	Failure to close on demand	Stroke test	1 month	Experience	4E-3
RSW Pump Circuit Breakers	Failure to trip pump on demand	Breaker trip test to assure trip on demand	6 months	Judgment	1E-3
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	2.0E-2
dc Div 1 Distribution Panel	Panel failure	Panel function is demonstrated by system test	3 months	Experience	5.8E-6
Div 1 EMS Transmission Network	Network failure	System functional test	3 months	Experience	5.9E-4
		Complete system test, error check	2 years	Experience	
Combustion Turbine Generator (CTG)	Failure to start and run	Start and operate CTG at rated speed and load for 1 hour	Monthly	Experience	5.0E-2
		Check oil levels, check for leaks	Monthly	Experience	
		Sample, analyze oil	3 months	Experience	
		Replace oil and filters; inspect inlet and outlet plenums and entire assembly	R/M outage	Experience	

TABLE 19K.11-1

FAILURE MODES & RAP ACTIVITIES (Continued)

COMPONENT	FAILURE MODE/CAUSE	RECOMMENDED MAINTENANCE	TEST OR MAINTENANCE INTERVAL	BASIS	UNAVAIL-ABILITY, FAILURE RATE
Structures of ac Power EDGs, 480Vac Transformers & MCCs; dc Batteries, Battery Racks & Inverters; PSW Pumps, HXs, & A/C Units; SLC Tank & Pumps; & Pumps of HPCF, ACIWA & RHR	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	Judgement	N/A
Single Train of HPCF System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgement	N/A
Single Train of CUW System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgement	N/A
Single Train of RSW System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgement	N/A
Single Train of ac Electrical System	Common mode type failure	System walkdown to identify CCF type problems	R/M outage	Judgement	N/A

APPENDIX 19M

TABLE OF CONTENTS

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 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 1.0E-06 per year. 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.

APPENDIX 19M.3

CONTENTS (Continued)

TABLES

ILLUSTRATIONS

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 SSAR Figure 19M.3-1. They were then screened out by the Step 2 path of the procedures outlined in FIVE Figure 6.0, which is included as SSAR Figure 19M.3-2, 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 1.0E-6 per year.

(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.3-1. Note that the core damage frequencies per year are less than 1.0E-6 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 1.0E-6, even though credit was taken for local manual operation of the RCIC. 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 8.94E-07. 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 this 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 Sections 19M.4 through 19M.6.

Table 19M.3-1

FIRE RISK SCREENING ANALYSIS SUMMARY

Initiators And Conditions	Fire Ignition Frequency	Core Damage Frequency Per Year
Safety-Related Buildings		
Division 1 Fire	8.15E-02	4.34E-08
Division 2 Fire	8.35E-02	1.11E-07
Division 3 Fire	8.92E-02	5.95E-08
Control Room Fire With Remote Control of 4 SRVs and RCIC	4.40E-02	8.94E-07
Turbine Building Fire	1.85E-01	2.19E-07

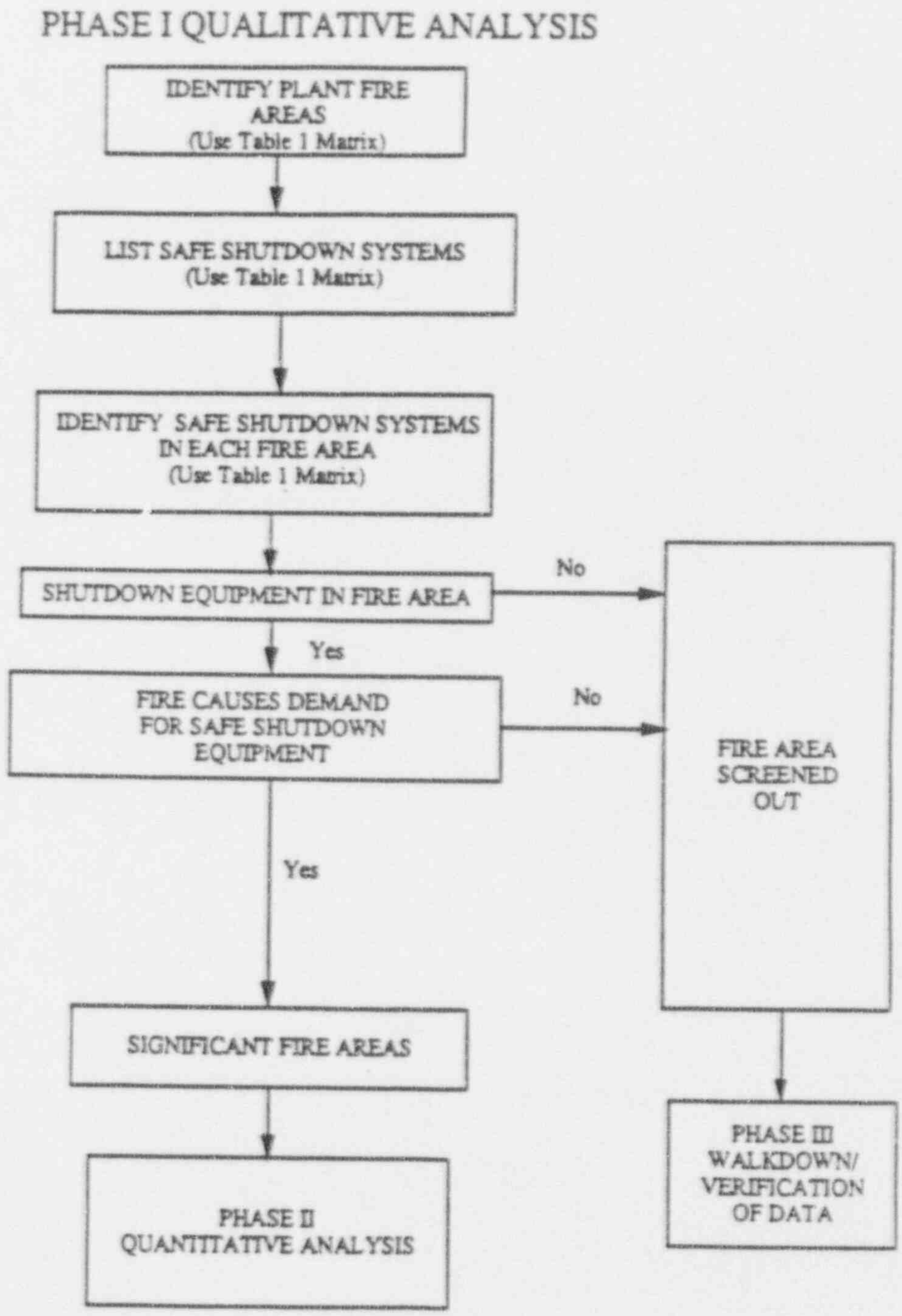

FIGURE 5.1

Figure 19M.3-1 PHASE I QUALITATIVE ANALYSIS FLOW CHART

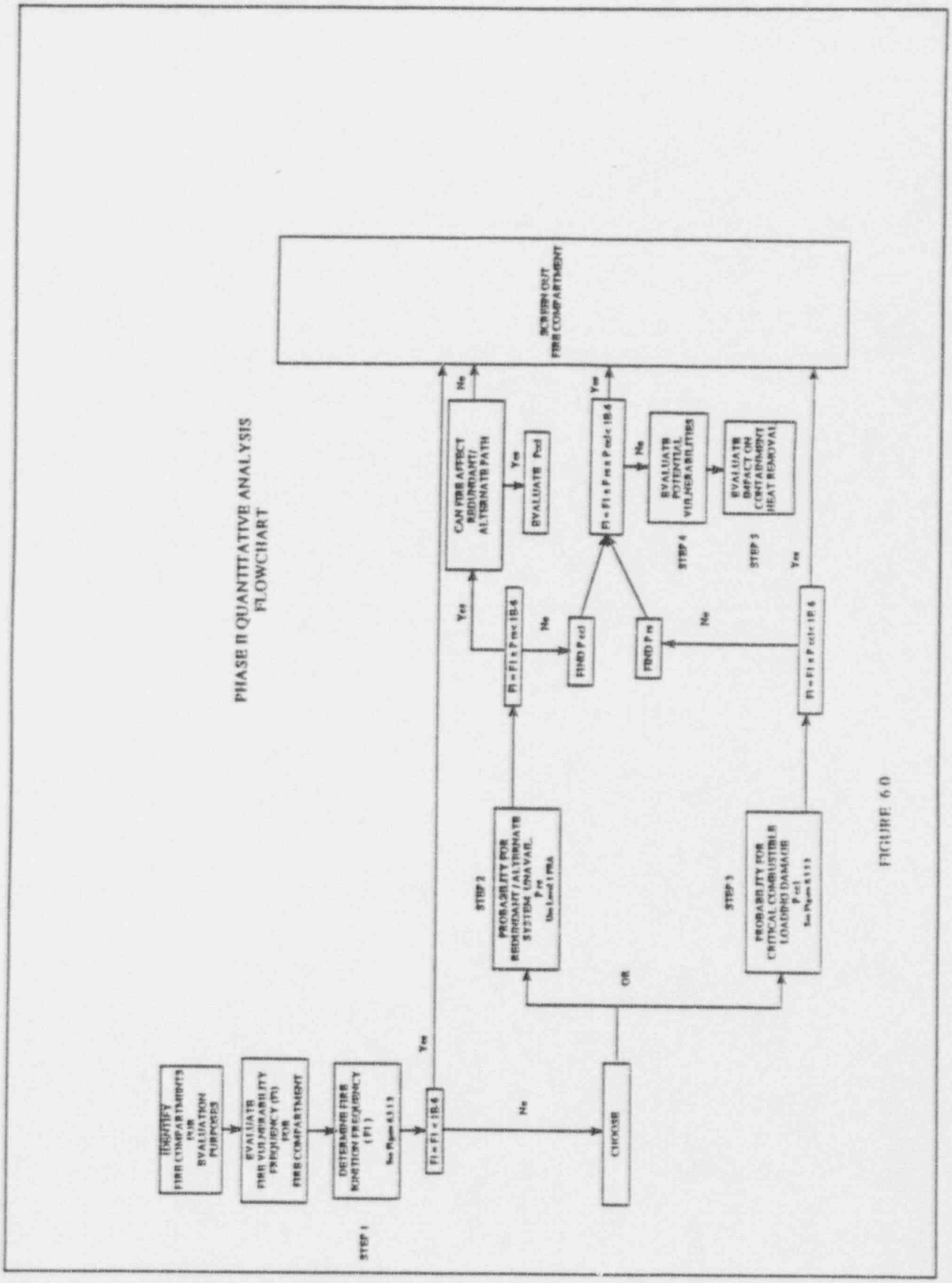

Figure 19M.3-2 PHASE II QUALITATIVE ANALYSIS FLOW CHART

APPENDIX 19M.6

TABLE OF CONTENTS

TABLES

ILLUSTRATIONS

APPENDIX 19M.6

ILLUSTRATIONS (Continued)

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.4-3, 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.4-4 through 19D.4-10, 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 Consequence	Annual Frequency
Fire disabling electrical Division 1	8.15E-02
Fire disabling electrical Division 2	8.35E-02
Fire disabling electrical Division 3	8.92E-02
Control room fire limiting ECCS control to remote shutdown panel	4.40E-02
Turbine building fire resulting in loss of off-site power	1.85E-01

19M.6.2 Results

Calculated core damage frequencies for each of the initiating events are summarized in Table 19M.6-1. Breakdowns of core damage frequency by accident class for each initiator are provided in Tables 19M.6-2 and 19M.6-3. It should be noted that the core damage frequencies for Class II events are reduced by a factor of 1.0E-04 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 Section 19D.5.

Event trees used for the ABWR fire risk screening analysis for divisional and control room fires are illustrated in Figures 19M.6-1 through 19M.6-4. Those for the turbine building fire event are given in Figures 19M.6-5 through 19M.6-11. As indicated in Tables 19M.6-2 and 19M.6-3, 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 criterion of 1.0E-06 per year.

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 pressure injection and manual depressurization 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 over two orders of magnitude 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, remote shutdown panel 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.6-4. It can be seen that neither of the above actions by itself satisfies the screening criterion of 1.0E-06. 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.

Table 19M.6-1

ABWR FIRE RISK SCREENING ANALYSIS SUMMARY

INITIATORS AND CONDITIONS	CORE DAMAGE FREQUENCY (per year)
Division 1 fire	4.34E-08
Division 2 fire	1.11E-07
Division 3 fire	5.95E-08
Control room fire with remote control of 4 SRVs and RCIC	8.94E-07
Turbine Building Fire	2.19E-07

Table 19M.6-2

DIVISIONAL AND CONTROL ROOM FIRE RISK W/REMOTE CONTROL OF RCIC & 4SRVs FOR CR FIRES

INIT. Event	Accident Class 1A	1B-1	1B-2	1B-3	1C	1D	II	IIIA	IIID	IV	TOT. CDF
D1D	7.69E-09	---	---	---	---	3.56E-08	7.08E-11	---	---	---	4.34E-08
D2D	4.42E-08	---	---	---	---	6.67E-08	6.87E-11	---	---	---	1.11E-07
D3D	8.59E-09	---	---	---	---	5.08E-08	7.34E-11	---	---	---	5.95E-08
CR	2.73E-07	---	---	---	---	6.21E-07	1.32E-10	---	---	---	8.94E-07

Table 19M.6-3

SUMMARY OF ABWR RISK SCREENING ANALYSES FOR TURBINE BUILDING FIRE

INIT Event	ACCIDENT CLASS 1A	1B-1	1B-2	1B-3	1C	1D	II	IIIA	IIID	IV	TOT. CDF	PERCENT
TE2	8.21E-09	---	---	---	---	3.61E-11	---	---	---	---	8.25E-09	3.75E+00
TE8	5.28E-09	---	---	---	---	1.72E-11	1.18E-12	---	---	---	5.30E-09	2.41E+00
TE0	1.06E-09	---	---	---	---	1.91E-09	9.44E-11	---	---	---	3.06E-09	1.39E+00
BE2	2.67E-12	---	---	---	---	1.24E-07	---	---	---	---	1.24E-07	5.64E+01
BE8	---	4.76E-08	---	---	---	---	---	---	---	---	4.76E-08	2.17E+01
BE0	---	---	3.00E-08	1.64E-09	---	---	---	---	---	---	3.16E-08	1.44E+01
TOT. CDF	1.46E-08	4.76E-08	3.00E-08	1.64E-09	0.00E+00	1.26E-07	9.56E-11	0.00E+00	0.00E+00	0.00E+00	2.19E-07	1.00E+02
PERCENT	6.64E+00	2.17E+01	1.36E+01	7.46E-01	0.00E+00	5.73E+01	4.35E-02	0.00E+00	0.00E+00	0.00E+00		1.00E+02

Table 19M.6-4

ABWR CONTROL ROOM FIRE RISK SCREENING ANALYSIS SUMMARY

CONDITIONS OF THE CONTROL ROOM FIRE ANALYSIS	CORE DAMAGE FREQUENCY (per year)
Remote control of 3 SRVs	6.27E-05
Remote control of 4 SRVs	1.27E-05
Remote control of 3 SRVs and RCIC	3.89E-06
Remote control of 4 SRVs and RCIC	8.94E-07

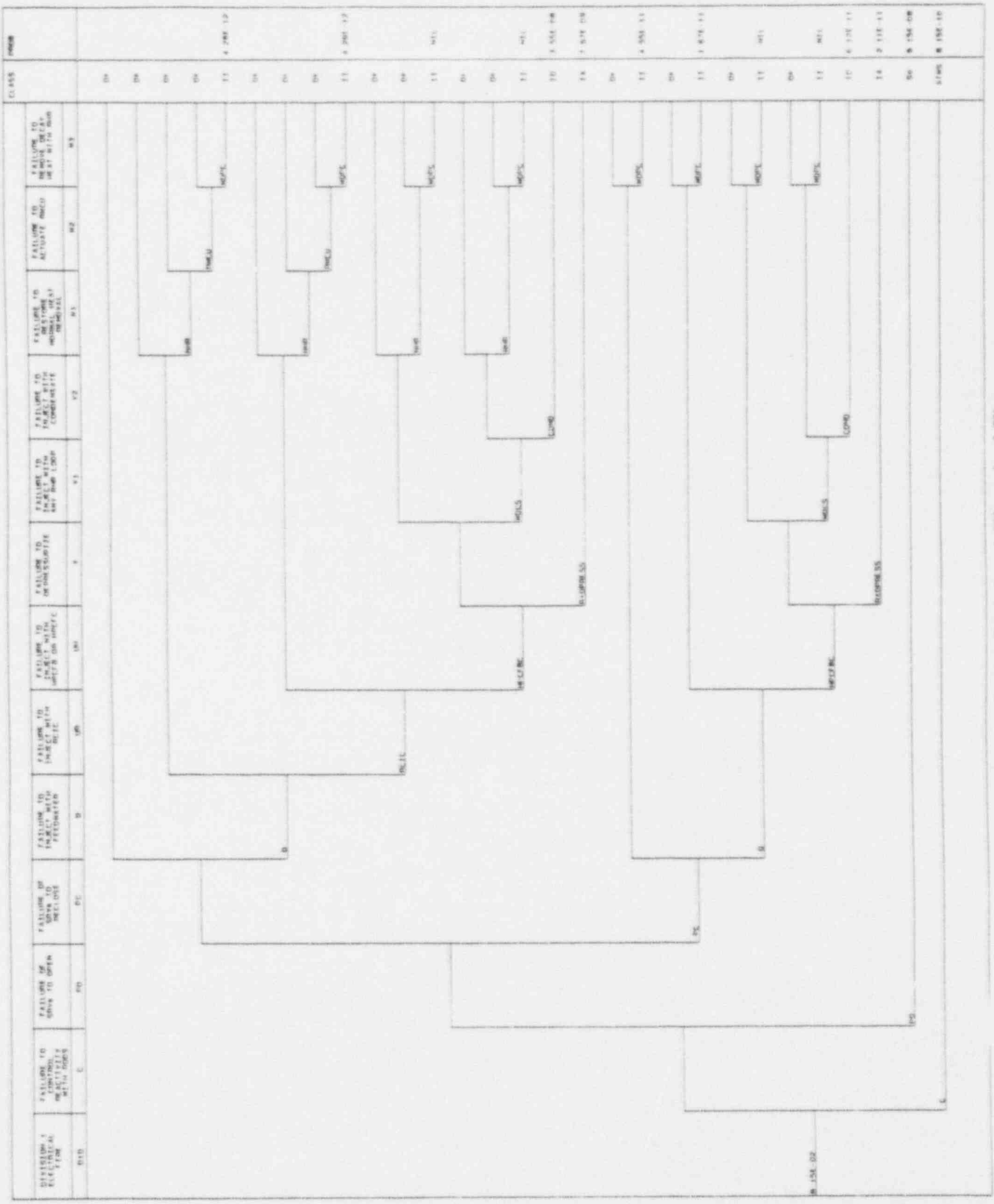

Figure 19M.6-1 Division 1 Electrical Fire

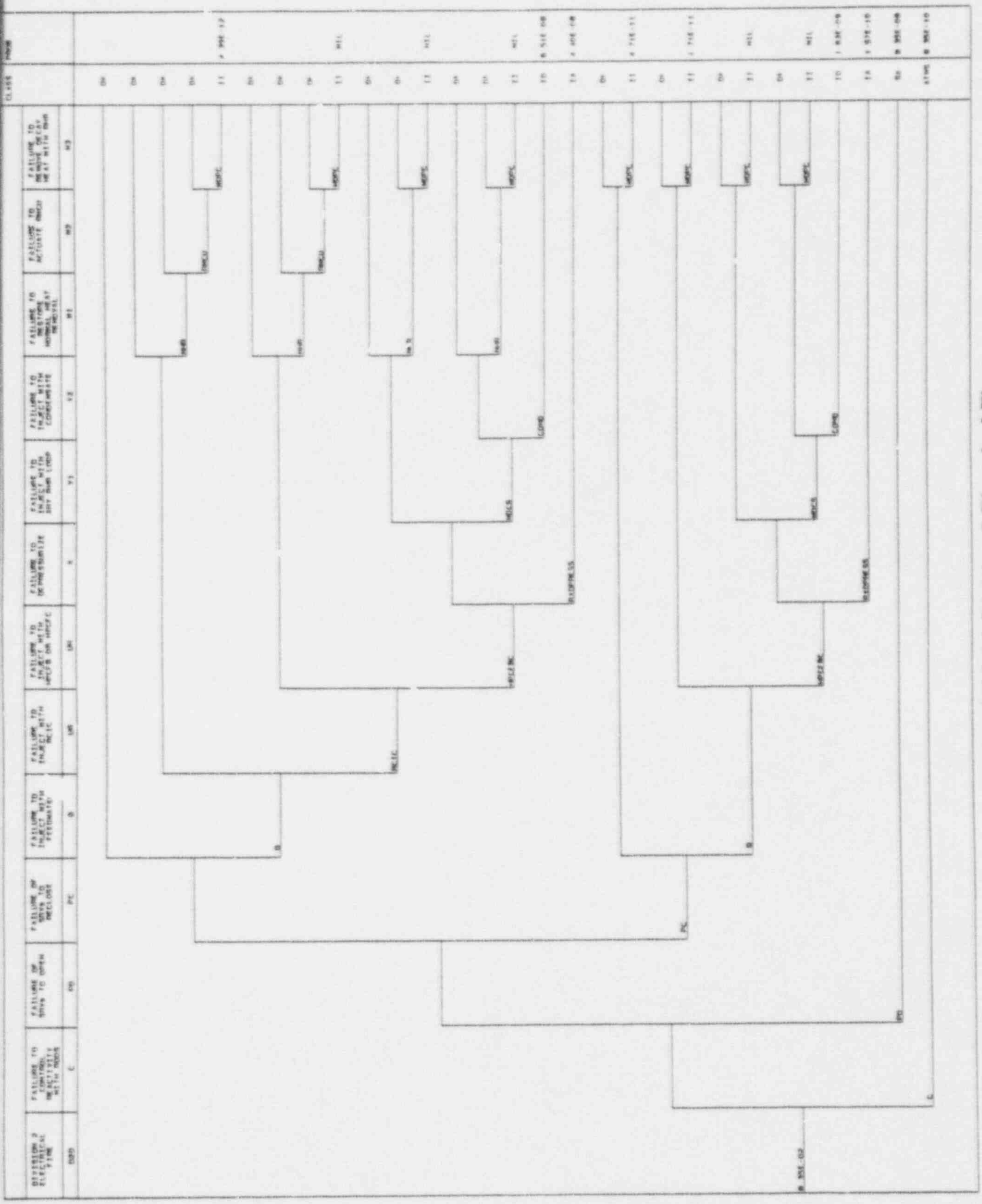

Figure 19M.6-2 Division 2 Electrical Fire

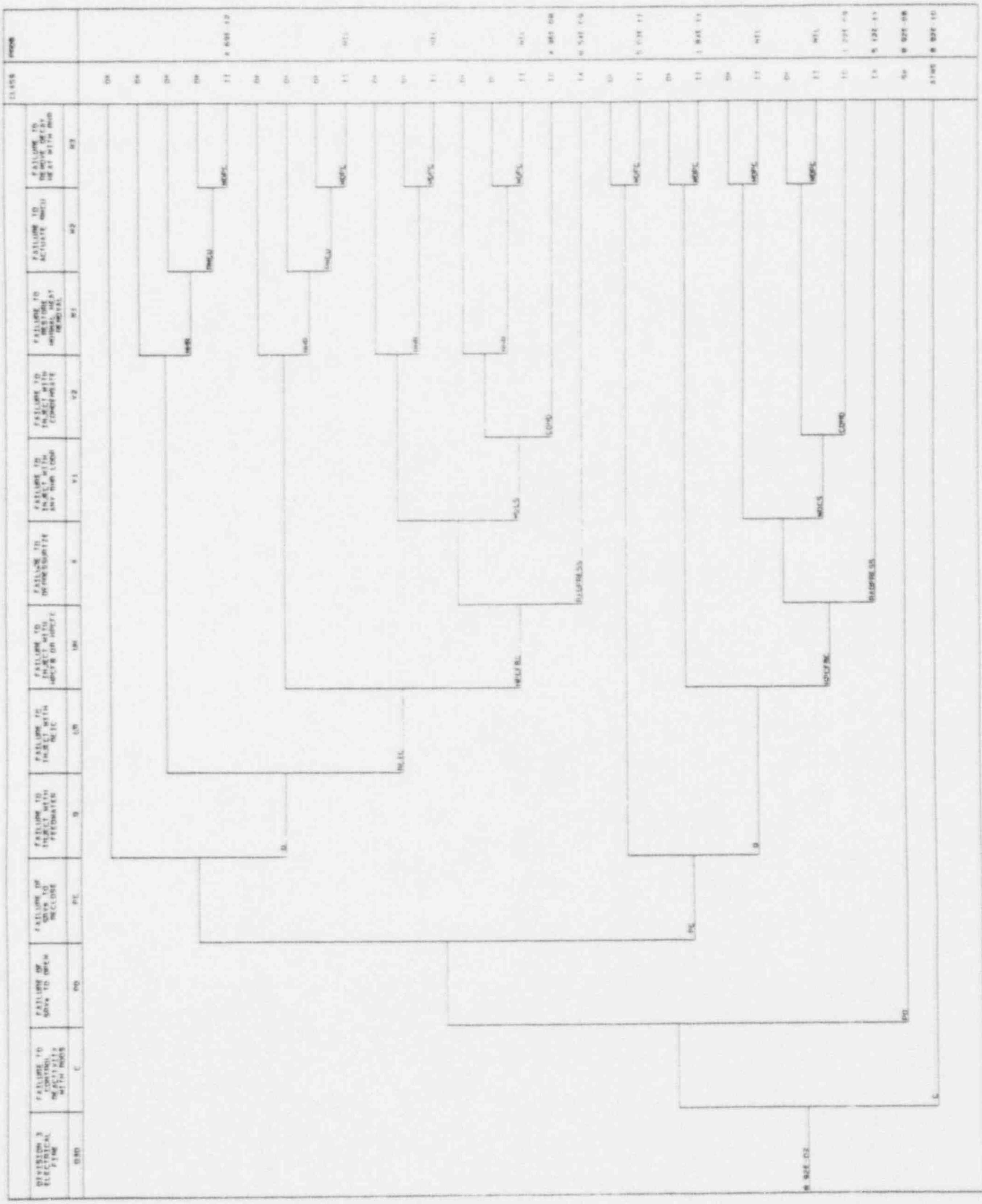

Figure 19M.6-3 Division 3 Electrical Fire

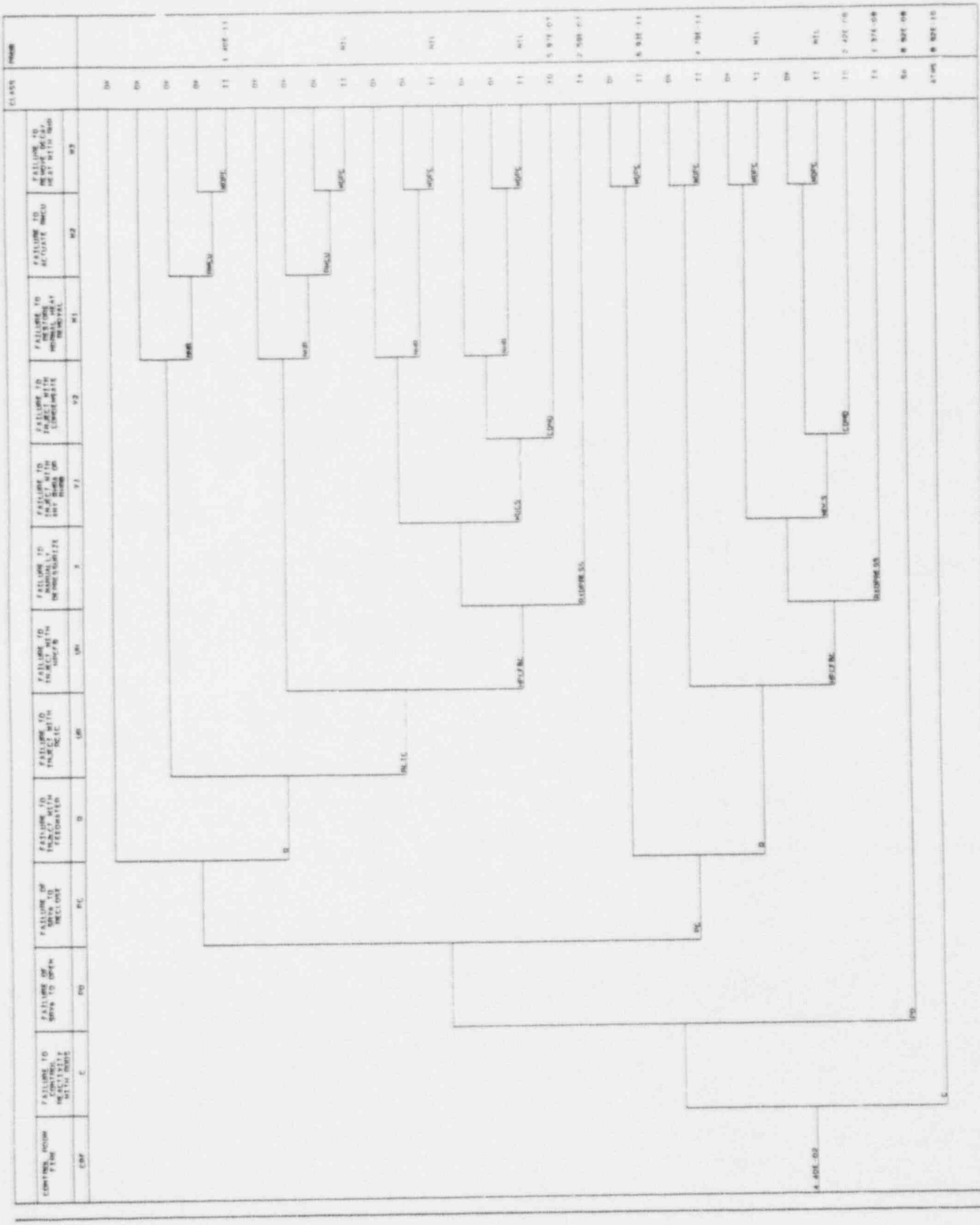

Figure 19M.6-4 Control Room Fire

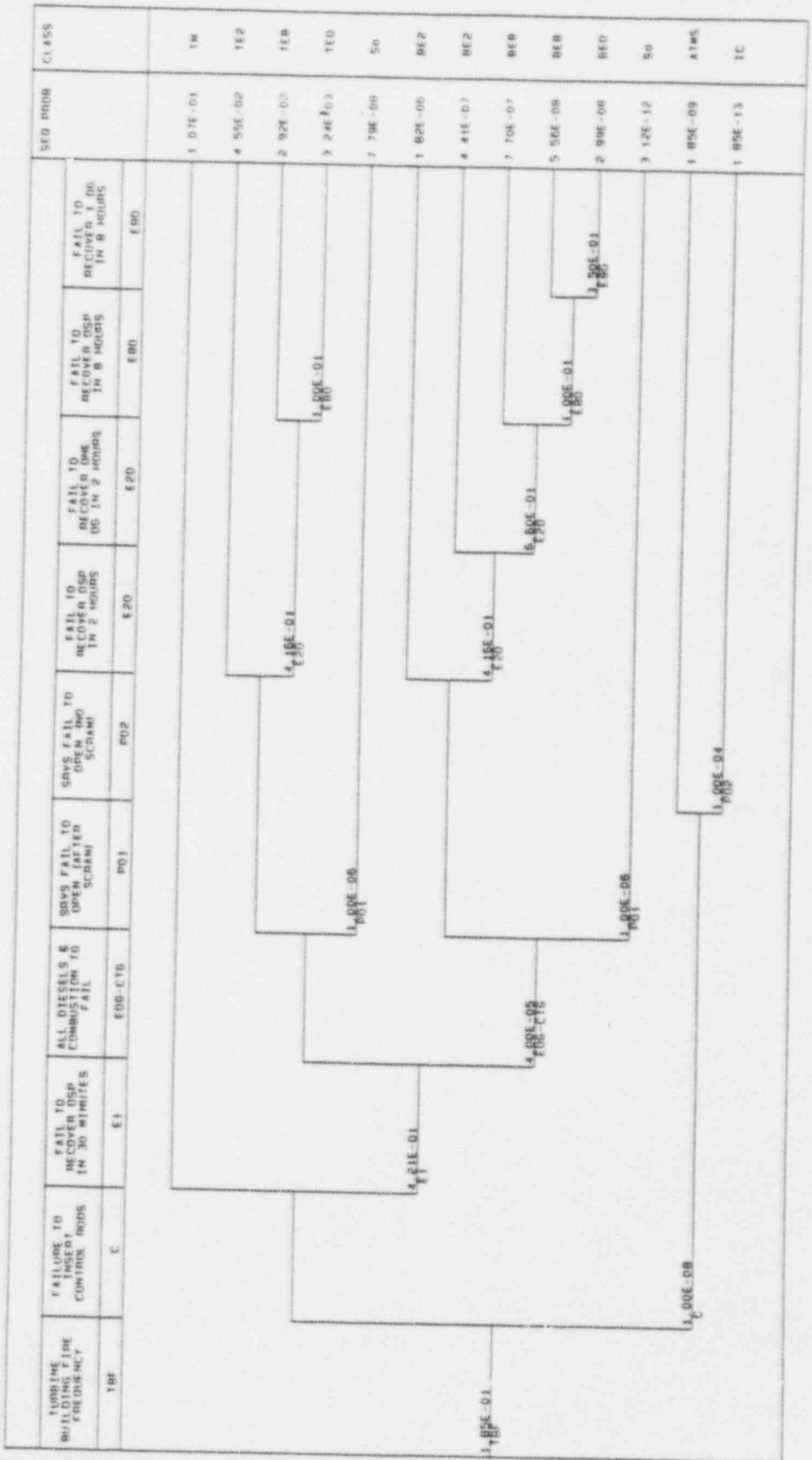

Figure 19M.6-5 Turbine Building Fire (Loss of Offsite Power and Station Blackout Event Tree)

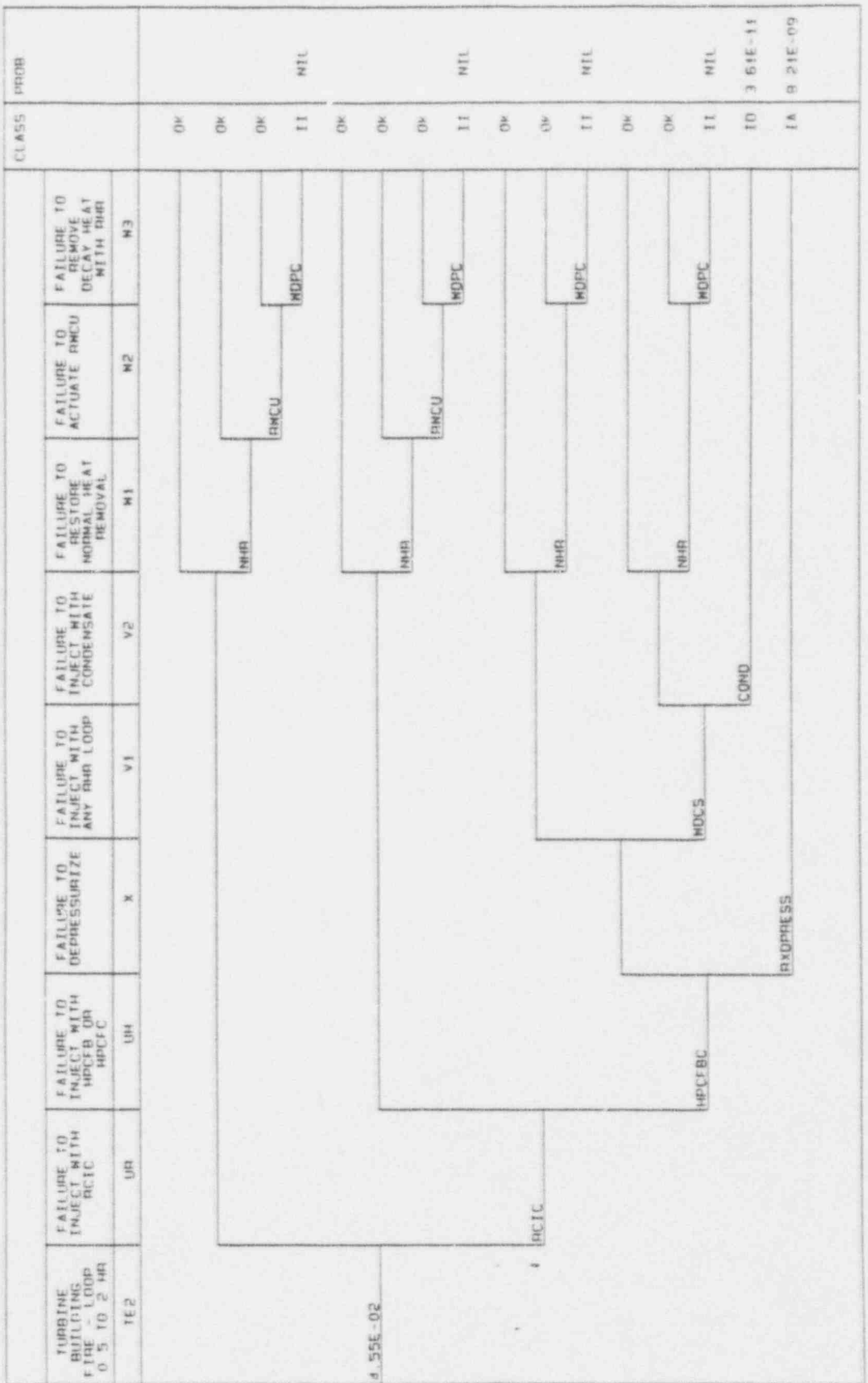

Figure 19M.6-6 Turbine Building Fire - Loss of Offsite Power Event Tree
(Recovery time: 30 min < t < 2 hrs)

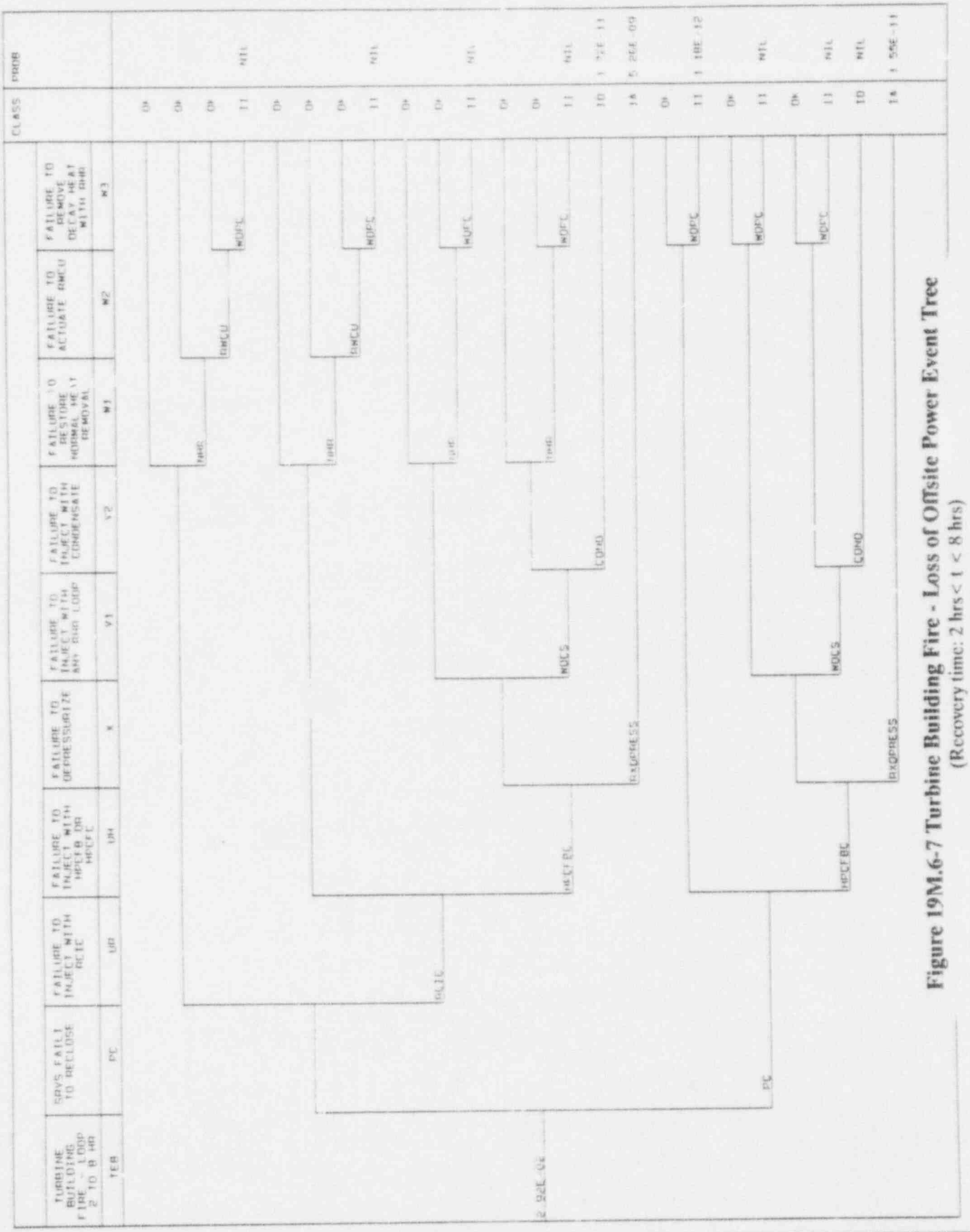

Figure 19M.6-7 Turbine Building Fire - Loss of Offsite Power Event Tree
(Recovery time: 2 hrs < t < 8 hrs)

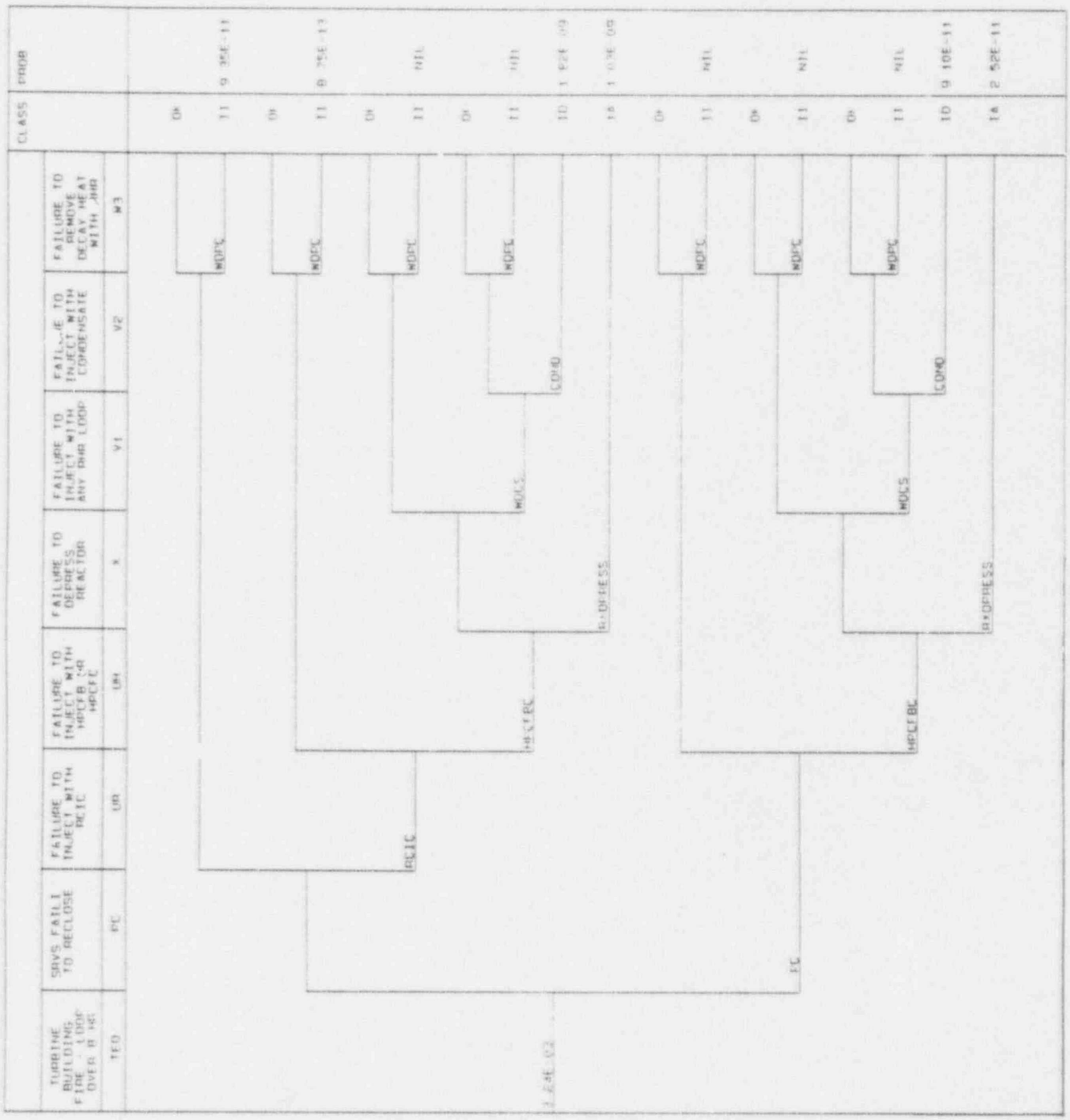

Figure 19M.6-8 Turbine Building Fire - Loss of Offsite Power Event Tree
(Recovery time: t > 8 hrs)

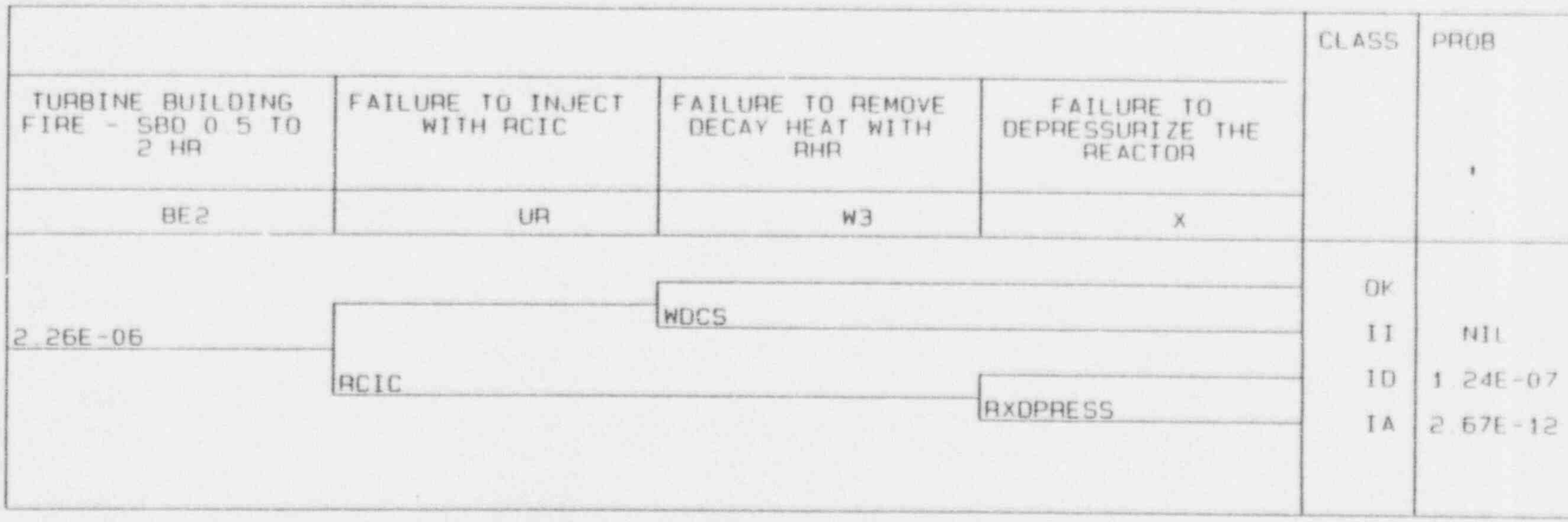

Figure 19M.6-9 Turbine Building Fire - Station Blackout Event Tree
(Recovery time: 30 min < t < 2 hrs)

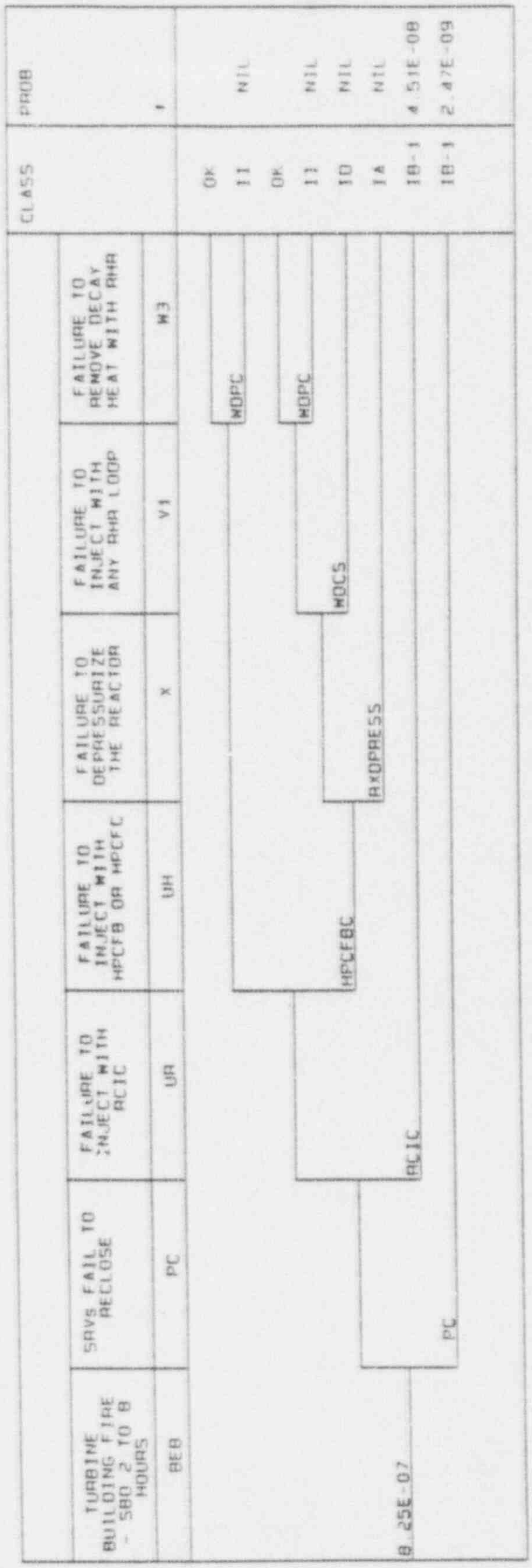

Figure 19M.6-10 Turbine Building Fire - Station Blackout Event Tree
(Recovery time: 2 hrs < t <8 hrs)

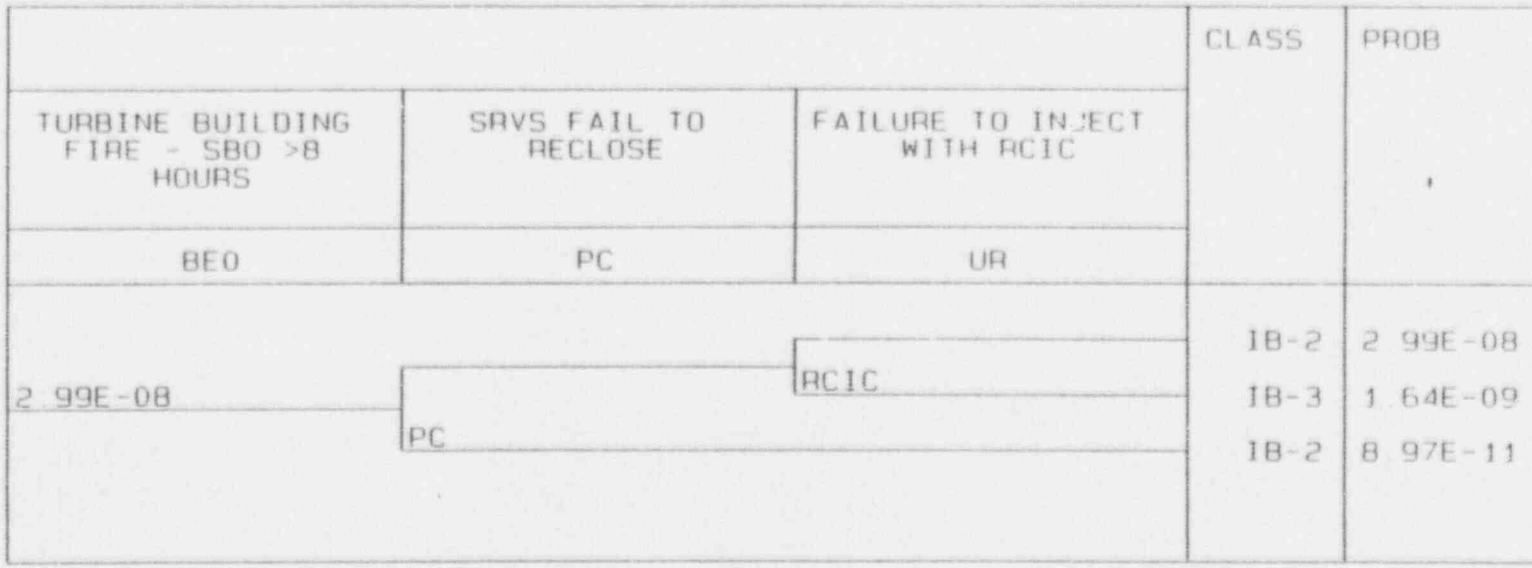

Figure 19M.6-11 Turbine Building Fire - Station Blackout Event Tree
(Recovery time: t > 8 hrs)

APPENDIX 19P
EVALUATION OF POTENTIAL MODIFICATIONS TO THE ABWR DESIGN

APPENDIX 19P

TABLE OF CONTENTS

APPENDIX 19P

TABLE OF CONTENTS (Continued)

SECTION 19P.1

CONTENTS

19P.1 INTRODUCTION AND SUMMARY

This section provides a description of an evaluation of potential changes to the ABWR design in order to determine whether further modifications can be justified.

19P.1.1 Background

The U.S. Nuclear Regulatory Commission's policy related to severe accidents requires, in part, that an application for a design approval comply with the requirements of 10CFR50.34(f). Item (f)(1)(i) requires "performance of] a plant site specific [PRA] the aim of which is to seek improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant". Chapter 19 provides the base PRA of the ABWR plant.

To address this requirement, a review of potential modifications to the ABWR design, beyond those included in the Probabilistic Risk Assessment (PRA), was conducted to evaluate whether potential severe accident design features could be justified on the basis of cost per person-rem averted.

This appendix summarizes the results of GE's review and evaluation of the ABWR design. Improvements have been reviewed against conservative estimates of risk reduction based on the PRA and minimum order of magnitude costs, to determine what modifications are potentially attractive.

19P.1.2 Evaluation Criteria

The benefit of a particular modification was defined to be its reduction in the risk to the general public.

Offsite factors evaluated were limited to health effects to the general public based on total exposure (in person-rem) to the population within 50 miles of the site. Five representative US regions were evaluated for selected individual ABWR sequences by the CRAC02 code. The regional results were then averaged to determine the exposures. Consistent with the standard used by the NRC to evaluate radiological impacts, health effect costs were evaluated based on a value of $ 1000 per offsite person-rem averted due to the design modification.

The offsite costs for other items such as relocation of local residents, elimination of land use and decontamination of contaminated land were not considered. Reductions in the risk of incurring onsite costs including economic losses, replacement power costs and direct accident costs are considered in this evaluation as credits against in the cost of the modification.

Based on the PRA results (Section 19P.2) 82% of the offsite risk results from very low probability events which have high consequence. The maximum justifiable cost of a modification was determined to be $183. Therefore, based on this methodology, no modifications are justifiable. However, a variety of modifications were reviewed to establish the relative attractiveness of potential changes.

19P.1.3 Methodology

The overall approach was to estimate the benefit of modifications in terms of dollar cost per total person-rem averted. Underestimated costs and overestimated benefits were assessed in order to favor modifications. Because of the uncertainties in the methodology and the desire to address severe accidents with sensible modifications, this basis is judged to be acceptable for purposes of this study.

19P.1.3.1 Selection of Modifications

Potential modifications were identified from a variety of previous industry and NRC sponsored studies of preventative and mitigative features which address severe accidents. Based on this composite list of modifications considered on previous designs, potential modifications were selected for further review based on being 1) applicable to the ABWR design and 2) not included in the reference PRA. Additional detail on the selection of modifications is provided in Section 19P.3.

19P.1.3.2 Costs Basis

Rough order of magnitude costs were assigned for each modification based on the costs of systems and system improvements determined by GE. These costs represent the estimated incremental costs that would be incurred in a new plant rather than costs that would apply on a backfit basis. Section 19P.5 defines the cost estimates for each of the modifications.

Even for a new plant such as the ABWR, relatively large costs (several million dollars) can be expected for some modifications if they involve modifications of the building structures or arrangement. This is because the cost of labor and material is often a function of the building area required. For other modifications which involve minor hardware addition, the cost is often dominated by the need for procedure and training additions which can amount to hundreds of thousands of dollars.

The costs estimates were intentionally biased on the low side, but all known or reasonably expected costs were accounted for in order that a reasonable assessment of the minimum cost would be obtained. Actual plant costs are expected to be higher than indicated in this evaluation. All costs are referenced to 1991 U.S. dollars.

For modifications which reduce the core damage frequency, the costs of modifications (see Section 19P.5) were further reduced by an amount proportional to the reduction present worth of the risk of averted onsite costs. Onsite costs include replacement power costs, direct accident costs (including onsite cleanup) and the economic loss of the facility. Evaluation of this credit included the following considerations:

(1) Accidents were assumed to occur at any time during the 60 year life of the plant. All onsite costs associated with the accident were evaluated as to their value at the time of the accident. The economic risk of such onsite costs was evaluated as a function of time based on the onsite costs and the core damage frequency determined by the PRA. The plant core damage frequency was considered to be constant over the life of the plant. The economic risks were then evaluated based on the present worth of the time dependent economic risks.

(2) Replacement power was based on a rate of $.013/kwh differential as bar cost. The differential rate was assumed to be constant over the remaining life of the plant.

(3) The economic value of the facility at the time of the accident was based on a straight line depreciated value. The initial invested cost was taken at $1.4 Billion based on DOE cost guidelines.

(4) Accident costs for onsite cleanup and facility were evaluated based on escalated costs to the time of the accident. Reference accident costs to the facility were assumed to be $2 Billion.

(5) The economic evaluations were based on a discount rate of 8% and escalation factor of 3%.

19P.1.3.3 Benefit Basis

The cumulative risk of accidents occurring during the life of the plant was used as a basis for estimating the maximum benefit that could be derived from modifications. A particular modification's benefit was based on its effect on the frequency of events or

associated offsite dose summarized in Tables 19P.2-1 and Table 19P.2-2. Dominant contributing failure probabilities were identified based on the PRA. Changes in these probabilities were estimated to evaluate the benefit of modifications. This basis is consistent with the approach taken in previous NRC evaluations. The cumulative offsite risk was evaluated over a 60 year plant life with no escalation in the evaluation criteria of $1000/person-rem.

Section 19P.4 summarizes each concept and estimated benefit for each individual potential modification. For each modification the cost per person-rem averted was evaluated to obtain the results of the individual evaluations. These conclusions are provided in Section 19P.7.

19P.1.4 Summary of Results

Potentially attractive modifications were selected based on previous evaluations of potential prevention and mitigation concepts applicable during severe accidents. Of the modifications applicable to the ABWR design and which were not already implemented, twenty one were selected for additional review.

None of the modifications considered met the $1,000/person-rem averted criteria. The low evaluated frequency of core damage and subsequent release of radioactive material does not support modification to the ABWR based on costs in relationship to the benefit of averted exposures.

Since the most beneficial modification was evaluated to be several orders of magnitude higher than the criteria, it was concluded that no additional modifications are warranted in the ABWR design to address severe accidents. Furthermore, due to its magnitude it can be calculated that this conclusion will not be sensitive to variations in the assumptions used in the PRA results.

SECTION 19P.2

TABLES

19P.2 SEVERE ACCIDENT RISK OF ABWR

The reference design for this study was the ABWR PRA as presented in the treated in the internal events PRA [Section 19.0]. This evaluation accounts for features which were included in the current ABWR design specifically to address severe accidents. These are further discussed in Section 19.3.1.5. These features and the reference description include:

	SSAR References
1) Firewater pump crosstie	5.4.1.1.10
2) Passive containment flooder	19.5.12
3) Gas turbine generator	9.5.11
4) Overpressure Protection	6.2.5.2.6

A summary of the core damage frequency and offsite exposure frequency with these features included is shown in Table 19P.2-1. Event frequencies used in this evaluation were the same as assumed in the base PRA.

The offsite exposures shown in Table 19P.2-1 were calculated by the CRAC2 code for release cases with similar consequences. Although discussed in Appendix 19E.3.2, the cases can be characterized as follows:

Case 1 Core Melt arrested in vessel or in Containment with actuation of containment rupture disk.

Case 2 Low Pressure Core Melt with suppression pool bypass and actuation of containment rupture disk.

Case 3 High Pressure Core Melt with drywell Head failure and fire water spray initiation.

Case 4 Suppression Pool Decontamination reduction (Not used).

Case 5 Large Break LOCA without recovery and with actuation of containment rupture disk.

Case 6 High Pressure Core Melt with Drywell Head failure and no firewater spray initiation.

Case 7 Low Pressure Core Melt with Drywell Head failure and no mitigation.

Case 8 High Pressure Core Melt with Early Containment failure.

Case 9 ATWS event with Drywell Head failure.

NCL Normal Containment Leakage to Reactor Building.

The offsite exposures for each case shown in Table 19P.2-1 were calculated by the CRAC2 code for five representative US regions for the selected individual ABWR sequences as discussed in Section 19E.3.

Table 19P.2-2 provides additional detail on the individual contributors to the total core damage frequency. As indicated on Table 19P.2-2, the core damage frequency is dominated by low pressure transient events (LCLP) (61.4%), followed by high pressure transient events (LCHP) (28.1%) and station blackout sequences (SBRC) (10.3%).

Review of Table 19P.2-1 also indicates that the dominant contributors to the ABWR offsite exposure risk are the relatively low probability (less than 4E-10/yr), high consequence events (Cases 6 through 9) which contribute about 82% of the offsite exposure risk.

Table 19P.2-1

OFFSITE ACCIDENT CASES

CASE*	FREQUENCY (per yr)	MANREM EXPOSURE** (per event)	CONTRIBUTION (per 60 yrs)	(%)
Case 1	2.0E-08	11,500	.014	7.7
Case 2	7.8E-11	8,328	3.9E-5	0.02
Case 3	1.3E-12	371,400	2.8E-5	0.02
Case 4	0	206,400	0	0
Case 5	6.3E-12	93,380	3.5E-5	0.02
Case 6	1.2E-10	2,416,000	.018	9.8
Case 7	3.7E-10	2,726,000	.061	33.2
Case 8	2.1E-10	3,202,000	.040	21.9
Case 9	1.5E-10	3,312,000	.031	16.8
NCL	1.4E-07	2,300	.019	10.4
TOTAL	1.6E-07		.183	100

* For case descriptions see Table 19E.3-6; frequencies are based on Table 19P.2-2.

** Average of regional values used; see Section 19E.3.

Table 19P.2-2

Core Damage Frequency Contributors

INIT. EVENT	1A	1B1	1B2	1B3	1D	II	IIID	IV	TOTAL	% CONTRIB
				EVENT SEQUENCE						
SCRAM	1.1E-08				4.3E-10	9.5E-13			1.1E-08	7.3
TURB TRIP	6.8E-09				2.7E-10	3.7E-11			7.1E-09	4.5
ISOLATION	1.8E-08				7.1E-10	1.1E-11			1.9E-08	11.9
LOOP2	4.1E-09				1.5E-11	4.2E-13			4.1E-09	2.6
LOOP8	2.4E-09				9.6E-12	1.4E-12			2.4E-09	1.5
LOOP8+	5.8E-10				1.1E-09	6.0E-11			1.7E-09	1.1
SBO2	6.6E-12				6.7E-08				6.7E-08	42.9
SBO8		2.6E-08							2.6E-08	16.7
SBO8+			1.5E-08	8.9E-10					1.6E-08	10.3
IORV	1.1E-09				2.0E-10	9.5E-13			1.3E-09	0.8
SB LOCA							2.5E-10		2.5E-10	0.2
ATWS								1.5E-10	1.5E-10	0.1
TOTAL	**4.4E-08**	**2.6E-08**	**1.5E-08**	**8.9E-10**	**7.0E-08**	**1.1E-10**	**2.5E-10**	**1.5E-10**	**1.57E-07**	**100**

OFFSITE RELEASE GROUP*	LCHP	SBRC	LCLP	LHRC	LBLC	ATWS	TOTAL CASE
CASE 1	3.4E-09	7.9E-10	1.6E-08		5.1E-11		2.0E-08
CASE 2			7.8E-11				7.8E-11
CASE 3	1.3E-12						1.3E-12
CASE 4							0
CASE 5					6.3E-12		6.3E-12
CASE 6	1.2E-10						1.2E-10
CASE 7	1.1E-10		2.6E-10				3.70E-10
CASE 8	2.1E-10						2.1E-10
CASE 9				1.1E-12		1.5E-10	1.5E-10
NCL (N)	4.0E-08	1.5E-08	8.0E-08		2.0E-10		1.4E-07
TOTAL	**4.4E-08**	**1.6E-08**	**9.6E-08**	**1.1E-12**	**2.5E-10**	**1.5E-10**	**1.57E-07**
CONTRIB %	28.1	10.3	61.4	0.122	0.2	0.1	100

* For description see Section 19E.2.2

SECTION 19P.3

TABLES

19P.3 POTENTIAL ABWR MODIFICATIONS

Potential modifications to the ABWR design were derived from a survey of various studies indicated in references 19P.8.1 through 19P.8.7 and the ABWR design process discussed in Section 19.7. From these, a composite list of modifications was established. This list of potential modifications was reviewed to identify concepts which were already included in the ABWR design or which are not applicable.

Table 19P.3-1 summarizes the complete list of modifications and their classification according to the following categories:

1. Modification is applicable to ABWR and already incorporated in the ABWR design. No further evaluation is needed. (Table 19P.3-1 provides a cross reference to the supporting section of the SSAR.)

2. Modification is applicable to ABWR and not incorporated in ABWR design. (Table 19P.3-2 lists the Category 2 modifications which are evaluated further in this report.)

3. Modification is not applicable to the ABWR design due to the basis provided.

4. Modification is applicable to ABWR and is incorporated with the referenced modification

Table 19P.3-1

MODIFICATIONS CONSIDERED

	Modification	Category	Basis (SSAR Reference)	Reference (19P.8-X)
1.	**ACCIDENT MANAGEMENT**			
a.	Severe Accident EPGs/AMGs	2		1
b.	Computer Aided Instrumentation	2		1
c.	Improved Maintenance Procedures/Manuals	2		1
d.	Preventive Maintenance Features	4	See 1c	1
e.	Improved Accident Mgt Instrumentation	4	See 1b	1
f.	Remote Shutdown Station	1	(7.4.2)	1
g.	Security System	1	(13.6.3)	1
h.	Simulator Training for Severe Accidents	4	See 1b	1
2.	**REACTOR DECAY HEAT REMOVAL**			
a.	Passive High Pressure System	2		1
b.	Improved Depressurization	2		1,2,3,5
c.	Suppression Pool Jockey Pump	2		1
d.	Improved High Pressure Systems	1	(6.3)	1
e.	Additional Active High Pressure System	1	(6.3)	1,5
f.	Improved Low Pressure System (Firepump)	1	(5.4.7.1.1.10)	1,2,3
g.	Dedicated Suppression Pool Cooling	1	(6.2.2)	1,2
h.	Safety Related Condensate Storage Tank	2		1
i.	Extended Station Blackout Injection	4	see 10e	1
j.	Improved Recirculation Mode	3	PWR	3
3.	**CONTAINMENT CAPABILITY**			
a.	Larger Volume Containment	2		1,4
b.	Increased Containment Pressure Capacity	2		1
c.	Improved Vacuum Breakers	2		1
d.	Increased Temperature Margin for Seals	1	(19F.3.2.2)	1,4
e.	Improved Leak Detection	1	(7.3.2)	3
f.	Suppression Pool Scrubbing	1	(19E.3.2)	5
g.	Improved Bottom Penetration Design	2		
4.	**CONTAINMENT HEAT REMOVAL**			
a.	Larger Volume Suppression Pool	2		1
b.	RWCU Decay Heat Removal	1	(19.3)	2
c.	High Flow Suppression Pool Cooling	1	(6.2.2)	1,4
d.	Passive Overpressure relief	1	(6.2.5.2.6)	1,2,5
5.	**CONTAINMENT ATMOSPHERE MASS REMOVAL**			
a.	High Flow Unfiltered Vent	3	Mark III	1,4
b.	High Flow Filtered Vent	3	Mark III	1,4
c.	Low Flow (filtered) Vent	2		1,2,3,4
d.	Low Flow Vent (unfiltered)	1	(6.2.5.2.6)	1,2,3,4

Table 19P.3-1

MODIFICATIONS CONSIDERED (Continued)

	Modification	Category	Basis (SSAR Reference)	Reference (19P.8-X)
6.	COMBUSTIBLE GAS CONTROL			
a.	Post Accident Inerting System	3	Inerted	1,4
b.	Hydrogen Control by Venting	3	Inerted	1,4
c.	Preinerting	1	Inerted	1,4
d.	Ignition Systems	3	Inerted	1,3,4,6
e.	Fire Suppression System Inerting	3	Inerted	1,4
7.	CONTAINMENT SPRAY SYSTEMS			
a.	Drywell Head Flooding	2		2
b.	Containment Spray Augmentation	1	(5.4.7.1.1.10)	1,2,3,6
8.	PREVENTION CONCEPTS			
a.	Additional Service Water Pump	2		3
b.	Improved Operating Response	1	(7.7.2)	1
c.	Diverse Injection System	4	See 2a	1
d.	Operating Experience Feedback	1		1
e.	Improved MSIV/SRV Design	1	(5.4.5,5.4.13)	1
9.	AC POWER SUPPLIES			
a.	Steam Driven Turbine Generator	2		1
b.	Alternate Pump Power Source	2		1
c.	Deleted			
d.	Additional Diesel Generator	1	(8.3.1)	1,3
e.	Increased Electrical Divisions	1	(8.3.1)	1
f.	Improved Uninterruptable Power Supplies	1	(8.3.1)	1
g.	AC Bus Cross-ties	1	(8.3.1)	1
h.	Gas Turbine	1	(9.5.11)	1
i.	Dedicated RHR (bunkered) Power Supply	4	See 2g	1
10.	DC POWER SUPPLIES			
a.	Dedicated DC Power Supply	2		1
b.	Additional Batteries/Divisions	4	See 10e	1
c.	Fuel Cells	4	See 10e	1
d.	DC Cross-ties	1	(8.3.2)	1
e.	Extended Station Blackout Provisions	1	(19E.2.1.2.2)	1,3
11.	ATWS CAPABILITY			
a.	ATWS Sized Vent	2		2,4,6
b.	Improved ATWS Capability	1	(19.7.2(2),(4))	1,5,6
12.	SEISMIC CAPABILITY			
a.	Increased Seismic Margins	1	(19I)	9
b.	Integral Basemat	3	Mark III	1
13.	SYSTEM SIMPLIFICATION			
a.	Reactor Building Sprays	2		2
b.	System Simplification	1	(1.3.5)	1
c.	Reduction in Reactor Bldg Flooding	1	(19.7.3(4))	5

Table 19P.3-1

MODIFICATIONS CONSIDERED (Continued)

	Modification	Category	Basis (SSAR Reference)	Reference
14.	**CORE RETENTION DEVICES**			
a.	Flooded Rubble Bed	2		1,2,4,6
b.	Reactor Cavity Flooder	1	(19.5.12)	3
c.	Basaltic Cements	1	(19.7.3(4))	1,4

Table 19P.3-2

MODIFICATIONS EVALUATED

1 ACCIDENT MANAGEMENT	a.	Severe Accident EPGs/AMGs
	b.	Computer Aided Instrumentation
	c.	Improved Maintenance Procedures/Manuals
2 DECAY HEAT REMOVAL	a.	Passive High Pressure System
	b.	Improved Depressurization
	c.	Suppression Pool Jockey Pump
	d.	Safety Related Condensate Storage Tank
3 CONTAINMENT CAPABILITY	a.	Larger Volume Containment
	b.	Increased Containment Pressure Capacity
	c.	Improved Vacuum Breakers
	d.	Improved Bottom Penetration Design
4 CONTAINMENT HEAT REMOVAL	a.	Larger Volume Suppression Pool
5 CONTAINMENT ATMOSPHERE GAS REMOVAL	a.	Low Flow Filtered Vent
7 CONTAINMENT SPRAY	a.	Drywell Head Flooding
8 PREVENTION CONCEPTS	a.	Additional Service Water Pump
9 AC POWER SUPPLIES	a.	Steam Driven Turbine Generator
	b.	Alternate Pump Power Source
10 DC POWER SUPPLIES	a.	Dedicated DC Power Supply
11 ATWS CAPABILITY	a.	ATWS Sized Vent
13 SYSTEM SIMPLIFICATION	a.	Reactor Building Sprays
14 CORE RETENTION DEVICES	a.	Flooded Rubble Bed

SECTION 19P.4

CONTENTS

SECTION 19P.4

CONTENTS (Continued)

TABLES

19P.4 RISK REDUCTION OF POTENTIAL MODIFICATIONS

This section provides evaluations of the benefits of potential modifications to the ABWR design identified in Table 19P.3-2. For each modification the basis for the evaluation and the concept is described. Table 19P.4-1 summarizes the benefit in terms of person-rem averted risk for each of the evaluated modifications.

19P.4.1 Accident Management

Accident management is a current topic under generic development within the Industry through the development of Accident Management Guidelines (AMGs) and revisions to Emergency Procedure Guidelines (EPGs). The following modifications are based on implementation of such generic activity.

19P.4.1.1 Severe Accident EPGs/AMGs

The symptom based EPGs, were developed by the BWR Owners Group following the accident at Three Mile Island, Unit 2. Currently the EPGs are under revision and accident management guidelines (AMGs) are being developed for severe accidents. These should provide a significant improvement which reduces the likelihood of a severe accident. Elements of these guidelines (such as containment pressure and temperature control guidelines) also deal with mitigating the effects of accidents.

In the ABWR PRA, Emergency Operating Procedures (EOPs) are based on these guidelines. Additional extensions of the EPGs and EOPs could be made to address arrest of a core melt, emergency planning, radiological release assessment and other areas related to severe accidents.

Since the existing EPGs cover preventive actions and some mitigative actions, the incremental benefit of this item would be primarily mitigative. It was judged that the reliability of manual actions associated with mitigation could be improved by 10%, especially in use of core melt arrest processes. Failure rates for manually initiated mitigative systems were decreased by 10%, to estimate the benefit. The resulting offsite risk reduction is about .015 person-rem over 60 years.

19P.4.1.2 Computer Aided Instrumentation

Computer aided artificial intelligence can be added which provides attention to risk issues in man-machine interfaces. Significant computer assisted display and plant status monitoring is already part of the ABWR control room design. Additional artificial intelligence could be designed which would display procedural options for the operator to evaluate during severe accidents. The system would be an extension of ERIS to provide human engineered displays of the important variables in the EPGs and AMGs.

Operator actions are made significantly more reliable by new features such as Emergency Procedure Guidelines, Safety Plant Parameter Displays (SPDS), and training on simulators. If the improvements described in Subsection 19P.4.1.1 are assumed to be implemented, the incremental benefit of additional improvements is expected to be low. The reliability of manually initiated preventive systems was increased by 10% to estimate the benefit. The estimated incremental benefit over severe accident EPGs (Subsection 19P.4.1.1) is about 3% in core damage frequency (CDF). Because the improvement affects all release cases, the incremental benefit is about .01 person-rem.

19P.4.1.3 Improved Maintenance Procedures/Manuals

For the GE scope of supply this item would provide additional information on the components important to the risk of the plant. As a result of improved maintenance manuals and information it would be expected that increased reliability of the important equipment would occur. This item would be a preventative improvement which would address several system or components to different degrees.

Based on a 10% improvement in the reliability of the High Pressure Core Flooder (HPCF), Reactor Core Isolation Cooling (RCIC), Residual Heat Removal (RHR) and Low Pressure Core Flooder (LPCF) systems, the CDF is reduced by about 9% which has a corresponding estimated person-rem reduction of about .016.

19P.4.2 Decay Heat Removal

Significant improvements in the reliability of ABWR high pressure systems have been made. Among these are RCIC restart (NUREG 0737, II.K.3.13) and isolation reliability improvements (NUREG 0737, II.K.3.15). Additionally, the redundant HPCF is an improvement over early product lines which used the single HPCS system.

19P.4.2.1 Passive High Pressure System

This concept would provide additional high pressure capability to remove decay heat through a diverse isolation condenser type system. Such a system would have the advantage of removing not only decay heat, but containment heat if a similar system to that under consideration for the Simplified BWR (SBWR) is employed.

The benefit of this system would be equivalent to an additional diverse RCIC system in addition to an additional containment heat removal system. The added system was assumed to be 90% reliable, designed to operate independent of offsite power and to be capable of in-vessel core melt arrest. Based on a reduction in the RCIC failure rate, the benefit is estimated at about .069 person-rem averted.

19P.4.2.2 Improved Depressurization

This item would provide an improved depressurization system which would allow more reliable access to low pressure systems. Additional depressurization capability may be achieved through manually controlled, seismically protected, air powered operators which permit depressurization to be manually accomplished in the event of loss of DC control power or control air events.

The ABWR high pressure core damage events represent about 28% of the total core damage frequency, but about 46% of the offsite exposure risk. The success of manual initiation was assumed to be improved by 50% and therefore the depressurization failure rate was reduced by a factor of 2. Based on this estimate of benefit offsite person-rem is reduced by about 23% and the estimated benefit is about .042 person-rem.

19P.4.2.3 Suppression Pool Jockey Pump

This modification would provide a small makeup pump to provide low pressure decay heat removal from the Reactor Pressure Vessel (RPV) using suppression pool water as a source. The return path to the suppression pool would be through existing piping such as shutdown cooling return lines.

The benefit of this modification would be similar to that provided by the firewater injection and spray capability, but it would have the advantage that long term containment inventory concerns would not occur.

If the system could make low pressure coolant makeup systems 10% more reliable, significant reductions in CDF would not be achieved because other low pressure systems are already highly reliable. The estimated benefit is that CDF is reduced 2% and the averted risk would be .002 person-rem.

19P.4.2.4 Safety Related Condensate Storage Tank

The current ABWR design consists of a standard non-seismically qualified Condensate Storage Tank (CST). This modification would upgrade the structure of the CST such that it would be available to provide makeup to the reactor following a seismic event.

This modification only benefits the risks of core damage following seismic events. However, because the suppression pool provides an alternate suction source and the HCLPF for the suppression pool is relatively high (see Appendix 19I), the dominant failure modes are not limited by water availability. Therefore the benefit of this modification is considered small. A benefit of 0.1 person-rem averted was arbitrarily chosen for an upgraded CST.

19P.4.3 Containment Capability

The ABWR containment is designed for about 45 psig internal pressure and includes a containment rupture disc which would relieve excessive pressure if it develops during a severe accident. By providing the release point from the wetwell airspace, mitigation of releases are achieved through scrubbing of the fission products in the suppression pool.

19P.4.3.1 Larger Volume Containment

This modification would provide a larger volume containment as a means to mitigate the effects of severe accidents. By increasing the size the containment could be able to absorb additional noncondensible gas generation and delay activation of the containment rupture disc or early containment failure.

This item would mitigate the consequence of an accident by delaying the time before the severe accident source term is released and allowing more time for radioactive decay and recovery of systems. However, if recovery does not occur, eventual release is not prevented and if operation of the containment overpressure rupture disc does not occur, ultimately the containment will fail due to the long term pressurization caused by core concrete interaction and skam generation.

If sequences involving drywell head failure were eliminated (Cases 3, 6, 7, 8 and 9), the offsite risks would be reduced by about 82% and about .15 person-rem would be averted.

19P.4.3.2 Increased Containment Pressure Capacity

The design pressure of the ABWR containment is 45 psig. The containment rupture disc pressure and ultimate capability are significantly higher. By increasing the ultimate pressure capability of the containment (including seals), the effects of a severe accident could be reduced or eliminated by delaying the time of release. If the strength exceeded the maximum pressure obtainable in a severe accident, only normal containment leakage would result.

This modification would mitigate the event, not change the core damage frequency and the increased pressure capability may not be sufficient to contain the long term pressurization caused by core concrete interaction and steam generation. However, if it were able to prevent all severe source term release except for normal containment leakage, the person-rem risk would be about .02 person-rem/60 years. Therefore, the benefit would be about .16 person-rem.

19P.4.3.3 Improved Vacuum Breakers

The ABWR design contains single vacuum breaker valves in each of eight drywell to wetwell vacuum breaker lines. The PRA included failure of vacuum breakers in "Case 2" assuming operation of wetwell spray. This modification would reduce the probability of a stuck open vacuum breaker by making [illegible] redundant in each line and elimina[illegible] operator action.

If Case 2 sequences were eliminated, the benefit of this modification would be about .00003 person-rem averted.

19P.4.3.4 Improved Bottom Head Penetration Design

The ABWR design includes a 2" stainless steel drainline from the bottom of the RPV which is used to prevent thermal stratification in the RPV during operation and to provide cleanup of the bottom head by the RWCU system. A carbon steel transition piece connects the drain line to the RPV. During a severe accident this transition piece may be susceptible to melting and may provide the earliest path for release of molten core material from the RPV to the containment.

The penetrations for the fine motion control rod drives in the ABWR also may provide a pathway for release from the RPV following a severe accident. Failure of the internal blowout supports on the lower core plate, provided to eliminate the support structure in current generation BWRs, and welds of the drives at the bottom of the vessel may allow the CRDs to be partially ejected into the drywell during the severe accident which would provide a small pathway for release to the containment.

The modification is to change the transition piece material to Iconel or Stainless Steel which has a higher melting point. By so doing, additional time would be available for recovery of core cooling systems. This modification also would establish external welds or restraints on the CRDs external to the vessel so that the drives would not be ejected following failure of the internal welds. The concept

would be to make such external welds and supports small enough that the benefit is not lost from eliminating the support beams in current generation BWRs. The benefit of these modifications would be to reduce the probability of in-vessel arrest failure (NO IV). Based on consideration of the heatup rate of the bottom head, it has been estimated that making these changes could provide up to two hours additional time for recovery of systems. It is estimated, based on engineering judgement, that this time could result in the invessel arrest failure probabilities being reduced by a factor of two. The resulting benefit is about .057 person rem averted.

A potential negative aspect of the modifications is that RPV failure could occur at another unknown location such as the bottom head itself. Although the time of vessel failure would be extended, the failure mode from these other locations could be potentially more energetic and lead to unevaluated consequences.

19P.4.4 Containment Heat Removal

The ABWR design contains 3 divisions of suppression pool cooling and provisions for a containment rupture disc for decay heat removal. In addition, modifications have been made to use the RWCU heat exchangers to the maximum extent possible. Consequently, loss of containment heat removal events contribute only .1% of the total core damage frequency and offsite exposures. Additional modifications are not likely to show substantial safety benefits.

19P.4.4.1 Larger Volume Suppression Pool

This item would increase the size of the suppression pool so that the heatup rate in the pool is reduced. The increased size would allow more time for recovery of a heat removal system.

Since this modification primarily affects LHRC events (see Table 19P.2-2), the maximum benefit would be elimination of the LHRC contribution to the Case 9 sequences. These events are mitigated by the containment rupture disc and only contribute about .0002 person-rem to the base case risk. The assessed maximum benefit is therefore about .0002 person-rem.

19P.4.5 Containment Atmosphere Mass Removal

The ABWR design contains a containment rupture disc which provides containment overpressure protection from the wetwell airspace and utilizes the suppression pool scrubbing feature of the suppression pool to reduce the amount of radioactive material released. One additional modification was considered.

19P.4.5.1 Low Flow Filtered Vent

Some BWR facilities, especially in Europe, recently have added a filter system external to the containment to further reduce the magnitude of radioactive release. The systems typically use a multi-venturi scrubbing system to circulate the exhaust gas and remove particulate material. In the ABWR, because of the suppression pool scrubbing capability, a significant safety improvement is not expected due to this modification.

The release of radioactive isotopes from the ABWR following severe accidents occurs through the containment rupture disc for Cases 1, 2 and 5. These sequences total about 8% of the exposure risk. The remaining sequences involve drywell head failure or early containment failure which would not be affected by this modification. The maximum benefit of the external vent system is therefore about .014 person-rem assuming perfect initiation of the filtered containment vent system.

19P.4.6 Combustible Gas Control

No additional modifications to the ABWR were identified in this group.

19P.4.7 Containment Spray Systems

19P.4.7.1 Drywell Head Flooding

This concept would provide intentional flooding of the upper drywell head such that if high drywell temperatures occurred, the drywell head seal would not fail. Additionally, if the seal were to fail due to overpressurization of the drywell, some scrubbing of the released fission products would occur. This system would be designed to operate passively or use an AC independent water source.

If an extension of the fire pump to drywell spray crosstie were considered for manual initiation of upper head flooding, additional reduction in the high temperature containment failure sequences (Case 8) would result. Additionally, a reduction in the high consequence drywell head failure sequences (Cases 6 and 7) could be achieved. If Case 8 sequences were eliminated and Case 6 and 7 source terms were reduced to a level similar to Case 3, the conservative benefit would be .12 person-rem. The estimated benefit of this is about .06 person-rem assuming a 50% reliability of initiation.

19P.4.8 Prevention Concepts

The ABWR design contains an additional division of high pressure makeup capability to improve its capability to prevent severe accidents other features such as the fire pump injection capability and the combustion gas turbine have been included in the design to enhance the plant capability to prevent core damage. The following additional concepts were considered:

19P.4.8.1 Additional Service Water Pumps

This item addresses a reduction in the common cause dependencies through such items as improved manufacturer diversity, separation of equipment and support systems such as service water, air supplies, or heating and ventilation (HVAC). The HPCF, RCIC, and LPCF pumps are diverse in the ABWR design since they are either supplied by different manufacturers or have different flow characteristics. Equipment is separated in the ABWR design in accordance with Regulatory Guide 1.75. Thus, no further improvement is expected with regard to separation.

Common cause dependencies from support systems such as service water systems, could conceivably reduce the plant risk through an improvement in system reliability. The concept for this item would be to provide dedicated support systems for each of the four diverse injection systems identified above.

The current design provides support to these systems from one of three divisions. Thus, the effect of this change would be to include additional support systems. In addition, diversity in instrumentation which controls these systems could be included so that redundant indication and trip channels would rely on diverse instrumentation.

A 10% increase in the reliability of the four systems was assumed which is the same improvement that may be derived from improved maintenance (Subsection 19P.4.1.3). This results in an estimated benefit of about .016 person-rem.

19P.4.9 AC Power Supplies

The current ABWR electrical design is improved through application of a gas-turbine generator to augment the offsite electrical grid. The following concepts were considered for additional on-site power supplies.

19P.4.9.1 Steam Driven Turbine Generator

A steam driven turbine generator could be installed which uses reactor steam and exhausts to the suppression pool. The system would be conceptually similar to the RCIC system with the generator connected to the offsite power grid.

The benefit of this item would be similar to the addition of another gas turbine generator, but would be somewhat less due to the relative unreliability of the steam turbine compared with a diesel generator and its unavailability after the RPV is depressurized. If it were sized large enough, it could have the advantage of providing power to additional equipment.

If the system has a 80% availability for all events, the benefit is similar to an 80% reduction in the diesel generator common mode failure rate. Evaluation of the PRA indicates that the resulting benefit is about .052 person-rem.

19P.4.9.2 Alternate Pump Power Source

The ABWR provides separate diesel driven power supplies to the HPCF and LPCF pumps. Offsite power supplies the feedwater pumps. This modification would provide a small dedicated power source such as a dedicated diesel or gas turbine for the feedwater, or condensate pumps so that they do not rely on offsite power.

The benefit would be less dependence on low pressure systems during loss of offsite power events and station blackout events. If the feedwater system were made to be 90 % available during loss of offsite power events and station blackouts, the benefit would be similar to adding an additional RCIC system (refer to Section 19P.4.2.1). The resulting benefit would be about .069 person-rem.

19P.4.10 DC Power Supplies

The ABWR contains 4 DC divisions with sufficient capacity to sustain 8 hours of station blackout (with some load shedding). This represents an improvement over current operating plant designs.

19P.4.10.1 Dedicated DC Power Supply

This item addresses the use of a diverse DC power system such as an additional battery or fuel cell for the purpose of providing motive power to certain components. Conceptually a fuel cell or separate battery could be used to power a DC motor/pump combination and provide high pressure RPV injection and containment cooling. With proper starting controls such a system could be sized to provide several days capability.

Providing a separate DC powered high pressure injection capability has a benefit of further reducing the station blackout and loss of offsite power event risks which represent about 75% of the total CDF, but only a small fraction of the offsite risk. If the effective unavailability of the RCIC is reduced by a factor of 10 due to the availability of a diverse system, one benefit would be similar to adding a power supply for feedwater (19P.4.9.2) and the benefit would be about .069 person-rem.

19P.4.11 ATWS Capability

The current ABWR design provides improvements in containment heat removal and detection of ATWS events to limit the impact of this class of events. The PRA indicates that ATWS events contribute about .1% of the core damage frequency (Table 19P.2-2) and about 17% of the offsite risk (Case 9).

19P.4.11.1 ATWS Sized Vent

This modification would be available to remove reactor heat from ATWS events in addition to severe accidents and Class II events. It would be similar to the containment rupture disc (which is currently sized to pass reactor power consistent with that generated during RCIC injection), but it would be of the larger size required to pass the additional steam associated with LPCF injection. The system would need to be manually initiated.

The benefit of this venting concept is to prevent core damage and to reduce the source term available for release following ATWS events. The evaluation shows that an ATWS sized vent manually initiated with a 100% reliability would have a maximum benefit of reducing the offsite dose by about .03 person-rem by reassigning the consequences from case 9 to case 1.

19P.4.12 Seismic Capability

The current ABWR is designed for a Safe Shutdown Earthquake of .3g acceleration. The seismic margins analysis (see Appendix 19I) addresses the margins associated with the seismic design and concludes that there is a 95% confidence that existing equipment has less than a 5% probability of failure at twice the SSE level. This capability is considered adequate for the ABWR design and no additional changes are considered.

19P.4.13 System Simplification

This item is intended to address system simplification by the elimination of unnecessary interlocks, automatic initiation of manual actions or redundancy as a means to reduce overall plant risk. Elimination of seismic and pipe whip restraints is included in the concept.

While there are several examples of redundant systems, valves and features on the ABWR design which could conceivably be simplified, there are several areas in which the ABWR design already has been improved and simplified, especially in the area of controls and logic. System interactions during accidents were included in this category. One area was identified in which simple modification of an existing system could provide some benefit.

19P.4.13.1 Reactor Building Sprays

This concept would use the firewater sprays in the reactor building to mitigate releases of fission products into the reactor building following an accident. The concept would require additional valving and nozzles, separate from the fire protection fusible links, to spray in areas vulnerable to release, such as near the containment overpressure relief line routing.

The benefit of this modification could be to reduce the impact of events which do not involve the operation of the containment rupture disk. Such events release fission products from the containment into the reactor building. Releases from normal containment leakage and cases 3, 6, 7, 8 and case 9 sequences could potentially be reduced. If 10% of these releases from these cases were arbitrarily mitigated by this method, the benefit would be about .017 person-rem.

19P.4.14 Core Retention Devices

Core retention features are incorporated into the ABWR Design. As discussed in Appendix 19E.2, if a severe accident has resulted in a loss of RPV integrity, accident management guidance specifies that drywell sprays be initiated which will cause the suppression pool to overflow into the lower drywell after a few hours and quench the debris bed. After the molten core has been quenched, no further ablation of concrete is expected and the decay heat can be removed by normal containment cooling methods such as suppression pool cooling. If sprays can not be initiated, the Lower Drywell Flooder System described in Section 9.5.12 cools a debris bed by flooding over the molten core in the lower drywell with water from the suppression pool. This system is similar to the "Post Accident Flooding" concept included in Reference 19P.8.4. One additional concept from Reference 19P.8.4 is included.

19P.4.14.1 Flooded Rubble Bed

This concept consists of a bed of refactory pebbles which fill the lower drywell cavity and are flooded with water. The bed impedes the flow of molten corium and increases the available heat transfer area which enhances debris coolability. The use of thoria (ThO_2) pellets in a multiple layer geometry has been shown to stop melt penetration; thus, preventing core-concrete interaction. Drawbacks to using thorium dioxide include cost, toxicity, and the radiological impact of radon gas release into the lower drywell via the radioactive decay of thorium. Other refractories such as alumina slow corium penetration but may fail to stop core-concrete contact. Other refractories may be susceptible to chemical attack by the corium and may melt at lower temperatures. Pebbles composed of refractories other than thoria also may be susceptible to floating because they have lower density than the corium. A major drawback common to all flooded rubble bed core retention systems is the need for further experimental testing in order to validate the concept in BWR applications.

The benefit of this modification lies in the potential elimination of core-concrete interaction and a corresponding decrease in non-condensable gas generation. Appendix 19E, Attachment C indicates a 90% certainty that debris on a concrete floor covered with water will be coolable in the current ABWR design.

Only sequences in which no liquid injection to the drywell occurs will result in core-concrete interaction. A conservative estimate of the benefit of this concept over the existing design would be elimination of sequences with core-concrete interaction except those with containment cooling failure. A review of Appendix 19E.2 indicates that this would effect about 1% of Cases 1, 6 and 7.. This corresponds to about .001 person-rem averted.

Table 19P.4-1

SUMMARY OF BENEFITS

Modification	Potential Manrem Averted
1 ACCIDENT MANAGEMENT	
1a. Severe Accident EPGs/AMGs	0.015
1b. Computer Aided Instrumentation	0.010
2 DECAY HEAT REMOVAL	
2a. Passive High Pressure System	0.138
2b. Improved Depressurization	0.042
2c. Suppression Pool Jockey Pump	0.002
2d. Safety Related Condensate Storage Tank	0.01
3 CONTAINMENT CAPABILITY	
3a. Larger Volume Containment	0.15
3b. Increased Containment Pressure Capacity	0.02
3c. Improved Vacuum Breakers	0.00003
3d. Improved Bottom Penetration Design	0.057
4 CONTAINMENT HEAT REMOVAL	
4a. Larger Volume Suppression Pool	0.0002
5 CONTAINMENT ATMOSPHERE GAS REMOVAL	
5A. Low Flow Filtered Vent	0.014
7 CONTAINMENT SPRAY SYSTEMS	
7a. Drywell Head Flooding	0.060
8 PREVENTION CONCEPTS	
8a. Additional Service Water Pump	0.016
9 AC POWER SUPPLIES	
9a. Steam Driven Turbine Generator	0.052
9b. Alternate Pump Power Source	0.069
10 DC POWER SUPPLIES	
10a. Dedicated DC Power Supply	0.069
11 ATWS CAPABILITY	
11a. ATWS Sized Vent	0.03
13 SYSTEM SIMPLIFICATION	
13a. Reactor Building Sprays	0.017
14 CORE RETENTION DEVICE	
14a. Flooded Rubble Bed	0.001

SECTION 19P.5

CONTENTS

SECTION 19P.5

CONTENTS (Continued)

TABLES

19P.5 COST IMPACTS OF POTENTIAL MODIFICATIONS

As discussed in Section 19P.1.3.1, rough order of magnitude costs were assigned to each modification based on the costs of systems determined by GE. These costs represent the incremental costs that would be incurred in a new plant rather than costs that would apply on a backfit basis. Credit for the onsite costs averted by the modification are discussed in Section 19P.1.3.2. For each modification which reduces the core damage frequency an estimate of the impact was made and then applied to the potential averted offsite cost. This section summarizes the cost basis for each of the modification evaluated in Section 19P.4. This basis is generally the cost estimate less the credit for onsite averted costs. Table 19P.5-1 summarizes the results.

The costs were biased on the low side, but all known or reasonably expected costs were accounted for in order that a reasonable assessment of the minimum cost would be obtained. Actual plant costs are expected to be higher than indicated in tnis evaluation. All costs are referenced to 1991 U.S. dollars based on changes in the Consumer Price Index.

19P.5.1 Accident Management

19P.5.1.1 Severe Accident EPGs/AMGs

The cost of extending the EPGs would be largely a one-time cost which should be prorated over several plants if accomplished by the BWROG. Current industry activity is addressing this as part of Accident Management Guidelines (AMG). If plant specific, symptom based, severe accident emergency procedures were to be prepared based on AMGs, the cost would be at least $ 600,000 for plant specific modifications to EOPs.

19P.5.1.2 Computer Aided Instrumentation

Additional software and development costs associated with modifying existing Safety Plant Display Systems are estimated to cost at least $600,000 for a new plant. This estimate is based on assumed additions of isolation devices to transmit data to the computer and in-plant wiring. Because this modification reduces the frequency of core damage events, a present worth of $400 onsite costs are averted and the cost basis is $599,600.

19P.5.1.3 Improved Maintenance Procedures/Manuals

The cost of at least $ 300,000 would be required to identify components which should receive enhanced maintenance attention and to prepare the additional detailed procedures or recommended information beyond that currently planned. Credit for reduction in onsite costs reduces the cost basis to $299,000.

19P.5.2 Decay Heat Removal

19P.5.2.1 Passive High Pressure System

The cost of an additional high pressure system for core cooling would be extensive since it would not only require additional system hardware which would cost at least $ 1,200,000, but it would also require additional building costs for space available for the system. Assuming the system could be located in the reactor building without increasing its height, building costs are estimated to be another $550,000. The credit for averted onsite costs is about $6,000 which brings the cost basis to $1,744,000.

19P.5.2.2 Improved Depressurization

The cost of the additional logic changes, pneumatic supplies, piping and qualification was estimated for the GESSAR design (reference 19P.8.1). A similar cost would be expected for the ABWR design. The cost is estimated to be at least $600,000 for an improved system for depressurization. This estimate assumes no building space increase for the added equipment. The credit for averted onsite costs was evaluated to be $1,400 which makes the cost basis $598,600.

19P.5.2.3 Suppression Pool Jockey Pump

The cost of an additional small pump and associated piping is estimated at more than $ 60,000 including installation of the equipment. It is assumed that increases in power supply capacity and building space are not required. Controls and associated wiring could cost an additional $ 60,000 for a total cost of at least $120,000. A credit of $200 for averted onsite costs makes the cost basis $119,800.

19P.5.2.4 Safety Related Condensate Storage Tank

Estimating the cost of upgrading the CST structure to withstand seismic events requires a detailed structural analysis and resultant material. It is judged that the final cost increase would be in excess of $ 1,000,000. No credit for onsite cost averted was assumed for this modification.

19P.5.3 Containment Capability

19P.5.3.1 Larger Volume Containment

Doubling the containment volume requires an increase in the concrete and rebar. If structural costs of the containment can be made for $ 1,200 per square foot, doub ng the containment volume without increasing its height, the cost would be at least $ 8,000,000. This estimate does not include reanalysis and other documentation costs. Since this modification is mitigative, no credit for onsite averted costs was assumed.

19P.5.3.2 Increased Containment Pressure Capacity

The cost of a stronger containment design would be similar in magnitude to increasing its size (see Subsection 19P.5.3.1). If the costs are primarily due to denser rebar required during installation and additional analysis, an estimate of at least $12,000,000 could be required. Since this modification is mitigative, no credit for onsite averted costs was assumed.

19P.5.3.3 Improved Vacuum Breakers

The cost of redundant vacuum breakers including installation and hardware is estimated at more than $10,000 per line. Instrumentation associated with this modification is not included. For the eight lines the cost of this modification is more than $100,000. Since this modification is mitigative, no credit for onsite averted costs was assumed.

19P.5.3.4 Improved Bottom Penetration Design

The cost increase of using a stainless or inconel transition piece as opposed to carbon steel would be expected to be small in comparison to the engineering and documentation change costs associated with the change. Costs, associated with external welds and support for the CRDs is judged to be at least $1000 per drive. In addition about $500,000 of analysis would be required to develop the changes. This would dominate the cost of this modification when applied to all 205 drives. Such changes are estimated to be at least $750,000.

Since this modification is mitigative, no credit for averted onsite costs applies.

19P.5.4 Containment Heat Removal

19P.5.4.1 Larger Volume Suppression Pool

This concept would result in similar costs as item Subsection 19P.5.3.1 for providing a larger containment. An estimate of $ 8,000,000 is assigned to this item.

19P.5.5 Containment Atmosphere Mass Removal

19P.5.5.1 Low Flow Filtered Vent

The cost of added equipment associated with the FILTRA system (excluding a test program) was estimated to be about $5,000,000 in Reference 19P.8.4. Although a detailed estimate was not prepared for the ABWR, an estimate of $3,000,000 has been assumed for the purpose of this evaluation.

Since this modification is mitigative, no credit for averted onsite costs applies.

19P.5.6 Combustible Gas Control

No additional modifications to the ABWR were identified in this group.

19P.5.7 Containment Spray Systems

19P.5.7.1 Drywell Head Flooding

An additional line to flood the drywell head using existing firewater piping would be a relatively inexpensive addition to the current system. Instrumentation and controls to permit manual control from the control room would be needed. It is estimated that the total modification cost would be at least $ 100,000 for the engineering, piping, valves and cabling.

Because this modification is mitigative, no credit for averted onsite costs has been applied.

19P.5.8 Prevention Concepts

19P.5.8.1 Additional Service Water Pump

The use of diverse instrumentation would not presumably have a significant equipment cost, but there would be an increased cost of maintenance and spare parts due to less interchangeability and less standardization of procedures.

These costs, however, are probably low in comparison with the extra support systems for air supply and service water. Equipment, power supplies and structural changes to include these new systems are estimated to cost at least $ 6,000,000. A small credit for averted onsite costs makes the cost basis for this item $5,999,000, based on the benefits discussed in Subsections 19P.4.1.3 and 19P.5.1.3.

19P.5.9 AC Power Supplies

19P.5.9.1 Steam Driven Turbine Generator

The cost of the system should be similar to that for the RCIC system, but additional cost would be needed for structural changes to the reactor building plus the generator and its controls. This item is expected to cost at least $ 6,000,000.

With credit for averted onsite costs, the cost basis for this item becomes $ 5,994,300.

19P.5.9.2 Alternate Pump Power Source

A typical feedwater pump for an ABWR sized plant could require a 4000 Kwe sized generator, at $300 per Kwe, a separate diesel generator and the supporting auxiliaries could cost at least $ 1,200,000. This cost would include wiring and installation of the alternate generator, but does not assume additional structural costs.

With credit for averted onsite costs, the cost basis for this item becomes $ 1,194,000.

19P.5.10 DC Power Supplies

19P.5.10.1 Dedicated DC Power Supply

Fuel cells are largely a developmental technology, at least in the large size range required for this application. In addition the process involves some risk of fire. To address these concerns a cost of at least $ 6,000,000 would be expected.

A separate battery would be less expensive than fuel cells, but would involve additional space requirements which could make this modification more expensive than adding a diesel generator as discussed in Subsection 19P.5.9.2. A battery bank capable of supplying 400 Kwe would be about 50 times larger in capacity than the emergency batteries. This number of batteries would require at least 5,000 square feet of space, assuming extensive stacking and without concern for seismic response. At $500/square feet construction cost, the additional space required would amount to $2,500,000 for this modification. Additional costs would be required for DC pumps, cabling and instrumentation and controllers. A total cost would be at least $3,000,000.

19P.5.11 ATWS Capability

19P.5.11.1 ATWS Sized Vent

Larger piping and additional training would be required to extend the existing rupture disk feature to be available during an ATWS event. Additional instrumentation and cabling would be required to make the vent operable from the control room. It is estimated that the incremental cost would be at least $ 300,000.

19P.5.12 Seismic Capability

No modifications were considered for this group.

19P.5.13 System Simplification

19P.5.13.1 Reactor Building Sprays

The cost of this modification is judged to be similar to the concept of drywell head flooding (Subsection 19P.5.5.1) if it only involves piping and valves which are tied into the firewater system. An estimate of $ 100,000 has been assigned to this item.

Onsite cleanup costs also could be affected by this modification. If the cleanup costs were eliminated an averted cost would conservatively be about $ 5,000.

19P.5.14 Core Retention Devices

19P.5.14.1 Flooded Rubble Bed

Reference 19P.8.4 estimated that the refractory material needed for this modification would cost approximately $ 1,000 per pound. If the lower drywell were filled with about 1.5 feet of this material, which would remain well below the service platform, at least 1250 cubic feet of material would be required. If it weighs 15 pounds/cubic feet, the material cost alone would amount to $ 18,750,000.

Table 19P.5-1

SUMMARY OF COSTS

Modification	Estimated Cost
1 ACCIDENT MANAGEMENT	
1a. Severe Accident EPGs	$ 600.0K
1b. Computer Aided Instrumentation	$ 599.6K
1c. Improved Maintenance Procedures/Manuals	$ 299.0K
2 DECAY HEAT REMOVAL	
2a. Passive High Pressure System	$ 1744.0K
2b. Improved Depressurization	$ 598.6K
2c. Suppression Pool Jockey Pump	$ 120.0K
2d. Safety Related Condensate Storage Tank	$ 1000.0K
3 CONTAINMENT CAPABILITY	
3a. Larger Volume Containment	$ 8000.0K
3b. Increased Containment Pressure Capacity	$12000.0K
3c. Improved Vacuum Breakers	$ 100.0K
3d. Improved Bottom Penetration Design	$ 750.0K
4 CONTAINMENT HEAT REMOVAL	
4a. Larger Volume Suppression Pool	$ 8000.0K
5 CONTAINMENT ATMOSPHERE GAS REMOVAL	
5a. Filtered Containment Vent	$ 3000.0K
7 CONTAINMENT SPRAY SYSTEMS	
7a. Drywell Head Flooding	$ 100.0K
8 PREVENTION CONCEPTS	
8a. Additional Service Water Pump	$ 5999.0K
9 AC POWER SUPPLIES	
9a. Steam Driven Turbine Generator	$ 5994.3K
9b. Alternate Pump Power Source	$ 1194.0K
10 DC POWER SUPPLIES	
10a. Dedicated RHR DC Power Supply	$ 3000.0K
11 ATWS CAPABILITY	
11a. ATWS Sized Vent	$ 300.0K
13 SYSTEM SIMPLIFICATION	
13a. Reactor Building Sprays	$ 100.0K
14 CORE RETENTION DEVICES	
14a. Flooded Rubble Bed	$18,750.0K

SECTION 19P.6

TABLES

19P.6 EVALUATION OF POTENTIAL MODIFICATIONS

A ranking of the modifications by $/person-rem averted is shown in Table 19P.6-1 based on the results and estimates provided in Sections 19P.4 and 19P.5.

The lowest cost/person rem averted modification is more than 1600 times the target criteria of $ 1,000 per person rem averted. Clearly none of the modifications is justifiable on the basis of costs for person-rem averted. This can be attributed to the low probability of core damage in the ABWR with the modifications to reduce risk already installed.

Table 19P.6-1

Summary of Results

Modification		Cost/Person-rem Averted	
7a.	Drywell Head Flooding	$	1,667K
13a.	Reactor Building Sprays	$	5,882K
11a.	ATWS Sized Vent	$	9,882K
2d.	Safety Related Condensate Storage Tank	$	10,000K
3d.	Improved Bottom Penetration Design	$	13,158K
2b.	Improved Depressurization	$	14,173K
9b.	Alternate Pump Power Source	$	17,351K
1c.	Improved Maintenance Procedures/Manuals	$	18,298K
2a.	Passive High Pressure System	$	25,158K
1a.	Severe Accident EPGs	$	39,834K
10a.	Dedicated DC Power Supply	$	43,478K
3a.	Larger Volume Containment	$	53,333K
1b.	Computer Aided Instrumentation	$	60,000K
2c.	Suppression Pool Jockey Pump	$	62,130K
9a.	Steam Driven Turbine Generator	$	114,726K
5a.	Low Flow Filtered Vent	$	214,286K
8a.	Additional Service Water Pump	$	375,000K
4b.	RWCU Decay Heat Removal	$	425,000K
3b.	Increased Containment Pressure Capacity	$	600,000K
3c.	Improved Vacuum Breakers	$	3,329,000K
14a.	Flooded Rubble Bed	$	18,750,000K
4a.	Larger Volume Suppression Pool	$	40,000,000K

19P.7 SUMMARY OF CONCLUSIONS

Potentially attractive modifications were identified from previous evaluations of potential prevention and mitigation concepts applicable during severe accidents and discussion with the NRC staff. Potential modifications were reviewed to select those which are applicable to the ABWR design and which have not already been implemented in the design. Of these modifications, twenty one were selected for additional review.

The low level of risk in the ABWR is demonstrated by the total 60 year offsite exposure risk of .183 person-rem. At this level only modifications which cost less than $ 183 can be justified. Based on this low level no modifications are justified for the ABWR.

Based on the PRA results, none of the modifications provided a substantial improvement in plant safety.

19P.8 REFERENCES

1. Evaluation of Proposed Modifications to the GESSAR II Design, NEDE 30640, Class III, June 1984.

2. Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2, NUREG-0974 Supplement, August 16, 1989

3. Issuance of Supplement to the Final Environmental Statement- Comanche Peak Steam Electric Station, Units 1 and 2, NUREG 0775 Supplement, December 15, 1989

4. Survey of the State of the Art in Mitigation Systems, NUREG/CR 3908, R&D Associates, December 1985

5. Assessment of Severe Accident Prevention and Mitigation Features, NUREG/CR 4920, Brookhaven National Laboratory, July 1988.

6. Design and Feasibility of Accident Mitigation Systems for Light Water Reactors, NUREG/CR 4025, R&D Associates, August 1985

7. Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG 1150, January 1991.

8. Technical Guidance for Siting Criteria Development, NUREG/CR 2239, Sandia National Laboratories, December 1982.