52-001

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CEB92-54

Tue, Nov 3, 1992

S07-1132
11 H3

To: Chet Poslusny, NRC
Bob Palla, NRC
Tony D'Angelo

From: Carol E. Buchholz CEB

Subject: Response to Questions from September 29 - October 1 Meeting

The following issues were raised in the discussions we had during the subject meetings. I have not yet been able to address all of the issues that were raised but I wanted to get this information out to you. I hope it is useful in resolving any questions you may have as you write the SER.

Issue 1:

Confirm the type of concrete used in the pedestal.

Response 1:

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 considered in the uncertainty analysis for core concrete interaction which was developed in CEB-92-

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X. 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 the analysis presented in CEB-92-X 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 in CEB-92-X. 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.

Issue 2:

Confirm mass of material used in core concrete interaction calculations.

Response 2:

Plant Design Data:

> Mass UO2 = 171600 kg
>
> Mass Zr = 7?550 kg (cans and clad)

Parameters computed/input in MAAP:

> Mass UO2 = 171600 kg
>
> Mass Zr = 72575 kg

The total UO2+Zr mass is closer to 244,000 kg in the MAAP analyses than the UO2+Zr+Steel mass as discussed in the meeting. The amount of steel that MAAP adds from the RPV is approximately 20,000 kg which represents the mass of the lower core plate. Due to the prompt penetration failure, there is not sufficient time to melt the lower plenum steel (i.e. CRD housings). In addition, a significant amount of iron is added as a result of the rebar in the concrete. The results of the FMX1P calculation indicate approximately 2000 kg/hr of iron added to the melt.

Issue 3:

Determine the time of Zr depletion for the bounding sequence for the core concrete interaction sequence.

Response 3:

For case FMX1P, the Zr is completely depleted by about 20,000 seconds into the event. The plots provided show that the onset of CO production is coincident with the depletion of Zr. During the early stages of CCI, the following exothermic reactions are depleting the Zr mass:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

$$Zr + CO_2 \rightarrow ZrO_2 + C$$

Therefore, the total hydrogen gas generation is not equivalent to the gas which would be generated from a 100% metal water reaction even though the Zr is depleted.

Issue 4:

Provide additional justification for the use of 100 kW/m^2 as a limiting value for the upward heat transfer limit.

Response 4:

The heat transfer between the water and the debris can be limited by:

a) Conduction within the debris,

b) Critical heat flux, or

c) 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 a credible limit to 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:

$$C_1 = 0$$

$$C_1 2 = 1550K$$

$$T(\delta_{lim}) \approx 450K$$

where: δ_{lim} = debris thickness.

Substituting into Equation (2), we have for the limiting debris thickness for coolability:

$$\delta_{lim} = 0.08m$$

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:

$$q'' = q'''\delta_{lim} = 100MW / m^2$$

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.

Issue 5:

Provide a discussion of the susceptability of the RIPS to failure as a result of contact with molten core material during a severe accident.

Response 5:

See ABWR SSAR Figure 5.4-2 for a pictoral description of the location of the RIPS in the RPV. Figure 5.4-1 shows more RIP detail.

Since the core melt progression is expected to contain the corium inside the core shroud, debris would not approach the RIP impellers or RPV RIP nozzles which are located outside the shroud. However, if the shroud is perforated by the corium, the corium might then enter the top of RIP impellers and possibly enter the stretch tube/shaft annulus. This is extremely unlikely since this annulus thickness decreases in the downward direction to 1.5mm (The variance between the 215mm diameter RIP shaft and the 218mm inside diameter of the stationary stretch tube.). Any molten material relocating through the RIP would quickly freeze or flow through the pump rather than flowing along the pump shaft.

In the event that 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. ABWR SSAR Figure 1.2-3b shows the two RIP vertical restraints which connect the bottom of each RIP motor housing to the RPV bottom head. These restraints prevent the RIP/motor from dropping out of the RPV in case the motor housing weld fails for any reason. Therefore, in the exceedingly unlikely event of RIP failure, the pump will not fall from the vessel, and the penetration through the vessel would be small.

Nevertheless, the corium is expected to freeze and, consequently, not flow down the annulus into the motor housing. Therefore, the RPV RIP nozzle motor housing Reactor Coolant Pressure Boundary would not be breached.

Issue 6:

Indicate the impact of suppression pool bypass on the probability of recovery for the RHR system.

Response 6:

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 1 (see attached). 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 differenct 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.

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 standard exponential non-recovery formula for systems (Section 19D.5.4):

$$P_f = e^{(-T/MTTR)}$$

where: P_f = Probability of failure to recover system (for one division),

T = Time

MTTR = Mean Time to Repair (19 hours)

For the three redundant divisions of RHR and considering potential common cause failures and limitations on the availability of operators, the probability of failure to recover 1 of 3 RHR divisions was estimated using:

$$P_{f(1/3)} = 0.5P_f$$

Hence, with T = 5 hours and MTTR = 19 hours:

$$P_{f(1/3)} = 0.4$$

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 byp... by adding an additional term to account for sequences which would other...e 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. Noting that the total pool bypass probability (for large and small bypass events) is 0.022 and assuming all pool bypass events would decrease the recovery time to 5 hours, the adjusted RHR non-recovery probability is estimated as:

$$P_{f-RHR-PB} = P_{f-RHR} + (1 - P_{f-RHR}) \times 0.022 \times 0.4$$

where: P_{f-RHR} = the existing non-recovery probabilities shown in the CETs and in Table 1,

$P_{f-RHR-PB}$ = the modified non-recovery probabilities,

0.022 = the probability of pool bypass,

0.4 = the non-recovery probability for sequences with bypass.

The modified RHR non-recovery probabilities are shown on Table 1 for comparison with the existing values.

<u>Sequences with In-vessel Core Damage Mitigation</u>

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:

$$3.87E-12 \times 0.05 = 1.9E-13$$

which is negligible.

The probability of class IIID sequences with core melt arrest in-vessel is:

$$2.1E-10 \times 0.9 = 1.89E-10$$

Class IB1

	Existing <u>Results</u>	Modified <u>Results</u>

No Cont Leak	2.11E-10	2.08E-10
RD Open	4.43E-11	4.66E-11
DW Head Fail	8.94E-13	9.40E-13

Class ID

	Existing Results	Modified Results
No Cont Leak	5.07E-11	5.01E-11
RD Open	1.82E-11	1.88E-11
DW Head Fail	3.67E-13	3.79E-13

Conclusions

The release category frequencies were modified to account for the changes in IB1 and ID sequences. The overall frequency of COPS operation is 2.08E-8 and that for late drywell head structural failure is 5.25E-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.

Issue 7:

Indicate the number of alloy blends which will be used in the passive flooder system.

Response 7:

The passive flooder contains a fusible plug which is a blend of several different metallic alloys. A blend is required to obtain a plug melting temperature of approximately 260 C. A single blend will be used. No need exists for multiple blends because sample valves will be tested before installation and during periodic refueling outages to ensure that the desired melting temperature is achieved. The sentence in CEB92-X which erroneously referred to multiple blends will be removed.

Issue 8:

Confirm that there are no openings in the support skirt which could allow water to flow from the upper drywell to the lower drywell.

Response 8:

There are no penetrations in the ABWR RPV support skirt.

Issue 9:

Discuss the use of the firewater spray to provide flooding of the lower drywell. In particular, explain the potential for the operator to use the firewater in drywell spray mode as opposed to vessel injection mode.

Response 9:

The primary goal of the ac-independent water addition system (firewater addition system) is to provide cooling water to the vessel in the event that all other means of vessel injection are inoperable. If the vessel has failed, water added to the vessel will drain into the lower drywell to allow for the quenching of the debris. If not required to assure adequate core cooling, the ac-independent water addition system may be operated in drywell spray mode to provide primary containment control. Drywell sprays can be used to reduce high drywell temperature, reduce high suppression chamber pressure, and provide steam inerting in the event of high containment hydrogen concentration. The only time the ac-independent water addition system will be used for primary containment protection when adequate core cooling is not assured is when containment failure is likely to occur without immediate corrective action.

This operational philosophy of the ac-independent water additional system is discussed in the Emergency Procedure Guidelines (EPGs) contained in the ABWR SSAR Appendix 18A. The text in the EPGs will be modified slightly to ensure that the ac-independent water system is not removed from the service of assuring adequate core cooling unless containment failure is likely without immediate corrective action.

Issue 10:

Provide justification for the probability of power recovery in the interval from 8 to 10 hours in the IB-2 sequences.

Response 10:

The conditional probability of failing to recover offsite power in the two-hour interval from 8 to 10 hours is obtained from the EPRI KAG 9(7/88 draft) Table D2-2 (see attached). A value of 0.6 is obtained by dividing the non-recovery probabilities for 8 and 10 hours.

Issue 11:

Provide additional information to indicate that the use of the firewater addition system represents the most probable mechanism for the flooding of the lower drywell.

Response 11:

The following text will be added to the discussion of the passive flooder system sequences in the Appendix 19E of the SSAR. Similar discussions will be added to the summary discussions in Chapter 19.

> The passive flooder system is designed to cause the lower drywell to be flooded when there is no water overlying core debris in the lower drywell. If there is no overlying water pool the fusible material in the valve will heat up, and melt the fusible plug. If there is water overlying the debris pool, the lower drywell will not heat up sufficiently to cause the passive flooder to open. Examination of the Containment Event Trees (Subsection 19D.5.11) shows that the firewater addition system is expected to operate in most of the accident sequences. Therefore, the passive flooder is not needed in the majority of accidents. Rather, the lower drywell flooder is viewed as a passive backup system which floods the lower drywell, in order to keep the temperature in the drywell low, and in order to allow quenching of the core debris.

__Issue 12:__

Confirm that the teflon disk is designed to slide out of the passive flooder rather than melting If necessary, consider the potential for the teflon disk to become stuck in the exit of the flooder line.

Response 12:

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 ^{o}C and a maximum continuous operating temperature of 288 ^{o}C both of which are above the plug melting temperature of 260 ^{o}C. Furthermore, teflon has high chemical resistance and will not adhere to the stainless steel plug nor the fusible plug.

__Issue 13:__

Provide information about the pressurization of the containment in the event of SRV discharge line failure. Note this was DCH Question 2 of the Staff's earlier letter:

> Provide justification that the reactor depressurization system is highly reliable during seismic events, and will assure a very low absolute frequency of high pressure reactor vessel failures in seismic events. This should include discussion of: (1) the impact of SRV discharge pipe failures on the ability to depressurize (indicated to be a concern in draft section 19E.2.3.3.4), and (2) quantitative estimates of the availability of wetwell sprays in these events.

TABLE 1

Accident Subclass	Core Melt Arrest in RPV	Active Injection Lower DW	Existing RHR Non-Recovery Probability (P_{f-RHR})	Modified RHR Non-Recovery Probability $(P_{f-RHR-PR})$
IA_1, IIIA_1	CM ARREST		0.05	0.06
	NO ARREST	INJECT	0.01	0.02
	NO ARREST	NO INJECT	0.1	0.11
IB1_0	CM ARREST		0.0	0.01
	NO ARREST	INJECT	0.01	0.02
	NO ARREST	NO INJECT	0.1	0.11
IB2_0	CM ARREST		0.05	0.06
	NO ARREST	INJECT	0.1	0.11
	NO ARREST	NO INJECT	0.1	0.11
IB3_0	CM ARREST		0.0	0.01
	NO ARREST	INJECT	0.05	0.06
	NO ARREST	NO INJECT	0.1	0.11
ID, IIID	CM ARREST		0.2	0.21
	NO ARREST	INJECT	[illegible]	0.21
	NO ARREST	NO INJECT	0.2	0.21

Response 13:

In the unlikely event failure of the discharge piping does occur, the ability of the reactor to depressurize will not be affected. The ADS system would automatically initiate when the water level in the vessel reacned Level 1. There are no directions to the operator which would direct the ADS to be inhibited. Rather, failure of the discharge piping would lead to a LOCA-type condition except that the break would be downstream of the SRV.

The issue addressed in part (2) of this question will be discussed in a later submittal.

Issue 14:

Investigate the difference between the consequence results calculated by GE and BNL.

Response 14:

Hal Careway at GE and Art Tingle at BNL have been in contact. In their discussions it was determined that the discrepancy in the consequence results appears to be the result of differences in the MACCS and CRAC codes. BNL ran a supplemental CRAC calculation which essentially duplicated the results of GE's evaluation. BNL is now investigating the differences between their MACCS and CRAC results. GE will support further dialog as required to resolve this issue.

Issue 15:

Provide additional information to support the analysis of the pedestal integrity if concrete ablation has occured.

Response 15:

Discussions were held between Gary Ehlert and Gutam Bagchi. In that meeting it was agreed that Gutam would look further at the GE submittals to date and indicate to GE if further information was required. GE will support further information and dialog as required to support this issue.

Table D2-2

CUMULATIVE NON-RECOVERY PROBABILITIES

Time (hr)	Probability of not recovering power	Time (hr)	Probability of not recovering power
0.5	0.48	13	7.3×10^{-3}
1	0.36	14	5.3×10^{-3}
2	0.22	15	3.9×10^{-3}
3	J.14	16	2.8×10^{-3}
4	0.11	17	2.1×10^{-3}
5	0.079	18	1.5×10^{-3}
6	0.064	19	1.1×10^{-3}
7	0.050	20	8.0×10^{-4}
8	0.036	21	5.9×10^{-4}
9	0.021	22	4.3×10^{-4}
10	0.021	23	3.1×10^{-4}
11	0.012	24	2.3×10^{-4}
12	0.010		

- Data estimated from licensee-event reports, and reported in NUREG/CR-1363[5] for valves, NUREG/CR-1205[6] for pumps, and NUREG/CR-1362[7] for diesel generators;
- Additional data compiled for diesel generators and reported in NUREG/CR-2989;[8]
- The data for diesel generators reported in NSAC/108;[9]
- The data base compiled for the Accident Sequence Evaluation Program,[10] which is based largely on data from the Reactor Safety Study;[11]
- The data provided for the Northeast Utilities system, as reported in the draft version of the ALWR PRA Key Assumptions and Groundrules Document;[12]
- Military data for non-nuclear installations reported in NPRD-2;[13] and
- The data for some electrical components and instrumentation reported in IEEE-500.[14]

In addition, raw data were extracted from available sources for several specific plants. These sources included the following:

- The plant-specific experience summarized in the Oconee PRA,[3]
- The data raported for Indian Point Units 2 and 3 in the Indian Point PSS,[15]
- The e‹.dence for Zion in the Zion PSS,[16]

19E DETERMINISTIC EVALUATIONS

.1 INTRODUCTION

.2 DETERMINISTIC ANALYSIS OF PLANT PERFORMANCE

2.1 METHODS AND ASSUMPTIONS

(as originally submitted)

2.1.1 Code Description

2.1.1.1 MAAP3.0B
2.1.1.2 ABWR Modifications

2.1.2 ABWR Configuration Basis

2.1.2.1 ABWR Configuration Assumptions
2.1.2.2 Station Blackout Performance

2.1.2.2.1 Summary
2.1.2.2.2 Core Cooling
2.1.2.2.3 Primary Containment Vessel (PCV) Integrity
2.1.2.2.4 Operator Actions
2.1.2.2.5 Recovery Following Restoration of AC Power
2.1.2.2.6 Conclusions

2.1.3 Phenomenological Assumptions

2.1.3.1 Steam Explosions
2.1.3.2 Degree of Metal-Water Reaction
2.1.3.3 Suppression Pool Bypass due to Additional Failures
2.1.3.4 Effect of RHR Heat Exchanger Failure in a Seismic Event
2.1.3.5 Radiation Heating of the Equipment Tunnel
2.1.3.6 Basemat Penetration
2.1.3.7 Hydrogen Burning and Explosions

2.1.4 Definition of Base Case Assumptions

2.1.4.1 Core Melt Progression and Hydrogen Generation
2.1.4.2 In-Vessel Recovery
2.1.4.3 System Recovery After Vessel Failure and Normal Containment Leakage
2.1.4.4 Early Drywell Head Failure
2.1.4.5 Consequences of Suppression Pool Drain
2.1.4.6 Stuck Open Vacuum Breaker
2.1.4.7 Containment Structural Failure Pressure
2.1.4.8 Overpressure Relief Rupture Disk

2.2 ACCIDENT SEQUENCES

(Submitted in CEB92-39 on June 30)

2.2.1 LCLP
2.2.2 LCHP
2.2.3 SBRC
2.2.4 LHRC
2.2.5 LBLC
2.2.6 NSCL
2.2.7 NSCH
2.2.8 NSRC
2.2.9 Summary

2.3 JUSTIFICATION OF PHENOMENOLOGICAL ASSUMPTIONS

2.3.1 Steam Explosions

(as originally submitted)

2.3.1.1 The Steam Explosion Process
2.3.1.2 Previous Studies
2.3.1.3 Theoretical Considerations
2.3.1.4 Application to ABWR

2.3.2 100% Metal-Water Reaction

(as originally submitted)

2.3.3 Suppression Pool Bypass Paths

(Don Knecht's work)

2.3.3.1 Introduction
2.3.3.2 Identification and Description of Suppression Pool Bypass Pathways
2.3.3.3 Evaluation of Bypass Probability
2.3.3.4 Suppression Pool Bypass Resulting from External Event Analysis

2.3.4 Effect of RHR Heat Exchanger Failure in a Seismic Event

(as originally submitted)

2.3.4.1 RHR Equipment Room Flooding
2.3.4.2 Dynamic Loads Induced by Chugging
2.3.4.3 RHR Equipment Room Structural Integrity

2.3.5 Potential for Flashing During Venting

(submitted in CEB92-12 on April 2)

2.3.5.1 Critical Time Constants for Blowdown Response
2.3.5.2 Pool Swell

2.3.5.2.1 Pool Swell due to Suppression Pool Flashing
2.3.5.2.2 Pool Swell due to Flow From Drywell
2.3.5.2.3 Steam Source
2.3.5.2.4 Application to ABWR

2.3.5.3 Carryover due to Entrainment

2.4 ADDITIONAL SEQUENCE ANALYSES [NAME CHANGE]

(as originally submitted, duplicates with section 19E.2.6 will be removed)

2.4.1 Core Melt Progression and Hydrogen Generation
2.4.2 In-vessel Recovery
2.4.3 System Recovery after Vessel Failure and Normal Containment Leakage
2.4.4 Early Drywell Head Failure
2.4.5 Suppression Pool Drain
2.4.6 Stuck Open Vacuum Breaker
2.4.7 Containment Structural Failure Pressure
2.4.8 Effect of Overpressure Relief Rupture Disk
2.4.9 Effect of Debris Coolability in the Lower Plenum

2.5 IDENTIFICATION AND SCREENING OF PHENOMENOLOGICAL ISSUES

(new - Submitted in CEB92-12, starting p. 10. Note title change)

2.5.1 Review of NUREG/4551 Grand Gulf and Peach Bottom Analysis

2.5.1.1 Grand Gulf
2.5.1.2 Peach Bottom
2.5.1.3 Application of NUREG/CR-4551 Results to ABWR

2.5.2 Review of NUREG-1335
2.5.3 Review of Recommended Sensitivity Analyses for an Individual Plant Examination using MAAP 3.0B (EPRI)
2.5.4 Review of ALWR Requirements Document
2.5.5 Summary and Conclusions

2.6 SENSITIVITY ANALYSIS AND SCOPING STUDIES FOR PHENOMENOLOGICAL ISSUES

(new - Submitted in CEB92-X)

2.6.1 Core Melt Progression and Hydrogen Generation
(new - Submitted in CEB92-X)

2.6.2 Fission Product Release from Core
(new - Submitted in CEB92-36)

2.6.3 CsI Re-evaporation
(new - Submitted CEB92-36)

2.6.4 Time of Vessel Failure
(new - Submitted CEB92-36)

2.6.5 Recriticality During In-Vessel Recovery
(new - Submitted CEB92-X)

2.6.5.1 Potential for Recriticality
2.6.5.2 Implications of Recriticality
2.6.5.3 Conclusions

2.6.6 Debris Entrainment and Direct Containment Heating
(new - Submitted in CEB92-36)

2.6.7 Fuel Coolant Interactions
(Put Summary Here - Details in Attachment 19EB)

2.6.8 Core Concrete Interaction and Debris Coolability
(new - Submitted in CEB92-36)

2.6.9 Fission Product Release Location
(new - Submitted in CEB92-36)

2.6.10 Fission Product Release Flow Area
(new - Submitted in CEB92-X)

2.6.11 Suppression Pool Bypass
(transferred to 19EE)

2.6.12 High Temperature Failure of Drywell
(new - Submitted in CEB92-X)

2.6.13 Suppression Pool Decontamination Factor
(new - Submitted in CEB92-X)

2 DETAILED PHENOMENOLOGICAL UNCERTAINTY STUDIES

2.7.1 Direct Containment Heating
(Put summary here - details in Attachment 19EA)

2.7.2 Debris Coolability
(Put summary here - details in Attachment 19EC)

2.7.3 Suppression Pool Bypass
(Put summary here - details in Attachment 19EE)

2.8 SEVERE ACCIDENT DESIGN FEATURE CONSIDERATIONS

2.8.1 Containment Overpressure Protection System
(submitted in CEB92-X)

2.8.1.1 Pressure Setpoint Determination
2.8.1.2 Variability in Rupture Disk Setpoint
2.8.1.3 Sizing of Rupture Disk
2.8.1.4 Comparison of ABWR Performance with and without COPS
2.8.1.5 Suppression Pool Bypass
2.8.1.6 Summary

2.8.2 Lower Drywell Flooder
(submitted in CEB92-X, Doug revising primarily for clarity)

2.8.3 Corium Protection for Lower Drywell Sump [Corium Shield]
(Put summary here - details in Attachment 19ED)

2.9 REFERENCES

3. CONSEQUENCE ANALYSIS

3.1 SITE ASSUMPTIONS
- **3.1.1 Meteorology**
- **3.1.2 Population**
- **3.1.3 Evacuation**

3.2 CRAC INPUT DATA
- **3.2.1 Input Which Differs From Standard CRAC Assumptions**
- **3.2.2 Input to CRAC from Performance Analysis**

3.3 COMPARISON OF RESULTS TO GOALS
- **3.3.1 Goals**
- **3.3.2 Results**

3.4 REFERENCES

EA. DIRECT CONTAINMENT HEATING

(submitted in CEB92-39 revised on July 15))

1 SUMMARY DESCRIPTION

2 DESCRIPTION OF EVENT TREE ANALYSIS

2.1 Event Headings
- 2.1.1 Containment Pressure Prior to RPV Failure (CONTPRES)
- 2.1.2 RPV Pressure at RPV Failure (RVPRES)
- 2.1.3 Mode of RPV Failure (MODRVFAIL)
- 2.1.4 Fraction of Core Inventory Molten in Lower RPV Head (RVCORMASS)
- 2.1.5 High-Pressure Melt Ejection (HPME)
- 2.1.6 Fraction of Entrained Debris Fragmented and Transported to the Upper Drywell (FRAG)
- 2.1.7 Peak Containment Pressure Following RPV Failure
- 2.1.8 Drywell Head Fails Following Vessel Failure

3 DETERMINISTIC MODEL FOR DCH

3.1 Debris Dispersal in the ABWR
- 3.1.1 Velocity Required to Transport Debris Particles
- 3.1.2 Argonne Experiments on Debris Dispersal
  - 3.1.2.1 Experiment on Zion Configuration
  - 3.1.2.2 Experiments on Grand Gulf Configuration
  - 3.1.2.3 Application to GE ABWR Configuration

3.2 Pressurization due to DCH

3.3 Calculation of Vent Clearing Time

3.4 Calculation of Dispersal Time Constant

3.5 Application of DCH Model to ABWR

3.6 Sensitivity to Various DCH Parameters
- 3.6.1 Base Case
- 3.6.2 Dispersal Time Constant
- 3.6.3 Debris Temperature
- 3.6.4 Nodalization

3.6.5 Zr Oxidation Ex-Vessel
3.6.6 Initial Drywell steam Fraction
3.6.7 Hydrogen Combustion
3.6.8 Vent Clearing

4 CONTAINMENT ULTIMATE STRENGTH AND UNCERTAINTY
4.1 Ultimate Strength
4.2 Uncertainty in the Failure Pressure

5 SUMMARY OF RESULTS
5.1 Quantification of Decomposition Event Trees
5.2 Impact on Containment Failure Probability
5.2.1 Sensitivity of Containment Failure Probability to Assumptions
5.3 Impact on Offsite Dose

6 CONCLUSIONS

7 REFERENCES

EB. FCI
(submitted in CEB92-X)

1 INTRODUCTION

2 APPLICABILITY OF EXPERIMENTS
2.1 Fuel Coolant Interaction Tests
2.2 Experiments With a Stratified System
2.3 BETA V 6.1
2.4 High Pressure Melt Ejection Experiments

3 EXPLOSIVE STEAM GENERATION
3.1 Phenomenology
3.2 Bounding Analysis

4 IMPULSE LOADS
4.1 Maximum Impulse Pressure
4.2 Impulse Duration
4.3 Pedestal Capability
4.3.1 Elastic-Plastic Calculation
4.3.2 Comparison to NUREG-1150 Grand Gulf Pedestal
4.4 Capability of the ABWR to Withstand Pressure Impulse

5 WATER MISSILES
5.1 Maximum Rise Height
5.2 Available Rise Height
5.3 Capability of ABWR to Withstand Water Missiles

6 CONTAINMENT OVERPRESSURIZATION
6.1 Methodology
6.2 Maximum Steam Generation Rates
6.2.1 Water Added to Debris
6.2.1.1 Water Inventory from Lower Plenum
6.2.1.2 Passive Flooder Flow

EC. CORE CONCRETE INTERACTION AND DEBRIS COOLABILITY

(submitted in CEB92-X)

ED. CORIUM SHIELD

(submitted in CEB92-47 on August 7)

EE SUPPRESSION POOL BYPASS

(combination of submittals in CEB92-36 and CEB92-X)