52-001

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CEB92-41

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Subject: Clarification / Additional Information Needed for Closure of Level 2 PRA Issues

## DCH

**Question 1:**

The Level 1 analysis does not consider depressurization of the sequences in accident classes IB-1, IB-2, IB-3, yet the Level 2 analysis (Table 1, CEB-92-39) reports that the bulk of these sequences are depressurized. Provide supporting analyses and/or revised Level 1 event trees which demonstrate that these sequences will in fact be depressurized. Identify and discuss the specific guidance provided to the operator in the EPGs for these sequences.

**Response 1:**

The depressurization system in the ABWR is automatic and does not rely on AC power. As a backup to the automatic actuation, the emergency procedure guidelines (EPGs), contained in subsection 18A of the SSAR, require the operator to manually initiate the ADS system when the water level reaches the top of active fuel. Since the EPGs are symptom based, there is no differentiation between station blackout events and other events. Therefore, there is no difference in the reliability for this class of events. The operability of the ADS system is discussed in subsection 19E.2.1.2.2, where it is shown that there is adequate DC power and nitrogen supply to actuate the ADS system during a blackout event.



D050 1/1

Based on the above discussion, the probability for depressurization used for station blackout events (class IB-1, IB-2 and IB-3 sequences) should the same as that used in the Level 1 event trees for non-station blackout events. In CEB92-39 the values for manual depressurization had conservatively been used instead of the values for automatic depressurization which are appropriate since a bypass timer was added to the reactor design. The appropriate corrections to Table 1 of CEB92-39 are attached. Based on the very low frequency of high pressure events as indicated on the table, high pressure melt sequences may be neglected.

**Question 2:**

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.

**Response 2:**

The response to this question will be provided at a later date.

**Suppression Pool Bypass**

**Question 1:**

Quantification of the failure probability of vacuum breakers in the pool bypass CET/DET is based on vacuum breaker operating data collected over a ten year period. It is our understanding that this includes surveillance (stroke) test data as well as leak rate test data, and is used to quantify several branches in the DET. Please provide a summary of the operating data, e.g., summary tables showing the component operating time, number of tests of each type, failures to open, failures to reclose, failures of leak rate tests.

**Response 1:**

The data includes failures and abnormalities encountered during surveillance testing and leak rate tests. The data was based on BWR operating experience from 4/81 to 3/91. The failure probabilities were based on the following data.

| | |
|---|---|
| Cumulative component time: | 2.66E7 hours, $\Delta T$ |
| Number of occurrences: | |
| failure to close: | 18, $N_{close}$ |
| failure to open: | 10 |
| leakage: | 45, $N_{leak}$ |
| other: | 8 |

Failures to close were detected during stroke capability test which are performed every month ($T_{close}$ = 720 hours). Vacuum breaker leakage was detected during leak rate testing performed during every refueling outage ($T_{leak}$ = 13,140 hours). The ABWR will contain eight vacuum breakers ($N_{valves}$ = 8). The failure probabilities were calculated as follows.

$$P(\text{VB Leak}) = N_{leak}T_{leak}N_{valves}/\Delta T = 0.18$$

$$P(\text{VB Stuck Open}) = N_{close}T_{close}N_{valves}/\Delta T = 3.9E\text{-}3$$

Failures in the "other" category do not effect actual vacuum breaker operation. Examples are failure of the disk position indicator lights when the disk was in the desired position and failure of the air cylinders which are used to perform the stroke capability tests. The air cylinders (which are not included in the ABWR design) are used only during testing and do not affect normal vacuum breaker operation. It should also be noted that the probability of failure to close and leak are conservative since the closing forces during accident conditions will be at least an order of magnitude greater than those present during testing and normal operation. Additional closing force will enhance sealing and overcome some, if not all, closing resistances.

**Question 2:**

Provide additional discussion of the criteria and rationale for excluding failures to open and failures of local leak rate tests from the database. It would appear that valves which fail to open during a surveillance test (perhaps due to binding on the shaft) might still open during an accident if the differential pressures are greater than used during the surveillance test. They would then be likely to stick open. Valves which fail to pass local leak rate tests even though their indicator switch indicates "close" may be a precursor to binding on the shaft, and may exhibit a similar tendency to fail to reclose. Given these uncertainties, and the lack of data on vacuum breaker performance under actual accident conditions, provide an assessment of the effect of retaining these failures in the database on the probability of a stuck open vacuum breaker (event VB) and the probability of vacuum breaker leak (event VB LEAK).

**Response 2:**

Twelve failures to open occurred in the ten year observation period. Eight of these failures were due to vacuum breakers drifting out of calibration. This type of failure would not affect vacuum breaker closure ability once it had opened. Two of the failures were attributed to worn or broken magnets. Again, this would not prevent the vacuum breaker from closing during an accident after it had opened.

The other two failures were due to: 1) a loose set screw on the flapper pivot pin, and 2) excessive clearance between the valve shaft and disk. The force

required to open these valves was greater than technical specification limits and greater than the force required to open the other vacuum breakers tested in the same sequence. In the ABWR design, depressurization transients which lead to opening of the vacuum breakers are very mild. Therefore, if either of the failure conditions existed during an accident, the affected valves would not open because the other vacuum breakers would open and relieve the wetwell pressure. Therefore, these failures were excluded in the probabilities of vacuum breaker leakage and sticking open.

Failures of local leak rate tests were included in the database. Failures attributed to worn gaskets, pitted seating surfaces or slight misalignment were included in the probability of VB LEAK. Failures due to mechanical binding or excessive closure force were included in the VB STUCK OPEN failure probability.

Failures of the position indicating lights were not included in the failure probabilities of VB LEAK or VB STUCK OPEN.

Based on the above discussion the treatment of the various failure modes is appropriate. Therefore, no additional assessment of the impact of vacuum breaker failures must be performed.

## COPS

**Question 1:**

Section X.4.1 provides a comparison of sequences with and without COPS. This assessment is insufficient to fully resolve the issue regarding net risk impact of COPS (O-14). Specifically, the net risk impact of COPS, and the effect of suppression pool bypass, CCI, etc. on this result cannot easily be ascertained by comparing results from MAAP calculations with and without COPS. Rather, the net risk impact should be assessed based on considering the impact of the system on the CET results, i.e., by assessing the risk profile or CET end states with and without COPS. In this way, any effects that system would have on shifting releases from one release category to another, or any interactions between phenomena/events would be accounted for. The information provided in Section X.4.1 of CEB-92-X should be supplemented in this regard to resolve the issue.

**Response 1:**

Section X.4.1.6 will be modified as follows:

X.4.1.6 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 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 COPS-1.1 and Figure COPS-1.1. 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 6 [of CEB-92-X]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 COPS-1.1 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 COPS-1.1 shows the probability of exceedance 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.

Table COPS-1.1

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 |

**Question 2:**

Provide a breakdown of the frequency of containment venting in terms of time to vent, e.g., the frequency of venting early (such as < 12 h), intermediate (such as 12-24 h), and late (such as >24 h).

**Response 2:**

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

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

2. The time between the release of fission products from the core and the time of release from containment (residence time) affects the removal in containment. For releases through the COPS 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 or 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 final important measure of time is the time available for offsite evacuation, should it be necessary. Discussions with several utilities indicate that evacuation of their Emergency Planning Zones (EPZ) can be completed in less than 8 hours, even in the worst weather conditions. Experience has also indicated that ad hoc planning can successfully evacuate a region in about 24 hours (Reference: WASH-1400, Appendix 6J).

Based on the forgoing, four time frames were selected in determining the ti[illegible] to fission product release, either via the rupture disk or directly from the drywell. Table COPS-2.1 summarizes the results which were obtained by using the probabilities given in Figure 2 of CEB92-X (submitted June 30, 1992) and

assigning them to a time and mode of release based on the accident analysis contained in subsection 19E.2.2.

Table COPS-2.1
Frequency of Fission Product Release

| Time of Release | Release Frequency | |
|---|---|---|
| No Release | 1.34E-7 | |
| | Release via Rupture Disk | Release via Drywell |
| > 24 hours | 2.1E-8 | 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 |

Passive Flooder System

Question 1:

Provide the assessment of net risk impact of the passive flooder system identified in O-15. As discussed above for COPS, net risk impact should be assessed using the modified CETs/DETs as the basis for demonstrating how the design feature influences the risk profile for the ABWR.

Response 1:

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 PFS-1.1, 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.

Table PFS-1.1

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 | 6.78E-8 | 6.78E-8 | 6.78E-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 |

**Question 2:**

In the ITAAC submittal (June 30, 1992), the minimum acceptable passive flooder flow rate is indicated to be 10.5 l/sec per valve. Based on the analyses presented in Section X.4.2.1 [of CEB-92-X, the expected flow rate for each valve under accident conditions, using Bernoulli's equation, is approximately 11.0 l/sec. Because the minimum acceptable flow rate is very close to the maximum theoretical flow rate possible under accident conditions, lodging of the Teflon disc in the valve, or small amounts of fusible material/alloy remaining in the valve after actuation may cause the valve flow to be unacceptably low. Furthermore, the analyses in Section X.4.2 suggest that 8 valves would be required to remove all the decay heat available at the time they would be actuated. (This is based on all of the core participating, but also does not include heat from exothermic reactions in the debris bed.) In view of the fact that a significant number of the valves would be required to operate in order to fulfill the system function, and the uncertainty in individual valve operability, the probability of successful passive flooder operation assumed in the PRA (0.999) appears overly optimistic. In this regard, please provide an assessment of the impact of reduced passive flooder system reliability on the

ABWR risk profile. A recommended approach for addressing this concern is to requantify the CETs/DETs assuming lower probabilities of successful system operation.

**Response 2:**

The intent of the submittal on the flow per valve was to identify the basis for the passive flooder design. As conceived, the fusible material would melt allowing the Teflon disk to be ejected from the pipe. Therefore, there would be little variation in the flow rate through the valve.

The analysis reported in Section X.4.2 [of CEB92-X] for the sizing of the valve does not represent the minimum acceptable flow condition. Rather, the flow calculated there is the basis for sizing the system to allow for rapid quenching of the debris. In order to clarify this point, section X.4.2 will be modified and expanded as shown below. As the results of this calculation show, the minimum flow rate may be accomplished by two fully open valves.

**X.4.2 Lower Drywell Flooder**

**X.4.2.1 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 (LD) flooder provides water to cool any core debris which relocates from the vessel into the LD and to establish 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 mass as its temperature decreases. Quenching prevents or mitigates core concrete interaction (CCI). An overlying water pool scrubs fission products which are released from the debris bed.

The minimum acceptable flow rate for the flooder system corresponds to the flow rate which can just absorb the maximum heat generated in the debris bed. Minimum acceptable flow is calculated in Section X.4.2.2. The expected flow rate in the flooder system can be obtained by applying Bernoulli's equation to the flooder geometry. The expected flooder flow is calculated in Section X.4.2.3.

**X.4.2.2 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 to the ABWR SSAR, 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 by 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.

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. NUREG-5565 indicates that a typical ablation rate for concrete is two inches per hour. The generation rate, assuming that the $H_20$ and $CO_2$ released during ablation completely react with zirconium, is 3.6 MW. Combining these two sources of heat yields a maximum 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 *l*/sec). Assuming a four inch throat as discussed in Section X.4.2 [of CEB-92-X submitted to the staff on June 30, 1992], 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 *l*/sec.

**X.4.2.3 Expected Flow Rate**

same as X.4.2.1 in CEB-92-X

**X.4.2.4 Time for Initial Flooding of the Lower Drywell**

similar to X.4.2.2 CEB-92-X, but modified to reflect that using expected flow rate not minimum acceptable flow

**X.4.2.5 Consequences of One Flooder Line Opening First**

same as X.4.2.3 in CEB-92-X

**X.4.2.6 Valve Opening Time**

same as X.4.2.4 in CEB-92-X

The response to the first question on the passive flooder assesses the impact of reduced passive flooder system reliability. The CETs were requantified assuming reliability of 0.99, 0.9 and 0. The results of this study indicated that the risk profile is not affected by the assumed reliability of the passive flooder.

<u>PRA Input to Severe Accident Closure Chapter</u>

**Question 1:**

Provide a sequence-by-sequence comparison of accident frequency between the ABWR and operating BWRs, and an explanation of specific reasons for differences. To the extent possible, this discussion should indicate the specific impact of the plant features (which account for the differences) on key PRA models or assumptions.

**Response 1:**

The response to this question will be provided separately.

**Question 2:**

Provide estimates of CCFP for the ABWR based on the revised CETs/DETs. Also provide separate estimates of CCFP for alternative definitions of containment failure, e.g., CCFP if containment venting after 12h, 18h, 24h is considered a success.

**Response 2:**

The containment overpressure protection system (COPS) provides for a reclosable release from a controlled location. Therefore, operation of COPS is not equivalent to containment failure. GE and the NRC have agreed upon two alternate definitions of containment failure. The first of these definitions is based on the functional performance of the containment to prevent a large release of fission products in the event of an accident. Although no precise definition of "large release" has been established, 25 rem at the site boundary has been selected as the basis for the measurement since there are no observable health effects for this type of release. Based on this definition $CCFP_{25\ rem}$ is 0.002.

Alternately, CCFP can be defined in terms of the structural integrity of the containment. Based on this definition, all sequences which lead to release via the drywell in the ABWR contribute to the CCFP. Using this definition and the information contained in Table COPS-2.1, $CCFP_{SI}$ is 0.005.

**Question 3:**

Provide a breakout and discussion of the contribution/effect of key Level 2 issues on CCFP and risk. Specifically address what the PRA results say about the importance of the individual issues/phenomena, including DCH, pool bypass, and CCI. Quantitative rather than qualitative arguments should be

used. This information may be embedded in the recent GE submittals, but a more concise and focused discussion of the role of these issues in the ABWR risk profile is needed.

**Response 5:**

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 DCH, pool bypass, and CCI each impact the containment failure probability and risk profile.

DCH:

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.

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 net impact of pool bypass events on fission product release frequency may be assessed by examination of Figure 2 [in CEB92-39]. 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.

<u>CCI:</u>

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.

Credit for Firewater Addition

**Question 1:**

Considerable credit is taken for recovery of core damage in-vessel for certain subclasses (e.g., IB-2, ID, and IIID), however, the bases for the assigned probabilities is vague. Specifically, it is not clear how much of the credit is due to (1) recovery of AC power, (2) recovery of previously failed systems, or (3) use of previously unavailable systems such as fire water. For each accident subclass, please identify the specific systems being credited, and the credit taken for each, so as to support the probability values used in the analysis.

**Response 1:**

The following summarizes the assumptions used to assess the probability of in-vessel recovery from a core damage accident. Injection into the vessel using the firewater addition system is not credited in the Level 1 core damage analysis. Availability of the firewater injection system has been considered only in the Level 2 analysis. Consequently, all accident sequence classes enter the Level 2 analysis with availability of the firewater system not determined. Since the firewater system is powered by its own direct drive diesel, its reliability is not affected by a station blackout event.

Classes IA_0, IA_1, IIIA_0 and IIIA_1

These classes of accidents are initiated either by a transient or by a LOCA. High pressure injection failed and RPV depressurization was not successful prior to RPV failure. In order to prevent RPV failure, recovery of a high pressure injection system is required in less than 1 hour. A recovery probability of 0.05 is calculated assuming a mean time to repair (MTTR) of 19 hours for the failed system.

Class IB2_0

This accident class consists of SBO sequences with operation of the RCIC system for eight hours. After failure of the RCIC, the RPV is depressurized. If in-vessel injection is re-established within about two hours of loss of RCIC, core damage can be arrested and vessel failure can be prevented. In vessel injections is re-established if either the offsite power is recovered in 2 hours or the firewater system is initiated. The conditional probability of recovering power in this 2 hour period is 0.6. The firewater system is assigned a failure probability of 0.01 based on operator error probability. This yields a combined failure probability of 0.006. The operator is expected to monitor the availability of DC power during the blackout, so there will be approximately 10 hours of warning time before use of the firewater addition system is necessary.

Classes ID and IIID

These accident classes consist of low pressure sequences with loss of low pressure injection. If in-vessel injection is established within about half an hour of initiation of core uncovery, core damage arrest in vessel is likely. For

these accident classes, the probability of in-vessel injection is based on alignment of the firewater system. A probability of 0.1 is assigned for failure to establish firewater injection.

Other sequences excluding Class II

For all other sequences in which the vessel failed before any presumed damage to the containment, core damage arrest in-vessel was conservatively neglected.

**Question 2:**

Clarify how the use of firewater was treated in the revised PRA. (It is our understanding that no credit has been taken for severe accident prevention (i.e., in the Level 1 analysis), and that credit is taken only in the Level 2 analysis.)

**Response 2:**

Your understanding is essentially correct. No credit was taken for the firewater system in preventing severe accidents for most classes of events since firewater addition must be initiated relatively quickly to prevent core damage. There is, however, one exception. Class II events, which have successful core cooling but no containment heat removal develop very slowly so it is judged that the operator has adequate time to use the firewater system to prevent core damage. In these cases, credit was taken for the firewater to prevent core damage.

**Question 3:**

Provide references to SSAR sections or GE submittals in which details regarding use of the AC-independent firewater addition systems are provided. This should include specific human actions required to connect the diesel-driven pumps and the fire trucks, locations that these actions would be taken, emergency procedures guiding these actions, necessary spool pieces, tools, etc. and design details such as pump head curves, pressure capacity of fire hose/piping, and in-line check valves to assure that rapid RCS pressurization will not result in a breach of the injection path.

**Response 3:**

The use of the firewater system for injection to both the vessel and the drywell sprays was identified as being important to the PRA in Appendix 18F. The use of the firewater system is addressed in the symptomatic emergency procedure guidelines (EPGs). A listing of the locations in which the firewater addition system is mentioned in the EPGs is provided below in the response to the question on modelling of operator actions in the Level 2 analysis. The specific, detailed human actions which are required for initiation of the firewater addition system and the requirements for tools and spool pieces will be developed by the COL applicant.

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 is specified in section 2.15.6 of the ITAAC.

## Modelling of Operator Actions in the Level 2 Analysis

**Question 1:**

Provide references to specific sections of the EPGs and SSAR which address the following:

A. operator actions in response to failure of SRV discharge line in seismic events,

B. operator actions following rupture disc opening,

C. operation of drywell sprays as alluded to on page 19E.2-11 of draft Section 19E.2.2,

D. operation of wetwell sprays alluded to in insert to June 4, 1992 GE markup of Section 19E.2.3.3.4.(1),

E. hookup of diesel-driven fire sprays, and fire truck for core/spray injection, and

F. operator response to RWCU line breaks alluded to in Insert 3 to June 4, 1992 GE markup of Section 19E.2.3.3.3.(4).

**Response 1:**

The ABWR EPGs are symptom based. Therefore, there are no event based procedures in the ABWR SSAR. Any event specific procedures will be determined in the plant specific procedures. Therefore, the discussions below relate the event described to the observable symptoms. Then, based on the symptoms, the proper procedures are referenced.

A. Failure of an SRV discharge line would result in increased pressurization and elevated temperatures in the wetwell. The vacuum breakers would open and the drywell temperature and pressure would also rise. Therefore, the operator would enter the procedures for drywell

temperature control DW/T (page 18A.5-4) and PC/P (page 18A.5-7). More details on these procedures are shown in Appendix 18F, Table 18F-3.

B. Procedure PC/P-5 (page 18A.5-9.1)indicates that the rupture disk path should not be closed until directed to do so by post accident recovery procedures. Development of these procedures will be the responsibility of the COL applicant.

C. The drywell spray actuation described in the draft of Section 19E.2.2 is specified in the drywell temperature control procedure, DW/T (page 18A.5-3).

D. The operation of wetwell spray referenced is specified in response to increasing pressure. The appropriate steps are specified in steps PC/P-1, PC/P-2 and PC/P-6 (pages 18A.5-7, 8 and 10).

E. Initiation of the firewater addition system is specified in the ABWR EPGs as the final option each time vessel injection or containment spray is required. Table OA-1 indicates each of the procedure steps in which the firewater addtion system is mentioned in the EPGs. The location of the step in the inventory of emergency operation information and controls is also given in Table OA-1.

F. The early stages of the response to an RWCU line break will be performed in accordance with the symptom based procedures. The operator will take the appropriate steps to shut the plant down, control water level and control containment pressure. The recovery actions identified in the referenced section are deferred to the event specific procedures to be developed as a part of the plant specific procedures.

Table OA-1
References to the Firewater Addition System in the ABWR EPGs

| Procedure Step | Location in Inventory Table | Location in Inventory Page |
|---|---|---|
| Vessel Inventory Control | | |
| RC/L-2 | 18F-2 | 18F-63 |
| Drywell Temperature Control | | |
| DW/T-2 | 18F-3 | 18F-146 |
| Containment Pressure Control | | |
| PC/P-1 | 18F-3 | 18F-156 |
| PC/P-2 | 18F-3 | 18F-159 |
| PC/P-6 | 18F-3 | 18F-163 |
| PC/H-4.1 | 18F-3 | 18F-198 |
| PC/H-4.4 | 18F-3 | 18F-201 |
| C1-2 | 18F-6 | 18F-268 |
| C4/3.1 | 18F-9 | 18F-306 |
| C5/3.2 | 18F-10 | 18F-325 |
| C6-2 | 18F-11 | 18F-333 |
| C6-4 | 18F-11 | 18F-337 |

## Level 2 Results

**Question 1:**

Figure 2 in CEB-92-39 appears to play a key role in integrating the results of the individual CETs for each accident subclass/PDS, and establishing frequencies for each release class/case in the Level 3 analysis. However, the submittal provides no discusssion of the role of this figure, how it was developed, and how it is used to support the frequencies of the various releases in the Level 3 analysis. A detailed discussion of this figure and how it is used is needed.

**Response 1:**

The following description of Figure 2 will be incorporated into 19D.5.11 of the SSAR.

**19D.5.11.2 Level 2 Results**

The logic diagram shown in Figure 2 groups the set of Level 2 sequences into release categories based on similar sequence characteristics judged to be important to definition of the offsite source term and consequences. Five parameters are used to define the release categories. This grouping resulted in the definition of 53 distinct release categories. The characteristics of each release category are determined by the branch attributes for the pathway through the [illegible]. The five grouping parameters are discussed below.

**[illegible].1 Initiator Code (INITCODE)**

[illegible]meter groups the sequences based on the accident sequence [illegible] accident-sequence-type definition is described in Section [illegible]. Note that sequence types NSCL (class IC) and NSCH (class IE) [illegible] of such low probability that they were truncated prior to performing the Level 2 analysis and are not included in the grouping diagram. In addition, as a result of the low probability of Class IV ATWS sequences (sequence type NSRC), they were also not evaluated in the Level 2 model although the Class IV frequency is shown in the logic diagram (STC 53).

**19D.5.11.2.2 Core Melt Arrested In-Vessel (IV)**

This parameter groups sequences based on whether late in-vessel cooling is successful in preventing vessel failure.

**19D.5.11.2.3 Mode of Release (REL MODE)**

This parameter groups sequences based on the mode of any fission product release from the containment. The following important characteristics are considered.

**19D.5.11.2.3.1 Normal Containment Leakage**

Containment pressurization is terminated, so there is no containment failure or COPS operation. These sequences have very small releases to the environment as a result of no[illegible]l containment leakage.

**19D.5.11.2.3.2 Rupture Disk**

Operation of the COPS leads to nearly complete release of the noble gases. Other fission product releases are negligible.

#### 19D.5.11.2.3.3 Drywell Head Failure

Long-term steam and non-condensable gas production leads to over pressurization of the containment. The drywell head fails before the COPS opens in these sequences.

#### 19D.5.11.2.3.4 Penetration Over-temperature Failure

High temperatures lead to failure of the large penetration seals in the drywell.

#### 19D.5.11.2.3.5 Early Containment Failure

Overpressure failure of the drywell head occurs at the time of RPV failure.

#### 19D.5.11.2.4 Pool Bypass (POOL_BP)

This parameter groups sequences based on whether radionuclides released into the drywell gas space bypass the suppression pool prior to release from containment. All drywell containment failure modes result in eventual pool bypass and no branching is required. For sequences without containment failure, this parameter is irrelevant. Hence, branching under this heading is only significant for the COPS release mode.

#### 19D.5.11.2.5 Drywell Spray (SPRAY)

Operation of the drywell sprays can be effective in mitigating the release of radionuclides. However, for sequences where vessel failure has not occurred and sequences where pool bypass has not occurred, operation of the sprays has no significant impact since suppression pool scrubbing will effectively mitigate radionuclide releases. Therefore, branching is only considered for sequences with pool bypass. Note that for sequences with drywell penetration overtemperature failure, the drywell sprays are not operating and no branching is necessary.

**Question 2:**

Provide a description of the process used to assign release characteristics to each of the end states of Figure 2 in CEB-92-39 is needed, and to group these releases for subsequent Level 3 analysis. Also identify: (1) the accident sequence group assigned to each of the 58 end states/STC#s, and (2) the frequencies assigned to each accident in Table 1-1 of the updated ABWR consequence analysis (June 30, 1992 fax from J. Duncan).

**Response 2:**

The important release characteristics for each of the severe accident sequences are summarized in Figure 2 of CEB92-39. 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 L2-1 below identifies the deterministic accident sequence associated with each of the end states in Figure 2 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.

STC#53 in Figure 2 [of CEB92-39] was binned with Case 9 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.

The deterministic sequences are then binned according to the characteristics of the fission product release. Table 1-1 [of the ABWR PRA Consequence Analysis Submittal, June 30 fax from J. Duncan] indicates combination of the deterministic sequences into release bins. Column F(i) of Table 1-1 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.

Table L2-1
Binning of Containment Event Tree Results

| STC # | Deterministic Bin | Consequence Bin | |
|---|---|---|---|
| 1 | NCL | NCL | |
| 4 | NCL | NCL | |
| 5 | LCHPPFP | Case 7 | |
| 6 | LCHPFSR | Case 1 | |
| 8 | LCHPPBR | Case 8 | See notes |
| 10 | LCHPPBD90 | Case 7 | See notes |
| 12 | LCHP00E | Case 8 | |
| 13 | NCL | NCL | |
| 14 | LCLPFSR | Case 1 | See notes |
| 15 | LCLPFSR | Case 1 | See notes |
| 16 | NCL | NCL | |
| 18 | LCLPFSR | Case 1 | See notes |
| 19 | LCLPFSD90 | Case 7 | |
| 21 | LCLPFSD90 | Case 7 | |
| 25 | NCL | NCL | |
| 26 | LCLPFSR | Case 1 | See notes |
| 28 | NCL | NCL | |
| 30 | SBRCPFR | Case 1 | |
| 37 | NCL | NCL | |
| 38 | LBLCFSR | Case 1 | See notes |
| 40 | NCL | NCL | |

Notes

Sequences 8 and 10: Releases taken from vacuum breaker study

Sequence 14, 26 and 38: Sequence is arrested in vessel indicating high probability of the use of the firewater addition system.

Sequence 15: This sequence is binned with those which have releases through the rupture disk since any fission products which are released from the vessel will be scrubbed through the suppression pool.

Sequence 30: No credit taken for firewater system since a long time was available to prevent core damage but the operator failed to do so.

**Question 3:**

Based on our initial review, it appears that core concrete interactions should be included as a top event in Figure 2. P ovide justification for not including it.

**Response 3:**

Section X.3.2 provides a detailed investigation into the impact that CCI has on the overall ABWR risk. As described as part of a previous respon--, the following conclusions are pertinent to CCI:

1. The frequency of sequences expected to experience dry C' is insignificant due to the high reliability of water addition to the lower drywell. For dominant core melt sequences that release core material into the containment 90% result in no significant CCI since the floor area of the lower drywell is large and water is present.

2. Even for those low frequency cases with significant CCI, radial erosion remains below the structural limit of the pedestal. Thus containment failure will not occur even for these sequences.

3. Regardless of assumptions about debris coolability, fission product release is dominated by the release of noble gasses because the containment overpressure protection device opens, forcing all fission products through the suppression pool.

Figure 2 is a release category grouping tree and includes only the most significant events relative to fission product release. Since the majority of the ABWR sequences involve water on the debris ex-vessel, CCI does not impact the source term.

**Question 4:**

The treatment of Class 2 accidents in the Level 2 analysis is limited to the information presented in Figure 9 in CEB-92-39. This figure is not discussed in the text, and the bases for the branch point probabilities are not presented. Furthermore, several of the probability values appear extremely optimistic. In particular, the assumption that continued core cooling is assured after rupture disc actuation does not acknowledge the potential for failure of injection due to decreased NPSH and the potential for random failure during the mission time. The assumption that gross containment failure leads to loss of core cooling with a probability of only 0.001 is also extremely optimistic given that containment failure can affect long term operability via radiation and temperature effects and access, as well as the two concerns noted above. In view of the importance of this event tree in virtually eliminating Class 2 sequences, a detailed discussion of the Class 2 analysis is needed, along with justification for the probability values assumed.

**Response 4:**

The following description of Class 2 will be incorporated into 19D.5.11 of the SSAR.

**19D.5.11.4 Decomposition Event Trees for Class II**

The containment event tree (CET) for Class II sequences is shown in Figure 9 [of CEB92-39]. The supporting DETs are shown in Figures 27 through 29 [also in CEB92-39]. This CET is substantially different from those for the Class I events. Class II consists of sequences with loss of containment heat removal (CHR) but with successful in-vessel injection. If CHR is not recovered within about 20 hours, the containment pressure will exceed the COPS rupture disk setpoint pressure (90 psig).

The first event in the CET assesses the probability of recovery of the RHR system prior to COPS operation (or containment overpressure failure) given that RHR was not recovered within 20 hrs. If the RHR system is successfully recovered, containment pressure will decrease and the event will be terminated. The probability of this is estimated assuming a mean time to repair of 19 hours for the system.

The second event in the Class II CET assesses the probability that the COPS rupture disk opens prior to drywell head failure for sequences without recovery of RHR. The probability of this event is consistent with the value described in Subsection 19D.5.11.3.12.6 with no pressure difference between the wetwell and drywell.

Given failure of the COPS rupture disk to open prior to drywell head failure, the third event in the CET assesses the probability that drywell head failure will result in loss of in-vessel injection and core damage.

A discussion of the considerations and assumptions used to estimate these event probabilities is provided below.

**19D.5.11.4.1 Loss of In-vessel Injection Given Venting with COPS**

The COPS is designed to vent the wetwell gas space when the wetwell pressure exceeds 90 psig. As discussed below, high suppression pool temperatures or loss of NPSH will not threaten the ability of in-vessel injection systems to operate for an extended period of time after COPS initiation. In addition, random failures of the in-vessel injection systems during their mission time have been considered in the Level 1 analysis. Consequently, it was estimated that there was a negligible probability of failure of in-vessel injection given successful COPS.

### 19D.5.11.4.2 Loss of In-vessel Injection Given Containment Failure

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

For cases in which the core is successfully cooled but the containment is not, the containment will pressurize. If the rupture disk fails to open, 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 will be more severe. Therefore, this potential was reviewed. The following general areas were reviewed and are briefly discussed below.

(1) drywell head failure,

(2) high temperatures in the suppression pool,

(3) high drywell temperatures.

The most likely containment failure location is the drywell head. 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 because 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 extremely unlikely that more than one division of core cooling would be lost as a result.

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°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. In the class II sequences derived from the Level 1 PRA, firewater availability had not been considered. Firewater can be used as an additional source of water following containment failure. The firewater system is much less vulnerable to containment failure. The combined failure probability of conventional cooling and firewater is estimated to be 0.0001, but a value of 0.001 was used for conservatism.

**Question 5:**

In the various CETs in CEB-92-39, the top event dealing with active injection to the lower drywell (LDWI) appears to assume that injection via firewater sprays (branch "FW SPRAY") assures that water will be added to the lower drywell. As a result, the potential for failure of the passive flooder system is not assessed in the subsequent branch. This treatment is inconsistent with our understanding that the lower drywell will only be flooded after a significant amount of water is added with this system, and only after a significant delay. Please address this apparent inconsistency.

**Response 5:**

Your understanding of lower drywell flooding is correct. The use of the firewater addition system will not lead to early flooding of the lower drywell. The apparent inconsistency is a result of a simplification made to the containment event trees. If the operator follows the emergency procedures, the firewater system will be initially configured to add water to the vessel. The alignment will be changed to firewater spray mode only if high temperatures are present in the drywell. However, modelling this as two separate actions adds considerable complexity to the containment event trees. Since very little insight can be gained by modelling the actions separately, it was decided to combine the two separate actions into one node.

**Question 6:**

In CEB-92-39, accident subclass IB2-1 is discussed in several locations in the text and is depicted in Figure 16. However, it is our understanding (based on information on page 1 of that submittal) that an event tree for this event was not developed based on its low frequency. Thus, the split fraction information for this subclass presented in the submittal (e.g., on page 3 of the submittal) appears irrelevant. Please clarify this.

**Response 6:**

The split fraction information associated with subclass IB2-1 was developed before the very low frequency of these events was identified. References to subclass IB2-1 will be deleted from the text and Figure 16.

**Question 7:**

In CEB-92-39, significant credit is taken for recovery of RHR prior to fission product release, however, little information or bases are provided for the values selected. Please identify: (1) the actions to restore RHR that are credited in the analysis, and (2) the measures that are assumed to be taken by the COL applicant prior to the accident to assure that these actions can in fact be implemented. Such measures would include accident management measures, storage of spare parts, installation of flanges or cross-connect capabilities, etc. The time available to implement these actions, and the accessibility to the necessary areas in the reactor building should be explicitly addressed for each accident subclass.

**Response 7:**

Recovery of the RHR system is described in 19D.5.4. The time available to repair the system is dependent on the use of the firewater addition system. There is little or no dependence on accident class. For sequences in which the firewater system is not used, the time available for repair is about 20 hours. For sequences in which the firewater system is used, the thermal mass of the suppression pool is increased, and the time available for recovery of RHR increases to about 30 hours.

**Revised MAAP Calculations**

**Question 1:**

In CEB-92-X and previous communications, GE indicated that the probability of a flooded lower drywell cavity at the time of reactor vessel failure is extremely low because the firewater system would need to inject for about 11h in order to overflow the suppression pool into the lower drywell. However, in the revised MAAP analyses provided in draft Section 19E.2.2 the reactor cavity is calculated to be flooded in cases NSRC-PF-R-N and SBRC-PF-R-N. Please provide a discussion which reconciles this conflicting information. Also provide a quantitative estimate of the probability of a flooded cavity at the time of reactor vessel failure based on the revised PRA.

**Response 1:**

In the NSRC-PF-R-N sequence, the RCIC system is used to inject water from outside the containment into the vessel. The water is boiled in the vessel and the steam passes into the suppression pool where it is quenched. This causes the water level in the pool to rise. Eventually, the water in the pool rises above the suppression pool return path, and the lower drywell begins to fill with water. Flooding of the lower drywell begins at about 2 hours. Since the RCIC flow is approximately six times the firewater system flow, these results are in good agreement.

In sequence SBRC-PF-R-N, water is introduced into the lower drywell only as a result of discharge of the lower plenum inventory after vessel failure.

The NSRC-PF-R-N sequence is a subset of the Class IV sequences. Class IV sequences represent sequences with successful vessel injection but failure to scram. The probability of all Class IV sequences is less than 1.66E-10. Since the initiating event frequency is very small (about 0.1% of the frequency of core damage events), containment event trees were not developed. However, since class IV sequences have injection available, it is judged that only a small fraction of these will result in core damage. Therefore, the contribution of these sequences to core damage with water present in the lower drywell may be neglected. The probability of a flooded cavity at the time of vessel failure is about 0.2% as discussed in CEB-92-X.

**Question 2:**

With regard to Figure 19E.2-6E, please provide an explanation for the lower drywell water mass increasing over a 10h period (apparently due to suppression pool overflow), while suppression pool mass continues to decrease.

**Response 2:**

The sequence depicted in Figure 19E-6 represents a sequence where a considerable amount of water has been added to the containment. The water reaches the suppression pool via the SRV discharge from the vessel. Therefore, the suppression pool temperature at this time is high. At approximately 10 hours the water level in the suppression pool has reached the level of the suppression pool return path. This allows water in the lower drywall to spill into the lower drywell. Mass and energy flow continue from the vessel. Since the SRVs discharge low in the pool the temperature in the pool continues to rise. This causes a slight volumetric expansion of the water. Since the water level in the suppression pool is already at the overflow point, the expansion results in flow into the lower drywell and leads to a slight decrease in suppression pool mass.

**Question 8-1:**

Provide the rationale for establishing the time of drywell spray initiation. In some cases analyzed, sprays are not considered to be started until 2h after reactor vessel failure. Discuss the reasons for this delay.

**Response 8-1:**

The drywell spray initiation times used in the analysis are simply assumptions used for the purpose of the analysis. 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 she does not initiate the system in drywell spray mode 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 -- [Update

to 19E.2.2, submitted June 30, 1992], 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 the actuation of the passive flooder in the lower drywell. This would quench the debris and keep the drywell cool. If at some later time the firewater system is initiated, the thermal mass of the suppression pool would be increased as in other sequences with firewater addition. Since the containment water level would reach the bottom of the vessel later than in the nominal case, the firewater injection would be terminated later, leading to later opening of the rupture disk. Although this 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.

**Question 3-2:**

Provide a detailed chronology of the "FS" cases which are identified in Table 19E.2-16 but not discussed in the text. Along with other events of significance, please include the time to: suppression pool overflow, lower drywell dryout, passive flooder opening, drywell spray start and stop, and firewater start and stop

**Response 3-2:**

The detailed chronologies of all "FS" cases are presented in the text. Until the sprays are initiated, the sequence is identical to the corresponding "PF" cases. Table M-3.1 gives the section and page numbers (of the markup version of 19E2.2) where the description of the time after the initiation of the firewater system for each case begins.

Table M-3.1
Location of Sequence Chronologies

| Sequence | Section | Page |
|---|---|---|
| LCLP-FS-R-N | 19E.2.2.1 (b) | 19E.2-13 |
| LCHP-FS-R-N | 19E.2.2. (b) | 19E.2-15 |
| LBLC-FS-R-N | 19E.2.2.5 (b) | 19E.2-18.1 |
| NSCL-FS-R-N | 19E.2.2.1 (b) | 19E.2-13 |
| NSCH-FS-R-N | 19E.2.2.6 (b) | 19E.2-19.1 |
| NSRC-FS-R-N | 19E.2.2.7 (b) | 19E.2-20 |

**Question 4:**

The reactor vessel failure times in the revised MAAP calculations appears to be delayed about 1 hour relative to the times predicted in the original calculations, however, no explanation for this change is presented. Please discuss the reasons for these differences.

**Response 4:**

The times of vessel failure between the updated calculations and the original analyses are compared in the following table.

Table M-4.1
Comparison of Vessel Failure Times

| Sequence | Original | Updated |
|---|---|---|
| LCLP | 1.0 | 1.8 |
| LCHP | 1.4 | 2.0 |
| SBRC | 10.9 | 12.3 |
| LHRC | n/a | n/a |
| LBLC | 0.8 | 1.4 |
| NSCL | 0.8 | 1.3 |
| NSCH | 0.8 | 1.3 |
| NSRC | 6.1 | 5.6 |

These differences are due primarily to changes that were made to the MAAP ABWR parameter file. The most significant change involved the core peaking factors. In the original PRA calculations, the core included one radial region that was highly peaked and, therefore, resulted in a faster heatup and vessel failure. The revised parameter file has a much more evenly distributed radial power. Therefore, the core heats up slower and the vessel fails later.

**Question 5:**

The release fractions for similar accidents are much lower in the revised MAAP calculations than in the original calculations, e.g., the CsI release fraction for LCHP-PF-P-M is decreased from 0.39 in the original analysis to 0.088 in the revised analysis, and the releases for the other cases are decreased from <1E-5 in the original analyses to <1E-7 in the revised analyses. Please discuss the reasons for these differences.

**Response 5:**

In general, the operation of the containment overpressure protection system prevents drywell failure and results in pool scrubbing of all fission product releases (except noble gasses). The change in the CsI release fraction to the environment from <1E-5 to <1E-7 is not significant. In either case, the contribution of CsI to dose is negligible.

For case LCHP-PF-P-M, the CsI release is lower for the revised calculation due to the difference in the containment failure mode. The original PRA analysis assumed drywell failure at a pressure of 90 psig at a temperature of 700 F. The drywell head was subsequently strengthened. The updated PRA used a drywell head failure criteria of 134 psig at a temperature of 500 F. This difference in failure criteria resulted in the original case failing the drywell head, whereas the updated case merely leaks through the penetrations. The updated analysis shows the containment pressure maintained at an elevated pressure while, due to drywell head failure, the original calculation resulted in rapid containment depressurization and increased CsI release. The same behavior was seen in case NSCH-PF-P-M.

**Question 6:**

The source terms predicted by MAAP for vented sequences are far lower than predicted by other code calculations. This may be a result of models/assumptions regarding suppression pool scrubbing. In view of the importance of this release class, provide an assessment of the impact that higher release fractions for these sequences would have on the ABWR risk profile, and compliance of the design with the ALWR design goal regarding 25 rem dose at the boundary. This will be a critical issue in the staff's review.

**Response 6:**

In CEB92-X Section X.2.13, a sensitivity study was performed using a very conservative decontamination factor for suppression pool scrubbing. The

resulting release of CsI to the environment increased four orders of magnitude above the best estimate analyses presented in 19E.2.2 to approximately 1.5E-3. The offsite dose for the base case and the sensitivity study were presented in Figure 10 of CEB92-X. The figure indicates that even for this very conservative case, the probability of exceeding 25 rem at the site boundary (1/2 mile) is 4%. Therefore, it is judged that the uncertainties in release fraction will not have a significant impact on meeting the ALWR design goal of 25 rem at the site boundary.

**LOCAs Outside Containment**

esponses to these questions will be provided separately

**Level 3 Analysis**

Question 1:

The warning times used in the analysis (0.8 h for essentially all ABWR sequences) appear unrealistic in view of the fact that in certain accidents, the event classification (emergency action level) will not be escalated to the point that evacuation would be recommended until late in the accident. In this regard, provide justification for the warning time used for each accident sequence on which the various Level 3 cases were based. Discuss the consistency of these estimates with the estimated times at which evacuation recommendations would be made based on emergency action levels.

**Response 1:**

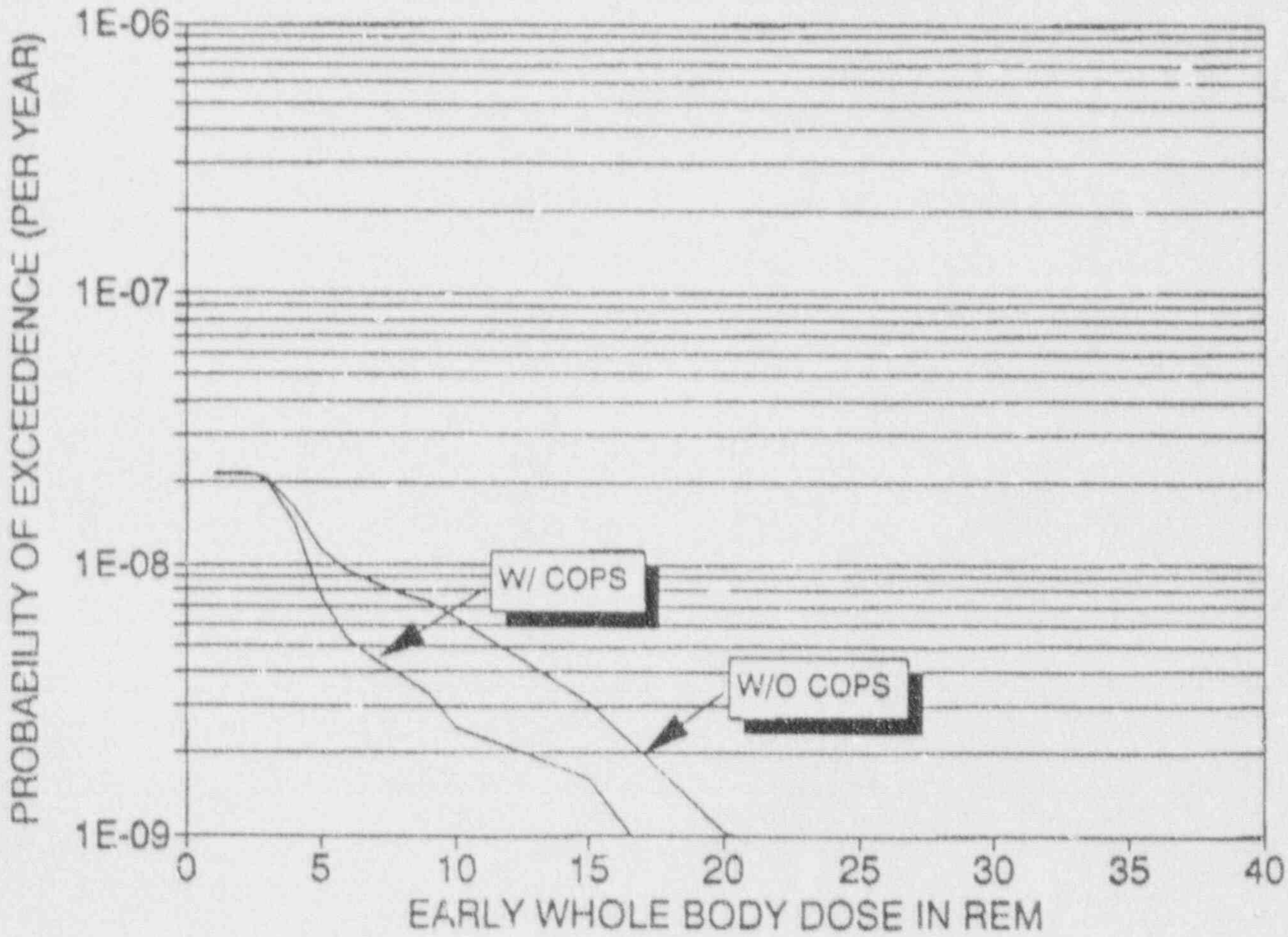
The basis for the warning time in each of the ABWR sequences is the onset of severe core damage. The emergency action levels specified in NUREG-0654 "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Prep dness in Support of Nuclear Power Plants", Appendix 1 require a site emergency be declared when "Degraded core with possible loss of coola geometry..." occurs. This is consistent with the warning time used in the ABWR Level 3 analysis.

**ITAACs for Level 2 Design Features**

GE would like to discuss these questions with the staff at the meeting on October 1.

Figure COPS-1.1

IMPACT OF COPS ON RISK



GE Nuclear Energy Proprietary Information

Table 1
Division of Accident Subclasses

| Accident Class | Subclass | Subdivided Subclass | RPV Pressure | RHR | Frequency |
|---|---|---|---|---|---|
| I | IA | IA_0 | High | Yes | 4.21 E-8 |
| | | IA_1 | High | No | 4.25 E-10 |
| | IB1 | IB1_0 | Low | No | ~~2.55 E-8~~ 2.57E-8 |
| | | IB1_1 | High | No | ~~1.54 E-10~~ 2.57E-12 |
| | IB2 | IB2_0 | Low | No | 1.62 E-8 |
| | | IB2_1 | High | No | ~~3.24 E-11~~ 1.62 E-12 |
| | IB3 | IB3_0 | Low | No | ~~8.81 E-10~~ 8.86E-10 |
| | | IB3_1 | High | No | ~~6.32 E-12~~ 8.86E-14 |
| | ID | ID | Low | No | 6.95 E-8 |
| II | II | II | n/a | n/a | 1.10E-6 |
| III | IIIA | IIIA_0 | High | Yes | 3.83 E-10 |
| | | IIIA_1 | High | No | 3.87 E-12 |
| | IIID | IIID | Low | No | 2.10 E-10 |
| IV | IV | IV | n/a | n/a | 1.66 E-10 |

< TRANSACTION REPORT >

09-24-1992(THU) 22:30

[ RECEIVE ]

| NO. | DATE | TIME | DESTINATION STATION | PG. | DURATION | MODE | RESULT |
|---|---|---|---|---|---|---|---|
| 7564 | 9-24 | 22:12 | 4089251193 | 36 | 0'18'01" | NORM.E | OK |
| | | | | 36 | 0'18'01" | | |

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10E4

GE Nuclear Energy

ABWR

To Bob Palla, NRC
J. Jo, BNL

Date 9.24.92

Fax No.

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Subject Responses to Questions for 10/1 Meeting

Message Happy Reading!