GE Nuclear Energy

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May 25, 1993

Docket No. STN 52-001

Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Review Schedule - **Closure of April 25, 1993 Meeting Issues**

Dear Chet:

Enclosed is Carol Buchholz's letter closing all of the subject issues (except Insights/Tier2/Tier 1) which were discussed in the April 25, 1993 meeting with the NRC.

Please provide a copy of this transmittal to John Monninger.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Carol Buchholz (GE)
Jack Duncan (GE)
Norman Fletcher (DOE)

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CEB93-16

May 21, 1993

To: J. N. Fox

From: Carol E. Buchholz

Subject: Closure of April 26, 1993 Meeting Issues

The following letter is intended to close all severe accident closure issues (except Insights/Tier 2/Tier 1) which were discussed in the April 26, 1993 meeting with the NRC in Washington. I have compared my responses with the meeting summary prepared by John Monninger and transmitted to us on May 10, as well as my notes from the meeting to confirm that all issues have been considered.

I anticipate that the submittal on Insights will be submitted on May 31, as agreed in the April 26 meeting.

Containment Sump Design

GE was requested to modify the containment sump protection discussion to indicate that the sump protection will be designed as described in Attachment 19ED of the SSAR. Enclosure 1 contains the markup of the SSAR which accommodates this request.

Suppression Pool Bypass (Design Basis Accident)

If a suppression pool bypass occurs during a design basis accident the vapor suppression function may fail which can lead to rapid pressurization of the containment. This in turn may cause opening of the COPS system or containment failure. The design basis calculations for suppression pool bypass were transmitted to the staff and were discussed in meetings during October and December 1991. On May 1, 1992 GE provided the staff with the results of analyses which investigated the impact of containment heat sinks on the bypass capability over the entire break spectrum. Recently, a series of discussions have been held which discuss the actual capability of the ABWR containment to withstand suppression pool bypass as it relates to design basis accidents. This summarizes that information.

First the results of the transient bypass capability calculation are presented to determine the maximum allowable effective area of the leakage path assuming that the vessel is depressurized at rate of 100 F/hr. This calculation was performed in accordance with SRP 6.2.1.1.C. The results of this analysis are compared to the results of similar analysis for a Mark II design. An analysis is then presented which considers the capability of the design to withstand bypass assuming the operator uses the wetwell sprays to limit the peak containment pressure. The potential for increase in the system capability by modifying the spray flow rates is also presented. A discussion is provided which describes the limits of a credible scenario which could result in a stuck open vacuum breaker and discusses the peak containment pressure which could result from such an event. Finally, a brief discussion is provided which compares the effective area to a physical area.

TRANSIENT BYPASS CAPABILITY ANALYSIS FOR VESSEL COOLDOWN SCENARIO

Transient bypass capability analysis for ABWR design, in accordance with the SRP Section 6.2.1.1.C, was performed to determine maximum allowable area of the leakage path for ABWR for the condition where the operator only takes action to cool the vessel at the rate of 100 F/hr. In the event of an accident with substantial bypass, the operator would be instructed to initiate sprays or cool the vessel more rapidly as needed to minimize containment pressurization. Analysis results were compared with similar analysis results for Mark II designs.

INTRODUCTION

The concept of the pressure suppression reactor containment is that any steam released from a pipe rupture in the primary system will be condensed by the suppression pool and will not have an opportunity to produce a significant pressurization effect on the containment. This is accomplished by channeling the

steam into the suppression pool through a vent system. If a leakage path were to exist between the drywell and the wetwell air space, the leaking steam would produce undesirable pressurization of the containment.

Large primary system ruptures (i.e. large break LOCAs) generate high pressure differentials across the assumed leakage path which, in turn, gives proportionately higher leakage flow rates. However, large primary system breaks also rapidly depressurize the reactor and terminate the blowdown. Once this has occurred, there will no longer be a pressure differential across the drywell leakage path, so the leakage flow and containment pressurization will cease. Since leakage into the wetwell is of limited duration, the maximum allowable area of the leakage path for large break LOCAs is expected to be large.

Small primary system ruptures, on the other hand, will result in low pressure differential across any leakage path. However, smaller breaks are expected to result in an increasingly longer reactor blowdown period which, in turn, results in longer duration of the leakage flow. The limiting case is a sufficiently small primary system break which will not automatically result in reactor depressurization. For this case it is assumed that the response of the plant operator is to shut the reactor down in an orderly manner at 100 F per hour cooldown rate. This would result in the reactor being depressurized and the break flow being terminated within approximately 6 hours. During this 6-hr period, the blowdown flow from the reactor primary system would have swept all the drywell air over into the wetwell air space. The blowdown steam would be condensed in the suppression pool. But, in order for this to occur, the water level in the vertical vents would have to remain depressed to the top of upper row of vents. This continuous pressure differential, combined with a 6-hr duration, will result in the most severe drywell leakage requirement.

ANALYSIS

The maximum allowable leakage path area was determined based on the most limiting accident scenario described above. Accordingly, it was assumed that plant operator initiates and completes a normal plant shutdown (100 F/hr) in 6 hours, and there is continuous bypass leakage over the entire 6-hr period. A stratified atmosphere model, which assumed only steam flows through the leakage paths, was used to ensure a conservative result. Credit for structural heat sinks, and actuation of wetwell spray was not taken.

Simplified end-point calculations were done to determine maximum allowable area of the leakage path. Key steps included in this procedure are:

1. Compute, M_{NC}, mass of noncondensable gas initially in the drywell and the wetwell air space.

2. Compute, ΔP_V, pressure difference between drywell and wetwell air space needed to keep top vent cleared.

3. Compute, P_{WM}, the maximum allowable pressure in the wetwell air space.

$$P_{WM} = P_{DES} - \Delta P_V ,$$

where P_{DES} is the containment design pressure.

4. Compute $(P_{WM})_{AIR}$, and $(P_{WM})_{STEAM}$ components of P_{WM}. Assume that wetwell airspace temperature equal to accident maximum pool temperature, and complete carryover drywell noncondensable gas into the wetwell air space.

$$P_{WM} = (P_{WM})_{AIR} + (P_{WM})_{STEAM}$$

5. Compute, M_S, mass of steam corresponding to $(P_{WM})_{STEAM}$. This defines allowable steam bypass leakage mass into the wetwell air space.

6. Define the leakage path flow rate of steam, M_{dot}, as follows:

$$M_{dot} = (A/K)\sqrt{(2g_c(\Delta P_V)/v)},$$

where v is the drywell steam specific volume.

7. Compute the maximum allowable leakage path effective area, A/K, as follows:

$$M_{dot}\ \Delta t = M_S\ ,$$

where t = 6 hours

Therefore, $A/\sqrt{K} = [(M_S)/(\sqrt{\{2g_c(\Delta P_V)/v\}}\,\Delta t)]$

RESULTS

Using the procedure outlined above, the calculated maximum allowable leakage path effective area for ABWR is found to be 0.005 ft². Maximum allowable leakage path effective areas as calculated for typical Mark II designs are found to be 0.003 ft². This difference between the ABWR and Mark II values can, primarily, be attributed to the WW air space to DW volume ratio parameter. A higher volume ratio is expected to result in an improvement in maximum allowable leakage path area. In comparison, this volume ratio parameter is roughly 0.8 for ABWR; and 0.7 for Mark II.

IMPACT OF WETWELL SPRAY OPERATION ON BYPASS CAPABILITY

The design basis calculations for the ABWR are based on operator action to operate the wetwell sprays half an hour after initiation of the event. Design basis calculations for Mark II plants are typically based on 10 minute operator action times. Thus, the design basis for the ABWR is more stringent than that for Mark IIs. The design basis calculations have been previously transmitted to the staff and were discussed in meetings during October and December 1991.

INTRODUCTION

The analysis presented above does not take credit for the operator to take action to reduce the rate of containment pressurization. Rather, it considers the cooldown of the vessel at 100 F per hour as the only means of terminating the bypass event.

However, were a bypass event to occur, the containment pressure would rise, and the operator would be instructed to initiate containment sprays.

In the following sensitivity study, the impact of the timing of spray initiation on the bypass capability was examined. Also considered as part of this study was the impact of wetwell spray flow rate on the capability. Finally, design considerations associated with increasing the wetwell spray flow rate were considered.

ANALYSIS

The SUPERHEX code was used to determine the maximum bypass area which could be accommodated for a given time of spray initiation on spray flow rate. The bypass area was specified by requiring the peak containment pressure to be below the design basis pressure of 45 psig. The spray initiation time was varied from the design basis value of 30 minutes to approximately 250 seconds, the earliest time at which the sprays would be initiated if the operator followed the emergency operation procedures. An additional calculation was also performed in which the wetwell spray flow rate was increased to its maximum value.

BASIS FOR INCREASED WETWELL SPRAY FLOW RATE

In order to determine the maximum flow conditions available for wetwell spray, the design options available for the RHR system were investigated. It was assumed that the wetwell spray header pipe size would not be altered. The flow rate for the spray system is then limited by the maximum fluid velocity in the pipe. Using the same maximum fluid velocity as that used in the design of previous BWR designs (35 ft/sec), the maximum spray flow rate is 1170 gpm.

There are, however, some design and operational challenges associated with the higher flow rate. If the drywell spray flow rate is retained, the combined operation of wetwell and drywell sprays is not possible. However, the full rated flow of 4200 gpm is required to ensure adequate decay heat removal in all modes of RHR operation. To accomplish this function a multi-step initiation procedure would be required:

1. Initiate pool cooling with the normal rated flow rate (4200 gpm)

2. Throttle the pool flow using the heat exchanger throttle valve to reduce the flow to about half the rated flow (2000 gpm)

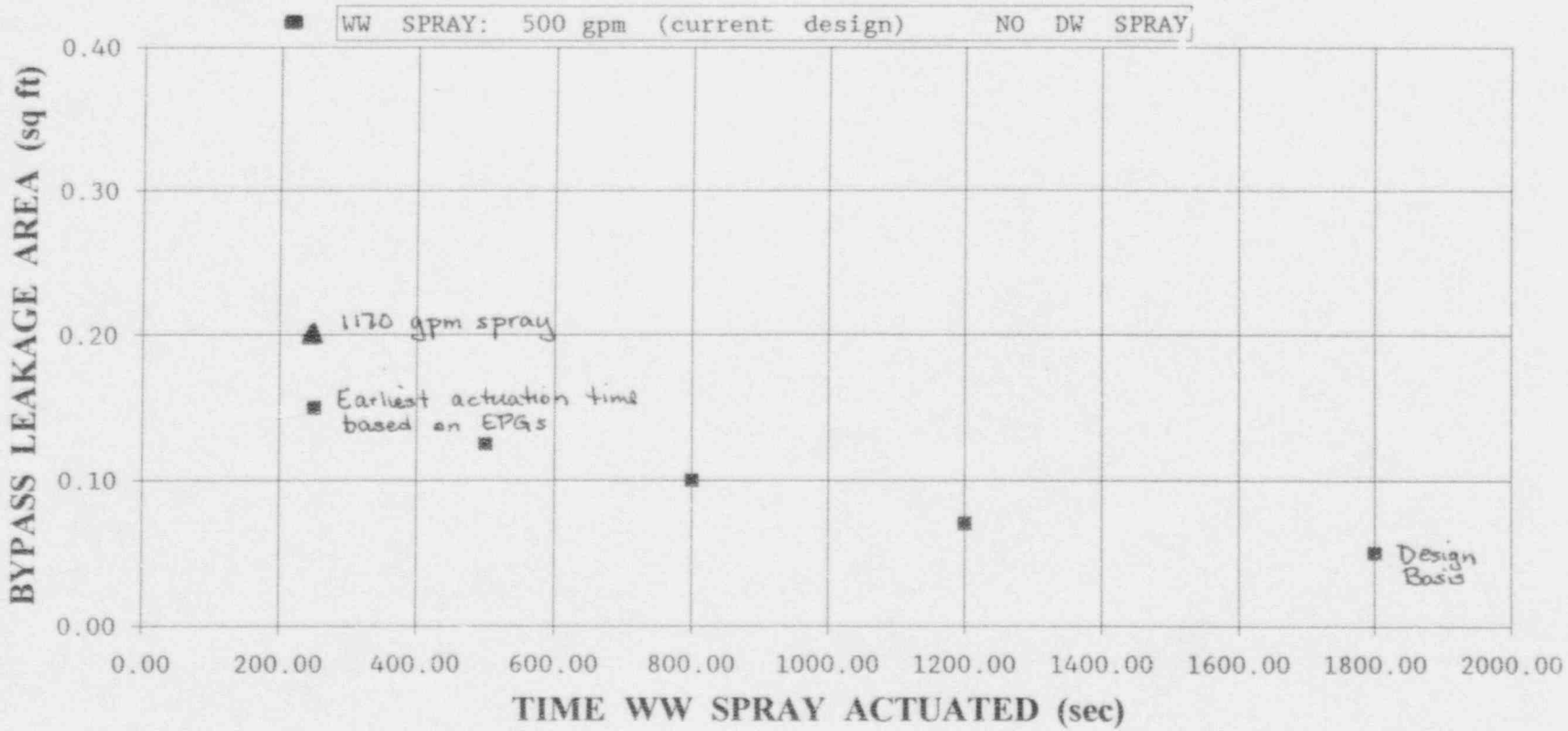
3. Initiate the wetwell spray, adding about 1000 gpm to the total flow rate

4. Bring the combined wetwell spray and pool cooling to the full rated level of 4200 gpm.

RESULTS AND CONCLUSIONS

The results of the SUPERHEX evaluations are shown in the following figure. The analysis indicates that the bypass capability of the ABWR design is 0.15 ft^2, assuming only that the operator initiates wetwell sprays on high containment

ABWR STEAM BYPASS LEAKAGE CAPABILITY



pressure as specified by the EPGs. While not analyzed here, the bypass capability could be further increased if credit is taken for the drywell spray function. If the wetwell spray flow rate is increased as specified in the above discussion, and a 250 second spray initiation time is assumed, the bypass capability rises to 0.2 ft^2.

There is significant complexity added to the operations of the plant and a significant reduction in operational flexibility is the wetwell spray flow rate is increased beyond its current design value. Therefore, while an increased flow rate could somewhat increase the bypass capability, the design change does not appear to be a net improvement to the design.

CONSIDERATIONS FOR A STUCK OPEN VACUUM BREAKER

INTRODUCTION

ABWR design includes vacuum breaker valves which open passively due to negative differential pressure (WW airspace pressure greater than the DW pressure) across the valve. These valves require no external power for their actuation, and closure force is gravity. These valves are installed horizontally locating in WW air space, one valve per penetration (through pedestal wall) opening into lower drywell. Therefore, there are no significant pool swell loads. The vacuum breakers are not expected to open during reactor blowdown period under all LOCA conditions, with the exception of break in one of the two main feedwater lines. The challenges associated with this event are described below.

Following a double-ended rupture in any of the feed water lines, the feedwater system side of the flow will enter directly into the drywell. Earlier analysis results have indicated a rapid decrease in the drywell pressure due to cold feedwater flow into the drywell. This decrease in drywell pressure creates negative differential pressure condition (roughly for about 20 seconds) in the drywell causing vacuum breaker valves to open. The valves will close after the negative differential pressure condition in the drywell has ceased.

Though it is considered to be a highly unlikely event, it is postulated that one vacuum breaker valve may stick open after it is first called upon to open. This postulated condition will result in creating a 20-in steam bypass leakage path. Analyses were performed to evaluate severity of this highly unlikely event. Results indicate that this postulated event will not impact ABWR containment structure integrity.

ANALYSIS

ABWR ECCS configuration includes three independent divisions:

Division A: RCIC + 1 RHR(LPFL) (RHR injects through FWL A)

Division B: 1 HPCF + 1 RHR(LPFL/SPRAY) (Separate RHR injection nozzle)

Division C: 1 HPCF + 1 RHR(LPFL/SPRAY) (Separate RHR injection nozzle)

There are no interlocks in the RHR system design, and the design permits switching RHR from LPFL to SPRAY mode through operator action.

In the event of a break in FWL A, LPFL mode of Division A will be disabled and, instead, the RHR flow will enter directly into the drywell producing drywell spray type steam condensation effect.

Different cases representing different ECCS configurations, including 1 RHR in WW/DW spray mode, were evaluated. It was assumed that operator will throttle 1 LPFL flow to maintain reactor water level at L8.

RESULTS

Analysis results are summarized in the following table. The peak pressure for the case with no credit for wetwell sprays (Case 1) indicates a peak pressure of 106 psia, marginally exceeding the rupture disk setpoint. Therefore, even with no credit for operator action, the containment will be protected from structural failure. If the operator initiates the spray after one-half hour (Case 2A) the peak containment pressure will just reach the rupture disk setpoint. However, for all cases where the operator takes the actions recommended by the EPGs, these results indicate that containment pressure will not exceed the rupture disk pressure set point (of 105 psia) or service level C, thus assuring ABWR containment integrity. This demonstrates that the ABWR design can accommodate the effects of a credible accident with the failure of a vacuum breaker to reclose.

COMPARISON OF EFFECTIVE AREA TO PHYSICAL AREA

Although the allowable leakage areas and bypass capability are expressed in terms of an effective area, it is instructive to understand the relationship between the effective leakage area and the physical area Bypass paths with leakage areas on the order of that described in the allowable leakage area calculations described above will typically be crack shaped gaps at the penetrations. The leak paths are generally unidentified, however one can make assumptions which should be typical for these pathways.

A through flow loss coefficient for an orifice is 1.2. This value would be conservative for most imagined bypass pathways. If an inlet loss coefficient of 0.5, and exit loss coefficient of 1.5 are also assumed, K will be approximately 3. Typical values for the vacuum breakers range from 2.7 to 2.8. Since the effective area and physical area are related by the square root of the loss coefficient, the use of 3 for K is a good approximation for the ABWR. This value of K is also consistent with that used in Mark II and Mark III studies.

TABLE 1: ABWR STEAM BYPASS LEAKAGE SENSITIVITY STUDY

FWL BREAK EVENT:-- ONE VACUUM BREAKER FAIL TO CLOSE TO CLOSE AFTER IT FIRST OPENED

SUMMARY RESULTS

CASE	Peak Pres (psia)	RHR (Available	HPCF (Available)	RCIC (Available)	Sprays (WW & DW)	REMARKS
1	99	1 LPFL	1 HPCF	RCIC	YES	1 WW Spray, 500 gpm (Current Capacity) @ 250 sec
						1 LPFL Throttled to Maintain L8
2	105	1 LPFL	1 HPCF	RCIC	YES	SAME AS CASE 1, Except Spray Actuation @ 1800 sec
3	96	1 LPFL	1 HPCF	RCIC	YES	1 WW Spray, 1140 gpm (Operator Control) @ 250 sec
						1 LPFL Throttled to Maintain L8
4	94	1 LPFL	1 HPCF	RCIC	YES	1 WW Spray, 500 gpm AT 250 seconds
						1 DW Spray, 3700 gpm (Current Capacity)
						1 LPFL Throttled to Maintain L8
5	92	2 LPFL	2 HPCF	RCIC	NO	1 LPFL Throttled to Maintain L8

Containment Ultimate Pressure

The issues regarding the containment ultimate pressure capability are being addressed by Gary Ehlert. A discussion was held by phone on April 28, 1993. As a result of this discussion GE intends to submit the following information to the NRC:

1. Summary Report on concrete tensile strength reduction from 100 psi to 10 psi (preliminary comparison provided).

2. FINEL comparison to test dta for SIT test.

3. Linear tearing study.

We anticipate submitting this information by May 31.

Suppression Pool pH Control

The following material will be incorporated into the SSAR as Subsection 19E.2.6.14.

> The chemical form of iodine may be affected by the acidity of the suppression pool. If the pool becomes acidic (pH < 7) the formation of volatile and organic forms of iodine may be enhanced. Experiments have indicated the potential for the radiolytic formation of nitric acid (HNO_3) in the suppression pool. This can then lead to the conversion of I^- to I_2 in the pool. The gas species remain in equilibrium with the I_2 in the pool, with the relative amounts governed by a partition fraction between the water- and gas-borne species. Reference A states, "If the pH is controlled so that it stays above 7, a reasonable value for the I- converted to I2 is 3.E-4. ...[Calculation] indicates a small production of volatiles for PWRs but virtually none for BWRs".
>
> Calculations were performed following the methods of Reference A to determine the potential for the formation of nitric acid to lead to an acidic suppression pool. These sequences differed by consideration of varying initial suppression pool pH, caused by the transport of CsOH to the pool as a result of the accident. In each calculation, the pH of the pool is monitored over time as nitric acid is formed radiolytically. The results of two calculations which bound the expected transfer of CsOH to the pool are given below. In the both cases, the transfer of CsI to the pool is assumed to be the same as that for CsOH.

Initial CsOH fraction in pool	10%	80%
Time (hr)	pH	pH
0	9.65	10.56
1	9.65	10.56
10	9.63	10.53
24	9.59	10.49

The results of these calculations indicate that the pH of the suppression pool will not drop to the acidic range within 24 hours of accident initiation. Therefore, nitric acid formation due to radiolysis will not have a significant impact on the source term. No further consideration of this phenomenon is necessary.

Reference A: NUREG/CR-5832, "Iodine Chemical Forms in LWR Severe Accidents", Final Report, January 1992.

Hydrogen Burning or Detonation in the COPS system

After the discussions of April 26, GE considered the plant costs associated with the design of piping to withstand the potential hydrogen detonation. Based on this study, we have altered our approach to the resolution of this issue. The following material will be added to the SSAR as Subsection 19E.2.8.1.6. The current section of this number (Summary) will be renumbered to 19E.2.8.1.7.

19E.2.1.6 Potential Impact of Hydrogen Burning and Detonation

Hydrogen burning and detonation are not a concern for the ABWR containment because the containment is inerted with hydrogen. There could be a potential for burning in the COPS system and the stack after the rupture disk opens. However, due to the design and operation of the COPS system, this issue does not have an impact on risk.

Hydrogen burning and detonation will be precluded in the piping associated with the COPS system. The piping will be inerted during operation with rupture disk located at the inlet of the stack. This, combined with initial purging of the piping, will ensure that the inertion of the containment will extend out to the stack, and prevent burning of hydrogen in the portion of the COPS system which is within the reactor building. Therefore, there will be no concern of the leading edge of the containment atmosphere mixing with the gas in the piping and causing a burn. After passing of the leading edge of the gas flow, the mixture in the piping will be identical to that in the containment. The gas flow through the system will prevent the backflow of air into the COPS piping.

Hydrogen burning could occur in the plant stack as the gas flow enters the stack. The stack is a non-seismic structure located on top of the reactor building. Because of this configuration, the reactor building has been designed to withstand the loads associated with the collapse of the plant stack. Furthermore, no credit is taken in the analysis for the plant stack to reduce the offsite dose by providing for an elevated release. All releases were presumed to occur at the elevation of the top of the reactor building. Therefore, hydrogen burning or detonation in the stack will have no impact on the consequences of a severe accident as modeled in this analysis.

No burning will occur within the COPS piping. Furthermore, no credit was taken for the plant stack to reduce the source term to the environment and the reactor building can withstand the collapse of the plant stack. Therefore, hydrogen burn or detonation in the COPS system will have no impact on risk and no further consideration of this phenomenon is required.

Hydrogen Burning or Detonation in the Reactor Building

Hydrogen burning and detonation are not a concern for the ABWR containment because the containment is inerted with hydrogen. However, there is a potential for burning in the reactor building in the event of an unisolated LOCA outside containment which leads to a severe accident.

In examining the potential impact of hydrogen in the reactor building, it is important to understand that this is only a concern for accidents in which a severe accident occurs with direct path to the reactor building. Hydrogen is not generated unless there is severe fuel damage. Therefore, the impact of hydrogen generation does not affect the analysis of the frequency of core damage with bypass.

No credit is taken in the source term or consequence analysis for the presence of the reactor building in reducing the dose associated with a core damage event. The only affect the reactor building can have on the analysis is through the assumptions for system recovery. While credit was taken for recovery of systems in the base analysis, no recovery was considered in the bypass study of Subsection 19E.2.3.3. Therefore, the conclusion of the analysis is not affected by potential burning or detonation in the containment: containment bypass sequences will contribute less than 10% to plant risk.

Impact of FMCRD Platform Grating

A concern that the grating could have an impact on fuel coolant interactions and debris coolability was raised during the March 18, 1993 ACRS meeting. GE does not believe there is any potential for negative impact of the grating on the analyses of these phenomena. The following information will be added to the sections of the Appendix 19E as indicated below.

19EB.4.1.1 Impact of FMCRD Platform Grating [on FCI]

The FMCRD platform grating is located in the lower drywell at the elevation of the access tunnel. This rotating platform is circular and mounted on the

rotating rail under the reactor vessel. There is an opening area at the center of the platform which is provided with traveling rail for the CRD handling device. Gratings will be installed on both sides of the rail for maintenance personnel. Typically, the grating consists of 1-inch by 3/8-inch metal slats mounted edge-wise to form a grid with a grid size on the order of 1-inch by 2-inch.

The presence of the grating could provide some increased fragmentation of the debris as the leading edge of the debris enters a pre-existing water pool. This will tend to increase the voiding of the pool. Because the structure of the grating is very open, there will be no significant limitations on the venting of steam generated below the grating. Increased voiding in the water pool will reduce the impulse loading from an FCI. This in turn will decrease the potential for early containment failure from FCI.

The grating will be ablated as the debris passes through it, in the same manner as the ablation of the bottom head. Therefore, the grating will have no impact on the severe accident performance after the initial debris relocation. Any late debris relocation would be a slow, drip-like relocation which would fall straight through the ablated region of the platform.

19EC.6.2 Impact of FMCRD Platform Grating [on Debris Coolability]

The FMCRD platform grating is located in the lower drywell at the elevation of the access tunnel. This rotating platform is circular and mounted on the rotating rail under the reactor vessel. There is an opening area at the center of the platform which is provided with traveling rail for the CRD handling device. Gratings will be installed on both sides of the rail for maintenance personnel. Typically, the grating consists of 1-inch by 3/8-inch metal slats mounted edge-wise to form a grid with a grid size on the order of 1-inch by 2-inch.

However, it is expected that the grating will quickly ablate due to the flow of debris. This is much the same as the ablation of the vessel bottom head as the debris leaves the vessel. Any late debris relocation would be a slow, drip-like movement which would fall straight through the ablated region of the platform. Therefore, debris will not be retained above the platform and there will be no impact on containment performance.

Equipment Survivability

In order to ensure the operability of the equipment during a severe accident as modeled in the ABWR PRA, a review of the severe accident analysis was undertaken to identify all equipment considered in the analysis. The environmental conditions for each piece of equipment were then identified. These conditions were then compared to the equipment qualification standards. The equipment qualification standards were not used as a strict measure. Rather, they were used to provide a measure of confidence that the equipment would survive the expected conditions.

Depressurization System

During a core damage event, the SRVs must be able to remain open during the in-vessel phase of the accident to ensure that any potential vessel failure occurs at low pressure. After vessel failure, SRV operability is not required. The depressurization capability will not be degraded due to radiation exposure because the design basis radiation exposure for the SRVs is higher than that predicted using best estimate severe accident methods. In addition, the thermal loads on the elastomers in the valve actuator, which limit the temperature capability of the valve, will be similar to that used for equipment qualification of the SRVs. Therefore, the operability of the valves will not be degraded by the conditions in the containment.

The SRVs are held open with a nitrogen actuator. The nitrogen supply is located outside of containment. As discussed in Subsection 19E.2.1.2.2 of the SSAR, the nitrogen supply will be adequate to assure SRV operability over a full range of hypothetical accidents.

Residual Heat Removal System (RHR)

The RHR system may be called upon to remove decay heat from the containment during a severe accident. Either shutdown cooling mode or suppression pool cooling mode may be used. The pressure and temperature of the suppression pool are not expected to exceed the system design pressure and temperature during any accident sequence. Shutdown cooling will only be used when the RPV is at relatively low pressures which are below the capability of the RHR system, approximately 1.0 MPa (135 psig).

The integrated radiation exposure to the RHR equipment will not reach the qualification limits for several days after system initiation during a severe accident. The RHR control system is outside of the containment and will not be significantly affected by a severe accident. The integrity of the system piping, spray headers and injection headers within the containment will not be adversely affected during an accident.

Firewater System

The firewater system may be called upon to inject water into the vessel or through the drywell sprays during a severe accident. The system is manually initiated. All flow in the system is from outside the containment. Thus, accumulation of radioactive material in the firewater pumping system will not occur. All components of the firewater system are outside of the containment and will not be significantly affected during a severe accident. Inside the containment, the firewater system utilizes RHR piping and spray headers which are not adversely affected by a core melt event.

Passive Flooder

The passive flooder may be needed to provide a water flow path from the suppression pool to the lower drywell after vessel failure. The flow path is opened as a direct result of high temperatures in the lower drywell which occur after debris relocation

from the vessel. This system does not contain any active systems, instrumentation or controls. Additionally, the system components are not hindered from performing their functions due to high radiation levels which might exist in the lower drywell after debris relocation from the vessel. Therefore, the system is expected to operate under the required conditions.

Containment Overpressure Protection System (COPS)

The COPS may be needed during a severe accident to relieve high containment pressure. The system contains piping, a rupture disk and two valves which are normally open and fail open. To relieve containment pressure, the rupture disk must burst. Activation will not be adversely affected by the temperature and radiation in the wetwell airspace during a severe accident. The sensitivity of rupture disk activation to wetwell temperature is discussed in Subsection 19E.2.8.1.2.

Vacuum Breakers

The vacuum breakers may need to open during a severe accident to relieve high differential pressures between the wetwell and drywell. Vacuum breakers are passive in nature and have no instrumentation and control other than position indication. Since the vacuum breakers are located on stub tubes high in the wetwell airspace, they will not be subjected to pool swell loads. There is no direct means for debris to reach the vacuum breakers. Therefore, they are not expected to be adversely affected during a severe accident.

RIP Vertical Restraints

The vertical restraints on the RIPs prevent the pumps from being ejected if the RIP attachment welds are destroyed during a core melt event. The restraints are attached to the outer vessel surface and do not experience the severe conditions within the vessel during core melting. Therefore, the integrity of the vertical restraints is not jeopardized.

Containment Isolation Valves

The containment isolation valves are expected to remain closed during a severe accident. The pressure capability of the isolation valves does not limit the containment response to overpressurization events. The radiation exposure to which containment isolation valves are designed to are higher than that which is expected to occur during a severe accident. Thus, the isolation valves will not fail during a severe accident.

Vessel Water Level Instrumentation

A severe accident can be terminated in-vessel if water injection to the vessel is recovered prior to vessel failure. For these sequences, the water level inside the vessel should be monitored to maintain core coverage. The ABWR contains RPV water level instrumentation which provides indication of conditions leading to inadequate core cooling, see Subsection 1A.2.16 of the SSAR "Identification of and Recovery From Conditions Leading to Inadequate Core Cooling [II.F.2]".

Containment Water Level

Precise water level determination in the containment is not necessary. However, the operator must monitor the approximate water level in order to avoid overfilling the containment and potentially damaging the COPS system. Level sensors in the wetwell will provide adequate indication of containment water level. The level sensors in the wetwell are expected to be operable during the entire course of a severe accident because the benign conditions in the suppression pool. The sensors located in the lower drywell above the floor may also be operable. Thus, the operator will be provided with indication to allow termination of a severe accident involving debris relocation from the vessel.

Drywell Temperature Instrumentation

If the drywell temperature exceeds the degradation temperature of the elastomers used in the containment penetrations, excessive containment leakage could occur. This can be prevented by use of the drywell sprays. Drywell sprays are manually initiated upon indication of high drywell temperature. Control of the drywell temperature so as to prevent excessive leakage should ensure temperature sensor operability. Further, the radiation levels in the containment will be below the design basis for the instrumentation. If the sensors may go off scale, drywell spray initiation will be indicated.

Severe Accident Suppression Pool Bypass

Questions have been raised concerning the applicability of the Morewitz plugging correlation to the vacuum breakers during hypothesized severe accident conditions in the ABWR. The following discussions will be added to the SSAR to address the technical bases for the aerosol plugging model and the potential impact of uncertainties in the model on the consequence analysis.

Addition to Subsection 19EE.3.1

The applicability of the Vaughn Plugging Model to the conditions of the vacuum breakers was examined by consideration of various test data provided by Morewitz (Reference 3). The data surveyed includes a variety of experiments involving orifices as well as pipes. The data for orifice plugging indicates that the plugging coefficient is comparable to that for small piping.

Morowitz also discusses the impact of steam on the plugging of leakpaths. He indicates test data which indicated that "leak paths quickly plugged when steam was introduced in the containment atmosphere". He also notes that densification effects such as condensation of water on hydroscopic deposits could increase the rate of plug formation. In the ABWR, hydroscopic CsOH particles form a significant fraction of the aerosol. A large portion of the aerosol mass is expected to be made up of tin (Sn) which is released during the core degradation phase of the accident. Tin is insoluble in water (Reference 4) and therefore, any plug created with tin would not be expected to be affected by the presence of steam. If continued core-concrete interaction

is predicted, a major contributor to the aerosol mass would be SiO2 which is also insoluble in water (Reference 4).

Most of the experimental evidence cited by Morowitz involves systems with very high pressure differences across the plug. For example, in the orifice test data noted above, the differential pressures ranged from 30 to 1000 psid. Morowitz indicates that a "either solid or porous plugs formed" in these experiments. Reference 3 also describes a test of a small, concrete, tilt-up-panel building at Atomics International in the early 1960's. The building was overpressurized and cracked so that it leaked badly. In order to plug the leaks, a sodium fire was lit inside the building and observers were stationed around the outside. No smoke was seen issuing from the building. Upon pressure testing of the building, no gas leaks could be detected. Reference 3 describes several other situations with lower pressure differences in which termination (or significant reduction) in gas flow rates was observed. In the ABWR the maximum pressure difference across the plug will be limited to the head of water above the first row of horizontal vents (about 3 psid assuming normal water level). Therefore a complete blockage is expected. Any small gas leakage would have an insignificant affect on the wetwell pressurization.

References:

3. "Leakage of Aerosols from Containment Buildings", H.A. Morewitz, Health Physics Volume 42, Number 2, February 1982.

4. Handbook of Chemistry and Physics, 53rd edition, CRC Press, 1972-1973.

19EE.4.4 Impact of Aerosol Plugging on Integrated Offsite Risk

The quantification of the environmental source term presented in the decomposition event trees considers a substantial reduction in the frequency of bypass release due to plugging of the release path as described in Subsection 19EE.3.1. There is some uncertainty associated with the gas permeability of the plug under these conditions which may affect the source term timing and magnitude. In order to asses the affect of this uncertainty, a sensitivity study has been performed.

The impact of aerosol plugging on offsite risk can be assessed by eliminating the credit taken for plugging in the suppression pool bypass DET (Figure 19EE.2-1) and recalculating the source term release category frequencies in Figure 19D.5-3. The change in source term category frequencies is propagated into the consequence analysis as described in Subsection 19E.3. The results of eliminating the credit taken for aerosol plugging in the suppression pool bypass DET are listed below.

DET branch probability	w/ plug credit	w/o plug credit

No Pool Bypass	0.9777	0.8168
Pool Bypass *	0.0223	0.1792

* includes large and small leaks

The results of recalculating the frequencies for the affected STC#s are shown in Table 19EE.4-1. Events with frequencies less than E-11 do not significantly contribute to the offsite risk. Therefore, only STC#s 6, 8, 18, 19 and 30 are risk significant.

The significant STC#s are binned in Table 19D.5-7 into Case 1, Case 7 and Case 8 for the consequence analysis performed in Subsection 19E.3. The total frequencies of these two cases must be re-evaluated using the frequency changes in Table 19EE.4-1. The base case frequencies appear in Table 19E.3-6. A comparison of the base case frequencies (with plugging credit taken) and the case frequencies without plugging credit is summarized below.

	Frequencies (events/year)	
Case Number	w/ plugging credit	w/o plugging credit
1	2.08E-08	1.96E-08
7	3.91E-10	1.05E-09
8	4.05E-10	9.21E-10

The impact of neglecting credit for aerosol plugging on offsite risk is presented in Figure 19EE.4-1. Probabilities of exceedence less than 1E-9 are considered insignificant in the consequence analysis. Divergences in the 1E-8 range are well within the level of confidence of the consequence study. Therefore, while aerosol plugging can significantly reduce the amount of fission products released from the containment given a bypass event, the phenomenological assumption has negligible impact on offsite risk.

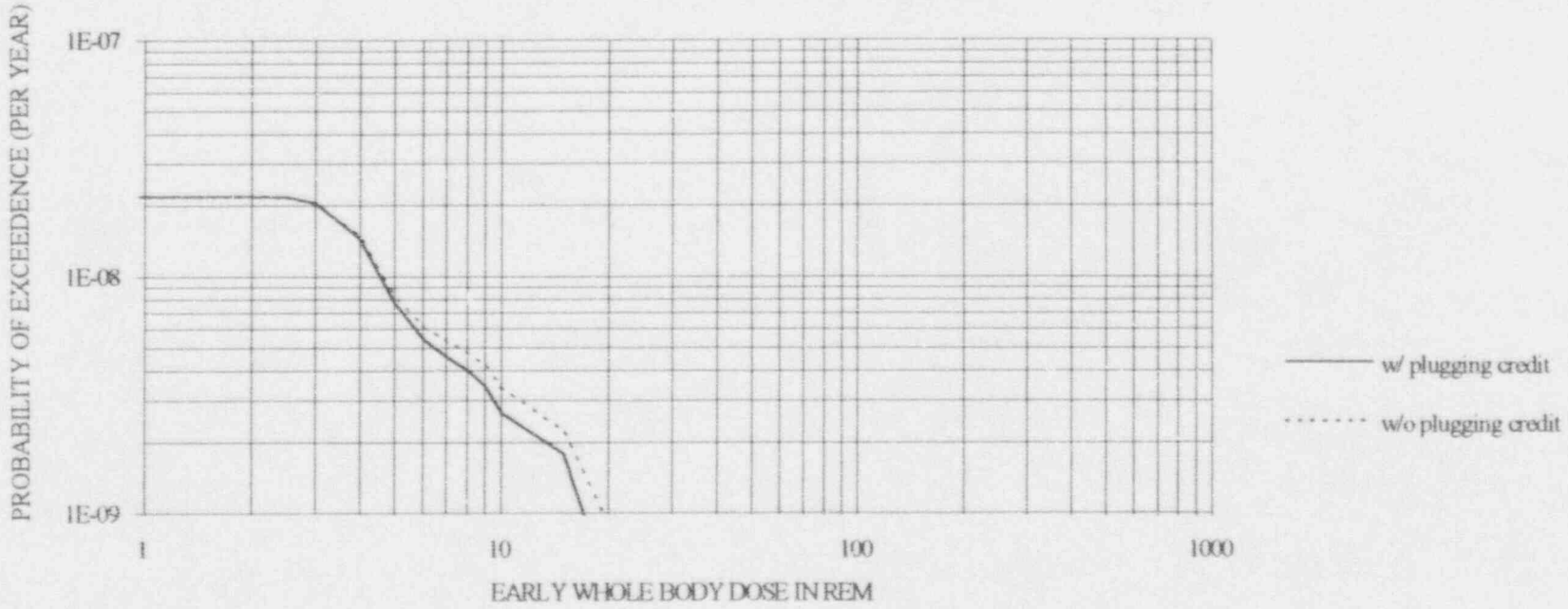
Table 19EE.4-1

EFFECT OF ELIMINATING AEROSOL PLUGGING CREDIT ON SOURCE TERM CATEGORY FREQUENCIES

STC # *	Frequency (w/ plugging credit)	Frequency (w/o plugging credit)
6	3.29E-09	2.75E-09
7	8.87E-13	7.13E-12
8	7.33E-11	5.89E-10
18	4.10E-09	3.43E-09
19	9.43E-11	7.58E-10
20	8.73E-13	7.02E-12
30	1.59E-11	1.33E-11
31	3.59E-13	2.88E-12
32	1.05E-14	8.44E-14
42	5.22E-12	4.36E-12
43	1.20E-13	9.64E-13
44	9.05E-16	7.27E-15

* see source term category grouping diagram, Figure 19D.5-3

Figure 19EE.4-1 IMPACT OF AEROSOL PLUGGING CREDIT ON OFFSITE RISK MEASURED BY WHOLE BODY DOSE AT 1/2 MILE AS PROBABILITY OF EXCEEDENCE



19ED.2 ~~PROPOSED~~ CORIUM SHIELD DESIGN

A protective layer of refractory bricks—a corium shield ~~could~~ will be built around the sumps to prevent corium ingression. The shield for equipment drain sump ~~would~~ will be solid except for the inlet and outlet piping which ~~would~~ will go through its roof. The shield for the floor drain sump ~~would~~ will be similar except that it must have channels at floor level to allow water which falls onto the LD floor to flow into the sump. The height of the channels ~~would~~ will be chosen so that any molten debris which reaches the inlet will freeze before it exited and spilled into the sump. The width and number of the channels ~~would~~ will be chosen so that the required water flow rate during normal reactor operation is achievable. A sketch of ~~a concept for~~ the floor drain sump shield is shown in Figure 19ED.2-1.

The walls of the equipment drain sump shield (solid shield) only have to be thick enough to withstand ablation, if any is expected to occur, for the chosen wall material. The walls of the floor drain sump shield (channeled shield) must ~~be thicker so~~ enough that molten debris flowing through the channels has ~~enough~~ sufficient residence time to ensure debris solidification.

Both shields ~~would~~ will extend above the LD floor to an elevation greater than the expected maximum height of core debris. Thus, no significant amounts of debris will collect on the shield roofs. The solid shield can be placed directly on top of the LD floor. The channeled shield will have refractory bricks embedded into the LD floor beneath the shield to prevent core-concrete interaction involving the molten debris in the channels.

The analyses presented in Sections 19ED.4 and 19ED.5 provide a basis for sizing the ~~proposed design~~ ~~of the~~ floor drain sump corium shield.

19ED.3 SUCCESS CRITERIA FOR ~~PROPOSED DESIGN~~ THE CORIUM SHIELD

For the proposed design to be considered successful, it must satisfy the following requirements:

(1) Melting Point of Shield Material Above Initial Contact Temperature

The shield wall material shall be chosen so that its melting temperature is greater than the interface temperature between the debris and the shield wall.

(2) Channel Length

The length of the channels in the shield must be long enough to ensure that a plug forms in the channel before debris spills into the sump. The freezing process is expected to take on the order of seconds or less to complete.

(3) Shield Height, H_{uw}, Above Lower Drywell Floor

The shield height above the lower drywell floor shall be chosen to ensure long term debris solidification. The freezing process will be complete during the time frame when the shield walls are behaving as semi-infinite solids. In addition, the shield must be tall enough to prevent debris from accumulating on the roof of the shield.

(4) Shield Depth, H_{lw}, Below Lower Drywell Floor

The shield depth ~~of the~~ below the lower drywell floor shall be chosen to ensure long term debris solidification.

(5) Water Flow Rate

The total flow area of the shield channels shall be great enough to allow water flow rates stated in the Technical Specifications regarding ~~was~~ water collection during normal operation without causing excessive water pool formation in the lower drywell.

(6) Chemical Resistance of Shield Walls

The wall material chosen for the corium shields must have good chemical resistance to siliceous slags and reducing environments. Resistance can be determined to a first degree by comparing the Gibb's free energy of the oxides which make up the shield wall and the oxides present in core debris.

(7) Seismic Adequacy

The seismic adequacy of the corium shields will be determined in the detailed design phase. Adequacy should be easily meet because the shields are at the lowest point in the containment. Missile generation is not an issue because the shields are not near any vital equipment.

Section 19ED.6 contains an example of success calculations for requirements (1) through (4) for a chosen channel height of 1 cm.