GE Nuclear Energy

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February 10, 1992

MFN No. 034-92
Docket No. STN 50-605
EEN-9220

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Robert C. Pierson, Director
Standardization and Non-Power Reactor Project Directorate

Subject: **GE Responses to the Request for Additional Information on ABWR Design for Severe Accidents**

Reference:
1. Letter, Chester Poslusny to P.W. Marriott, "ABWR Design for Severe Accidents," dated January 10, 1991, MFN No. 011-92
2. Letter, Chester Poslusny to P.W. Marriott, "ABWR Design for Severe Accidents," dated January 16, 1991, MFN No. 018-92

Enclosed are thirty-four (34) copies of the GE responses to the Reference 1 and 2 requests for additional information. These responses reflect the discussions at the January 28-29, 1992 GE/NRC meeting in San Jose, California.

A copy of this material was telecopied to the NRC on February 7, 1992.

Sincerely,

P.W. Marriott

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cc: F. A. Ross (DOE)
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Response to Request for Additional Information

Pertaining to ABWR Design for Severe Accidents

General:

There appear to be typographical errors in the RAI. There are also instances where nomenclature was used which is inconsistent with that used for the ABWR. In the following statements of the questions italics have been used to indicate where word changes have been made to better understand the question. Also, the four questions on accident management have been renumbered as Questions 18 to 21

Question 1:

The staff has used the guidance provided in SECY 90-016 to determine the degree of compliance with the Commission's Severe Accident Policy Statement for Evolutionary Designs and specially the ABWR design. Since the Commission has approved the guidance in the SECY, it represents one of the more important regulatory guides. In SECY 90-016 the staff concluded one method available to the designer to minimize the rapid containment pressure rise due to a High Pressure Core Melt Ejection was to have a reactor cavity design with features to contain ejected core debris. Within the ABWR documentation, there are commitments to meet the reactor cavity design criteria as specified in SECY 90-016. However, since the guidance provided in the SECY paper is quite general in nature, it is unclear as to how the ABWR design has achieved the goals. Therefore, so that the staff can understand how the ABWR design meets the above criteria, the following additional information is requested.

Provide scaled engineering type drawings of the *lower drywell*. The drawings should be sufficiently complete so as to show the following design details.

a. General dimensions of the *lower drywell* internals including the size and type of *reactor* vessel insulation

b. Location and size of any ledge-like surfaces

c. Location and configuration of all penetrations

d. Location and configuration of all openings to the *upper* drywell compartment

e. Size and configuration of the *lower drywell* floor

f. Size, elevation, and configuration of the floor vents

Response 1:

The primary means of preventing direct containment heating in the ABWR is the highly reliable depressurization system. The reliability and capability of this system are the subject of Question 16. The ABWR design has extraordinary ability to prevent severe accidents if the vessel is depressurized. This results in a very low total core damage probability. Thus, although the frequency of high pressure core melt events for the ABWR is small, these events comprise a significant fraction of the core damage events.

The arrangement elevation drawing of the ABWR containment and reactor building is shown in the ABWR SSAR Figures 1.2-2 and 1.2-2a. The lower drywell elevation is shown in Figure 1.2-3b and 1.2-3c. The arrangement plan is depicted in Figures 1.2-13e through 1.2-13h. The vessel skirt of the ABWR is solid, there are no openings in it which could connect the upper drywell to the lower drywell. The only pathway connecting the lower drywell to the upper drywell and wetwell is through the ten drywell connecting vents. The configuration of the drywell connecting vents may be best seen in Figure 3.8-18 of the SSAR.

Since the vessel skirt is solid, there is no flow path alongside the vessel to the upper drywell. Therefore, the details are of the vessel insulation are not necessary to evaluate the retention of the debris in the lower drywell.

Question 2:

Provide a discussion of the specific design features which are felt to be responsible in achieving the goals of SECY 90-016. In addition, provide the basis for each of the design features ability to achieve the goals. Where possible, provide any insights relative to the sensitivity to dimensional changes in maintaining the design objective.

Response 2:

As agreed in discussions between GE and the NRC staff of January 28, 1992, this questions will be subsumed in the discussion of question 15.

Questions 3:

Due to the status of the ABWR design, there may be design features which are important to the overall design but cannot be finalized at this time. For this reason, provide all interface requirements which are felt necessary to assure that all design objectives relative to retaining core debris are met.

Supplement to Question 3:

As agreed in the discussions between GE and the NRC staff on January 23, 1992, the scope of this question has been expanded to include the passive flooder and the rupture disk.

Response 3:

No direct credit is taken for the detailed configuration of the lower drywell in reducing the potential for entrainment out of the lower drywell in the very unlikely event of a high pressure core melt scenario. Therefore, no interface requirements are required relative to the retention of core debris in the lower drywell.

The detailed design of the passive lower drywell flooder has not been performed at this time. The detailed design will have to meet the requirements specified in subsection 9.5.12 of the ABWR SSAR. Any additional insights which may be revealed as a result of the sensitivity and uncertainty analyses now in progress will be included in the interface requirements.

The containment overpressure protection system is described in subsection 6.2.5.2.6. The setpoint of the rupture disk will be modified to account for the increase in drywell head pressure capability. The uncertainties in rupture disk opening pressure and containment failure pressure are being considered in determining the new rupture disk setpoint. This information will be included in a future amendment of SSAR Appendix 19E.2. Insights from that study, such as the acceptable variation in rupture disk opening pressure at a given setpoint, will be included in the appropriate interface document.

Question 4:

Because of the low probability of severe accidents, analytical studies often prove to be an important factor in the overall support of a design feature. In this regard, provide the supporting analytical studies which have been performed to support the design. Of particular interest are the parametric studies performed and the assessment of the results of these studies. Identify those areas that are supported solely by these analytical studies.

Response 4:

The analytical studies which support the assessment of the ABWR performance during a postulated severe accident are contained in subsection 19E.2 of the ABWR SSAR. The calculations shown there in Amendment 10 will be supplemented by the screening, sensitivity and uncertainty analyses as discussed with the staff on January 22, 1992. An additional issue has been identified which will be added to the discussion in 19E.2: the ACRS raised a concern that there may be fission product release due to water carryover when the rupture disk is opened. Calculations indicate that there is no significant increase in the source term due to this mechanism.

Question 5:

Identify those experimental tests that are felt to support the cavity design of ABWR. Show the applicability of each test to the ABWR design. In addition, discuss the test results and show how the results demonstrate the ability of the ABWR design to retain core debris.

Response 5:

As stated in the response to question 16, the ABWR design provides a highly reliable reactor vessel depressurization system to preclude a possible vessel failure at high pressure. In the unlikely event of a high pressure melt ejection, the limited experimental data indicates that the possibility exists for substantial debris dispersal out of the cavity. A detailed investigation into the uncertainties involved in this phenomenon and their impact on the peak drywell pressure expected during a high pressure melt ejection is being incorporated into the on-going ABWR uncertainty analysis. The NRC and ACRS have been provided with presentations regarding this analysis. The experimental evidence which supports the uncertainty analysis and the application to ABWR are summarized below.

Some tests have been performed to investigate the entrainment phenomenon in an ABWR-like cavity. A simulant fluid experiment using Wood's metal was run at Argonne to investigate the dispersal of debris out of a typical Mark III cavity configuration. The results of this experiment are documented in "Hydrodynamic Sweepout Thresholds in BWR Mark III Reactor Cavity Interactions" ANL/LWR/SAF-84-1 by B. W. Spencer, et al..The configuration was somewhat similar to that of the ABWR. The experiment involved a sunken pedestal with openings at higher elevations. The ANL experiment demonstrated very efficient entrainment of debris out of the cavity for high gas velocities. The threshold velocity for entrainment, when scaled to a reactor situation, is exceeded by a reactor pressure vessel depressurizing from approximately 1000 psig. The entrainment phenomenon observed in the experiment was different than that studied for some PWR cavities. Due to the sunken cavity and the presence of debris on the floor prior to the depressurization event, the gas jet exiting the vessel was observed to undermine the debris and levitate the debris particles by a flooding mechanism. Once the debris was lifted to an elevation equal to the openings, the debris was discharged out of the cavity.

There are differences between the ANL test configuration and the ABWR lower drywell which will contribute to the retention or removal of the debris from the flow stream before it reaches the upper drywell. The ANL cavity was a cylinder with large unobstructed openings to represent the CRD openings. The top of the cylinder was flat with little space in which the debris could become trapped. The bottom of the vessel protrudes into the ABWR lower drywell providing a considerable space for the trapping of debris away from the gas flow streams in lower drywell. The flow path from the lower drywell to the upper drywell requires the flow to turn upward into the wetwell drywell connecting vents. The momentum of the debris as it passes into the connecting vents will tend to deposit it, then in order for the debris to be carried into the upper drywell it would have to be re-entrained. Furthermore, after the wetwell vents clear, most of the debris which reaches the vents will be carried downward into the suppression pool.

Question 6:

With respect to core debris coolability, there are a series of MACE tests underway. Several tests have recently been completed. Discuss the applicability of both the test parameters as well as the test results to the ABWR design. If the MACE tests are determined to be inapplicable, provide the basis for the current design and discuss the need for further experimental verification on the GE assumptions of core debris *coolability* in a relatively short period of time.

Response 6:

The issue of debris coolability is being addressed in the ABWR uncertainty analysis. Work performed by IDCOR and the DOE Advanced Reactor Severe Accident Program (ARSAP) indicate that water on core debris will provide for a substantial amount of cooling. However, experiments performed to date have not conclusively proven that the debris will be quenched and any core concrete interaction will be stopped if water is poured on molted core debris ex-vessel.

The MACE test series which is now being performed at Argonne National Laboratory is the latest in a series of core concrete interaction and debris coolability tests which have been performed over the last several years. The MACE tests are more prototypic of the ABWR than any tests performed to date. As yet, only the scoping test and one additional test (M1) have been completed. The M1 test was run late in 1991. Due to problems with the initial configuration of the corium simulant and instrumentation failures, this test was not prototypical for a hypothesized core melt. Final conclusions about that test are not yet available. Preliminary examination of the test section indicates that the UO2 powder did not become fully molten at the top of the test section. Instead, there is a thick layer of scintered material which formed a thick crust over most of the test section. This condition is clearly not prototypic for the ABWR. GE will continue to follow the MACE test series and other similar experiments and apply the lessons learned to the PRA.

Due to this lack of conclusive data, the ABWR uncertainty analysis will address three possible debris-water cooling rates. The first will represent the typical MAAP modelling and will result in upward heat losses of about 1 MW/m2. The SWISS-2 test observed upward heat fluxes of about this magnitude. The second type of heat removal considered is controlled by film boiling at the surface of the debris. The heat transfer rates for this mechanism are typically on the order of 300MW/m2. The third heat transfer rate will attempt to acknowledge the presence of an impermeable crust on top of the debris with an upward heat transfer rate of approximately 100MW/m2. This range of debris cooling rates should provide a means to investigate the possible uncertainties in debris cooling.

The ABWR design provides for a highly reliable passive system to flood the lower drywell with water from the suppression pool after core debris enters. Not only will this system provide for debris cooling, it will trap fission products that may be released from possible core concrete interaction and control the containment gas temperatures resulting from core concrete interactions.

In addition, in the event that the overlying pool of water cannot cool the debris, the ABWR rupture disk will prevent the containment structure from failing and further limit the fission product release. This reduces the sensitivity of the ABWR to the final resolution of the debris coolability issue.

Question 7:

One important aspect of determining ex-vessel core debris coolability is the calculation of the amount of debris that is considered to be on the floor. Provide the results of the parametric study that was used to determine the appropriate value to be used in assessing coolability. In addition, provide the rationale that was used in the selection process. Include the name of the code used in the above study as well as the key input parameters (mass composition, temperature of the debris in the lower head at time of lower head failure).

Response 7:

The ABWR uncertainty analysis currently being performed will address a range of possible debris masses, debris temperatures, and composition. The ranges of values are similar to those used in NUREG/CR-4551 for Grand Gulf, and spans the ranges typical of MAAP and MELCOR results. These ranges of conditions will then be used to predict the core debris behavior. The outcome of the detailed uncertainty study will be a better understanding of containment behavior related to core-concrete and core-water interactions. A two-dimensional core-concrete interaction model developed by ARSAP will be used for this analysis. This model is based in part on the MAAP DECOMP model. A description of this model can be obtained by contacting the DOE ARSAP project manager. Further information on the results of this effort will be provided to the staff as it becomes available.

Question 8:

Based on the above model, provide the depth of erosion into both the basemat as well as the reactor vessel pedestal for at least the first 24 hours or until the debris is quenched, whichever comes first. Provide the basis (i.e., calculations, assumptions, and test data) for the penetration rate that was used in the calculations. In addition, include the total thickness of the basemat as well as the maximum penetration into the pedestal that can be tolerated and still retain structural integrity.

Response 8:

The ABWR uncertainty analysis will provide the expected concrete erosion depths for a variety of debris masses, debris-water cooling rates, and debris temperatures. The capacity of the lower drywell wall has been estimated in a conservative analysis, based on the margins built into the design. Preliminary results, presented to the NRC staff on January 22, indicates that the pedestal can withstand at least 38 cm of ablation without loss of structural integrity. The final results of this analysis will be included in the sensitivity analysis being performed to supplement subsection 19E.2 of the ABWR SSAR.

Question 9:

For the calculation model used in the response to Question 8, provide the supporting containment pressure/temperature response profile. Include in the results both the integrated and rate of non-condensibles produced as a function of time. In addition, what is the containment integrity sensitivity to the core debris not cooling in the relatively short period of time assumed by GE (? hrs).

Response 9:

The model used for estimating the concrete erosion will also provide information regarding the non-condensible gas generation associated with a range of debris masses, debris-water heat transfer rates, and debris temperatures. Since the stand-alone DECOMP model does not represent the entire containment, a simple pool heatup calculation along with the ideal gas assumption will be used to estimate the containment pressure/temperature for the ranges of parameters discussed earlier.

The presence of the rupture disk essentially prevents the overpressure failure of the containment due to core concrete interaction since the pressurization rate is slow enough to allow the wetwell to participate in the containment pressurization. The time of rupture disk opening will be estimated based on the pressure/temperature calculation discussed above.

Question 10:

Please provide engineering drawings and performance specification requirements (expected and permitted variance in fusible plug performance, flow rate, etc.) for the passive flooder valves. Also include any analysis performed which would be the basis for the opening time of the valves, number of valves expected to open and expected flow rate and distribution of water over the debris bed.

Response 10:

A description of the lower drywell flooder design is given in subsection 9.5.12 of the ABWR SSAR. The details of the lower flooder must be developed per the specification given in 9.5.12, and has been identified as needing interface requirements.

The rate of water addition to the cavity from the flooder system will exceed that amount required to remove the debris stored heat, any possible oxidation heat resulting from core-concrete interactions, and provide for long term cooling of the debris. Detailed calculations of the flow rate will be provided later. The lower drywell of the ABWR is a flat open chamber. Therefore, there would be no inhibiting influences which would prevent the water from filling the lower drywell to a uniform depth. The equilibrium water level is sufficient to cover the debris. Again, details will be provided later.

Question 11:

Please discuss the lower drywell subcompartment analysis which may have been performed as a result of cavity pressurization resulting from water being introduced over a core debris bed by the *lower drywell* flooder system.

Response 11:

The pressure difference which drives the flow of water from the suppression pool to the lower drywell is the elevation difference of the water in the wetwell-drywell connecting vents. Any steam generated in the lower drywell is vented to the suppression and eventually returns back to the lower drywell. If the drywell begins to pressurize as a result of water addition to the lower drywell, the flow of water will stop. Therefore, by design, the passive flooder system cannot be the source of any long term drywell pressurization.

The lower drywell pressure could rise in the instant water is first introduced into the lower drywell. However, this increase will not challenge the containment capability. Detailed analyses of the potential for pressurization of the lower drywell is being considered in the ABWR sensitivity analysis which will be provided to the staff later. A complete discussion of the passive flooder as a source for pressurization will be included.

Question 12:

With respect to Accident Progression Group "C" with some form of core cooling being accomplished through the SRVs and T-quenchers, it appears that suppression pool hydrodynamic effects were not considered as the pool is being heated. Please provide discussion on how the pool loads were calculated as local and or bulk pool temperature increase during the SRV discharge in progression "C".

Supplement to Question 12:

The thrust of the question is to determine whether or not pool dynamic loads were considered in the evaluation of the containment structure. The reference to "Accident Progression Group "C" " was taken from Chapter 19. It refers to a group of high pressure sequences. This is the key to the question, since the concern focuses on whether SRV loads were included in the structural evaluation. For the high pressure sequences, the only way to control pressure is via the SRVs. Since the pool could be at elevated temperature, the SRV loads should be addressed. It may be that these loads are small in comparison with the loads that were considered. The important issue, however, is to document how this load was considered in the evaluation of the containment loads.

Response 12:

There is no mention of "Accident Progression Group "C" " in the ABWR SSAR. High pressure core melt sequences are designated IA and IE in the context of the containment event trees of Appendix 19D, and LCHPxxxx and NSCHxxxx in the context of accident progression analysis contained in Appendix 19E. I presume these sequences are those being referenced in this question.

The ABWR uses X-quenchers for the SRV discharge. The peak temperature in the suppression pool before the vessel fails in an NSCHxxxx sequence is approximately 57 C (135 F). This value bounds the LCHPxxxx sequence since the ATWS will result in more energy transfer to the pool than the non-ATWS case. The suppression pool heat capacity temperature limit, given in the SSAR Appendix 18A, is about 67 C. Thus, the severe accident case lies within the analyzed limits, and the hydrodynamic loads will not challenge containment.

Question 13:

In regard to drywell/wetwell bypass, it appears that GE estimates that the drywell/wetwell vacuum breakers have a low contribution to suppression pool bypass flow effects. Please provide engineering drawings of the valves and a discussion on valve attributes which would yield expected high performance.

Response 13:

The analysis of suppression pool and containment bypass is presented in subsection 19E.2.3.3 of the ABWR SSAR. In that analysis it was concluded that bypass had a small contribution to overall plant risk. The vacuum breaker path dominated this (relatively small) increase. A presentation was made to the staff on October 8, 1991 which described the vacuum breakers used in ABWR and the improvements in their design over that used in past plants. A sketch of the valve taken from that presentation is shown in Figure 1. The ABWR vacuum breaker is a swing check valve which opens passively due to differential pressure across the valve, and requires no external power to actuate. The force of gravity closes the valve. It is capable of accommodating the expected repeated cyclic loading which might occur due to chugging in the Mark I and Mark II designs. The performance of the valve has been demonstrated, it is highly unlikely to stick open.

Additionally, the ABWR vacuum breakers are mounted high in the suppression pool on the pedestal wall away from the wetwell/drywell connecting vents (see Figures 1.2-3c and 1.2-3k of the ABWR SSAR). They open into the lower drywell. This arrangement location precludes the vacuum breakers from being subjected to cyclic loading due to chugging during the later part of the LOCA blowdown phase. Analysis indicates that the vacuum breakers are not subjected to any duty during the LOCA blowdown period. They are expected to open only after the blowdown is complete and either sprays are initiated or subcooled water starts spilling out of the LOCA break.

Question 14:

The containment performance under severe accident conditions is greatly affected by containment pressure capability. The containment pressure capability can be evaluated through probabilistic structural reliability estimates or a deterministic criterion. Based on a review of Appendix 19F, "Containment Ultimate Strength," the ABWR containment system's weakest locations appear to be the drywell head and penetrations.

It appears that in evaluating the containment performance, you *chose* to use probabilistic reliability estimates supplemented by some testing as described in Appendix 19F *as the bases* for determining containment integrity success criteria during a severe accident. If that is a correct understanding, please describe the bases for establishing a pressure capability for the drywell head and penetrations at ambient and elevated temperatures as described in Appendix 19F. If *a* deterministic *criterion* was used to establish containment success criteria, please describe *it. Also, indicate if* uncertainties in the containment reliability estimates *were* evaluated and how.

Supplement to Question 14:

The thrust of the question is to determine how the containment was evaluated and the basis for that evaluation. There are three parts to the question. The first question is how was the containment performance computed; probabilistic or deterministic? The next question is what proved to be the weakest region of the containment and how was that determined. There is a discussion in Chapter 19 of a scaled test in Japan that showed the head to be the weakest region, but it is unclear as to whether the test was the basis or was it based on analysis. The final question is how was the failure of the containment defined? In Appendix 19F, a structural methodology is provided which defines the failure point of the head. If the head is indeed the weakest link, confirm that the methodology was indeed that used for defining containment failure. If another region was the weakest, what method was used to define failure.

Response 14:

The containment structural performance of the ABWR was evaluated by test and deterministic criteria. Two mechanisms for the structural failure of the containment were identified (in addition to the overpressure protection rupture disk design feature): failure of the concrete portion of the containment and failure of the drywell head. These possibilities were addressed separately.

The concrete portions of the containment were tested in two scale model tests as described in the ABWR SSAR section 19F.2. The 1/6-scale global model test identified two critical locations where overpressure failure of the concrete structure would be most likely: the cylindrical wall at the lower drywell access tunnel opening elevation and the top slab region. The results of this test were extrapolated by elastic methods and determined that the pressure associated with failure of the cylindrical wall is 180 psig, as described in subsection 19F.2.1. Additional testing using a 1/10-scale local model, described in subsection 19F.2.2, was used to determine the pressure capability of the top slab. The test results showed that the pressure capability of the top slab is 180 psig.

The drywell head, not included in the tests, was separately analyzed using deterministic methods. as described in subsection 19F.3.1. The best estimate pressure capability of the drywell head was found to be 134 psig at 500 F. Subsequently the uncertainty in the containment structural failure pressure was estimated for use in the ABWR uncertainty analyses. Following the standard practice for PRA analysis, a lognormal distribution was assumed to describe the uncertainties in modelling and material properties. The details of this analysis have been shared with the NRC and will be incorporated in a future draft of the Appendix 19F.

Additional question related to containment capability:

In addition to the above Question 14, the staff asked GE to confirm that the ABWR containment meets the requirements of SECY 90-016 as it regard containment structural performance and hydrogen generation. SECY 9ι 16 recommends adherence to 10 CFR 50.34(f) for containment performance.

Response to additional question:

The ABWR meets this requirement as documented in the SSAR subsection 19E.2.3.2.

Question 15:

Provide an identification of all ABWR design features which fulfill a core melt prevention function, or provide some capability for use in recovery sequences. Describe how in the design process these feature were selected, and provide some quantification of each feature's risk benefit worth. Identify which of these features were from existing designs and which were new or possess new capabilities. Describe the process which was utilized to decide which severe accident enhancements should be incorporated into the ABWR and which to exclude (if any).

Response 15:

The ABWR has a wealth of features which prevent, mitigate and allow recovery from a hypothesized severe accident. Many of these features were developed over the more than 30 years of BWR development by GE. Some features are newly developed for the ABWR. The initial PRA effort indicted that the ABWR had abundant means of preventing and mitigating severe accidents. However, key insights gained from the PRA led to the selection of additional features. As a result of the early design efforts and enhancements based on the PRA, the risk associated with severe accidents in the ABWR is extremely low.

Analysis has been performed for the ABWR to evaluate additional enhancements which could provide additional risk reduction. No additional enhancements were identified as being cost beneficial. This analysis is being incorporated in Appendix 19O of the ABWR SSAR. The key features responsible for the ABWR capability are given below, grouped by their ability to prevent and mitigate a severe accident. Recovery aspects of the various systems are discussed in either of these contexts.

Prevention Features:

> Emergency Procedure Guidelines: The ABWR employs symptom based procedures which are consistent with the BWR Owner's Group Revision 4 EPGs. Symptom based EPGs were developed collectively by the owner's group and the NRC and are approved by the NRC for generic application to BWRs. The EPGs do not require the identification of the causes of an accident in order to mitigate its consequences. The operator responds based on the symptoms (e.g. instrument readings) of the plant. The EPGs identify the best accident strategy to mitigate the conditions without requiring knowledge of the event initiator. The ABWR EPGs incorporate a few enhancements tailored for the ABWR.
>
> Combustion Turbine Generator: This design feature was added to the ABWR as required by the EPRI ALWR Requirements document. A combustion turbine generator (CTG) starts automatically. It is automatically loaded with selected

investment protection loads. Safety-grade loads can be added manually. This provides diverse power to the nuclear safety-related equipment if none of the three safety-grade diesel generators are available and offsite power is lost. This feature reduces the risk associated with station blackout events.

Batteries: The ABWR has four divisions of batteries for DC power. This reduces the probability of a severe accident resulting from a loss of all DC power. This results in a substantial benefit since the operator does not have the ability to monitor or control the plant under these conditions. Extra redundancy of DC power was included in the ABWR during the early design stage of the plant.

Emergency Core Cooling Systems: The ECCS network of the ABWR is an extension of that found in previous plants. It consists of three separate divisions, each of which has one high pressure injection system and one low pressure injection system. Any single pump is capable of preventing core damage. Diversity is available for the water source since the suppression pool and condensate storage tank may be used as the suction source for the high pressure pumps.

There also diversity available in the motive force for injection. The low pressure core flooders (LPCF) and high pressure core flooders (HPCF) have motor driven pumps. The reactor core isolation cooling system (RCIC) is driven by a steam turbine. In the ABWR, unlike past designs, the RCIC is a safety grade system.

Condensate Pumps: The ABWR condensate pumps can be powered by the diesel generators to supply core cooling to the reactor vessel. This feature was identified in the design stage.

Feedwater Pumps: The feedwater pumps in the ABWR are motor driven. This allows recovery of core cooling by the feedwater system after an isolation event. This reduces the probability of a core damage event following a transient.

Safety Relief Valves: The ABWR design utilizes a highly reliable automatic depressurization system to reduce the vessel pressure in the event of a loss of the normal and emergency high pressure injection systems. An enhancement to the ABWR as compared to some earlier BWRs is that the ABWR the nitrogen supply for the SRVs is based on the containment pressure. Therefore, the SRVs can be held open over the entire range of containment pressures.

Residual Heat Removal: The RHR system of the ABWR is enhanced as compared to earlier plants. There are three separate divisions, each of which can supply core cooling (the LPCF function) as well as containment heat removal. The ABWR also differs from previous plants in its ability to run with the heat exchangers in the loop while in LPCF mode. This substantially reduces the risk of loss of containment heat removal. The additional division of RHR also allows the operator more flexibility in mitigating LOCA or transient events since only one of three divisions of RHR is necessary to supply adequate core cooling for design basis events.

AC Independent Water Addition System: In the ABWR the fire protection system can be connected to one of the LPCF injection loops by manual valve operation to provide injection. The motive force for this firewater supply can be either the onsite fire pumps (one motor driven, one direct diesel), or a fire truck. Thus, the firewater addition system is independent of AC power. This provides additional redundancy and diversity to the core cooling network. Since piping and valves have high seismic capacities, the AC independent water addition system is particularly beneficial in seismic events. This feature of the ABWR design was identified as a result of the PRA.

Mitigation Features:

Inerted Containment: SECY 90-016 required that evolutionary plants meet the requirements of 10 CFR 50.34 (f) as it relates to combustible gas generation and hydrogen control. Containment inerting is a feature of the Mark I and Mark II containment designs. The nitrogen-inertion of the ABWR containment design will prevent combustion in the containment in the highly unlikely event of a core damage accident. This includes both the generation of hydrogen during the core heatup and degradation phase as well as any potential ex-vessel core-concrete interactions. 10 CFR 50.34 (f) also requires that the containment peak pressure remain below Service Level C for steel structures and the corresponding factored loads for concrete structures if the pressure associated with the oxidation of 100% of the active fuel cladding is added to the design basis LOCA loads. The ABWR also meets this requirement as described in subsection 19E.2.3.2 of the ABWR SSAR.

Reactor Vessel Depressurization System: The depressurization system has benefits in addition to those described above in the prevention discussion. In the unlikely event of a core melt accident, operation of the depressurization system will result in a low reactor vessel pressure at the time of core material discharge from the vessel. This will eliminate the potential for debris dispersal to the upper drywell region, reducing the potential for heatup of the upper drywell. Since depressurization ensures that the debris is in the lower drywell, the passive flooder, described below, is able to provide for debris cooling.

Passive Flooder: The ABWR is designed with a cavity floor space sufficient to spread the core debris. It is also designed with a passive lower drywell flooder system that will flood the cavity using suppression pool water in the unlikely event that core material enters the lower drywell. This system utilizes ten temperature sensitive fusible plug valves that will open to allow the cavity to be flooded. It is expected that this system will mitigate the consequences of any possible core-concrete interactions by quenching the core debris, reducing the generation of non-condensible gases, scrubbing any releases of fission products released from the core debris, and preventing containment heatup. This feature was added to the ABWR design as a result of insights gained from the PRA.

Imbedded Containment Liner: The lower drywell wall of the ABWR is designed with an imbedded liner to provide added protection against possible containment challenges. In the event of a severe core damage accident the containment boundary is protected from direct heating by the core debris. The inner meter of concrete is a sacrificial layer of concrete which provides protection to the containment boundary.

Basaltic Concrete: In the unlikely event of a severe core damage accident, molten fuel and clad material could be discharged from the reactor pressure vessel. In this event, the passive flooder is designed to open and allow for flooding of the cavity prior to any significant core-concrete interaction as described above. Due to a limited amount of applicable ex-vessel debris cooling data, the potential for core-concrete interaction, even in the presence of water, can not be eliminated. One of the primary containment challenges resulting from core-concrete attack is the amount of non-condensible gas generated. The ABWR cavity floor is specified to use basaltic-type concrete. This type of concrete releases small amounts of non-condensible gasses if it is decomposed. This limits the rate of pressurization of the containment.

AC Independent Water Addition System: In addition to the core damage prevention function described above, this system is also a powerful tool for the mitigation in the unlikely event of severe accident. The system can add water to the containment via the vessel as described above, or via the drywell spray header. In either case, the water provides additional thermal mass which delays the time of containment failure if containment heat removal is not restored. Additionally, if some of the corium were transferred to the upper drywell as a result of a high pressure core melt, then the spray system would keep the upper drywell cool, preventing the degradation of the penetration seals. This, combined with the containment overpressure protection system ensures that the fission product releases are scrubbed by the suppression pool.

Containment Arrangement: The relative elevations of the suppression pool and the lower drywell provides the driving potential for water flow through the passive flooder system. The lower drywell floor has been sized to allow good heat transfer between the overlying water pool and the core debris bed. The corss-sectional area of the lower drywell has also been sized to assure that the final drawdown level of the suppression pool after flooder operation will be sufficient to allow proper steam condensation and fission product scrubbing through the horizontal vents.

Vacuum Breaker Arrangement: The ABWR vacuum breakers are located at an elevation high in the upper drywell and wetwell airspace, away from the wetwell/drywell connecting vents. This results in a substantial reduction of loads on the vacuum breakers during LOCAs and accidents. The vacuum breakers which will be used in the ABWR are based on the loading observed in earlier plants. This results in additional margin in the valve design. Although not quantified, this increased margin will result in additional vacuum breaker reliability. Thus, the risk associated with suppression pool bypass will be reduced. This reduction in vacuum breaker loading was identified during an early phase of the ABWR design process.

Containment Overpressure Protection System: In the unlikely event that a loss of decay heat removal accident progresses to a point where the containment integrity is threatened, this pressure will be relieved from the wetwell through the containment overpressure protection system. This system is comprised of a relief line designed to 150 psig, and two rupture disks in series which open before the pressure rises to a level where the structural integrity could be challenged. This system substantially reduces the probability of containment failure. In addition, in the event of fission product release, the emitted release will be scrubbed by the suppression pool which results in a substantial reduction in source term. This feature was added to the containment as a result of insights gained from the PRA.

Question 16:

ADS system functionability and reliability are issues of particular importance. The ADS allows for a low pressure injection success path where *high pressure injection* fails. Also, for those sequences where no RCS makeup is available, ADS provides primary system depressurization prior to vessel failure, precluding DCH related containment challenges. This DCH prevention function is of special importance due to inherent uncertainties in demonstrating the capability of retentive cavities.

Please provide an evaluation of the ADS system, demonstrating that it possesses high functional reliability for those sequences where it would be called upon to depressurize the RCS prior to vessel failure. This evaluation should account for the systems and support system states for the sequence were ADS would be called upon. Also, describe how system design will allow for the ADS to remain open after actuation, for and adequate period, preventing possible later repressurization and high pressure vessel failure.

Response 16:

Subsection 19D.6.2.5 provides an evaluation of the ADS system reliability including the nitrogen, control and instrumentation systems. Additional information about the SRVs and the ADS system may be found in subsections 5.2.2, 7.3.1.1.1.1 and 19E.2.1.2.2.2.

Subsection 7.3.1.1.1.1 (3) (h) indicates that the signal cables, solenoid valves, safety/relief valve operators and accumulators are located inside the drywell and are designed to operate in the most severe accident resulting from a DBA LOCA, including the radiation effects. The conditions in the containment during the early stages of a severe accident (before vessel failure) which requires depressurization using the SRVs are less challenging than those specified by a DBA LOCA. Additional analyses of the ADS system capability were performed in support of station blackout performance analysis. This discussion is included in 19E.2.1.2.2.2. The conclusions of that analysis are that there is ample DC power for the operation of the SRVs for many days after the 8 hour capability required by the station blackout rule.

Section 5.2.5 indicates that the nitrogen accumulator capacity for each valve is designed to be sufficient to open for one actuation at drywell design pressure even if the air supply to the accumulators is lost. The risk significant severe accidents in the ABWR PRA remain below the design pressure of the containment in the time period before vessel failure. Valve operability at high containment pressure conditions are also discussed in subsection 19E.2.1.2.2.2 (2) (b). Based on the presence of the containment overpressure protection system, the maximum drywell pressure is approximately 100 psig. (Determination of a new setpoint is not completed following strengthening of the drywell head.) Subsection 19E.2.1.9.2.2 (2) (b) indicates the operator actions which could be taken to assure SRV operability under these conditions. The appropriate operator actions are specified in the ABWR EPGs. Since the containment pressurizes very slowly, over a period of about a day, there is ample time for the operators to take the appropriate actions.

Given the above discussions one may conclude that the ADS system will not be compromised before vessel failure in the unlikely event of a severe accident.

Question 17:

Please identify those features or design changes which were added to the design as a result of the initial PRA analysis which identified significant risk contributors. List those features or design changes which were or will be added to the design to eliminate or mitigate a weakness evaluated in the PRA.

Response 17:

As agreed in discussions between GE and the NRC staff of January 28, 1992, this questions will be subsumed in the discussion of questions 15.

Question 18:

Accident Management, as defined in SECY 89-012, involves actions taken by the plant staff to: (1) prevent core damage, (2) terminate progress of core damage and retain the core within the vessel, (3) maintain containment integrity, and (4) minimize offsite release. The present focus of the ABWR PRA is on the first of these objectives. However, the PRA can also be used as a tool to identify and assess potential risk reduction measures aimed at the latter three objectives of accident management. If identified at the design stage, specific provisions can be made in the plant design to facilitate (or eliminate the need for) such measures (e.g., automation of otherwise manual actions, or use of remote-manual rather than local manual valves). Please describe your plans to use the PRA to identify and assess additional accident management measures, and to expand the scope of the study for this purpose.

Response 18:

The ABWR PRA is a level 3 PRA which considers core damage prevention, mitigation and offsite release. Opportunities to improve the plant performance were identified and implemented throughout the PRA analysis effort. The response to question 15 identifies many improvements incorporated into the ABWR design as a result of the PRA. These improvements span all four types of accident management actions.

A final step in the use of the PRA to enhance plant safety is the identification of operator actions. These issues are then candidates for additional automation, improved procedures or hardware changes. The identification of key operator actions has been completed and further improvements to the ABWR design and procedures are being considered for these actions.

Question 19:

Please discuss the applicability and significance of each of the accident management strategies identified in Generic Letter 88-20, Supplement 2 to the ABWR design. Specifically identify any design features which eliminate the need for a strategy, or facilitate implementation of a strategy. Identify and discuss any other unique measures or strategies for dealing with potential severe accidents in the ABWR design.

Response 19:

The ABWR has an abundance of features which substantially reduce the risk of a severe accident. All aspects of accident management were addressed in the ABWR design. A summary of the features which have significant impact on accident prevention and mitigation are given in the response to Question 15. The strategies identified in Supplement 2 to Generic Letter 88-20 are given below using the numbering scheme of that document.

2.1 Strategy to Reduce Containment Spray Flow Rate to Conserve Water for Core Injection

This is a PWR strategy not applicable to ABWR.

2.2 Strategy to Enable Early Detection, Isolation, or Otherwise Mitigate the Effects of an Interfacing Systems LOCA

The ABWR design has increased the design pressure of the low pressure systems to 300 psig. Using the NRC methodology for pipe rupture this increase in design pressure allows the piping to withstand full reactor pressure. Discussions with the NRC staff to fully resolve this issue are nearly complete. This effectively eliminates interfacing system LOCA as a credible accident for the ABWR design. No additional measures are required.

2.3 Makeup to Emergency Storage Tank

The condensate storage tank (CST) of the ABWR serves to provide an emergency source of water in the event of an accident in which emergency core cooling is necessary. The Emergency Operating Procedures to be developed from the ABWR EPGs will specify monitoring of CST water level when ECCS pumps are taking suction from it. Additionally, CST water level is continuously displayed on a fixed display monitoring panel in the main control room and there is an alarm which indicates low water level in the CST. Normal plant procedures specify the means for refilling the CST.

2.4 Strategy to Ensure Appropriate Recirculation Switchover and Manual Intervention Upon Failure of Automatic Switchover

This is a PWR issue not applicable to ABWR.

2.5 Strategy to Ensure Adequate Plant Heat Removal Capability by Emergency Connection(s) of Existing or Alternate Water Sources

Injection to the ABWR vessel may be accomplished in a variety of means beyond the ECCS. The response to question 15 identifies several means of core cooling in the extremely unlikely event that all ECCS systems fail to operate.

3.1 Strategy to Extend Emergency Core Cooling System Availability by Switching Pump Suction

The ABWR vessel injection may be taken from a number of sources as noted in the response to Question 15. If the vessel is maintained at low pressure, the rising suppression pool water level is not a concern until the pool level is approximately equal to the level of the bottom of the reactor pressure vessel. This will not occur for approximately one day. This allows adequate time for the development of additional strategies.

3.2.1 Strategy to Enable Emergency Bypass or Change of Protective Trips for Injection Pumps

As discussed, the ABWR has a wide number of pumps and water supply sources which could be used to allow injection of coolant into the ABWR vessel during an accident. The additional means for core coolant injection reduce the probability that bypass of protective trips is warranted in light of the additional systems that may be available.

As noted in NUREG/CR-5474, there are many types of protective trips in BWR plants. The RCIC system has a high turbine exhaust trip which is designed to protect the RCIC turbine. The setpoint of this trip for the ABWR is higher than that in previous generations of plants. This increase, combined with the AC independent water addition system and combustion gas turbine effectively eliminate the chance that core cooling is lost due to station blackout.

High reactor water level trips on the ABWR are tied only to the systems capable of vessel injection at high pressure. If the water level were allowed to continue to rise to the elevation of the main steam lines, significant damage to the system could occur due to the loads associated with SRV cycling. Therefore, bypass of this signal is not considered to be appropriate.

ECCS pumps in the ABWR are located in in the reactor building. This allows the pumps to operate with suction from the suppression pool even if the pool becomes saturated. Combined with the variety of alternate injection sources available in the ABWR, it is not judged that further efforts to bypass the suction pressure trips are warranted due to the risk of damage associated with pump cavitation.

Other trips associated with such conditions as low lube oil pressure, low control oil pressure, thrust bearing wear, etc. can not be evaluated until the plant specific pump procurement since the risk of damaging the pump depends on the individual pump design.

3.2.2 Strategy to Extend Reactor Core Isolation Cooling System Availability by Pump Trip Function Bypass or Change

This strategy was discussed, in part, in item 3.2.1 above. The RCIC system design and operating procedures will provide for core cooling for at least 8 hours in a station blackout event with failure of the combustible gas turbine generator. No further actions are deemed necessary.

3.3 Core Injection by Non-Safety Related Pumps

Several non-safety related pumps which could provide vessel injection are discussed above in the response to Question 15. The use of CRD pumps is not perceived to produce additional benefit.

3.4 Strategy to Use Alternate Seal Injection (e.g., Hydrotest Pump) When Reactor Coolant Pump Seal Cooling is Lost

This is a PWR strategy not applicable to ABWR.

3.5 Strategy to Use Condensate Pumps or Startup Feedwater Pumps for Steam Generator Injection

This is a PWR strategy not applicable to ABWR.

4.1 Strategy to Conserve Battery Capacity by Shedding Non-Essential Loads

The ABWR has four division of DC power. Division A, which supples the RCIC system is capable of supplying the required power for at least 8 hours. Detailed procedures for load shedding will be identified as the detailed design of the ABWR progresses.

4.2 Strategy to Use Portable Battery Chargers or Other Power Sources to Recharge Station Batteries

The ABWR station batteries are capable of supplying the required power for at least 8 hours. Procedures for recharging batteries cannot be developed before equipment procurement.

4.3 Strategy to Enable Emergency Replenishment of the Pneumatic Supply for Safety Related Air Operated Components

Procedures to replenish the pneumatic supply cannot be developed before equipment procurement. A discussion of the types of actions that could be taken are given in subsection 19E.2.1.2.2.2 (2) (b).

4.4 Strategy to Enable Emergency Bypass or Change of Protective Trips for Emergency Diesel Generators

This strategy cannot be developed before equipment procurement.

4.5 Strategy to Enable Emergency Crosstie of AC Power Between Two Units or to an Onsite Gas Turbine Generator

The ABWR plant is being certified as a single unit with no onsite gas turbine generator. Use of the Combustible Gas Turbine Generator is discussed under item 4.6 below.

4.6 Strategy to Use a Diesel Generator or Gas Turbine Generator to Power a Control Rod Drive or Other Appropriate Pump for Core Injection

The ABWR has a combustible gas turbine generator which may be use to power any division of ECCS and feedwater/condensate pumps. This is a significant improvement in electrical system design for the ABWR. Further discussion is included in the response to Question 15.

4.7 Strategy to Use Diesel Driven Firewater Pump for BWR Core Injection, PWR Steam Generator Injection or Containment Sprays

The ABWR has a firewater injection system which is capable of providing core cooling injection or containment sprays. Further discussion is included in the response to Question 15.

5.1 Strategy to Re-open Main Steam Isolation Valves and Turbine Bypass Valves to Regain the Main Condenser as a Heat Sink

The ABWR EPGs call to use the main condenser as a heat sink in the unlikely event that all three divisions of RHR are unavailable or inadequate to remove the energy being generated.

6.1 Strategy to Provide Additional Supply of Borated Makeup Water for Long-Term Accident Control

The ABWR EPGs include the use of alternate injection systems to supply borated water to the vessel.

6.2 Strategy to Inject Borated Water in Case of Potential Core Damage and to Guard Against boron Dilution in the Core

The ABWR EPGs call for the the injection of boron sufficient to maintain the reactor in a cold shutdown condition, assuming all the control rods failed to insert fully. Boron injection is accomplished through the HPCF (B) injection line which injects into the core. Strategies for water control during and after boron injection assures that boron dilution will not occur to an extent to allow recriticality. If additional boron injection is required in case of core damage, additional boron can be mixed in the SLC boron tank and injected into the vessel. This strategy should be investigated as part of a severe accident recovery strategy to be developed later.

Question 20:

Provide a description of the design of the equipment hatch and its ability to be rapidly reclosed during shutdown, if necessary. Include a discussion of the need for AC power or any other support systems in order to effect closure, and the pressure seal arrangement, i.e., whether the hatch is pressure-seating as opposed to pressure-opening (which would require full bolting to accomplish sealing under pressure). Discuss any strategies/procedures for rapidly closing major penetrations during shutdown.

Response 20:

The ABWR equipment and personnel hatches are both pressure seating hatches. Neither one requires electrical power to close. The personnel hatch is a simple metal door. The equipment hatch is pulled down using a chain, positioned properly and bolted. Both doors can be closed within minutes. The only other major penetration on the ABWR containment which might be open during shutdown is the drywell head. Closing the head would take several hours. AC power would be required in order to move the equipment platform and lift the drywell head into position. The drywell head is pressure opening, so full bolting would be required to accomplish sealing under pressure. No strategies are being developed to close the drywell head quickly.

Question 21:

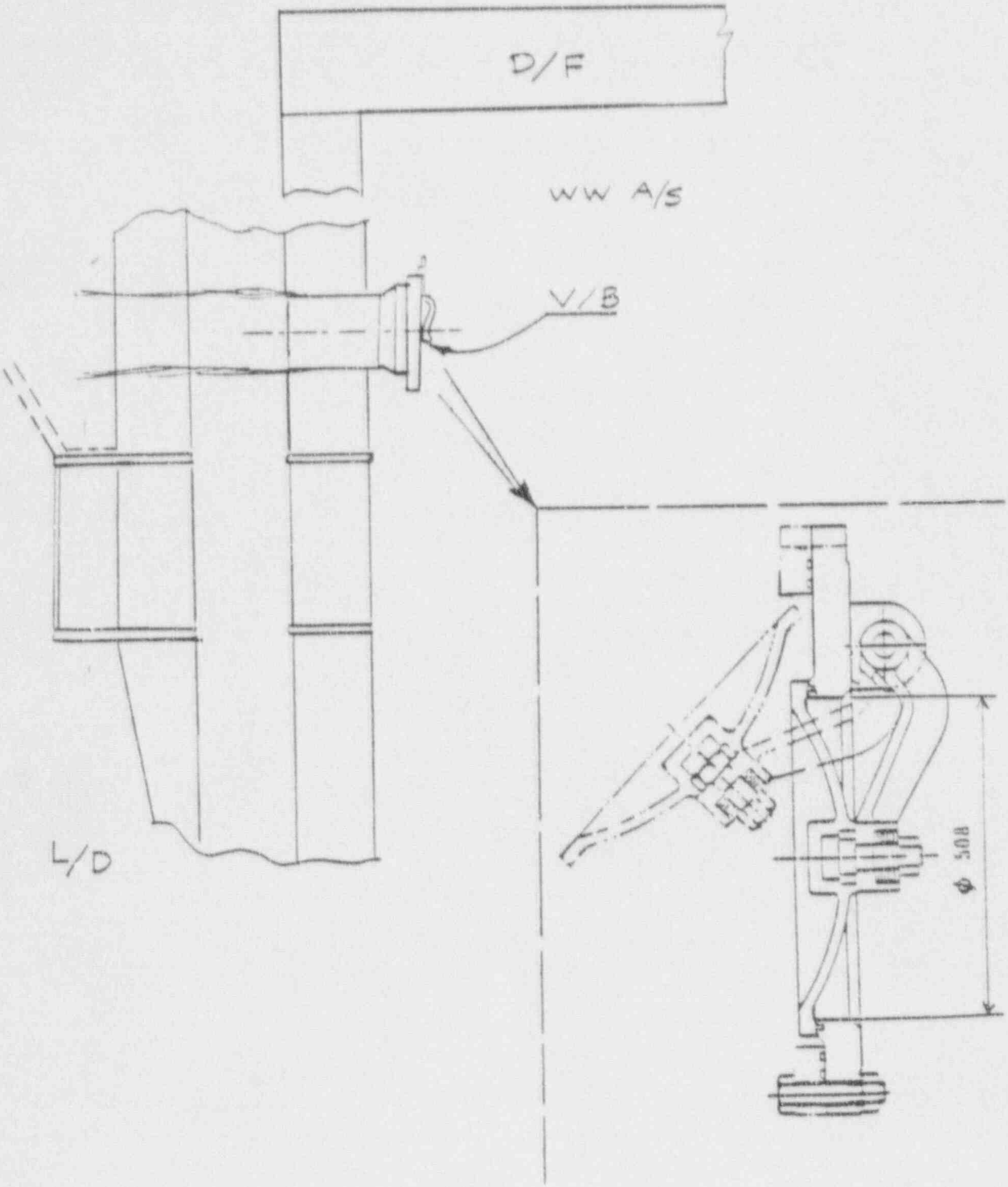
Please discuss GE's planned approach for assuring that each of the five elements of accident management defined in SECY 89-012 will be appropriately addressed by the vendor/licensee. Identify the respective responsibilities of GE and of the licensee for addressing each of the elements, and any methods and/or guidance that are expected to be used in this process (e.g., the "Process for Evaluating Accident Management Capabilities" developed by NUMARC, the "Severe Accident Management Guidance Technical Basis Report" developed by EPRI, or the accident management guidelines now under development by each of the reactor vendors as part of the industry Accident Management Program).

Response 21:

Accident management is a relatively new area being investigated by both the NRC and industry. GE is participating in the efforts of the BWR Owner's Group in the development of Accident Management Guidelines. These guidelines will address all five elements of accident management. The Owner's Group activity is making use of a wide variety of information sources.

Final responsibility for all areas of severe accident management lies with the applicant. GE has addressed many Accident Management Procedures in the ABWR PRA and EPGs. Accident management procedures can be developed by the applicant based on these procedures, as well as the Owner's Group assessments, when site specific details are available. Training in Severe Accidents, Accident Management Guidance and Decision making Responsibilities are the responsibility of the applicant. Again, GE expects that the applicant will follow the work of the BWR Owner's Group.

Instrumentation needs and equipment qualification are areas of active assessment by both the NRC and EPRI. Any insights or requirements which result from these studies will be incorporated into the ABWR during detailed design and equipment procurement.



WW/DW VACUUM BREAKER

FIGURE 1