July 13, 1994

Docket No. 52-001

Mr. Joseph Quirk
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Dear Mr. Quirk:

SUBJECT: RESOLUTION OF ITEMS REGARDING PROBABILISTIC RISK ASSESSMENT (PRA) FOR THE ADVANCED BOILING WATER REACTOR (ABWR)

This letter provides the resolutions of issues regarding PRA which were identified in Section 19.1 of the staff's draft safety evaluation report and the draft final safety evaluation report for the ABWR. As stated in the staff's final safety evaluation report and as documented in this letter, GE has provided adequate resolutions to all of these items.

Sincerely,

Original Signed By:

Thomas H. Boyce, Project Manager
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Enclosure:
As stated

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ABWR PRA DSER AND DFSER ITEM RESOLUTION

<u>Outstanding Item 1 in the DSER</u> In its DSER the staff noted that GE's use of the Clinton reactor protection system (RPS) unavailability estimate was inappropriate since the design of the Clinton RPS and the ABWR RPS were essentially different. The staff identified the assumed ABWR RPS unavailability estimate as Outstanding Item 1 in the DSER. On the basis of further discussions with GE and the low estimate of core damage frequency from ATWS events, whether using GE or staff reliability assumptions, the staff determined that no additional submittal was required to resolve this item. The staff concluded that any expected change in the original RPS unavailability assumption would not significantly change the relative insights from the ABWR PRA and therefore is acceptable. Therefore, the staff considers this Outstanding Item resolved.

<u>Confirmatory Item 1 in the DSER</u> GE's original estimate for the frequency of vessel isolation (including loss of feedwater events) and non-isolation trips was about 1 per reactor-year. In the DSER the staff took exception to this value. This was identified as Confirmatory Item 1 in the DSER. However, after discussions with GE and evaluating actual BWR reactor trip data collected by the staff over a recent 3-year period for U.S. plants, the staff found that the trip frequency for the ABWR is expected to be consistent with this value. The staff finds this acceptable. The staff considers this Confirmatory Item resolved.

<u>Confirmatory Item 2 in the DSER and Confirmatory Item 19.1.5.2-1 in the DFSER</u> In early ABWR SSAR submittals, GE estimated the frequency of having an inadvertent open relief valve (IORV) to be about 0.01 per reactor-year. The staff noted in the DSER that this estimate was substantially lower than the value (0.07 per reactor-year) used for the Limerick plant. Justification for this difference was identified as Confirmatory Item 2 in the DSER and Confirmatory Item 19.1.5.2-1 in the DFSER. GE modified the ABWR PRA to use an IORV frequency of 0.1 per reactor-year, which the staff finds is conservative and, therefore, acceptable. The staff considers this Confirmatory Item resolved.

<u>Outstanding Item 2 in the DSER and Open Item 19.1.5.2-1 in the DFSER</u> In early SSAR submittals, GE did not document the details of the contribution of support system failures (such as loss of dc power, loss of service water system) as initiating events. The need to provide this documentation was identified as Outstanding Item 2 in the DSER and Open Item 19.1.5.2-1 in the DFSER. GE subsequently provided an analysis of the contribution of support system failures as initiating events in Appendix 19D.3 in the SSAR. The staff reviewed this submittal and found it to adequately cover the support systems, as requested. The staff finds this acceptable. The staff considers this Open Item resolved.

<u>Outstanding Item 3 in the DSER and Open Item 19.1.5.2-2 in the DFSER</u> In early SSAR submittals, GE did not provide results of accident analyses of postulated interfacing system LOCA events as applicable to the ABWR design. This was identified as Outstanding Item 3 in the DSER and Open Item 19.1.5.2-2 in the DFSER. The staff's concern dealt with the possibility that multiple isolation valves in a line might fail and the inflow of reactor coolant system water would overpressurize the interfacing, low pressure line, causing it to fail. Subsequently, GE upgraded the quality of the piping that interfaces between high- and low-pressure systems such that the piping should be able to withstand the stresses from full reactor coolant system pressure. The NRC's review determined that this upgrade in combination with the existing isolation valves reduced the chances of interfacing system LOCAs to the point that they were insignificant contributors to core damage frequency and risk. Piping upgrades are checked in ITAAC 2.4.1, 2.4.2, 2.4.4, 2.2.2, 2.2.4, and 2.6.1. The staff finds this acceptable. This staff considers this Open Item resolved.

Outstanding Item 4 in the DSER and Open Item 19.1.5.2-4 in the DFSER In early SSAR submittals, GE did not document the results of the accident analyses of postulated LOCA events outside the containment (in particular, steam line breaks in the reactor core isolation coolant steam piping and the reactor water cleanup (RWCU) lines that normally experience full reactor coolant system pressure) in combination with failure of the isolation valves. This was identified as Outstanding Item 4 in the DSER and Open Item 19.1.5.2-4 in the DFSER. Subsequently, GE performed several evaluations of LOCAs outside of containment. These analyses provide a bounding rather than realistic representation of LOCAs outside of containment. The staff's independent analysis determined that even under the conservative assumptions about the frequency of these events and the potential consequences, the resulting estimates for core damage frequency and risk were several orders of magnitude less than the Commission's Safety Goals. The staff found that LOCAs outside of containment do not represent a disproportionate risk for the ABWR design. The staff considers this Open Item resolved.

Confirmatory Item 1 in the DSER and Confirmatory Item 19.1.5.3-1 in the DFSER Early in the review of the ABWR PRA, the staff questioned ambiguities in the writeup of the IORV event. This was identified as Confirmatory Item 1 in the DSER and as Confirmatory Item 19.1.5.3-1 in the DFSER. GE subsequently acceptably clarified the SSAR text and the accompanying table. The staff considers this Confirmatory Item resolved.

Confirmatory Item 2 in the DSER In the DSER the staff questioned the success criteria assumption used to determine the adequacy of one of the three residual heat removal (RHR) trains to remove heat from the suppression pool following a failure-to-scram event in combination with a vessel isolation event and the failure of boron injection as well as the minimum injection flow rate. This was identified as Confirmatory Item 2 in the DSER. In a subsequent meeting with the staff, GE presented analytical results that supported the assumption that one train of the RHR provides adequate heat removal. Based on these results, the staff found the success criteria assumptions to be acceptable. The staff considers this Confirmatory Item resolved.

Outstanding Item 5 in the DSER, Confirmatory Item 3 in the DSER, and Open Item 19.1.5.4-1 in the DFSER In the review of the ABWR PRA, the staff required GE to provide further justification that, (1) by calculating common-mode-failure at the train level, it was able to capture the full contribution to common-mode-failure probability in the analysis as if it had calculated at the component level and (2) the values assumed for the common-cause-failure probability were appropriate. These issues were identified as Outstanding Item 5 in the DSER, Confirmatory Item 3 in the DSER, and Open Item 19.1.5.4-1 in the DFSER. In further discussions with GE, the staff refined its request to require GE to requantify the PRA with and without taking into account certain additional areas of common mode failure that were not in the ABWR PRA as originally submitted. The results of this sensitivity study were used by GE to provide insights into additional systems, structures, and components that should be added to the reliability assurance program recommendations in SSAR Appendix 19K. The sensitivity study provided a reasonable and acceptable representation of common-cause failure safety insights. The staff considers this Open Item resolved.

Outstanding Item 21 in the DSER and Open Item 19.1.5.4-2 in the DFSER In the DSER the staff required GE to provide a list of systems not included in the GE nuclear island, the assumed reliability for each system, and any safety significant insights GE believes are important to designing the systems to meet the assumptions of the PRA. This was identified as Outstanding Item 21 in the DSER and as Open Item 19.1.5.4-2 in the DFSER. GE provided an acceptable listing of this information in an update to SSAR Section 19.10. The staff considers this Open Item resolved.

Open Items 19.1.5.4-4 and 19.1.11.4-2 in the DFSER After completion of the DSER, the NRC's Advisory Committee on Reactor Safeguards (ACRS) performed an independent investigation of the RWCU system. The ACRS determined that GE erroneously took credit for use of the RWCU system at high pressure during transients. It found that the RWCU system success criteria were suspect and that the system would isolate on high temperature, rather than act as a high-pressure, high-temperature heat removal path. These issues were identified as Open Items 19.1.5.4-4 and 19.1.11.4-2 in the DFSER. The ACRS determined that the RWCU system would isolate on high temperature and that if it did not, equipment would be damaged or would exceed its design temperature if the system were used as a high pressure decay heat removal system. GE subsequently acceptably corrected the design deficiencies by redesigning the isolation logic of the RWCU system, realigning the isolation configuration so that only the heat-vulnerable resin beds are isolated on high temperature, and limiting the total isolation of the RWCU to those periods when the containment isolation function is actuated. In addition, since the RWCU will only be put into operation by emergency procedure as a high pressure decay heat removal system in the event the RHR fails, cooling water will be diverted by procedure from the RHR heat exchangers to the RWCU heat exchanger to limit the temperature increase across the RWCU heat exchanger. GE calculated that this temperature increase would be only a few degrees above the heat exchanger design temperature and reasoned that this was acceptable because the RWCU is a backup system that only would have to be used in this configuration for very low probability, beyond-design-basis events. The staff accepts this reasoning and considers this Open Item resolved.

Staff Correction Item 11 in the DSER and Confirmatory Item 19.1.5.4-1 in the DFSER In the DSER the staff said GE should have taken credit for the use of the fire water system in the level 1 of the ABWR PRA, not just the level 2 portion. This issue was identified as Staff Correction Item 11 in the DSER and as Confirmatory Item 19.1.5.4-1 in the DFSER. Staff Corrections were changes in assumptions made by the staff in its evaluation of the risk of the ABWR design up to the time of the DSER that were identified to document the basis of the staff's DSER findings. GE subsequently stated that its position on modeling of the fire water system was conservative from the standpoint of core damage frequency. The staff found GE's position acceptable and did not pursue the matter, since the inclusion of the fire water system in the level 1 analysis would not change the safety insights and would only lower the estimated core damage frequency. The staff considers this Staff Correction resolved.

Outstanding Item 6 in the DSER In the DSER the staff questioned the reliability assumptions on the RHR and high-pressure core flooder pumps. This was identified as Outstanding Item 6 in the DSER. The staff reevaluated the net effect of assuming lower reliability values for this equipment and determined that the net effect on core damage frequency was negligible and did not change any safety insights. Therefore, the staff did not require GE to modify these reliability values. The staff considers this Outstanding Item resolved.

Confirmatory Item 4 in the DSER and Confirmatory Item 19.1.5.5.2-1 in the DFSER In the DSER the staff questioned the appropriateness of using GESSAR II design information for estimating the test and maintenance frequency for some equipment in the ABWR design. This was identified as Confirmatory Item 4 in the DSER and as Confirmatory Item 19.1.5.5.2-1 in the DFSER. Originally, the PRA assumed test and maintenance unavailabilities similar to those in many BWR PRAs. GE subsequently increased the assumed test and maintenance unavailabilities in the ABWR PRA by a factor of two. The staff expects this increased expected unavailability to be conservative. Based on engineering judgement, the staff finds that this assumed increase is acceptable for estimating the unavailabilities of the equipment in question. The staff considers this Confirmatory Item resolved.

Confirmatory Item 19.1.5.5.3-1 in the DFSER After the DSER was completed, the staff requested GE to perform a sensitivity study on the effect of varying surveillance intervals and outage times on core damage frequency. This was identified as Confirmatory Item 19.1.5.5.3-1 in the DFSER. GE subsequently stated that it does not intend to vary surveillance intervals, since it does not intend to change them from those used historically. The staff found this statement a sufficient and acceptable response to the concern about surveillance intervals. GE also subsequently submitted a sensitivity study on the effect of varying outage times. The results of this sensitivity study were factored into the ABWR reliability assurance program (SSAR Appendix 19K) and the safety insights derived from the ABWR PRA (SSAR Section 19.8). The sensitivity study provides a reasonable and acceptable representation of the effects on safety insights from varying outage times. The staff considers this Confirmatory Item resolved.

Confirmatory Item 19.1.5.7-1 in the DFSER Subsequent to the DSER, the NRC identified a number of minor errors in GE's quantification approach used in combination with its design-specific and generic data to quantify the sequence frequency estimates. This was identified as Confirmatory Item 19.1.5.7-1 in the DFSER. GE acceptably revised the ABWR PRA to correct these errors. The staff considers this Confirmatory Item resolved.

Interface Item 15 in the DSER and Open Item 19.1.5.9-1 in the DFSER In the DSER the NRC required GE to provide lists of the equipment to be included in the reliability assurance program and provide reliability targets for systems and/or components. This was identified as Interface Item 15 in the DSER and as Open Item 19.1.5.9-1 in the DFSER. GE provided this information in Appendix 19.K to the ABWR SSAR, which provided an acceptable listing of structures, systems, and components to be included in the reliability assurance program. The staff considers this Open Item resolved.

Open Items 11 and 18 in the DSER and Open Item 19.1.5.11.1-1 in the DFSER In the DSER the NRC stated that GE had to perform an uncertainty analysis for internally initiated events. This was identified as Open Items 11 and 18 in the DSER and as Open Item 19.1.5.11.1-1 in the DFSER. GE subsequently provided an acceptable uncertainty analysis for these events that is appropriate for the level of design detail available in the design certification process for the ABWR. The staff considers this Open Item resolved.

Open Items 19.1.2.2.1-1 and 19.1.5.11.2-1 in the DFSER After the NRC issued its DSER, GE modified the ABWR plant design. The need to update the ABWR PRA was identified as Open Items 19.1.2.2.1-1 and 19.1.5.11.2-1 in the DFSER. GE subsequently submitted modified sequences for internal events as part of its PRA update. The NRC reviewed these sequences and found them to be performed in a manner consistent with acceptable PRA methodology and with acceptable data and assumptions. The staff considers this Open Item resolved.

Confirmatory Item 2 in the DSER In the DSER the staff stated that it was investigating to determine the logical minimum injection flow to the vessel needed to avoid core damage following a vessel isolation event coupled with failure to scram and failure to provide poison injection. This was identified as Confirmatory Item 2 in the DSER. Subsequent calculations by GE determined that the flow rate would be adequate for this case. The staff found these calculations acceptable. The staff considers this Confirmatory Item resolved.

Outstanding Item 22 in the DSER and Open Item 19.1.2.2-1 in the DFSER In the DSER and the DFSER, the staff required GE to provide information describing (1) how PRA insights were used in the ABWR design process, (2) what ABWR design features, if any, were included as a result of PRA insights to reduce risk-significant sequences and phenomena, (3) how plant operating experience was factored into the ABWR PRA, and (4) how PRA insights were used to address severe accident phenomena. The staff identified these issues as Outstanding Item 22 in the DSER and Open Item 19.1.2.2-1 in the DFSER. These were all addressed by GE in amendments to Section 19.7 of the SSAR. The staff has reviewed these submittals and finds that they acceptably and satisfactorily describe the use of PRA insights in the design process. The staff considers this Open Item resolved.

Outstanding Item 22 in the DSER and Open Item 19.1.2.3-1 in the DFSER In the DSER the staff stated that GE's list of design features that are expected to significantly lessen core damage frequency and risk estimates was incomplete. This was identified as Open Item 22 in the DSER and as Open Item 19.1.2.3-1 in the DFSER. GE subsequently made an acceptable submittal of the information needed to address design features that are important to safety and design features that were modified based on PRA insights. The submittal modified SSAR Section 19.7. The staff considers this Open Item resolved.

Open Item 19.1.2.3-2 in the DFSER In the DFSER the NRC asked GE to compare the dominant sequences from applicable existing BWR PRAs to those of the ABWR PRA. This was identified as Open Item 19.1.2.3-2. GE updated SSAR Appendix 19D.11 with the requested information that made an acceptable comparison to sequences from the Grand Gulf PRA. The staff considers this Open Item resolved.

Open Item 19.1.2.2.2-1 in the DFSER In the DFSER the staff noted that GE did not provide an up-to-date analysis of the robustness of its design to withstand seismic events that are beyond the design basis. This was identified as Open Item 19.1.2.2.2-1. GE subsequently provided this analysis in SSAR Appendices 19H and 19I. The staff found the ABWR seismic margins analysis acceptable. The staff considers this Open Item resolved.

Open Item 19.1.2.2.2-2 in the DFSER In the DFSER the staff stated that physical separation (3-hour fire barriers) between the three safety divisions is the most important assumption in the ABWR fire PRA analysis. The need for assuring the existence of this separation was identified as Open Item 19.1.2.2.2-2 in the DFSER. GE acceptably included the requirement to confirm these barriers in Section 19.8 of the SSAR and in ITAAC 2.15.10, 2.15.12, 2.15.5c, and 2.15.6. The staff considers this Open Item resolved.

Open Items 19.1.2.2.2-3 and 19.1.6.4-5 in the DFSER In the DFSER the staff stated that GE must provide guidance or an ITAAC for COL applicants regarding assumptions in the fire PRA for parts of the ABWR design that are not in the design certification (e.g., the ultimate heat sink). This was identified as Open Items 19.1.2.2.2-3 and 19.1.6.4-5 in the DFSER. GE provided acceptable guidance in SSAR Section 19.10. The staff considers this Open Item resolved.

Open Item 19.1.2.2.2-4 in the DFSER In the DFSER, the staff requested GE to submit an analysis of the important systems, structures, or components (SSCs) in the ABWR design with regard to the severe accident fire analysis. This was identified as Open Item 19.1.2.2.2-4 in the DFSER. GE submitted an acceptable analysis in Appendix 19K of the SSAR that lists important SSCs to be added to the reliability assurance program. The staff considers this Open Item resolved.

Interface Item I-9 in the DSER and Open Item 19.1.2.2.2-5 in the DFSER In the DFSER the NRC required that GE provide an internal flooding analysis. This was identified as Interface Item I-9 in the DSER and as Open Item 19.1.2.2.2-5 in the DFSER. GE subsequently submitted an acceptable risk-based internal flooding analysis in SSAR Appendix 19R and a list of SSCs in Appendix 19K to the SSAR that should be added to the reliability assurance program to help assure that the internal flooding PRA assumptions are part of the design of the as-built, as-operated plant. The staff found that the probabilistic internal flooding analysis provides a reasonable representation of the risks and safety insights associated with internal floods in the ABWR design. The staff considers this Open Item resolved.

COL Action Item 19.1.2.2.2-1 in the DFSER In the DFSER the staff noted that GE had not performed an external flooding analysis for the ABWR design. This was identified as COL Action Item 19.1.2.2.2-1 in the DFSER. The NRC required GE to develop a COL Action Item for performance of a site-specific PRA for external flooding to search for vulnerabilities. GE subsequently provided a COL Action Item in Section 19.9 of the SSAR that calls for such an evaluation. The staff finds this acceptable.

Outstanding Item 20 in the DSER and Open Item 19.1.2.2.3-1 in the DFSER In the DSER the staff required GE to provide a systematic analysis of risk for the ABWR design during modes other than full power. This was identified as Outstanding Item 20 in the DSER and as Open Item 19.1.2.2.3-1 in the DFSER. GE submitted an acceptable risk-based shutdown evaluation as Appendix 19Q to the SSAR. This analysis provides a reasonable representation of the risks and safety insights associated with operation of the ABWR in modes other than modes 1 and 2. The staff considers this Open Item closed.

Open Item 19.1.2.4.2-1 in the DFSER In the DFSER the staff noted that GE had not provided design acceptance criteria (DAC) and ITAAC based on evaluation of the ABWR PRA for both internal and external events modeled by the PRA. This was identified as Open Item 19.1.2.4.2-1 in the DFSER. GE subsequently submitted certified design material (CDM) and ITAAC that reflected insights gained from performing the ABWR PRA. These insights were documented in Section 19.8 of the SSAR. The staff found these insights as discussed in the CDM and ITAAC to be acceptable. The staff considers this Open Item resolved.

Open Item 19.1.2.4.1-1 in the DFSER In the DFSER the staff required GE to provide a list of systems, structures, and components (SSCs) for both internal and external events that should be used by the COL applicant to help develop its reliability assurance program (RAP). This was identified as Open Item 19.1.2.4.1-1 in the DFSER. GE subsequently provided this information in Appendix 19K to the SSAR. The staff found this list to be an acceptable representation of SSCs that should be included in RAP. The staff considers this Open Item resolved.

Interface Item 1 in the DSER and COL Action Item 19.1.5.2-1 In early SSAR submittals in the ABWR PRA, GE estimated that the loss of offsite power frequency was about 0.1 per reactor-year. The need to confirm this frequency at the specific site an ABWR is built at was identified as Interface Item 1 in the DSER and COL Action

Item 19.1.5.2-1 in the DFSER. The staff and GE subsequently agreed that confirmation of the applicability of this value for a particular site is the responsibility of the COL applicant. GE developed a COL Action Item to cover this issue in SSAR Section 19.9. The staff considers this acceptable.

Open Item 19.1.5.2-3 in the DFSER In the DFSER the staff required GE to develop an ITAAC that confirms the upgraded quality of the pipes that interface between high- and low-pressure systems. This was identified as Open Item 19.1.5.2-3 in the DFSER. This requirement was met acceptably in ITAAC 2.4.1, 2.4.2, 2.4.4, 2.2.2, 2.2.4, and 2.6.1. The staff considers this Open Item resolved.

Open Item 19.1.5.4-3 in the DFSER The staff required GE to develop an ITAAC for COL applicants to demonstrate how its design for interfacing systems meets the reliability assumptions and design insights for the ABWR. This was identified as Open Item 19.1.5.4-3 in the DFSER. This requirement was met acceptably by the inclusion of important PRA-based safety insights from Section 19.8 of the SSAR into various ITAAC, COL Action Items, and insights for future COL applicants. Reliability targets are provided by Appendix 19K of the SSAR and are to be included in the DRAP program. The staff considers this Open Item resolved.

Open Item 19.1.6.3.2-1 in the DFSER In the DFSER the staff required GE to provide the updated seismic analysis. This was identified as Open Item 19.1.6.3.2-1 in the DFSER. GE submitted an acceptable updated seismic margins analysis in SSAR section 19I. This analysis provides a reasonable representation of the areas of the ABWR design that are most vulnerable to seismic events beyond the design bases. The staff considers this Open Item resolved.

Confirmatory Item 19.1.6.3.2-1 in the DFSER In the DSER the staff indicated that it believed the fuel assembly seismic capacity was optimistic. This was identified as Staff Correction 5. After discussions with GE, the staff determined that a capacity of 1.2g is achievable, acceptable, and reasonable. The staff required that justification for the 1.2g capacity be documented in the SSAR. This was identified as Confirmatory Item 19.1.6.3.2-1 in the DFSER. GE acceptably provided the required documentation. The staff considers this Confirmatory Item resolved.

Staff Corrections 6, 7, and 8 in the DSER and Open Item 19.1.6.3.2-2 in the DFSER In the DSER the staff indicated that it believed that the seismic capacities of flat-bottomed tanks, diesel generators, and electrical equipment were optimistic and would need to be evaluated at the time of plant construction. These were identified as Staff Corrections 6, 7, and 8 in the DSER and as Open Item 19.1.6.3.2-2 in the DFSER. Subsequently, GE reduced some of these capacity assumptions and committed in Section 19.9.4 of the SSAR to confirm the seismic capacities of some of the equipment modeled in the ABWR seismic margins analysis. The staff considers this Open Item resolved.

Staff Correction 10 in the DSER In the DSER the staff indicated that use of both the Lawrence Livermore National Laboratory and EPRI hazard curves was appropriate when evaluating seismic risk. This was identified as Staff Correction 10 in the DSER. Subsequently, the staff requested and received Commission concurrence, through SECY 93-087 and its SRM, that seismic core damage frequency estimates are not necessary for evaluating severe accident concerns at ALWR sites, if a combination of PRA-based methods and seismic margins methods are used to provide insights into design robustness. The staff considers this Staff Correction resolved.

Staff Correction 10 in the DSER In the DSER the staff stated that GE should consider seismic hazard uncertainties in its seismic PRA analysis. This was identified in Staff Correction 10 in the DSER. Subsequently, the staff's position changed in that it recommended the submittal of a seismic margins analysis in lieu of a seismic PRA. GE then submitted an acceptable PRA-based margins analysis, which does not make use of hazard curves. The NRC concurs that use of the PRA-based margins method obviates the need to perform an uncertainty analysis on the hazard curve. The staff considers this Staff Correction resolved.

Interface Items 10 and 14 in the DSER and Open Items 19.1.6.3.2-3 and 19.1.6.3.7-1 in the DFSER In the DSER the staff stated that the COL applicant must confirm the seismic capacities of structures, systems, and components modeled in the PRA and these capacities should be included in the design specifications for the equipment. This was identified in Interface Items 10 and 14 in the DSER and Open Items 19.1.6.3.2-3 and 19.1.6.3.7-1 in the DFSER. Confirmation of High Confidence, Low Probability of Failure (HCLPF) of the most important systems, structures, and components is detailed in a COL Action Item in Appendix 19H.5 of the SSAR. During a walkdown to be performed on the as-built plant by the COL applicant, the overall seismic capability of a selection of the as-built systems credited in the seismic margins analysis will be physically examined. The staff considers this Open Item resolved.

Interface Item 11 in the DSER In the DSER the staff stated that COL applicants must perform a site-specific seismic PRA that evaluates the potential for seismic-induced soil failures, such as liquefaction, differential settlement, or slope instability. This was identified as Interface Item 11 in the DSER. The need to perform a site-specific seismic analysis was reinforced by the Commission's SRM to SECY 93-087. Regarding the need for a seismic PRA, following the DSER the staff modified its position and recommended to the Commission in SECY 93-087 that PRA-based seismic margins methods be used instead. GE acceptably performed such an analysis, which was submitted as Appendix 19I. However, this analysis assumed that either no liquefaction would occur at the site during beyond design bases earthquakes or that liquefaction would have no effect on the HCLPF of the ABWR structures, systems, and components. To preclude the chance of liquefaction at a specific site, GE modified the ABWR SSAR (Sections 2.0 and 2.3.2.31) to limit site selection to those sites where no liquefaction would be expected at the site-specific SSE level (The site-specific SSE level may be lower than the 0.3 g SSE for the ABWR design, perhaps as low as 0.1 g). The staff recently questioned the effect of liquefaction on sequence and plant HCLPFs. In Amendment 35, GE included a statement in a COL Action Item to ensure that a seismic margins assessment be conducted at 1.67 times the site-specific SSE. This is acceptable. The staff considers this Interface Item resolved.

Interface Item 12 in the DSER and COL Action Item 19.1.6.3.2-1 in the DFSER In the DSER the staff stated that COL applicants must perform a plant walkdown to examine the as-built ABWR for potential seismic-related problems. This was identified as Interface Item 12 in the DSER and COL Action Item 19.1.6.3.2-1 in the DFSER. GE submitted guidance for the COL applicant on performance of a seismic walkdown (SSAR Appendix 19H.5) and the need for a walkdown is identified by COL Action Item 19.9.5 in the SSAR. The staff considers this acceptable.

Interface Item 13 in the DSER In the DSER, Interface Item 13 dealt with several issues. First, the staff stated that COL applicants must develop deterministic and probabilistic site-specific response spectra for all sites. This requirement is covered by Sections 2 and 3 of the ABWR SSAR that require the COL applicant to compare the seismic characteristics of a chosen site to those assumed in the ABWR design. Second, the COL applicants were to demonstrate that the seismic design response spectra for the plant enveloped probabilistic site spectra used in the ABWR PRA. Third, if the site-specific deterministic or probabilistic response spectra exceed the spectra assumed in the ABWR PRA, the COL applicant was to perform a site-specific seismic PRA to confirm that the dominant sequences identified in the ABWR PRA had not been significantly altered. Subsequent to the DSER, the Commission (through SECY 93-087 and its SRM) concluded that a seismic margins analysis was an acceptable and preferred method of evaluating safety insights from beyond design bases seismic events. This position supersedes the staff requirement for development of a seismic PRA. The Commission position does require COL applicants to update the ABWR PRA to include pertinent site-specific external event information. In addition, in Appendix 19H.5 of the ABWR SSAR, it states that if the site parameters are not enveloped by those in the ABWR PRA, then the site-specific HCLPF capacities for these structures and components need to be established. The staff finds these commitments acceptable. The staff considers this Interface Item resolved.

<u>Open Item 19 in the DSER and Open Item 19.1.6.3.2-4 in the DFSER</u> In the DSER the staff required that GE address in the ABWR PRA the failure of containment penetrations and isolation valves during a seismic event. This was identified as Open Item 19 in the DSER and Open Item 19.1.6.3.2-4 in the DFSER. GE subsequently provided a PRA seismic margins analysis to supersede the seismic PRA. The seismic capability of containment penetrations and isolation valves is addressed in the seismic margins analysis and the HCLPF was determined to be greater than 0.7g, which is significantly greater than the SSE (0.3g). The staff considers this Open Item resolved.

<u>Staff Correction 3 in the DSER and Open Item 19.1.6.3.2-5 in the DFSER</u> In the DSER the staff pointed out that the ABWR PRA incorrectly took too much credit for firewater as a mitigating system in the seismic Class II CET. This was identified as Staff Correction 3 in the DSER and as Open Item 19.1.6.3.2-5 in the DFSER. GE subsequently provided a PRA seismic margins analysis to supersede the seismic PRA. The staff determined that this problem has been rectified in the seismic margins analysis. The staff considers this Open Item resolved.

<u>Open Item 19.1.6.3.2-6 in the DFSER</u> In the DFSER the staff requested that GE document the process it used to discover potential seismic failures such as failure of the safety relief valve discharge lines (in the wetwell air space). This was identified as Open Item 19.1.6.3.2-6 in the DFSER. GE subsequently provided an acceptable discussion of its method of searching for these failures. The staff considers this Open Item resolved.

<u>Open Item 19.1.6.3.4.1-1 and Confirmatory Item 19.1.6.3.4.2-1 in the DFSER</u> Because the original ABWR PRA seismic analysis was based on an outdated plant model, the staff requested in the DFSER that GE resubmit its seismic analysis, based on a PRA-based seismic margins method. This was identified as Open Item 19.1.6.3.4.1-1 and Confirmatory Item 19.1.6.3.4.2-1 in the DFSER. GE resubmitted an acceptable PRA-based seismic evaluation in SSAR Appendix 19I. The seismic margins analysis provides a reasonable representation of the risks and safety insights for the ABWR design associated with beyond design bases seismic events. The staff considers these Open Items resolved.

<u>Open Item 12 in the DSER</u> In the DSER the staff stated that GE had to submit a fire PRA for the ABWR design. This was identified as Open Item 12 in the DSER. GE subsequently submitted an acceptable analysis of severe accident fires at full power based on integration of the fire-induced vulnerability examinations (FIVE)

methodology and the ABWR PRA. The fire analysis provides a reasonable representation of the risks and safety insights for the ABWR design associated with beyond design bases fires. The staff considers this Open Item resolved.

Open Item 19.1.6.4-1 in the DFSER In the DFSER the staff requested clarification of a GE statement that some penetrations between divisions would be qualified to a lower standard if the penetrations contained non-safety equipment. This was identified as Open Item 19.1.6.4-1 in the DFSER. GE subsequently provided clarification that there would be no differentiation between the capabilities of penetrations that house either safety or non-safety equipment to prevent the spread of a fire. The staff finds this clarification acceptable. The staff considers this Open Item resolved.

Open Item 19.1.6.4-2 in the DFSER In the DFSER the staff requested clarification about the capabilities of buildings other than the reactor building to limit the spread of smoke between divisions. This was identified as Open Item 19.1.6.4-2 in the DFSER. GE subsequently provided information about the capabilities of smoke control systems in other safety related buildings. The staff found this information acceptable to resolve this issue. The staff considers this Open Item resolved.

Open Item 19.1.6.4-3 in the DFSER In the DFSER the staff required GE to assure that opening more than one door between a division with a fire and one without a fire would not defeat the smoke control system to the point that sufficient smoke/hot gases could move between divisions to damage equipment or hamper recovery activities. This was identified as Open Item 19.1.6.4-3 in the DFSER. GE stated in the SSAR that the smoke control system is capable of preventing the migration of smoke and hot gasses between divisions with an open door to the extent that the smoke and hot gasses cannot adversely affect safe shutdown capabilities, including operator actions. GE also addressed this issue by adding to a COL Action Item the need for the COL applicant to test the capability of the smoke control systems (i.e., defining, in part, how the test of the smoke control system should be performed including testing the capability of the system with a single door open between divisions). This is acceptable. The staff considers this Open Item resolved.

Open Item 19.1.6.4-4 in the DFSER In the DFSER the staff required GE to develop an ITAAC for the assurance that (1) 3-hour fire barriers between safety divisions are built/installed properly, (2) smoke, hot gases, and fire suppressants will not migrate from a safety division with a fire to a safety division without a fire to the extent that the division without the fire could be adversely affected, (3) the smoke control system must be shown to be capable of preventing migration of smoke between divisions with either an open door between the divisions or the failure of a supply or exhaust line valve to close, and (4) water intrusion mitigation devices (e.g., curbs and shields) are in place. This was identified as Open Item 19.1.6.4-4 in the DFSER. GE acceptably resolved these issues by committing to Item (1) above through ITAAC 2.15.10 and 2.15.12 and to Items (2), (3), and (4) above through ITAAC 2.15.5c and 2.15.6. The staff notes that GE has modified the ABWR design so that there are no longer any curbs around doors in the Reactor Building. While suppressants now can migrate under doors, flooding is not a concern due to the acceptable drainage capabilities of the higher floors of the Reactor Building. The staff considers this Open Item resolved.

Open Item 19.1.6.4-6 in the DFSER In the DFSER the staff requested GE to provide an analysis of the important systems, structures, and components in the ABWR design with regard to the severe accident fire analysis. These components are to be factored into the reliability assurance program. This was identified as Open

Item 19.1.6.4-6 in the DFSER. GE acceptably provided this information in an update to SSAR Appendix 19K. The staff considers this Open Item resolved.

Interface Item 9 in the DSER and Open Item 19.1.6.5-1 in the DFSER In the DFSER the staff required GE to perform an internal flood PRA for the ABWR design. This was identified as Interface Item 9 in the DSER and Open Item 19.1.6.5-1 in the DFSER. GE subsequently submitted an acceptable internal flood PRA in Appendix 19R of the SSAR. The risk-based internal flood analysis provides a reasonable representation of the risks and safety insights associated with beyond design bases internal floods. The staff considers this Open Item resolved.

Open Item 19.1.2.4.3-3 in the DFSER: As required by 10 CFR 50.34(f), GE performed an evaluation of design alternatives for severe accident prevention and mitigation. Because the staff had not completed its review of this issue at the time of the DFSER, this was identified as DFSER Open Item 19.1.2.4.3-3. The staff's review has been completed and is documented in FSER Section 20.3. The results of this review indicate that none of the design alternatives analyzed by GE are justified on the basis of cost-benefit considerations. This resolved Open Item 19.1.2.4.3-3.

Open Item 19.1.2.4.3-4 in the DFSER: In the DFSER, the staff indicated that GE was to submit additional information to support the staff evaluation of severe accident mitigation design alternatives (SAMDAs) required by NEPA. This was identified as DFSER Open Item 19.1.2.4.3-4. This information has been submitted by GE, and has been used by the staff in its evaluation of SAMDAs for the ABWR. The staff evaluation is documented in the Environmental Appraisal for the ABWR. This resolved Open Item 19.1.2.4.3-4.

Open Items 19.1.5.2-4 and 19.1.11.4-1 in the DFSER: In the DSER, the staff noted that GE did not provide an assessment of LOCA events outside the containment (ex-containment LOCAs) in combination with failure of the isolation valves. Of particular interest were steam line breaks in the RCIC steam piping and the RWCU. This was identified as DSER Outstanding Item 4, and DFSER Open Items 19.1.5.2-4 and 19.1.11.4-1. In response to NRC and ACRS concerns, GE submitted an assessment of the frequency of core damage from events initiated by LOCAs outside containment, and provided additional clarification regarding the effects of a break in the RWCU suction line. The NRC expressed a number of concerns with GE's initial analysis during the interactive review. For example, the potential for an unisolated LOCA outside containment to result in eventual depletion of coolant inventory, or the failure of injection pumps as a result of high temperature and steam concentrations in the vicinity of the break was not reflected in GE's event trees. GE subsequently revised their analysis to address these and other concerns. The NRC has reviewed this analysis and finds that it provides a reasonable representation of the potential for LOCAs outside containment to lead to core damage. On the basis of this review, the NRC considers the open items regarding the frequency of LOCAs outside containment (DFSER Open Items 19.1.5.2-4 and 19.1.11.4-1, and DSER Outstanding Item 4) to be resolved.

Open Item 19.1.5.6-1 in the DFSER: The NRC's initial evaluation of the Human Reliability Analysis (HRA), documented in the DSER, was based on the review of 12 aspects of the analysis. These related to HRA documentation, the material available to support the HRA, the types of analyses performed, the quantification methods and performance shaping factors utilized, the completeness and types of human actions modelled, the sources of generic data used, and how the effects of the advanced technology on the operator's role/tasks are addressed in the HRA. Based on the initial review the NRC identified four Outstanding Items, one Confirmatory Item, and seven Interface Requirements in the DSER relevant to the HRA. In the DFSER the staff grouped these items into three general concerns: (1) a lack of details and documentation regarding GE's modelling and analyses of human errors (Open Item 19.1.5.6.1-1), (2) a lack of sensitivity or uncertainty analyses of human errors modelled in the ABWR PRA (Open Item 19.1.5.6.2-1), and (3) a lack of design details necessary to support a plant-specific HRA (Open Item 19.1.5.6.3-1). The resolution of these concerns is discussed below.

Open Item 19.1.5.6.1-1 in the DFSER: Three of the DSER outstanding items are closely related and involve the adequacy of HRA models and supporting documentation. The NRC concerns expressed in the DSER are summarized as follows:

- Adequacy and Completeness of the Documentation (DSER Outstanding Item 7)

  The HRA-related documentation provided in the SSAR was incomplete, and did not provide indication of the analyses used, how the HRA analysis methods were implemented, what performance models and performance shaping factors were considered, or how human error probabilities (HEPs) were quantified.

- Quantification Methods Used to Estimate Human Error Probabilities (DSER Outstanding Item 8)

  Discussion of quantification methods or detailed rationale for the HEP estimates was provided for only six of the human actions modelled. For most of the other errors the method used was identified in tables only.

- Generic Human Error Data Sources (DSER Outstanding Item 9)

  While the documentation in the SSAR provided general information on the source data used for HEP estimates, references to specific data tables within the referenced documents were generally not included.

These three items were identified as DFSER Open Item 19.1.5.6.1-1.

In response to the NRC concerns, GE provided additional documentation for an expanded set of human actions treated in the ABWR risk analyses. Human actions with greatest potential impact on core damage frequency (Level 1 PRA results) were identified by GE based on a review of leading cutsets in the PRA, as well as the results of sensitivity analyses and importance analyses. Important human actions for containment performance (Level 2 PRA results) were identified based on a review of those operator actions either modelled probabilistically in the containment event trees, or treated deterministically in the Level 2 analyses, i.e., operator actions that were assumed to be successfully accomplished, such as operating or securing containment sprays. A description of each of the important actions, and where applicable, the basis for quantification in the PRA were provided. Human actions

judged to be important to each of the deterministic analyses of external events were also identified, and further described.

The NRC considers the HRA-related documentation provided in the final SSAR to be a noticeable improvement over the original SSAR in terms of the completeness of human actions covered, and the bases for treatment in the PRA. In general, the documentation provides an adequate description of the HRA analyses used for those errors of greatest significance to risk results. The NRC notes that the state of the HRA documentation is essentially unchanged for many of the remaining human actions in the Level 1 PRA, but that these human actions are not expected to influence ABWR core damage frequency since they were not identified as significant contributors in the sensitivity analyses. In addition, the inconsistencies alluded to in the DSER have been resolved in the updated SSAR. Accordingly, the NRC finds that the additional information regarding the important human actions modelled in the risk analyses acceptably resolves the concerns raised in the DSER regarding documentation.

With regard to HRA methods, the NRC notes that the bulk of the human actions treated probabilistically in the risk analyses have been quantified based on a generalization of HEPs from previous HRAs, primarily the GESSAR II PRA, without explicit consideration of such aspects as performance shaping factors and the plant-specific human system interface. The use of these values was judged acceptable by GE because of improved human-system interface design and greater automation envisioned for the ABWR. The NRC recognizes that it is not possible to conduct a meaningful detailed human reliability analysis in the absence of such specific items as emergency operating procedures, control room design layout and staffing, and plant simulators. Moreover, the objectives of the HRA performed at the design certification stage are more limited than for a detailed HRA for a completed plant design, and focus on obtaining a nominal assessment of the human interactions to support the overall risk assessment. Given the limitations in design detail and the more limited objectives of the design certification HRA, the HEP values used by GE appear to be reasonable representations of operator performance. Based on sensitivity analyses performed by GE, the potential increase in core damage frequency for the ABWR given higher human error probabilities appears to be limited to about 2 orders of magnitude above the base case frequency. On the basis of the improved documentation, and the results of the supporting sensitivity analyses, the NRC concludes that the DSER issues related to HRA models and documentation have been acceptably resolved.

<u>Open Item 19.1.5.6.2-1 in the DFSER</u>: The NRC noted in the DSER that GE did not perform an assessment of human error probability uncertainty bounds (error factors) or the sensitivity of core damage frequency to HEP values. As a result, this was classified as Outstanding Item 10 in the DSER. In response to this item, GE performed an analysis of the impact of variations in HEPs on estimated core damage frequency. In these analyses, GE varied HEPs modelled in the Level 1 and 2 analyses either: (1) for each error individually, (2) for all errors simultaneously, and (3) simultaneously within categories of human errors. An analysis of human actions occurring in the leading cutsets of the ABWR internal events assessment was also performed. These analyses provide a number of insights into the impact of human error on ABWR core damage frequency estimates. In addition to the sensitivity analyses, GE considered the affect of uncertainties in HEPs on core damage frequency as part of the Level 1 uncertainty analysis. The NRC considers that the uncertainty and sensitivity analyses provided by GE acceptably resolve the concerns raised in the DSER. On this basis this issue is considered resolved.

<u>Open Item 19.1.5.6.3-1 in the DFSER</u>: In the DSER, the NRC identified a number of items for which resolution hinges on design details that will not be available until the COL-stage, e.g., control room design, plant-specific data, and human system interface for advanced technologies. The specific items and rationale for their initial classification as Interface Requirements is provided below:

- Material Available to Support the HRA (Interface Requirement 2)

  It did not appear that the material available to the HRA team at the time of the DSER was adequate for a detailed evaluation of human action or an estimate of the HEPs.

- Human-System Analyses Performed (Interface Requirement 3)

  System-level operating procedures and emergency procedure guidelines were used as a basis for task analyses (in support of human system interface design). However, the sample task analysis which was provided for review was not considered to be to a suitable level of detail to support a detailed HRA.

- Adequacy of Human Action Modelling (Interface Requirement 4)

  Although the ABWR human action modelling appeared to be reasonable in terms of conventional control boards, there was a concern that as the design becomes more detailed, additional human action modelling may be required to reflect its increased automation and advanced technology.

- Performance Shaping Factors Evaluated (Interface Requirement 5)

  Due to immaturity of the design, performance shaping factors (PSFs) were considered for only a limited number of human actions in the HRA, and important PSFs such as emergency operating procedures (EOPs) and human-system interface (HSI) design were not analyzed.

- Treatment of Advanced Technology (Interface Requirement 6)

  The changing role of the operator due to increased automation was not analyzed for its HRA implications.

- Generalization from Earlier PRAs (Interface Requirement 7)

  Most of the HEPs were taken from the GESSAR II PRA, and were not modified for use in the ABWR.

- Insights Gained from the Analysis (Confirmatory Item 5)

  The NRC considered GE's plan for usage of information and insights gained from the HRA to support the system/operational design to be reasonable, but that a final determination on acceptability could not be made until further design development.

These seven items were identified as DFSER Open Item 19.1.5.6.1-1.

Subsequent to issuance of the DSER, the NRC reevaluated its expectations for the HRA performed at the design certification stage. This reevaluation stemmed from the NRC's recognition that it is not practical or possible to conduct a meaningful detailed human reliability analysis in the absence of such specific items as emergency operating procedures, control room design layout and staffing, and plant simulators, and the belief that major insights offered by an HRA can be still be extracted from a scoping-type analysis. As noted in Chapter 1, Appendix A of the NRC review of EPRI's ALWR Utility Requirements Document (NUREG-1242, Volume 2, Pt. 1), the objectives of the HRA performed at the design certification stage are more limited than for a detailed HRA for a completed plant design, and focus on obtaining a nominal assessment of the human interactions to support the overall risk assessment. The assessment of human interactions required at the design certification stage, termed a scoping analysis or approximate quantification, should be sufficient to accomplish the following stated objectives: (1) identify the types of human interactions that may be important to risk for the ALWR design, (2) provide a nominal quantification of the human interactions sufficient to support the overall assessment of core damage frequency, and frequency of severe release, (3) provide a mechanism to investigate the potential effects of varying the reliability assessed for human interactions, and (4) establish a framework for performing a more extensive HRA when the design progresses to the state of an actual plant, with operating procedures, layout, etc. With these more limited objectives, the need for a detailed assessment of human reliability is, in effect, shifted from the design certification stage to the COL stage, when the design details are available. The acceptability of this approach is contingent upon satisfying the following four criteria at design certification: (1) all significant human actions are represented in the human reliability analysis, (2) reasonable, scoping-type human error rates are used for critical human actions, (3) sensitivity and uncertainty analyses are used to identify important human actions and further assess the impact of human error probability values, and (4) a process is established by which the COL-applicant will confirm the adequacy of the HRA treatment after the design details have been developed.

Based on an analysis and classification of the human actions identified in the fault and event trees, the NRC concludes that the variety and types of human actions modelled in the ABWR HRA are acceptable. While the completeness of the HRA for the detailed design cannot be confirmed at this time, the NRC has not identified any major omissions with regard to the human actions represented in the analysis. The staff notes that human error probability values used by GE generally ranged from 1E-3 to 0.1, with the bulk of the human actions represented by an error rate of 0.01 or greater. These values appear to be reasonable, and should be achievable given the emphasis on human factors principles and dynamic simulation as part of the detailed control room design process. The major exception is the HEPs assumed for sensor miscalibration, which were typically assigned values on the order of 1E-5. A special procedure governing sensor calibration will need to be adopted by the COL applicant to assure the validity of these values. This is addressed by SSAR Section 19.9.8. The NRC concludes that the HEP values used by GE are acceptable representations of operator performance, given the limitations in design detail and the more limited objectives of the design certification HRA. With regard to sensitivity and uncertainty analyses, GE has performed an analysis of the impact of variations in HEPs on estimated core damage frequency, and also included the affect of uncertainties in HEPs on core damage frequency as part of the Level 1 uncertainty analysis. The NRC considers that these analyses acceptably resolve the concerns raised in the DSER related to sensitivity and uncertainty analyses.

The remaining criterion concerns the establishment of a process by which the COL applicant will confirm the adequacy of the HRA treatment after the design details have been developed. The Human Factors Engineering (HFE) process, described in FSER Section 18 and controlled by the HFE process ITAAC/DAC, establishes several requirements that are relevant to the HRA, specifically: (1) as part of Design Commitment 1.b, an HFE Program Plan shall be developed and implemented which establishes methods for addressing the potential for operator error, (2) as part of Design Commitment 4, a Task Analysis Implementation Plan shall be developed and implemented which establishes methods for identification of tasks that are critical to safety in terms of importance for function achievement, potential for human error, and impact of task failure, and (3) as part of Design Commitment 7.a, a Human Factors Verification and Validation Implementation Plan shall be developed and implemented which establishes that the HFE performance measures to be used as the basis for evaluating the dynamic task performance results shall address operating crew errors and error rates. The conformance of the final design with each of these commitments will be verified in order for the COL-applicant to load fuel, as discussed in ITAAC 3.1. The NRC considers that these commitments provide reasonable assurance that the detailed task analyses called out in the DSER will be performed, and that human actions important to PRA results will be given explicit consideration as part of the control room design process. The process also provides assurance that the final control room design has not introduced any human engineering deficiencies which would either significantly increase the error rates for human actions modelled in the HRA, or the potential for additional, risk-significant errors not modelled in the HRA. On this basis, the staff considers the open items identified in the DSER and DFSER concerning plant-specific PRA have been acceptably resolved.

Open Item 19.1.7.2-1 in the DFSER: The staff noted in the DSER that although a certain amount of drywell-to-wetwell leakage is allowed in a BWR suppression containment, GE dismissed consideration of suppression pool bypass effects in the CETs because of the low estimated frequencies and risk. Citing past BWR operating experience, results of a simplified scoping calculation, and risk implications of containment bypass, the staff concluded that GE should factor the potential for drywell-to-wetwell bypass into the CETs. This was identified as DSER Outstanding Item 13 and DFSER Open Item 19.1.7.2-1. As part of DSER Outstanding Item 17, the staff also requested that GE consider uncertainty aspects of drywell-to-wetwell bypass. GE addressed the issue of suppression pool bypass during severe accidents by performing an assessment of potential suppression pool pathways, and modifying the CETs to address the potential for suppression pool bypass via the drywell/wetwell vacuum breakers. Suppression pool bypass due to leakage through vacuum breakers is included in the revised CETs as a top event, and is quantified using a supporting DET. The DET considers the potential for a vacuum breaker to stick open, the potential that vacuum breaker leakage occurs, and the potential that aerosols subsequently plug the leak path. In addition, GE performed a set of sensitivity analyses that investigated the effects of various assumptions on the time of containment overpressure protection system (COPS) actuation and fission product release.

The NRC has reviewed GE's screening of potential suppression pool bypass paths. The staff notes that GE's analysis appears to underestimate the likelihood of completely bypassing the suppression pool, for certain lines and scenarios, however, this does not undermine the key insights obtained from the analysis. The NRC concludes that GE's screening of potential suppression pool bypass paths is systematic and sufficiently detailed to identify the bypass paths of major risk significance. This screening adequately resolves that portion of Open Item 19.1.7.2-1 concerning which drywell-wetwell bypass paths have the greatest risk significance for the ABWR. With regard to the modelling of suppression pool bypass in the CETs, the NRC notes that GE's estimates of the probability of stuck open vacuum breakers and vacuum breaker leakage are based on a significant amount of operating data applicable to the ABWR vacuum breaker design, and are reasonable. While there is a considerable probability of vacuum breaker leakage in the ABWR design, the impact of this leakage on containment response would not be significant due to the high reliability of drywell and/or wetwell sprays in the ABWR. On the basis of the revised CET/DETs and supporting justification and sensitivity analyses provided by GE, the NRC concludes that the issues related to modelling of suppression pool bypass in the PRA have been adequately addressed by GE. This resolved DFSER Open Item 19.1.7.2-1, and DSER Outstanding Item 13, and that aspect of Outstanding Item 17 dealing with drywell-to-wetwell bypass.

Open Item 19.1.7.3-1 in the DFSER: The staff indicated in the DSER, that although the COPS is intended to provide protection against rare sequences in which containment integrity is challenged by overpressurization, there are potential adverse effects of the COPS. Therefore, the staff concluded that GE must justify the COPS and the pressure relief setpoint, considering the downside risks. This was identified as DSER Outstanding Item 14 and DFSER Open Item 19.1.7.3-1. In response to this concern, GE modified the CETs to represent the actuation of COPS as a separate top event, and performed additional assessments to justify the setpoint, sizing, and net risk impact for the COPS. In establishing the set point pressure for the final design of COPS, GE estimated the probability of actuating the system (and hence releasing fission products) when it might have been possible to prevent a release by recovering the RHR before containment failure, and showed this value to be small. The probability of drywell head failure before rupture disk opening was also estimated and reflected in the CETs. In determining the size of the rupture disk, GE selected a rupture disk flow area sufficient to assure enough steam flow to prevent further containment pressurization, while maintaining the flow area low enough that flashing and swell of the suppression pool would not introduce water into the COPS piping, potentially damaging the system. The associated rupture disk flow area is sufficient to deal with the decay heat levels associated with accidents with reactor scram, and also provides a marginal capability to deal with low power ATWS events. GE's assessment also indicates that suppression pool swell and flashing associated with COPS actuation will have no significant impact on COPS operability/survivability or fission product releases. To demonstrate the net impact of COPS on risk for the ABWR, GE provided a comparison of ABWR performance with and without the COPS, including comparisons of the time and magnitude of fission product release for the frequency dominant sequence for the ABWR, as well as an assessment of the impact of COPS on the frequency of core damage and drywell head failure for various accident classes. The results indicate that while COPS reduces the time of fission product release, the earlier times of release are offset by the fact that the releases through COPS are scrubbed by the suppression pool (without COPS, fission products would likely be released from the drywell head region of the ABWR containment, unscrubbed, upon reaching the structural limit of containment). GE also demonstrated that the probability of drywell head failure would increase significantly without COPS.

The staff concludes that the updated CETs provide an acceptable representation of the operation and potential failure of COPS, and that GE's estimates of the probability of rupture disk opening are reasonable and consistent with the results of GE's analysis of uncertainties in rupture disk actuation pressure and drywell head failure pressure. Based on the estimated uncertainties in commercially available rupture discs, and the NRC's assessment of the ABWR pressure capability provided in Section 19.2 of the FSER, the NRC considers that the setpoint of the system provides a reasonable balance between the competing goals for the system, specifically, minimizing the probability of drywell head failure while maximizing the time before fission product release to the environment, and is therefore acceptable. On the basis that the COPS system is capable of controlling containment pressurization for accident sequences with reactor scram, without incurring adverse impacts associated with suppression pool flashing, the NRC concludes that the sizing of the COPS is also adequate. The NRC concludes that the COPS system has a significant net benefit which outweighs the potential negative aspects of the system. The NRC further concludes that the design of the system, in terms of sizing and pressure setpoint, is acceptable, and that further refinement by GE is not necessary. This resolved the open item regarding the net risk impact of the containment overpressure protection system (DFSER Open Item 19.1.7.3-1, and DSER Outstanding Item 14).

<u>Open Item 19.1.7.3-2 in the DFSER</u>: In the independent NRC calculations of conditional containment failure probability (CCFP) reported in the DSER, containment failure was defined as either: (1) a breach of containment structural integrity sufficient to allow a large release path, or (2) a wholebody dose at 1/2 mile from the reactor in excess of 25 rem. Early in the ABWR review, the structural integrity definition

of containment failure was interpreted to include actuation of COPS before 24 hours. However, the NRC later recognized that venting in less than 24 hours should not be equated with containment failure, and in the DSER committed to address the issue of vent timing separately in the FSER. This was identified as DSER Staff Correction 9 and DFSER Open Item 19.1.7.3-2. Consistent with this commitment, containment venting was not assumed to constitute containment failure under the structural integrity definition in the staff's review of the updated Level 2 and 3 PRA, regardless of the time of venting. This resolved DSER Staff Correction 9 and DFSER Open Item 19.1.7.3-2.

Open Item 19.1.7.4-1 in the DFSER: The staff discussed the benefits of the passive flooder system in the DSER and also identified a number of technical concerns introduced by the system. These issues related to calculations and experiments that suggest: (1) significantly more concrete degradation of the pedestal wall may occur than predicted by the MAAP code, (2) the passive flooder system may actuate before significant amounts of the core debris enter the lower drywell, introducing the possibility of a subsequent fuel/coolant interaction (FCI), and (3) a fuel/coolant interaction may occur when water is poured onto core debris in the lower drywell, as during the intended operation of the passive flooder system. Recognizing that an effective flooder system is necessary to maintain safe temperatures within the lower drywell following a severe accident, the staff required that GE further examine these aspects of the design. This was identified as DSER Outstanding Item 15, and DFSER Open Item 19.1.7.4-1. In response to NRC concerns, GE modified the CETs to include water addition to the lower drywell via the passive flooder system as a top event, and provided additional information and analyses in support of the design and operation of the passive flooder system. This included PRA sensitivity analyses, design details of the passive flooder valves, estimates of the time required to actuate the flooder valves, criteria and analyses related to sizing of the flooder system, and an assessment of the potential for and consequences of premature actuation of the passive flooder valves.

The NRC considers the modelling of the passive flooder system in the ABWR CETs to be acceptable. Model quantification is also acceptable given the relatively small sensitivity of ABWR risk to the passive flooder system, and the fact that various alternative means of late water injection into the lower drywell exist in the ABWR design given passive flooder system failure. The NRC has reviewed GE's calculations related to the sizing of the passive flooder system, and considers the system to be adequate in terms of its capability to remove heat from the debris bed, even with only two flooder valves fully open. With regard to the concern that significantly more concrete degradation of the pedestal wall may occur than predicted by the MAAP code, staff calculations for the ABWR confirm GE's claim that the likelihood of reactor pedestal failure is unlikely within the first 50 hours, even if water is not added to the lower drywell. Thus, the modelling of the passive flooder system and its impact on debris coolability will not have a significant impact on reactor pedestal failure and risk results for the ABWR. Based on the design of the valves and the valve testing that will be performed by the vendor/COL-applicant, the staff concludes that the potential for passive flooder valve operation prior to reactor vessel breach is insignificant, and need not be considered in estimating the probability of a flooded lower drywell. However, even if the lower drywell is flooded at vessel failure, the results of calculations performed by both GE and NRC indicate that the ABWR reactor pedestal and containment are capable of withstanding the associated loads. The timeliness and rate of water addition does not appear to be a concern because the likelihood of reactor pedestal failure is extremely unlikely within the first 50 hours, and because FCIs do not pose a significant threat to the ABWR containment. The NRC concludes that all concerns raised in the staff's review of the passive flooder system have been adequately addressed, and that further refinement of the passive flooder system by GE is not necessary. This resolved the open item regarding the net risk impact of the passive flooder system (DFSER Open Item 19.1.7.4-1, and DSER Outstanding Item 15).

Open Item 19.1.7.5-1 in the DFSER: GE's original CET analysis assumed that pouring water on the core debris would lead to a coolable debris bed. The NRC noted in the DSER that calculations and experiments suggest that core-concrete interactions (CCI) could continue even after water is injected into the lower drywell from the passive flooder system. In the extreme, this could result in structural failure of the reactor pedestal and earlier actuation of the COPS due to production of additional non-condensable gases. Accordingly, the staff required GE to address the threat to containment integrity from a CCI in the event that core debris is not quenched by the overlying water pool and to also address the influence of continued concrete attack on source terms. This was identified as DSER Outstanding Item 16, and as part of Outstanding Item 17. This item was further divided into three separate items in the DFSER: treatment of CCI and reactor pedestal failure in the CETs (Open Item 19.1.7.5-1), structural analysis of the reactor pedestal (Open Item 19.1.7.5-2), and impact of CCI on source terms (Open Item 19.1.8-1). In response to staff concerns, GE modified the CETs to include CCI and reactor pedestal failure (PED) as separate top events in the revised CETs, and developed DETs to substantiate the branch point probabilities for these two top events. GE also provided justification for the selection of DET branch point probabilities, and parametric analyses of core concrete ablation rates using a modified version of the MAAP code.

The NRC has reviewed the details of GE's CET/DET analysis for CCI and reactor pedestal failure for consistency and completeness of the analysis relative to the current body of relevant severe accident information. GE's treatment of the probability of COPS actuation due to CCI appears to be reasonable. The NRC identified a number of areas in which it was not in agreement with GE's treatment of CCI in the revised CETs, e.g., the manner in which the event trees have been quantified appears to overstate the likelihood that CCI will be terminated by the addition of water. These problems, however, do not have a significant bearing on the ABWR risk profile, as follows: The structural capacity of the ABWR reactor pedestal, in conjunction with GE's and NRC's estimates of maximum concrete erosion depths indicate that reactor pedestal failure due to CCI will not occur until well after 24 hours. Thus, the contribution of reactor pedestal failure to risk for the ABWR will be very small. The NRC's offsite consequence calculations also indicate that the additional risk due to the additional ex-vessel source terms is not significant. On these bases, the NRC considers the open item regarding treatment of CCI in the CET (DFSER Open Item 19.1.7.5-1, and this aspect of DSER Outstanding Items 16 and 17) to be resolved.

Open Item 19.1.7.5-2 in the DFSER: GE presented the results of a structural analysis of the reactor pedestal which indicated that pedestal integrity is maintained for an ablation depth up to 1.55 m. However, this information was not submitted in time for evaluation in the DFSER. Accordingly, this was identified as DFSER Outstanding Item 19.1.7.5-2. NRC subsequently completed review of the analysis, as documented in Section 19.2 of the FSER. Based on the review, the NRC agrees with GE's characterization of the margin in the pedestal, and the use of a 1.55 m ablation depth as a pedestal failure criteria. Both GE and NRC-sponsored calculations indicate that ablation depths will be significantly less than this well after 24 hours. The considerable margins in the structural capacity of the ABWR pedestal, combined with the estimated radial erosion depths ensure that pedestal failure from CCI will not occur within the first 24 hours of an accident. This observation enables the NRC to concur with GE's conclusion that the possibility of pedestal failure due to CCI is very remote in the ABWR, and will not have a significant bearing on the ABWR risk profile. This resolved the open item concerning the structural capacity of the reactor pedestal (DFSER Open Item 19.1.7.5-2).

Open Item 19.1.7.6.1-1 in the DFSER: In the DSER, the staff noted that GE did not consider direct heating and fuel-coolant interactions as credible phenomena that could lead to failure of the ABWR containment and did not include these phenomena in the ABWR CET. The staff indicated that these and other phenomena can occur under certain circumstances and may possibly be of sufficient magnitude to threaten the integrity of the containment. Accordingly, the staff required GE to modify the CETs to consider the following severe accident phenomena: direct containment heating (DCH), fuel-coolant interactions (FCIs), continued core concrete interaction, and drywell/wetwell suppression pool bypass. This was identified as DSER Outstanding Item 17. This item was tracked as five separate items in the DFSER: direct containment heating (Open Item 19.1.7.6.1-1), rapid steam generation loads from FCIs (Open Item 19.1.7.6.2-1), energetic FCIs, or steam explosions (Open Item 19.1.7.6.2-2), core concrete interactions (Open Item 19[illegible]), and drywell/wetwell suppression pool bypass (Open Item 19.1.7.2-1).

In response to NRC concerns, GE modified the ABWR containment event trees to explicitly reflect the impact of DCH. Direct containment heating is addressed in the early containment challenge (CI) top event in GE's revised CETs. A main event tree and three supporting DETs are used to evaluate this top event. The NRC has reviewed the details of GE's CET/DET analysis for DCH for consistency and completeness relative to the current body of severe accident information. The NRC notes that GE's analysis treats, to varying degrees, all major processes/phenomena believed to have a major influence on DCH pressure loads via either: (1) explicit representation in the DET, (2) bounding assumptions in the methodology used to calculate the pressure loads, or (3) supporting sensitivity calculations. The NRC also notes the parameter values assigned to each branch of the DET are generally consistent with the results of existing severe accident analyses, and represent a reasonable attempt on the part of GE to reflect the ranges over which these parameters might be expected to vary. Finally, the split fractions assigned by GE to each of the branches appear to reasonable, and limited primarily by the lack of experimental data directly applicable to the ABWR configuration. On the basis of these observations, the NRC concludes that the structure of the CET/DET, and assigned split fractions are acceptable. During its review, the NRC identified a number of modelling assumptions which would lead to underestimates in GE's calculated DCH pressure loads. The principal concerns involved the following: (1) use of baseline containment pressures significantly lower than observed in NRC-sponsored BWR accident analyses (due primarily to differences in in-vessel hydrogen production), (2) lack of consideration of combustion/recombination of hydrogen with residual oxygen in containment, and (3) the assumption of a pure steam environment in the lower drywell prior to reactor vessel failure. In response to NRC questions on the DCH analysis, GE performed additional calculations to explore the sensitivity of DCH pressure loads to various DCH modeling parameters and assumptions, including those issues raised by the staff. In view of the remaining NRC concerns with GE's base case, as well as the large uncertainties inherent in estimating DCH pressure loads, NRC adopted the bounding GE sensitivity case as the basis for staff estimates of the contribution of DCH to CCFP and risk for the ABWR. Based on the containment's ability to withstand the pressure loads associated with the sensitivity case, the NRC considers GE's treatment of DCH in the CETs to be acceptable. This resolved DFSER Open Item 19.1.7.6.1-1 and that portion of DSER Outstanding Item 17 dealing with DCH.

Open Items 19.1.7.6.2-1 and -2 in the DFSER: In the DFSER, the staff indicated that it had not completed its review of GE's assessment of rapid steam generation loads, and would require GE to provide additional rationale for excluding energetic FCIs (steam explosions) from consideration in the PRA. This was identified as Open Items 19.1.7.6.2-1 and -2. In response, GE provided additional analyses to justify excluding rapid steam generation events and steam explosions from consideration in the PRA. GE's assessment is based on a deterministic evaluation of pressure loads and ABWR structural capability, combined with probabilistic evaluation of the likelihood of having a flooded drywell at the time of vessel breach. As part of the deterministic evaluation, GE assessed the maximum rates at which steam would be generated from fuel-coolant interaction in the lower drywell from various sources and estimated the corresponding peak pressures for the ABWR. GE also provided calculations of the loads associated with ex-vessel steam explosions. On the basis that peak containment pressures/loads were estimated to be within the structural capabilities of the containment, GE concluded that rapid steam generation events or steam explosions that lead to failure of the lower drywell are not credible. With regard to the probabilistic evaluations, GE provided an assessment of the potential for a pre-flooded cavity based on consideration of various ways in which water could be introduced into the lower drywell of the ABWR prior to reactor vessel failure. Based on this assessment, GE concluded that the total frequency of sequences in which the lower drywell would be flooded prior to reactor vessel breach would be extremely small, and would represent about 0.3% of all core damage sequences.

The staff's assessment of GE's deterministic calculations is provided in FSER Section 19.2, and supports GE's claim that the loads associated with FCIs are within the structural capabilities of the ABWR containment. NRC is also in agreement with GE's assessment of the probability of a flooded cavity. The staff notes that the potential contribution of ex-vessel FCIs to risk for the ABWR was assessed as part of the original NRC review documented in the DSER. In that review FCIs were estimated to potentially increase the overall conditional containment failure probability by about 1 percentage point. However, the original NRC estimate was based on a 50% probability of water in the lower drywell at the time of vessel failure. As stated previously, the probability of a flooded lower drywell cavity at the time of reactor vessel failure is significantly less than this. Accordingly, ex-vessel FCIs represent an even smaller challenge to the ABWR containment than suggested in the original evaluation. Although the probability of containment failure due in-vessel steam explosions has not been further evaluated by GE as part of their assessment of FCIs, the NRC expects that this failure mode is insignificant in terms of both its absolute frequency and contribution to CCFP. While the NRC does not agree with certain of the aspects of GE's phenomenological assessment of FCI loads (see FSER Section 19.2), the NRC concurs with GE's conclusion that the likelihood of containment failure due to FCIs is very small in the ABWR design, due principally to the small likelihood of water being present in the ABWR lower drywell region at the time of vessel breach. On this basis, the NRC concludes that FCIs need not be explicitly modelled in the ABWR CET. This resolved the DFSER open items regarding rapid steam generation (Open Item 19.1.7.6.2-1) and energetic FCIs (Open Item 19.1.7.6.2-2), and that portion of DSER Outstanding Item 17 dealing with FCI.

Open Items 19.1.7.7-1 and 19.1.8-1 in the DFSER: The staff noted in the DSER that GE did not perform an uncertainty analysis as part of the Level 2 PRA. Recognizing the large uncertainties inherent in calculations of severe accident phenomena and accident progression, the NRC concluded in the DSER that GE must provide a treatment of uncertainty as part of its overall analysis. This was identified as DSER Outstanding Item 18, and DFSER Open Item 19.1.7.7-1. The NRC also found GE's method for calculating source terms to be generally acceptable, but concluded in the DSER that GE should include a treatment of source term uncertainty as part of its overall analysis of uncertainties because large uncertainties are inherent in these calculations. This was identified as part of DSER Outstanding Item 18, and as DFSER Open Item 19.1.8-1.

The NRC implied in the DSER that GE should perform a full, statistical uncertainty analysis on a par with the methods used in NUREG-1150. However, the NRC has subsequently reconsidered and modified this position. In lieu of a full quantitative uncertainty analysis for the Level 2 portion of the PRA, the NRC and GE agreed to the following: (1) the implementation of a systematic process for identifying issues and phenomena of greatest risk significance for the advanced design, and (2) a more thorough treatment of those issues and phenomena and their associated uncertainties as part of the Level 2 analysis (e.g., in the CETs). GE has followed the agreed upon approach in performing the updated ABWR Level 2 analyses. As the first step, GE performed a systematic review of results of recent PRAs and severe accident analyses related to BWRs, with the goal of identifying key phenomena of relevance to the ABWR, and its associated uncertainties. Issues which could have a significant impact on the severe accident performance of the ABWR were singled out for further evaluation and sensitivity analyses. Uncertainty was addressed by GE by performing additional sensitivity analyses, treating the issue in a Decomposition Event Tree (DET), or both. Issues considered to be of greatest potential risk significance for the ABWR were treated by using DETs, which in turn were linked to the updated CETs. This included the issues of DCH, CCI and suppression pool bypass. With regard to source term uncertainties, GE identified a number of in-vessel issues which contribute to uncertainties in source terms for the ABWR. GE provided the results of sensitivity analyses in which they evaluated the effects of changes in these models/assumptions on the magnitude and time of fission product release. Based on the results of these analyses, GE concluded that the in-vessel issues either are conservatively represented in the base case ABWR (MAAP 3.0B) analyses, or do not have a significant affect on ABWR severe accident performance. On this basis, GE did not investigate any of the in-vessel issues further as part of CET/DET or source term analysis. A number of ex-vessel/containment performance issues which impact source terms were also identified through the issue screening process. As mentioned previously, several of these issues (direct containment heating, core concrete interactions, and suppression pool bypass), were addressed by GE explicitly within the CETs using decomposition event trees. For the issues of drywell head failure area, suppression pool decontamination factors, and source terms from core concrete interactions, GE performed additional sensitivity analyses. On the basis of these analyses GE concluded that these issues do not have a significant impact on ABWR risk, and therefore do not warrant further consideration.

The NRC is in general agreement with GE's conclusion that the ABWR risk profile is not significantly changed by variations in the aforementioned models/assumptions with one major exception, namely, suppression pool decontamination factors. To address this concern, the NRC has based its independent assessment of ABWR risk on lower suppression pool decontamination factors. In view of the NRC's use of its own more conservative assumption regarding suppression pool decontamination factors, and the results based on this assumption, the NRC considers the open item regarding source term uncertainty for the ABWR to be resolved (DFSER Open Item 19.1.8-1, and the source term portion of DSER Outstanding Item 18).

The NRC considers the overall approach taken by GE for addressing uncertainties (e.g., use of DETs for selected key issues) to be consistent with the objectives of the ALWR PRA, and therefore acceptable. The NRC acknowledges that while GE's approach allows the analyst to consider a range of issue outcomes within the CETs, the approach simply provides more justification for a single point estimate based on expanded event trees, rather than a complete uncertainty analysis. As a result, no statistical information on the uncertainty ranges is included in the CET quantification. The NRC recognizes this as a limitation of the analysis, but considers the approach to be acceptable for meeting the objectives established for the ALWR PRA. The implementation of this approach by GE, as described in previous sections of the FSER, resolves the open item concerning uncertainty in Level 2 analyses (DFSER Open Item 19.1.7.7-1, and DSER Outstanding Item 18).

Open Item 19.1.10-1 in the DFSER: In the DSER, the staff compared the integrated risk results for the ABWR to NRC quantitative health objectives and safety goals, NRC and EPRI requirements/goals for ALWRs, and GE's design goal. Based on these comparisons, the staff reached tentative conclusions on how the various objectives and goals were met. Subsequent to the DSER, GE modified the ABWR design, and submitted the results of the updated ABWR PRA (Levels 1, 2, and 3) reflecting modifications to the plant design, as well as modeling enhancements and corrections identified by GE and staff since the original PRA. The staff committed to report the results of its review of the integrated risk estimates in the FSER. This was identified as Open Item 19.1.10-1 in the DFSER. The staff's evaluation of GE's updated PRA analyses is presented in portions of FSER Section 19.1. Based on review of the updated PRA analysis, the staff has performed a limited update of the comparisons to the safety goals. These comparisons are presented in FSER Section 19.1.3.8.3. This resolved DFSER Open Item 19.1.10-1.

Open Item 19.1.11.4-1 in the DFSER: see Open Item 19.1.5.2-4 above.

Confirmatory Item 19.1.5.8-1 in the DFSER: In the DSER (Section 19.3.9), NRC noted some potential inconsistencies with regard to the classes to which certain sequences were assigned. In particular, in classifying the end-states of the functional event trees, GE had grouped some sequences with loss of coolant inventory makeup into Class IC (ATWS events with failure of boron injection and loss of high pressure coolant makeup), even though the amount of heat dumped to the suppression pool in these two types of sequences is significantly different. The inconsistency in accident classification was in fact an inconsistency in applying the accident class definition rather than a problem with the definition itself. The problem was limited to a misclassification by GE of two end-states in the ATWS tree as class IC, even though SLCS injection is successful. The NRC has reviewed this aspect of GE's updated analysis and finds that this inconsistency is still present in the revised analysis. However, the frequency of the end-states in question are on the order of 1E-14 per year and are insignificant to the core damage frequency and risk for ABWR. This resolved Confirmatory Item 19.1.5.8-1.

Confirmatory Item 19.1.7.5-1 in the DFSER: The staff indicated in the DFSER that GE intends to use concrete with a low rate of non-condensible gas production (i.e., basaltic or similar concrete) in the construction of the ABWR lower drywell in order to delay the time of COPS actuation, and that a Tier 1 requirement should be established to specify the type of concrete to be used. In Section 19EC.6.1 of the SSAR, GE indicated that basaltic concrete will be required for the floor of the lower drywell, but that the type of concrete to be used in the reactor pedestal will not be specified. GE noted that if non-basaltic (e.g., limestone) concrete is used for the pedestal, the sidewall erosion rate will be slower than presented in the revised SSAR analyses and the time of pedestal failure later. GE acknowledged that the non-condensable gas generation from the non-basaltic concrete in the pedestal would be higher, but that this would not have a significant impact on the containment response since the surface area of the sidewall will be only 10 to 15 percent of the floor area and because gases released from the sidewall will not fully participate with the debris pool. The NRC agrees with GE's rationale. Based on GE's commitment to specify that basaltic concrete be required for the floor of the lower drywell, the NRC considers DFSER Confirmatory Item 19.1.7.5-1 to be resolved.

Confirmatory Item 19.1.8-1 in the DFSER: In the DSER, the staff raised a concern regarding whether pool swell and flashing associated with COPS actuation would adversely impact fission product releases for the ABWR. The staff indicated that GE had made additional calculations showing that suppression pool flashing following venting does not lead to significant fission product releases, however, GE had not provided these calculations at the time of the DSER. This was identified as DSER Confirmatory Item 8 and DFSER Confirmatory Item 19.1.8-1. GE subsequently provided their calculations regarding pool swell and flashing. These calculations address both the impact of aerosol entrainment on fission product releases for vented sequences, as well as the potential effects of pool swell on COPS operability. GE's modelling approach for estimating the impact of pool flashing on fission product release consists of an equilibrium flashing model for depressurization, and a model for entrainment from the liquid pool by the vapor passing through the pool. GE estimated the fraction of fission products that can be carried out the COPS and concluded that suppression pool flashing will have no significant impact on fission product releases. GE calculated the pool swell height for limiting suppression pool conditions using a drift flux model, and estimated that the maximum pool swell height would be about 2 meters below the bottom of the COPS penetration. GE also compared the swell height obtained from the drift flux model to the results from a TRAC study for a Mark II containment, and found the drift flux model to be conservative. On this basis, GE concluded that pool swell will have no impact on COPS operability/survivability.

Based upon an examination of the available data, the NRC concludes that the aerosol entrainment function (ratio of flowrate of liquid mass entrained to flowrate of gas mass passing through the pool surface) could be significantly higher than the value predicted by GE. However, the impact of aerosol entrainment on the GE and NRC risk estimates is not significant in either case because the maximum quantity of fission products that could be released from the pool as aerosols during the flashing process is expected to be less than the fission product release fractions used to represent vented sequences in the PRA. As such, the impact of pool flashing on the ABWR source terms for vented sequences is considered to be bounded by the NRC-adjusted source term for vented sequences.

With regard to the impact of pool swell on COPS operability, the staff notes that GE did not consider the suppression pool conditions associated with Class II sequences in estimating maximum possible pool swell height, and that these conditions would be more limiting than those evaluated by GE. Specifically, the initial pool temperature used in GE's calculation (280°F) is based on the frequency-dominant core damage sequence LCLP, and is about 50°F below the saturation temperature at the COPS setpoint. In contrast, the suppression pool is at the saturation temperature in Class II sequences. The higher pool temperature would result in a higher maximum pool swell height and an increased potential for introducing liquid into the COPS line. The staff has considered the impact of a higher suppression pool temperature on risk for the ABWR and concludes that total risk would not be significantly increased. In order to impact risk, three conditions most occur: (1) pool swell must be sufficiently high to introduce liquid into the COPS line, (2) the COPS line must fail as a result of dynamic forces associated with the transport of liquid in the line, and (3) injection to the core must fail as a result of COPS failure. The staff considers the conservatisms in GE's drift flux model and the margins calculated for the less limiting pool conditions to be sufficiently large that if pool swell were recalculated for saturation conditions using more realistic models, there would still be remaining margin before significant amounts of liquid would be swept into the system. Due to the gradual nature of the depressurization process, the liquid that would enter the COPS would be in the form of a froth rather than slug, and would tend to be swept out of the system by the high velocity gas flow, without adverse consequences. Finally, the probability of losing injection to the core as a result of COPS failure would be very small in the ABWR since all safety related equipment located in the reactor building, including HPCF pumps, will be qualified to operate in a steam environment at 15 psig. On this basis, the NRC considers DFSER Confirmatory Item 19.1.8-1 and DSER Confirmatory Item 8 to be resolved.

Confirmatory Item 19.1.9-1 in the DFSER: Subsequent to issuance of the DSER, GE modified the ABWR design and updated the Level 2 portion of the PRA. This resulted in a change to the estimated source terms for each ABWR release class or case, and their respective frequencies. Additionally, GE identified an error in the way that weather data was read into the CRAC2 computer code used for the original consequence calculations. As a result of this error, wind speeds were systematically overestimated and offsite consequences underestimated in the ABWR results reported in the DSER. GE committed to correct this error in the revised consequence calculations that accompany the updated PRA. This was identified as DFSER Confirmatory Item 19.1.9-1. The results of GE's updated consequence analyses were submitted in Amendment 28 to Section 19E.3-1 of the SSAR. Additional information regarding the source terms and release class frequencies which form the basis for the updated consequence analyses was provided in a July 9, 1992 fax from GE. A copy of the CRAC-2 computer code input and output files were also submitted by letter dated October 28, 1992, and are documented in Section 2A of Amendment 30 to the SSAR.

The staff has compared GE's estimates of person-rem exposure per event with offsite consequences independently calculated using the MACCS code. The results are comparable with one major exception, i.e., the NRC's dose estimate is significantly higher for sequences with containment venting. Additional calculations were performed by the staff using both the CRAC2 and MACCS codes to explore the underlying reasons for the differences for the vented case. These analyses explored the sensitivity of results to: release fractions, presence of refractory materials, time of release, release duration, energy of release, and meteorological data. The staff concludes that for the same source term, the differences noted between GE and staff estimates are attributable to the differences in the consequence code used. Based on discussions with GE personnel and review of the additional information and analyses, it appears that the errors in the original GE analysis have been rectified, and are not a contributing factor. This resolved the Confirmatory Item 19.1.9-1.